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Docket Nos. 50-458/459

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

Dear Mr. Cahill:

Subject: Draft SER - River Bend Station, Units 1 and 2) (RBS-1, RBS-2)

Enclosed is a copy of the draft SER for the River Bend Station. The draft is incomplete, as indicated in the index of the SER. We will transmit to you the remaining portions as they become available.

Please review the draft carefully and prepare your responses to the open items, which are described in sufficient detail in the text of the draft. There are four categories of these open items:

(a) Staff review is stopped pending additional information from Gulf States Utilities Company (GSU).

(b) Staff review is complete. Issue will be closed when GSU amends the FSAR.

(c) Staff review is complete but no resolution has been reached between GSU and the staff.

(d) Staff review is complete and the findings will be reported at a later date.

Please prepare a detailed agenda aimed at resolving these open items. We are prepared to meet with you as necessary to discuss and resolve these items. Meetings between GSU and the staff to date have proven extremely effective in clarifying and resolving open items. We would prefer a draft response to each of these items prior to the meetings.

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If you have any questions, please contact NRC Project Manager, Edward Weinkam.

Sincerely,

Original signed by

T. M. Novak, Assistant Director
for Licensing
Division of Licensing

Enclosure:
As stated

cc w/enclosure:
See next page

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2 SITE CHARACTERISTICS

2.1 Geography and Demography

The River Bend site characteristics were reviewed in accordance with SRP 2.1.1, 2.1.2, and 2.1.3 (NUREG-0800).

2.1.1 Site Location and Description

The River Bend nuclear plant is located on a 3342-acre site in West Feliciana Parish, on the east bank of the Mississippi River, approximately 24 miles north-northwest of Baton Rouge, Louisiana. Figures 2.1 and 2.2 show the general region of the River Bend site; Figure 2.3 shows the immediate site area.

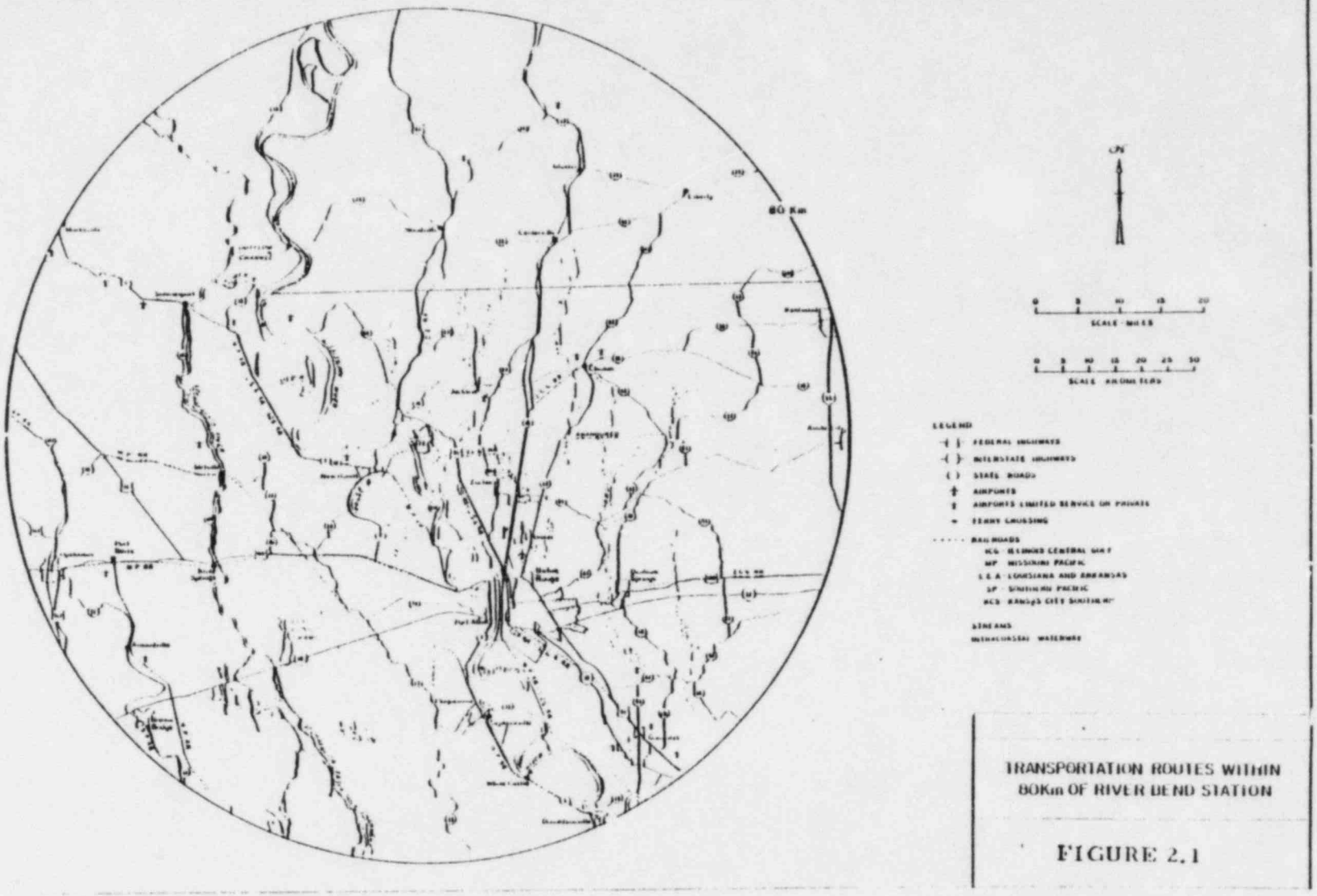
The site is a combination of heavily forested land and open fields with two general levels of elevation. The lower level, about 35 ft mean sea level (msl), is an alluvial flood plain along the banks of the Mississippi River. The upper level, about 100 ft msl, is a terrace on which the reactor and auxiliary buildings were constructed.

The coordinates of the River Bend Unit 1 are 30°45'26" north latitude and 91°19'64" west longitude. The universal transverse mercator coordinates are 3,403,705 m Northing, 659,678 m Easting.

2.1.2 Exclusion Area Authority and Control

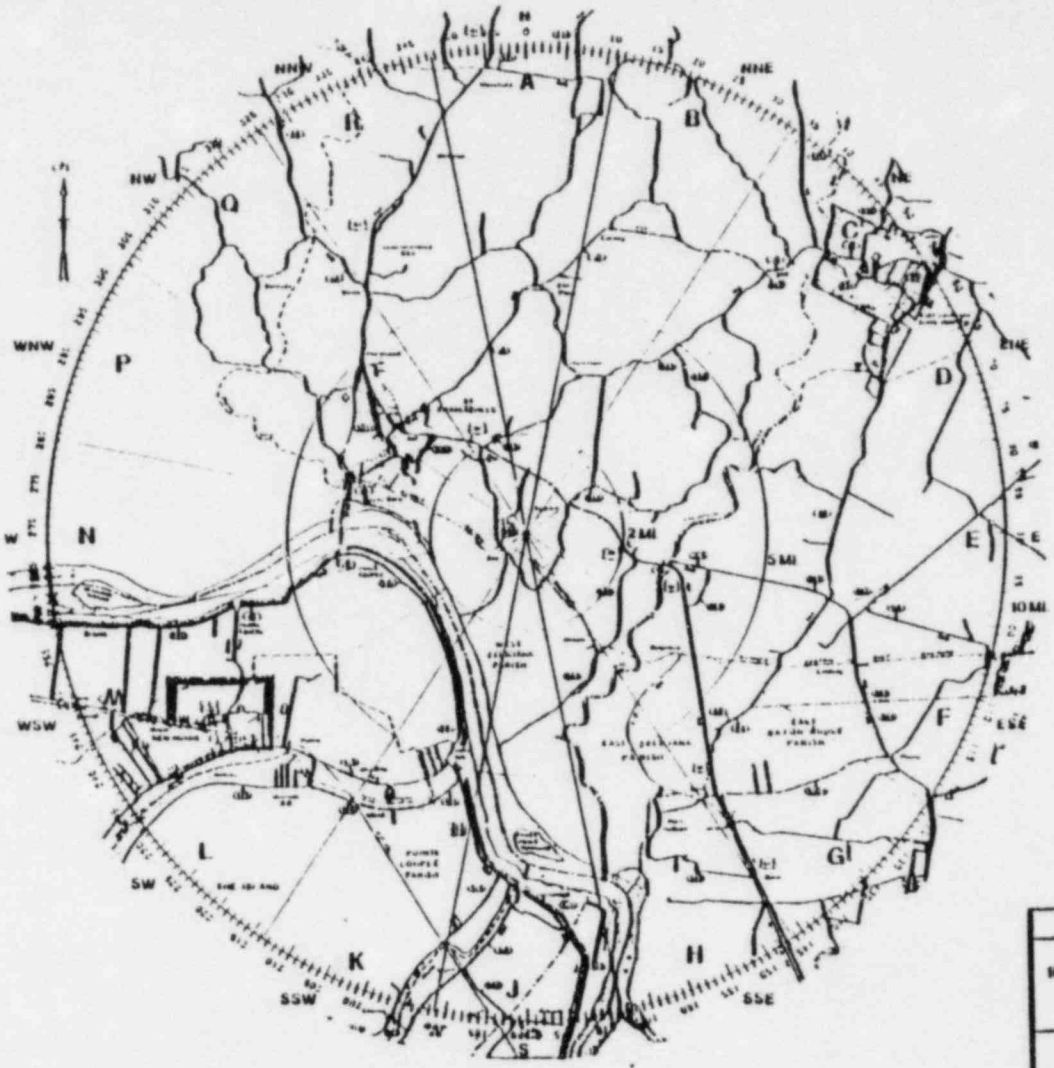
9/4

The applicant has defined the exclusion area as a circular area within a 3000-ft (-m) radius from the reactor to the site boundary. The plant boundary and the exclusion area are shown in Figure 2.4. The entire exclusion area, including mineral rights, is owned by the applicant. There are no residents or recreational areas within the exclusion area. The exclusion area is traversed by two roads, a railroad, and a railroad spur line. The applicant



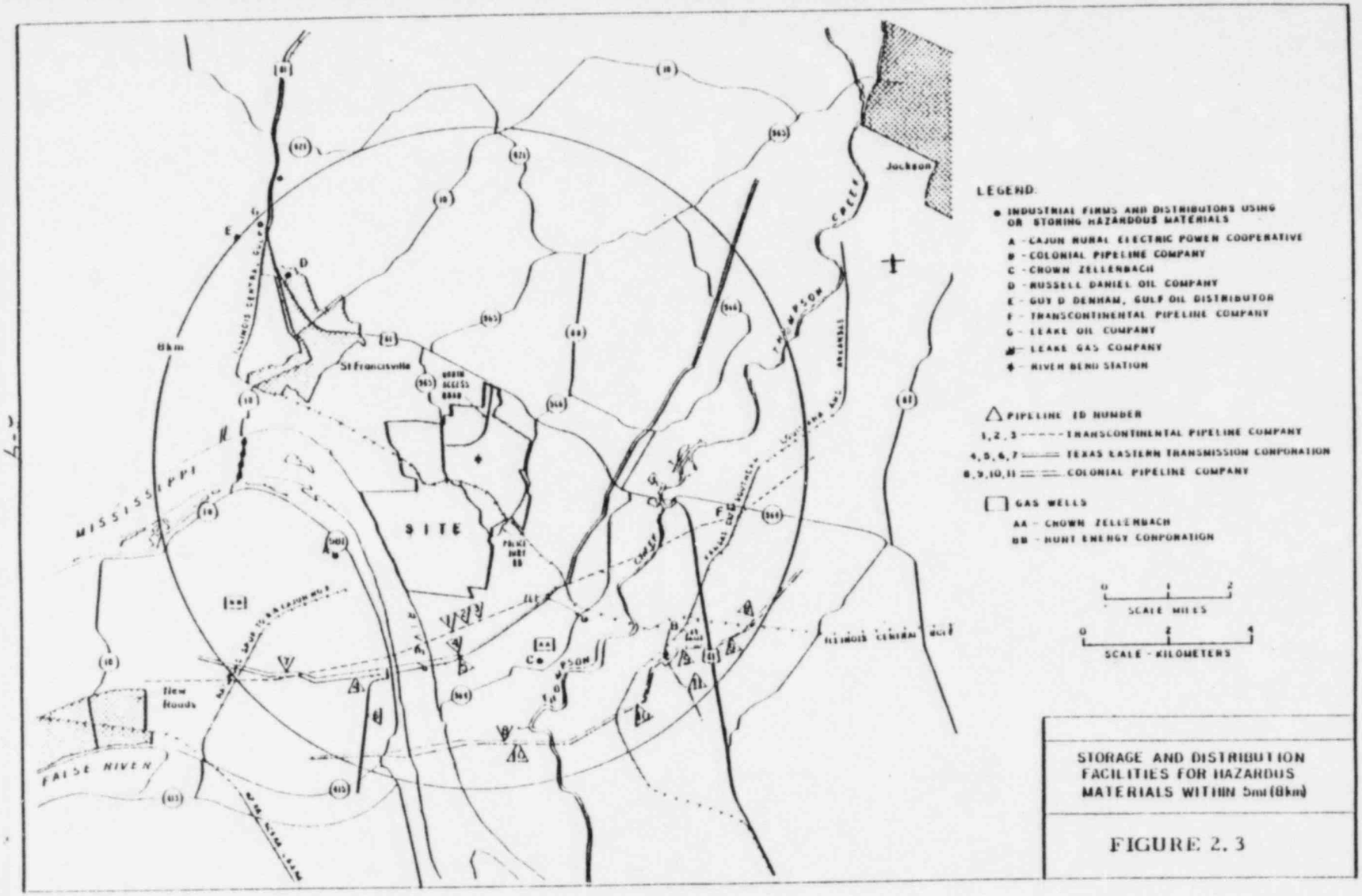
2.2

Figure 2.1 Transportation routes within 50 mi of the River Bend Station



10 MILE RADIUS-RIVER BEND STATION
SITE AND LOCAL AREA

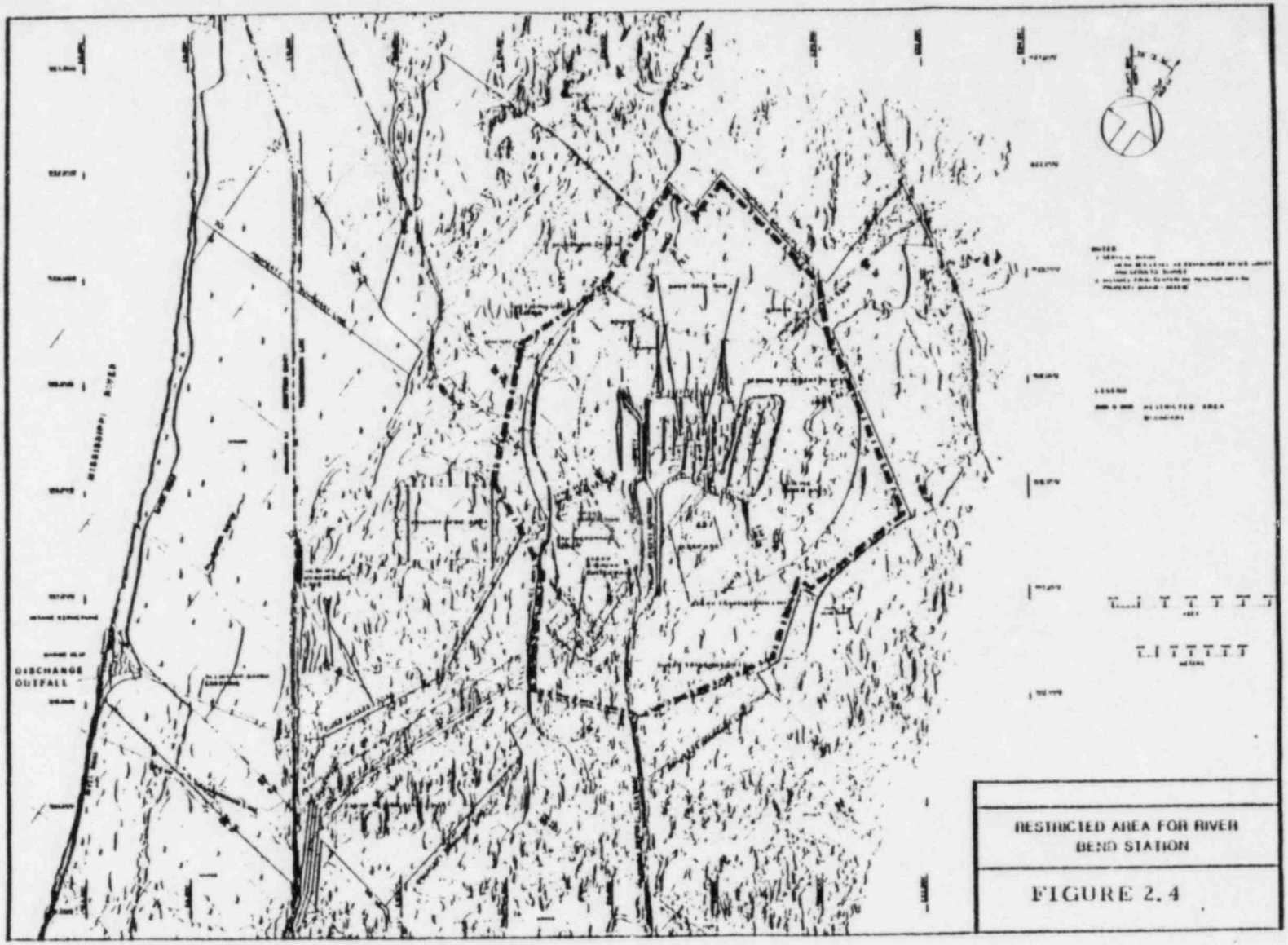
FIGURE 2.2



STORAGE AND DISTRIBUTION FACILITIES FOR HAZARDOUS MATERIALS WITHIN 5mi (8km)

FIGURE 2.3

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RESTRICTED AREA FOR RIVER BEND STATION

FIGURE 2.4

has established adequate emergency plans to control traffic on these routes in the event of an emergency. Within the property owned by the applicant there is a 75,000-ft^2 parcel of land owned by AT&T, which is used for a microwave tower. The tower (shown in Figure 2.4) is, however, located outside of the exclusion area boundary for the River Bend station.

On the basis of the ownership of the land within the exclusion area, including the mineral rights, and because suitable arrangements have been made to control traffic on the roads and railroads, the staff concludes that the applicant has the authority to control all activities, as required by 10 CFR 100.

2.1.3 Population Distribution

The River Bend site is located in a low population density area, and there are no residences within a 0.75-mile radius of the reactor buildings.

The largest community within 5 miles is St. Francisville, Louisiana, which had a 1980 population of 1284, a decrease from the 1970 population of 1452. St. Francisville is about 3 miles northwest of the site. Within 10 miles, Zachary City is the largest community, with a 1980 population of 7927 persons. Zachary is about 9 miles southeast of the site. The transient population within 6 miles is limited because of the rural character of the area.

The applicant has chosen a low population zone (LPZ) radius of 2.5 miles. The applicant has indicated that the nearest densely populated center of 25,000 or more (as defined by 10 CFR 100) persons is the city of Baton Rouge, Louisiana (1980 population of 219,486), whose nearest boundary is about 16 miles south-southeast of the River Bend site. The population center distance is at least 1-1/3 times the LPZ radius, as required by 10 CFR 100.

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2.1.4 Conclusion

On the basis of the 10 CFR 100 definitions of the exclusion area, low population zone, and the population center distance, the staff concludes that the applicant's values for these items are acceptable.

2.2 Nearby Industrial, Transportation, and Military Facilities

2.2.1 ~~Transportation~~ Routes

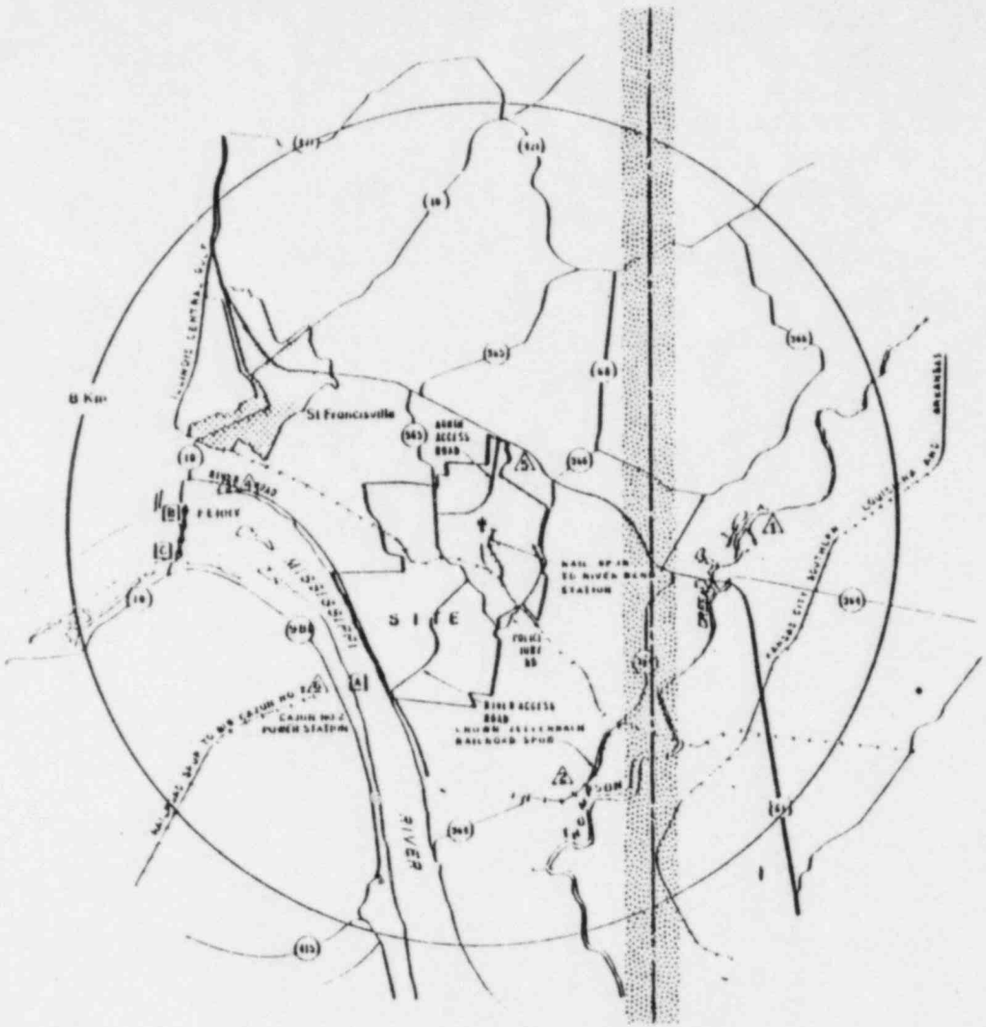
River Bend was reviewed in accordance with SRP 2.2.1, 2.2.2, 2.2.3, 3.5.1.5, and 3.5.1.6 (NUREG-0800). The results of this review are presented below.

The nearest major road to the site is U.S. Highway 61, which passes within 1 mile of the plant in the northern direction. Police Jury Road, a local route, passes through the site and intersects Route 61 in two locations, as shown in Figure 2.5. The daily traffic count for Highway 61 was approximately 7900 vehicles (both directions) in 1978. The other local roads had fewer than 2000 vehicles per day.

The 5000-ft distance from U.S. Route 61 is sufficient to preclude adverse effects on the plant from an explosion-type incident on this route.

The nearest railroad is the Illinois Central Gulf line, which passes through the site. A short spur line connects this line to the River Bend station. This railroad company has indicated that hazardous materials are not normally carried on this section of the line north of Crown Zellerbach (see Figure 2.5). The Missouri Pacific Railroad spur is used to transport materials to the Cajun No. 2 coal-fired generating station on the opposite side of the Mississippi River. This company has indicated that it had transported no hazardous materials in the past year.

The Kansas City, Southern Louisiana, and Arkansas Railroad Company would not indicate what hazardous materials, if any, were being transported on that line. The 4-mile separation distance from the line to the reactor site is sufficient, however, to preclude explosive overpressures or toxic gas releases endangering the nuclear facility.



LEGEND

△ INDUSTRIAL FIRMS
1 HOLLOWAY GRAVEL
2 CROWN ZELLENBACH
3 ARMY CORP OF ENGINEERS - ST FRANCISVILLE CASTING YARD
4 JOAN OF ARC COMPANY
5 LA[illegible] HOT-MIX
6 BIG CAJUN NO. 2 POWER PLANT

♦ RIVER BEND STATION

[ ] BARGE FACILITIES AND FERRY LANDINGS
A - BIG CAJUN NO 2 POWER PLANT
B - NEW ROADS FERRY - (WEST FELICIANA PARISH)
C - ST FRANCISVILLE FERRY - (POINTE COUPEE PARISH)

F A A VICTOR AIRWAY - V 71

RAILROAD
KCS KANSAS CITY SOUTHERN
L & A LOUISIANA AND ARKANSAS
MP MISSOURI PACIFIC
ICG ILLINOIS CENTRAL GULF

SCALE - MILES
0 1 2

SCALE - KILOMETERS
0 1 2 3 4

INDUSTRIAL FIRMS AND MAJOR TRANSPORTATION ROUTES WITHIN 5mi (8km)

FIGURE 2.5

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The Mississippi River, 2 miles from the site, is used for commercial barge traffic. In 1977, approximately 124 million tons of cargo were transported between Baton Rouge and the Ohio River. It is estimated that about 1/3 of this cargo is potentially hazardous material. The 2-mile distance between the plant and the river offers sufficient protection against an explosion-type incident during barge transportation.

The plant control room is provided with a chlorine gas detector to isolate chlorine in the event of transportation accidents (barges, trucks, railcars) or a release from storage at either the power station or the Crown Zellerbach plant within 5 miles of the site. The applicant has indicated that chlorine is the only toxic material in the vicinity of the site that represents a design basis for control room design features. However, the staff has determined that ammonia is currently being shipped in barges past the Grand Gulf site, upstream from the River Bend facility. Hence, the staff has requested that the applicant investigate and evaluate barge transportation of ammonia past the River Bend site. This is an open item.

Although the River Bend plant intake structure is located in an embayment, away from the main river channel, barge traffic potentially could enter this area. However, damage to the intake structure would not result in the loss of cooling water because there is sufficient water in the cooling towers to permit a safe shutdown without adding river makeup water. The accidental upstream release of oil or cryogenic liquids into the river will not present a hazard to plant cooling because the intake screens are located more than 10 ft below mean low water level and more than 20 ft below normal water level.

Any spill of corrosive materials into the river would be detected by pH detectors in the circulating water system. If required, the standby cooling towers could be used or safe shutdown.

2.2.2 Nearby Facilities [redacted]

There are no chemical plants, refineries, or military bases within 5 miles of the River Bend generating station. The principal industries within 5 miles of River Bend include the Big Cajun No. 2 power plant (coal fired), the Crown

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Zellerbach paper mill, and the Lambert Redi-Mix Company (see Figure 2.5). These plants do not store or produce explosive-type products. The major toxic chemical stored at two of these facilities is chlorine. The reactor control room is equipped with chlorine detectors (see SER Sections 6.4 and 2.2.1).

The closest pipelines are two 36-inch-diameter natural gas transmission pipelines that cross in a northeast-southwest direction, approximately 2 miles southeast of the site, as shown in Figure 2.3. The 2-mile distance provides adequate protection for the River Bend site in the event of a gas line break that could result in a fire and/or explosion.

The closest natural gas wells are located more than 3 miles from the nuclear plant. This distance provides adequate protection in the event of a fire and/or explosion at these sites.

Because of current oil and gas exploration in the region of the plant site, the staff will require the applicant to inform the Commission of any new wells and/or pipelines that may be located near (or within) the exclusion area boundary of the River Bend nuclear facility.

The nearest commercial airport to the River Bend station is the Ryan Airport, located at Baton Rouge, approximately 19 miles southeast of the site. There are no small public or private airports within 5 miles of the River Bend station. The applicant indicates that there is no military airspace use that might present a hazard to the River Bend plant. The staff confirmed this by checking the latest (December 1982) Department of Defense (DOD) area planning chart showing military training routes for the eastern United States. The chart indicates that there are no military training routes within at least 20 miles of the River Bend station.

The only commercial route within a 5-mile radius is Federal Airway V71, located approximately 2.5 miles east of the site. The centerlines of V222 and V114N pass 7 miles northwest of the site and 8.5 miles northeast of the site, respectively.

At the request of the staff, the applicant examined the traffic along these routes and performed an analysis of the probability of an aircraft impacting the plant. As a result of this analysis, the applicant concludes that the probability of an aircraft crash causing radiological consequences in excess of the guideline values in 10 CFR 100 is within the acceptance criteria of SRP 2.2.3 (NUREG-0800) (less than about 10^{-7} per year), and is, therefore, acceptable. Based on its review of this analysis, the staff concurs with this conclusion.

2.2.3 Conclusions

The staff review of the River Bend FSAR has been based on the criteria in GDC 4 and in SRP 2.2.3. The staff concludes, subject to satisfactory resolution of the issue regarding possible barge shipment of ammonia past the site, that the plant is adequately protected and can be operated with an acceptable degree of safety as a result of activities at nearby transportation, industrial, and military facilities.

2.3 Meteorology

Evaluation of regional and local climatological information, including extremes of climate and severe weather occurrences that may affect the design and siting of a nuclear plant, is required to ensure that the plant can be designed and operated within the requirements of Commission regulations. Information concerning atmospheric diffusion characteristics of a nuclear power plant site is required for a determination that radioactive effluents from postulated accidental releases, as well as routine operational releases, are within Commission guidelines. Sections 2.3.1 through 2.3.5 have been prepared in accordance with the review procedures described in the SRP, utilizing information presented in FSAR Section 2.3, responses to requests for additional information, and generally available reference materials as described in the appropriate sections of the SRP.

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2.3.1 Regional Climatology

The climate of east central Louisiana, where the plant is located, is described as humid subtropical, having generally warm temperatures, with an annual average of near 70°F (21°C) (Nat'l Clim. Ctr., 1973, 1979).

Average monthly temperatures in the area range from a low of 40°F (4°C) in January to a maximum in the 90°F (32°C) to 100°F (37°C) range during July. Precipitation, primarily rain, averages more than 54 inches (1372 mm) a year in this part of the state. The warm, moist, unstable air that covers the region produces showers and thundershowers of short duration throughout the year. In addition, the passage of frontal systems and tropical storms and hurricanes from the Gulf of Mexico contributes to the annual precipitation. Average snowfall totals less than 1 inch a year.

Thunderstorms occur in every month, with the average number of thunderstorm days per year more than 90. Hail with the thunderstorms is observed 1 to 2 days a year (Baldwin, 1973).

Lightning strikes to plant structures are possible, and their frequency can be determined as a function of the number of thunderstorm days and plant structure dimensions (Marshall, 1973). A recent study (NUREG/CR-2252) determined an annual average of 93 thunderstorms at Baton Rouge, based on data from the period 1948-1977. Following methods described in Marshall (1973), it was estimated that lightning may strike the plant every 176 days.

In addition to lightning, thunderstorms may produce tornadoes, as evidenced by a total of 304 statewide tornadoes during the period 1953 to 1971, an average of 16 per year (Baldwin, 1973). In the site vicinity (a 1-degree square), a toranado may be estimated every 3415 years, using methods outlined in WASH-1300. The design tornado for the plant is that given in RG 1.76 for Region I, with a wind speed of 360 mph (160.9 m/s), total pressure drop of 3 psi (0.21 kg/cm^2), and a rate of pressure drop of 2 psi/sec (0.14 kg/cm^2/sec). Large-scale air pollution episodes are not a problem in the area. Studies by Holzworth (1972) indicate there were about 10 days a year of high air pollution potential forecasts during the period 1960 to 1970.

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Winds are principally from the southeast during the year, with lesser frequencies from the north and northeast. Average monthly wind speeds are 10 mph (4.5 m/s) or less. Heavy fog that reduces visibility to 0.25 mile (400 m) or less occurs about 40 days a year.

As discussed above, the staff has reviewed available information relative to the regional meteorological conditions of importance to the safe design and siting of this plant. Based on this review, the staff concludes that the applicant has identified and considered appropriate regional meteorological conditions in the design and siting of this plant, and therefore, meets the requirements of 10 CFR 100 and GDC 2.

2.3.2 Local Meteorology

Meteorological measurements made at the National Weather Service office in Baton Rouge about 20 miles (32 km) southeast of the site provide information on normal and extreme meteorological events that represent conditions that effect the site.

Values of normal and extreme meteorological parameters measured at Baton Rouge are shown in Table 2.1.

Average monthly relative humidity ranges from a low of 59% at noon to a high of 88% at 6 a.m. The humidity data (wet bulb - temperature) from 1949 to 1979 at Baton Rouge were used to assume the design characteristics for the ultimate heat sink (mechanical draft cooling tower) for the immediate post-accident and subsequent 30-day periods in accordance with RG 1.27, [redacted].

As discussed above, the staff has reviewed availble information relative to local meteorological conditions of importance to the safe design and siting of this plant. Based on this review, the staff concludes that the applicant has identified and considered appropriate local meteorological conditions in the design and siting of this plant, and therefore, meets the requirements of 10 CFR 100.10 and GDC 2.

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Table 2.1 Mean and extreme meteorological values at Baton Rouge, Louisiana*

Parameter	Value
Mean	
Annual temperature, F (°C)	67.4 (19.7)
Annual precipitation, in. (mm)	54.05 (1372)
Annual snowfall, in. (mm)	0.2 (5.1)
Wind speed, mph (m/s)	7.9 (3.5)
Wind direction	southeast
Extreme	
Temperature, F (°C)	
Highest	100 (39)
Lowest	10 (-12)
Maximum monthly precipitation, in. (mm)	14.5 (368)
Minimum monthly precipitation	trace
Maximum monthly snowfall, in. (mm)	1.8 (45.7)
Fastest mile wind speed, mph (m/s)	58 (25.9)
Wind direction	east-northeast

*through 1979

2.3.3 Onsite Meteorological Measurements Program

A 150-ft tower on the site provides measurements of wind speed, wind direction, and dew point at the 30- and 150-ft levels, and the vertical temperature difference between the 30- and 150-ft levels. Ambient temperature is measured at the 30-ft level, and precipitation is measured 15 ft above ground level, on top of the meteorology instrument building. The onsite measurement program began in October 1972 and, following installation of new sensors in March 1977, the present program has produced data that are used in the 2-year joint frequency distribution from March 1977 to March 1979 to evaluate the consequences of accidental and routine gaseous releases. The measurements of wind speed and direction at the 30- and 150-ft levels as well as vertical temperature difference between these levels are transmitted to recorders in the plant control room.

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The emergency response meteorological program responsive to NUREG-0654 will be reviewed in a supplement to this SER. The onsite meteorological measurements system conforms to RG 1.23 and has provided adequate data to represent onsite meteorological conditions as required in 10 CFR 100.10. The onsite meteorological data provide an acceptable basis for making conservative estimates of atmospheric dispersion conditions used for estimating consequences of accident and routine gaseous releases to offsite locations.

2.3.4 Short-Term (Accident) Diffusion Estimates

Short-term (up to 30-day) accident releases from buildings and vents were evaluated by the applicant and the staff using the direction-dependent atmospheric dispersion model described in RG 1.145 with consideration of increased lateral dispersion during stable atmospheric conditions accompanied by low wind speeds. The staff used the onsite joint frequency distribution of wind speed by direction at 30 ft by atmospheric stability class from March 1977 to March 1979; a Gaussian dispersion model was evaluated at the 0.5-mi (894-m) exclusion area boundary (EAB) and 2.5 mi (4003-m) outer boundary of the low population zone (LPZ). As a result of this staff evaluation, the relative concentration (χ/Q) value for a 0 to 2-hour time period at the EAB, which is expected to be exceeded less than 0.5% of the time, was estimated to be 6.2×10^{-4} sec/m^3 in the west EAB direction. This direction was estimated to produce the most conservative χ/Q. The estimated χ/Q values at the LPZ distance for different time periods are given in Table 2.2. It should be noted that the results determined by both the applicant and the staff are generally comparable.

Table 2.2 River Bend LPZ (2.5 mi) relative concentration (χ/Q)

Time period	χ/Q (sec/m^3)
0-8 hours	7.4×10^{-5}
8-24 hours	4.9×10^{-5}
1-4 days	2.0×10^{-5}
4-30 days	5.7×10^{-6}

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Credit for building wake effect is given, along with lateral plume meander under the stable and neutral stability conditions during periods of light wind. The staff-calculated atmospheric dispersion estimates in this section are appropriate for assessments of the consequences of radioactive releases for design-basis accidents, in accordance with the requirements of 10 CFR 100.11.

2.3.5 Long-Term (Routine) Diffusion Estimates

Routine gaseous releases from the River Bend plant vent are evaluated at a number of points of interest around the plant. This evaluation was done using the onsite meteorological data collected from March 1977 to March 1979. The data were used by the applicant and staff with a straight-line transport and diffusion model described in RG 1.111. The relative concentration and relative deposition from normal continuous and periodic releases were determined at nearby receptor locations. The staff results of these calculations are given in Table 2.3. The results determined by both the staff and applicant are comparable. Therefore, the staff has confirmed the applicant's assessment of representative atmospheric dispersion estimates for demonstrating compliance with the numerical guides for doses contained in 10 CFR 50, Appendix I.

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2.4 Hydrologic Engineering

[redacted]

The staff has reviewed the hydrologic engineering aspects of the applicant's design, design criteria, and design basis of safety-related facilities for River Bend Station Units 1 and 2. The acceptance criteria used as a basis for the evaluations are set forth in the SRP (NUREG-0800) in sections 2.4.1 through 2.4.14 (Hydrologic Engineering). These acceptance criteria include the applicable General Design Criteria (10 CFR 50, Appendix A), Reactor Site Criteira (10 CFR 100), and Standards for Protection Against Radiation (10 CFR 20, Appendix B, Table II). Guidelines for implementation of the requirements of the acceptance criteria are provided in Regulatory Guides, ANSI Standards, and Branch Technical Positions identified in Sections 2.4.1 through 2.4.14 of the SRP. Conformance to the acceptance criteria provides the basis for concluding that the site and facilities meet the requirements of Parts 20, 50, and 100 of 10 CFR with respect to hydrologic engineering.

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2.4.1 Hydrologic Description

The River Bend Station site is located on the east bank of the Mississippi River at about River Mile 262, approximately 24 miles north-northeast of Baton Rouge, Louisiana. The station buildings and safety-related equipment are located on a bluff above the river flood plain. The alluvial flood plain at the base of the bluff varies from about 3000 ft to 4000 ft wide with an elevation of about 35 ft MSL. The finished plant grade will be 95 ft MSL. All safety-related equipment in buildings not sealed from floodwater are at a minimum elevation of 98 ft MSL. The plant layout is presented in Figure 2- .

The principal hydrologic features in the vicinity of River Bend Station are the Mississippi River and its tributaries. The area within the site boundary is drained by Grants Bayou on the east and Alligator Bayou on the west. The station buildings are located on the divide between Grants Bayou and one of its tributaries, West Creek. Grants Bayou enters Alligator Bayou to the south of the site. Alligator Bayou then flows into Thompson Creek, which enters the Mississippi River approximately 7 miles downstream of the River Bend Station embayment. Flood potential at the site due to West Creek, Mississippi River, and Grants Bayou was evaluated.

During normal operations, cooling water for River Bend Station is provided by the circulating water system -- a pumped closed-loop system utilizing four mechanical draft cooling towers as a heat sink. Emergency cooling is provided by a standby mechanical draft cooling tower and cooling water storage basin for each unit.

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Significant hydrologically related site and plant features are discussed in the following subsections:

2.4.1.1 Mississippi River and Tributaries

The Mississippi River and its tributaries have many flow control structures that affect river stage and discharge of the Mississippi River at the site location. The primary structures are the levee embankments and the floodway system which includes the Old River Diversion Structure for the Atchafalaya Floodways.

The levee system along the west bank (across the river from the site) extends almost uninterrupted from Cape Girardeau, Missouri (mile 956) to the Gulf of Mexico. On the east side of the river at the River Bend site only natural levees exist along the river bankline. These natural embankments, extending about 10 to 15 ft above the floodplain, were created by sediment deposits from flood overflows that have occurred over thousands of years.

2.4.1.2 Dams

There are no dams on the Mississippi River between the site and the Ohio River confluence upstream and no dams downstream of the site. The dams nearest the site are located on Indian and Cotile Creeks near Alexandria, Louisiana, more than 100 river miles northwest of the site. These are tributaries to the Red River which does not connect with the Mississippi but discharges its waters into the Atchafalaya distributary system.

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2.4.1.3 Local Streams

The River Bend site lies within the 15.3 sq miles Grants Bayou drainage basin with approximately 8.1 sq miles of the basin upstream from the plant facilities. The nearest stream to the plant facilities is West Creek, a tributary to Grants Bayou, which drains approximately 1 sq mile including the facilities. The only other stream near the plant is Grants Bayou. Flooding of these two streams is the primary concern for the site facilities. Grants Bayou joins Alligator Bayou just south of the Gulf States Utilities property.

The drainage area of the Alligator Bayou system north of the southern Gulf States property line is approximately 30.4 sq miles. Alligator Bayou joins Thompson Creek about 3 miles upstream of its confluence with the Mississippi River.

There are no control structures on Grants Bayou. The reach of West Creek adjacent to the plant facilities has been rerouted and confined in a 2850 ft long Fabriform-lined (grout filled nylon fabric) channel. Flows are directed into the channel through a drop structure at the upstream entrance to the lined channel segment.

2.4.1.4 Lakes

There were originally 24 small farm ponds existing within the site boundary prior to construction. Plant construction has reduced this to 20 ponds with an increase in total pond surface from 28.6 acres to 61.1 acres. The increase is due to the impoundment of the Wildlife Management Lake. Based

on the staff evaluation of the impoundments, failure of pond embankments do not present any significant flood hazards to site facilities.

2.4.2.5 Canal

The only canal at the station site is associated with the cooling water system. The canal serves as a cold water flume connecting the individual cooling tower water storage basins. The flume is non-safety related. If the flume were to fail, cooling water would be provided by the Standby Service Water System. The flume has a rectangular cross-section of reinforced concrete construction which is designed to maintain a water flow velocity from 3 to 5 fps during operation. The water surface elevation during operation will vary from 103 to 106 ft MSL.

2.4.2.6 Water Use

Domestic water use of Mississippi River water within 100 miles is limited to the Bayou Lafourche Fresh Water District. The intake pumps are located at River Mile 175.5, approximately 87 miles downstream of the River Bend effluent discharge location. The pumps convey river water to the Water District at a maximum rate of 254.2 mgd. There is no human consumption of water from any of the local streams.

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There are 74 private water wells, including 6 on the site property within a 2-mile radius of the plant site. Many of the private wells are unused, however, because of the extension of municipal water mains in the early 1970s. The only municipal water supply wells located within a 5-mile radius of the plant site are owned by the St. Francisville water supply system. These wells are between 1500 and 1800 ft deep. Onsite water needs are supplied by two 150 gpm wells that are 1800 feet deep. Water for fire protection is supplied by one 800 gpm well which is 150 feet deep.

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2.4.2 Floods ~~Flood Potential~~

The applicant has investigated the following potential sources of flooding of the River Bend site: Probable Maximum Flood (PMF) on the Mississippi River, seismically induced dam failure floods on the Mississippi River; floods on West Creek; PMF on Grants Bayou; and Local Probable Maximum Precipitation (PMP) on the plant site. We have reviewed the material in the FSAR using procedures described in SRP Sections 2.4.2 - 2.4.7 and agree that these are the only credible potential flood sources for the site.

2.4.3 Probable Maximum Flood (PMF) on Streams and Rivers

2.4.3.1 Mississippi River PMF

The applicant has estimated the Lower Mississippi River PMF event to be 5,500,000 cfs by assuming that the Corps of Engineers project design flood (PDF) of 3,330,000 cfs is 60% of the PMF. The justification for this estimate is based on the fact that the Standard Project Flood (SPF) is normally 40 to 60% of the PMF, the percentage generally increasing with drainage area. The Mississippi River PDF was determined using meteorological assumptions which can be considered more extreme than those for the ~~PDF~~ SPF. Hence the PDF can be conservatively considered to be at least 60% of the PMF.

In addition, the west bank levee grade is 57.5 ft msl. Therefore, the PMF discharge will probably overflow the levee system into the broad floodplain area between the Mississippi River and the Morganza and Atchafalaya floodways. The applicant considers the elevation 57.5 ft msl to be the limiting water level for Probable Maximum Flooding in the Mississippi River. The staff

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has reviewed the analysis based on the procedures in SRP Section 2.4.3 and concludes that the plant meets the guidelines of Regulatory Guides 1.59 and 1.102 and therefore meets the requirements of GDC 2 with respect to runoff produced flooding on the Mississippi River.

2.4.4 Seismically Induced Dam Failures

There are no dams on the main stem of the Mississippi River between the site and the confluence with the Ohio River. All other upstream dams are on tributaries to the Mississippi River and are more than 100 river miles upstream. The combined storage of these dams is small compared to the volume of the PMF hydrograph of the Mississippi River. Because of the diverse locations of the dams a single seismic event could only fail a few of them at most. The staff, in accordance with the procedures described in SRP Section 2.4.4, concludes that upstream dam failures do not present a danger to the site and that the site meets GDC 2 with respect to seismically induced dam failures.

2.4.3.2

2.4.3.3 West Creek Flood Potential

As discussed in Section 2.4.1.3 of the SER, West Creek is a small tributary to Grants Bayou which drains approximately 1 square mile including most of the plant site. The reach of West Creek which originally ran through the plant site has been rerouted and confined in a 2850 ft long Fabriform (grout filled nylon fabric) lined channel. Flows are directed into the channel through a drop structure at the upstream entrance to the lined channel segment.

2.4.3.3 from p 2

2.4.3.4 " p 2

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The applicant evaluated the West Creek flood potential for three conditions; the PMF, 1/2 PMF + OBE, and the 25-yr flood + SSE. The reason for evaluating these combinations was that either the Safe Shutdown Earthquake (SSE) or the Operating Basis Earthquake (OBE) in combination with a flood were assumed to fail the fabriform channel embankment so as to decrease the cross section area of the canal. The embankment was assumed to fail to a 20 H:1V slope.

The channel was assumed to remain intact for the PMF alone. At the construction permit (CP) stage, the applicant determined the 1/2 PMF + OBE to be the most severe condition, producing a maximum elevation of 98.49 ft MSL. For the CP stage, the PMF peak discharge was determined to be 5460 cfs. At that time, the staff agreed that the applicant's analysis was conservative. Since the peak water level was higher than the proposed level to which flood protection was to be provided, i.e., 98 ft MSL, the applicant committed that all appropriate doors would be closed during plant operation.

For the operating license (OL) stage, the applicant determined the PMF to be 4000 cfs. With the modified peak discharge, the applicant determined that the combined 1/2 PMF (2000 cfs) + OBE for West Creek would still produce the most critical flooding condition for the plant facilities. The maximum water surface elevation determined for this event was 95.1 ft MSL when coupled with simultaneous winds of 50 mph. The applicant did not provide supporting arguments or rationale for the reduction in the PMF and 1/2 PMF peak discharges

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from the CP to OL stage analyses. Staff evaluation of the applicant's analysis indicates that the West Creek unit graph provided in the FSAR has a volume of only 0.82 in., an 18% deficiency. The resulting PMF hydrograph, therefore, has a runoff volume equivalent to only 24.6 in. of precipitation, compared to the Probable Maximum Precipitation depth of 30.1 in. The insufficient volume of the unit graph largely accounts for the reduction in peak discharge. Thus, the staff does not accept the design basis maximum flood elevations presented in the FSAR.

During a field reconnaissance trip of January 1982, the Fabriform channel was observed to have accumulated enough sediment to support vegetation growth exceeding the channel top bank elevation in height. This observation indicates that sediment deposition in the Fabriform channel is a factor to be considered in evaluating the West Creek flood potential.

Considering sedimentation effects, the staff reevaluated the applicant's CP stage, PMF, and 1/2 PMF + OBE flood events. Considering the bankline failures that are assumed to occur along West Creek during high flows there would be significant debris accumulation in the West Creek channel. This debris would greatly increase channel roughness conditions. Based on staff analyses, the PMF event would result in the most severe flood conditions producing maximum water surface elevations exceeding 98 ft MSL, excluding wave runup. The deposition of debris could increase the channel bed elevation and the corresponding water surface elevation by more than 1.0 ft. This effect would need to be added to the effects of increased roughness, resulting in stillwater elevations exceeding 99 ft MSL.

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Furthermore, if flood elevations were to exceed the 100 of the railroad track (95 ft MSL), the open excavation awaiting construction of Unit 2 would be effected. This excavation may still be in existance when Unit 1 goes into operation. If West Creek were to overflow its banks, the excavation could be filled with water. This could lead to hydrostatic loads on Unit 1 structures exceeding those of the design basis ground-water level of 70 ft. Based on the above discussion, the staff does not consider the applicant's analysis of the West Creek flood potential to be conservative. The staff cannot conclude at this time that the plant meets GDC-2 with respect to flooding from local streams.

2.4.3.3 Grants Bayou PMF

The PMF peak discharge for Grants Bayou in the vicinity of the plant facilities and upstream of East Fork Grants Bayou was determined to be 24,000 cfs by the applicant. The maximum water level for the PMF on Grants Bayou was determined to occur between sections 10 and 11 (FSAR Figure 2.4-21) where the water level varies from 94.5 to 101.6 ft MSL. Since the adjacent cooling tower yard is at about 104 ft MSL, the Grants Bayou PMF was not a flood design consideration.

The staff has reviewed the applicant's analysis in accordance with the procedures described in SRP Section 2.4.3 and concludes that the plant meets GDC 2 with respect to runoff induced flooding in Grants Bayou.

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2.4.3.4

2.4.5 Effects of Local Intense Precipitation

The applicant evaluated the effect of runoff from the local probable maximum precipitation (PMP) on safety-related structures. The analysis assumed that the site drainage system is inoperable and that runoff from within the site drainage boundary leaves the plant area by passing over the railroad tracks along the east bank of West Creek. Considering flow over the railroad tracks at elevation 95 ft MSL as weir flow, the applicant calculated that a flow of 4900 cfs can pass out of the plant area into West Creek at water elevation 96 ft (1 ft above the top of the track and 2 ft below the minimum safety-related equipment elevation). The 4900 cfs flow is greater than the applicant's CL estimate of the PMF for the entire West Creek drainage area (1.0 sq miles). The applicant therefore concluded that safety-related equipment at elevation 98 ft MSL would not be effected by the site PMP.

Part of the basis for the applicant's analysis, as discussed in Section 2.4.3 of the FSAR, is that there is no flooding condition on West Creek which produces a water elevation above the railroad track (elevation 95 ft MSL); thus, water does not enter the plant area due to stream flooding. The applicant concluded that runoff is not restricted in leaving the plant area by West Creek. The staff position, as discussed in a previous section, is that during the design basis flood, water levels will exceed the elevation of the railroad tracks and water will flow into the plant area from West Creek. During the West Creek PMF or the 1/2 PMF coincident with the OBE seismic event, precipitation falling on the plant area will be superimposed

2.4.3.4 (cont)

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upon overflow from West Creek to produce the maximum water levels at safety-related facilities. Based on our analyses, these levels will exceed 98 ft MSL. Drainage into the Unit 2 excavation, as discussed in a previous section, is also a concern with regard to local intense precipitation. In conclusion, the applicant has not demonstrated that safety-related facilities are adequately protected against effects of the site PMP.

All building roofs are sloped and do not have parapet walls. Therefore, no potential exists for significant ponding on the roofs.

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2.4.4 Seismically Induced Dam Failures

There are no dams on the main stem of the Mississippi River between the site and the confluence with the Ohio River. All other upstream dams are on tributaries to the Mississippi River and are more than 100 river miles upstream. The combined storage of these dams is small compared to the volume of the PMF hydrograph of the Mississippi River. Because of the diverse locations of the dams a single seismic event could only fail a few of them at most. The staff, in accordance with the procedures described in SRP Section 2.4.4, concludes that upstream dam failures do not present a danger to the site and that the site meets GDC 2 with respect to seismically induced dam failures.

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2.4.5 Cooling Water ~~Canal~~ Canals and Reservoirs

2.4.5.1 Normal Cooling Water Supply

The normal cooling for each unit is provided by mechanical draft cooling towers. The source of makeup water for the cooling tower basins is the Lower Mississippi River.

Flow of the Lower Mississippi River is affected by diversion into the Atchafalaya River through the Old River diversion channel near Coochie, Louisiana (approximately 52.5 miles upstream of the plant site). Prior to construction of the Old River Diversion Structure in 1963, the recorded minimum flow for the Lower Mississippi River was 75,000 cfs. With the control of diversions into the Atchafalaya River, flows are not expected to be lower than 100,000 cfs. The estimated Mississippi River stage at the site at the time of the 75,000 cfs flow was 1.4 ft MSL. The top elevation of the river intake screens is -3.5 MSL, 4.9 ft below the minimum observed stage.

The cooling tower makeup water system is designed to deliver a maximum makeup water flow rate of approximately 32,000 gpm for the two station units. The system consists of three pumps each sized for 16,000 gpm. Two pumps are required for normal operations while the third is a spare to accommodate maintenance or failure of either of the two operating pumps. There is no safety-related equipment in the system, nor does loss of its normal operating capability adversely effect any safety-related components.

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Ice does not form in the Mississipot River in the vicinity of the site with recorded water temperatures at St. Francisville (River Mile 266.0) ranging from 34° to 88°F. Thus, adverse effects as a result of low water due to icing are unlikely.

The entrance of the Mississippi River makeup water pumphouse is at elevation 60.5 ft msl, 3 ft above the West Bank levee grade. Therefore, it is highly unlikely that the pumphouse will be affected by any flooding from the Mississippi River.

We have reviewed the information presented by the applicant and conclude that neither low water levels nor floods in the Mississippi River will unusually restrict the availability of cooling water for normal operations as required by GDC 44 (10 CFR 50, Appendix A).

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2.4.5.2 Emergency Cooling Water Supply

Emergency cooling will be provided by one standby seismic Category I mechanical draft cooling tower with an associated cooling water storage basin per each unit.

Each standby cooling tower is 158 ft, 6-in. inside diameter and is supported by a foundation above its storage basin. The structures are tornado missile protected and designed to seismic Category I requirements. The water storage basins are located 60% below grade and the mat and walls are constructed of watertight concrete, designed to prevent seepage. The cooling towers and walls of the water storage basins are above the probable maximum flood level. Each water storage basin has a capacity of 6,500,000 gallons.

The standby service water system is designed to provide sufficient cooling to permit shutdown and cooldown and to maintain reactor return water temperatures below 105°F for 30 days under adverse meteorological conditions when the normal service water system is unavailable.

Each separate cooling tower and basin is designed to provide emergency cooling for 30 days following a design basis accident (DBA) postulated as a double ended break of a reactor recirculation pump suction line coincident with a complete loss of offsite power. The applicant performed an analysis of the emergency cooling system's thermal performance following a DBA. Using the maximum mean 1-day wet bulb temperature over 29 years of data

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(1949-1973), the applicant computed a maximum cooling tower return water temperature of 90.6°F which is well below the maximum allowable return water temperature of 95°F, ~~which~~ required to keep the reactor return water temperature below 105°. The estimated maximum water losses, including drift and natural and forced evaporation during the 30 worst days (for evaporation losses from 29 years of record) is 6,295,000 gallons, leaving 205,000 gallons as a design safety margin.

The makeup water requirement after 30 days of operation is at a maximum rate of approximately 167,900 gal/day. Primary makeup water is provided by 2 wells capable of supplying a combined flow of 300 gpm (432,000 gal/day). Makeup to the basin is manually controlled to maintain the water level at elevation 133.3 ft. which is the maximum basin operating level. Should this makeup source become unavailable, makeup can be supplied by any of the following alternate means.

1) Use of tank trucks, to carry Mississippi River water from the plant barge slip to the standby cooling tower storage basin.

2) Use of a temporary pump and piping to pump Mississippi River water into the storage basin.

3) Use of an alternate well - because of the presence of a large shallow water table (approximately 30 ft. below grade) and the favorable geology a temporary well could be drilled and put into operation in a few days.

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The system is not susceptible to adverse weather conditions such as icing because of the mild winter climate. In addition, valves and other components essential to the operation of the standby cooling water system are located inside tunnels or inside the standby service water pumphouse. These structures prevent rainwater or snow from impinging on the components of the system, therefore protecting it from freezing or icing. The water pump suction is 51 ft 8 in. below the water level in the basin; therefore even in extreme low ambient temperatures it will not freeze up. The basin is partially below grade, hence ground temperatures contribute to maintain a water temperature above freezing.

The staff has not as yet completed its independent analysis of the thermal and hydrologic performance of the emergency cooling water supply. This will have to be completed before the staff can determine whether the plant meets the guidelines of Regulatory Guide 1.27 and the requirements of GDC 44 with respect to the emergency cooling water system.

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2.4.6 Flooding Protection Requirements

As discussed in Section 2.4.3, the design basis flood level (DBFL) has not yet been resolved between the applicant and the staff. Determination of flooding protection requirements for the site is contingent upon determination of the DBFL.

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2.4.7 Dispersion, Dilution, and Travel Times of Accidental Releases of Liquid Effluents in Ground and Surface Waters

The applicant analyzed two possible pathways for a postulated liquid radwaste tank failure. One pathway considered was a discharge directly into the Mississippi River. Based on the dispersion characteristics of the river at low flow the applicant calculated a minimum dilution factor of 10^3 between the site and the nearest downstream water user at Bayou LaFourche about 87 miles downstream.

The other pathway considered was the groundwater pathway from the radwaste building to the nearest downgradient groundwater user. Since the water in the small local streams is not used for human consumption, the case of a pathway from the groundwater to a local surface stream and then to water users on the Mississippi River is considered bounded by the case of the spill directly into the Mississippi River. For the groundwater spill the applicant identified two downgradient wells which could be affected and calculated travel times and dilution factors based on groundwater gradients, permeability, dispersion characteristics, and retardation factors due to ion exchange. The applicant determined a dilution factor of at least 10^5 for the nearest well and travel times for cesium 137 and strontium 90 of over 1,000 years for the well with the shortest travel time (not the nearest well).

The staff has ~~not yet~~ performed an independent analysis of radionuclide transport due to a radwaste tank failure. As a result of this analysis ~~performed herein~~ the staff has concluded that the plant meets the requirements of 10 CFR 100 with respect to potential accidental releases of radioactive effluents.

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2.4.8 Groundwater

2.4.8.1 Description

Three aquifer systems occur in the immediate vicinity of the River Bend Station; these are the Mississippi River Alluvial Aquifer, the Upland Terrace Aquifer, and the Tertiary Aquifer System which consists of three zones. All of the aquifers are unconsolidated sedimentary deposits, comprised predominately of sandy materials. Groundwater flow in the Upland Terrace and Alluvial aquifers is generally toward the Mississippi River; flow in the Tertiary Aquifer zones is generally in a southerly direction toward the Gulf of Mexico. Locally, flow in the Tertiary Aquifer is effected by large withdrawals in the vicinity of Baton Rouge, about 30 miles to the south.

The uppermost aquifer underlying the plant site is the Upland Terrace Aquifer which is utilized to provide fire protection water. In the vicinity of the plant, this unit is used to provide domestic water to scattered residences. The aquifer consists of unconsolidated sedimentary material ranging from clay to gravel and is overlain by 10 feet of loess. Water levels in this unit are at about 57 ft msl at the plant site.

Beneath the Upland Terrace Aquifer, separated by interbedded layers of clay several hundred feet thick, is the Tertiary Aquifer. Three distinct fresh water zones separated by clay layers have been identified extending down to about 2000 ft below land surface. Parts of the topmost zone are about 100 ft from the surface at the plant site, interbedded with the uppermost confining

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clay layers. These aquifers provide the major source of fresh water for the City of Baton Rouge, and West Feliciana and Pointe Coupee Parishes. Onsite water supply, plant makeup water, and domestic water supply is provided by two wells tapping the zone 3 aquifer at a depth of about 1800 ft. Extensive pumping in the vicinity of Baton Rouge has caused large drawdown in the lower sand layers of the zone 3 aquifer near the site.

The other major aquifer in the vicinity of the site is the alluvial aquifer which is restricted to the floodplain of the Mississippi River. The aquifer is locally semiconfined by a mantle of clay and silt. The alluvial aquifer is hydraulically connected to the Mississippi River and responds to changes in river stage. Near the site the alluvial aquifer is overlain by up to 85 ft of clay which acts as a local confining layer. Total thickness of the clay and the alluvial aquifer is about 200 ft. The aquifer is not utilized in the vicinity of the plant.

2.4.8.2 Design Basis Groundwater Level

Design basis groundwater level for hydrostatic loading at the site is 70 ft. msl. This design basis was reviewed and approved by the staff at the construction permit stage. This level is calculated to occur only in conjunction with the design basis flood on the Mississippi River. Normal ground-water elevation at the site is at approximately 57 ft msl. The staff has reviewed the data in the FSAR showing the changes in elevation in the

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Upland Terrace Aquifer over a seven year period (1972-1979) as well as the corresponding changes in Mississippi water level. The staff concludes that the design basis groundwater level of 70 ft msl selected by the applicant is conservative and that the plant meets the requirements of GDC 2 with respect to the design basis groundwater level for hydrostatic loading.

The applicant has not stated the groundwater level which was used in combination with other environmental loads, however.

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2.4.9 Technical Specifications and Emergency Operations Requirements

At this time the applicant has not provided any technical specifications or Emergency Operation Requirements to protect the plant from adverse hydrological events. Technical specifications may be proposed, however, as part of the resolution to some presently open items.

2.4.10 Conclusions

On the basis of its review, the staff concludes that the flood analysis for the plant is in conformance with the procedures in Regulatory Guide 1.59 "Design Basis Floods for Nuclear Power Plants" in regard to flooding from the Mississippi River. However, the staff does not agree with the applicant on the appropriate Design Basis Flood Level for flooding from local intense precipitation on West Creek and the plant site drainage system. The staff cannot conclude that the site meets the requirements of GDC 2 with respect to local flooding until the proper DBFL is resolved.

The staff has not as yet completed its independent analysis of the UHS with respect to emergency water supply and thus cannot conclude that the plant meets the requirements of GDC 44 with respect to emergency cooling water availability.

The staff concludes that the design basis groundwater levels chosen for evaluation of hydrostatic loading meets the requirements of GDC 2. However the applicant has not provided the water level to be used in combination with other environmental loads.

The staff has performed an independent analysis of radionuclide transport following a postulated liquid radwaste tank failure and thus concludes that the plant meets the requirements of 10 CFR 100 with respect to potential accidental releases of radioactive liquid effluents.

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2.5 Geology ~~and~~ Seismology and Geotechnical Engineering

In its review of the River Bend FSAR, the staff has considered pertinent information gathered since the SER-CP was issued.

The U.S. Geological Survey (USGS) served as the staff's geological and seismological advisor during the CP review. A staff consultant, Lawrence Livermore National Laboratory (LLNL), is currently performing a probabilistic seismic hazard analysis of the site. The results of this study will be discussed in the final SER.

Information considered since issuance of the SER-CP in September 1974 includes: (1) data gathered from both site and near-site investigations, including geologic and geophysical information acquired as a result of extensive oil/gas

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Table 2.3. Maximum value of River Bend releases, χ/Q* and D/Q

Location	Direction/ distance (mi)	Release** type	χ/Q (sec/m³)	D/Q 1/m²
Exclusion Area C***	NW/0.66	A	4.6-7**†	9.9-9
	W/0.60	B	4.2-5	6.5-8
	W/0.60	C	6.0-6	1.2-7
Exclusion Area R***	NW/0.72	A	4.4-7	9.0-9
	W/0.63	B	3.9-5	6.0-8
	W/0.63	C	4.8-6	9.8-8
Residence	NW/0.78	A	4.1-7	8.1-9
	NW/0.78	B	1.5-5	3.5-8
	NW/0.78	C	1.2-6	2.3-8
Garden	NW/0.78	A	4.1-7	8.1-9
	NW/0.78	B	1.5-5	3.5-8
	NW/0.78	C	1.2-6	2.3-8
Animal	NW/0.78	A	4.1-7	8.1-9
	NW/0.78	B	1.5-5	3.5-8
	NW/0.78	C	1.2-6	2.3-8
	ESE/0.68	C	1.2-6	2.7-8

*undecayed and undepleted

**A = continuous containment vent
B = radwaste continuous ground-level release
C = four 20-hour vent purges

***Exclusion Area C = exclusion area distance from the containment vent
Exclusion Area R = exclusion area distance from the radwaste vent

† 4.6-7 = 4.6 x 10-7

exploration in the site region; (2) discussions with individuals knowledgeable of the geology of the site and region; (3) a staff review of recently acquired literature; (4) a discussion, by the applicant, of the potential for geothermal-geopressured engergy resource development and seismicity that has been observed in areas where these wells exist; (5) new information, based on recent and ongoing studies on the extent of the subsurface geologic structure associated with 1811-1812 New Madrid earthquake; (6) a limited seismic hazard analysis assessing the probability of exceeding the operating basis earthquake (OBE) during the plant life; and (7) a compilation of observed seismicity in the vicinity of the site through 1978. As a result of its recent review and evaluation of the above geologic and seismologic information, the staff has

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determined that the SER-CP conclusions regarding the safety of the River Bend station from a geological and seismological standpoint remain valid. The staff is confident that the results of the applicant's ongoing efforts will confirm the adequacy of the seismic design for the site. The most relevant subjects presently being addressed by the applicant include: (1) a remote-sensing imagery study, (2) an interpretation of geophysical seismic reflection profiles within 5 miles (8 km) of the plant, and (3) an identification of the type of trap constituting an oil reservoir in the site vicinity.

The applicant's presentation in FSAR Sections 2.5.1, 2.5.2, and 2.5.3 has satisfied the requirements of 10 CFR 50, 10 CFR 100, and Appendix A to 10 CFR 100 by (1) performing post-CP site and near-site geological, seismological, and geophysical investigations/studies as needed, and (2) consulting with individuals knowledgeable of the local geology and seismology. The seismologic portion of FSAR Section 2.5, however, is somewhat outdated and should be revised to reflect state-of-the-art practice. This information will aid the staff in evaluating the applicant's conclusions regarding the favorable geologic and seismic characteristics of the River Bend site.

In addition, the applicant is in conformance with applicable portions of RGs 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants"; 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants," Revision 2; and 1.132, "Site Investigations for Foundations of Nuclear Power Plants."

The newly acquired information reviewed for the River Bend is discussed in Sections 2.5.1, 2.5.2, and 2.5.3 below

2.5.1 Geologic and Seismic Information

Since issuance of the SER-CP, the applicant has conducted geologic investigations both on and off site and has evaluated geophysical data obtained by others. These investigations and analyses have resulted in an increased understanding and confirmation of the surface and subsurface conditions of the River Bend site and vicinity. No new known information has been developed

indicating that the staff's earlier conclusions regarding the suitability of the River Bend site should be modified.

With the exception of three topic areas, the applicant has responded to all staff geology-related questions. Those areas not completely addressed by the applicant are: (1) performance of a remote-sensing imagery study, (2) interpretation of seismic reflection surveys recently conducted in the immediate vicinity of the site, and (3) identification of the trap from which the hydrocarbons are produced from a commercial oil well 3 miles (4.8 km) southeast of the site.

The principal newly developed geologic/geophysical information presented in the FSAR relative to the River Bend site and site vicinity (within 5 miles (8 km) of River Bend) has been derived from the following sources:

(1) mapping and photographing of the excavation for the plant structures
(2) acquisition and interpretation of several of the seismic reflection surveys conducted since issuance of the CP
(3) structural contour maps of geologic horizons underlying the site and vicinity prepared by the Geomap Company of Houston, Texas.

Geologic mapping of the plant excavation by the applicant, coupled with complete photographic coverage, has confirmed the geologic integrity of the plant foundations area. As predicted by the applicant's pre-CP investigations, the River Bend excavations as inspected, mapped, and photographed, show no evidence of either faulting or through-going deformation. The staff inspected the excavations for Units 1 and 2 on November 11, 1976, and concluded that no structural abnormalities were visible within either the mapped or then visible portions of the excavation.

The geologic and geophysical information derived from the pre- and post-CP hydrocarbon test wells and several of the seismic reflection profiling surveys within 5 miles (8 km) of the River Bend plant have been obtained and evaluated by the applicant. These data, as interpreted by the applicant, confirm the earlier (CP) conclusions relative to the near-site area that neither faulting nor through-going deformation appear to be present.

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As of August 1982, seven post-CP test holes (one of which has been commercially successful) and approximately 41 miles (66 km) of geophysical prospecting (seismic reflection profiling) have been completed within 5 miles (8 km) of the River Bend plant. This commercial well is 3 miles (4.8 km) southeast of the plant. Because of the great depth (in excess of 17,000 ft (5181 m)) and consequent low hydrocarbon reservoir porosity, potential geologic hazards (such as possible differential subsidence) that may be associated with the near-site development of shallow hydrocarbons do not exist.

Copies of the proprietary seismic reflection data have recently (January 12, 1983) been received from the applicant. The staff is evaluating these data to confirm the applicant's finding (not yet incorporated by amendment into the FSAR) that no faulting exists within 5 miles (8 km) of the River Bend plant.

Contouring of several Cretaceous horizons by Geomap (not affiliated with the applicant) and construction (by the applicant) of a 13-mile (20.9-km) long northwest-trending geologic cross-section extending through the plant area to depths in excess of 17,000 ft (5181 m) have confirmed the geologic integrity of the site (5-mile (8-km) radius) area. It should be noted, however, that Geomap's structural contour maps are not totally comprehensive in that the maps portray an evaluation of only geophysically logged test holes, and do not reflect Geomap's interpretation of the seismic surveys conducted within the area.

2.5.2 Vibratory Ground Motion

In its review of the seismological aspects of the plant site, the site has considered pertinent information gathered since its initial seismologic review made in conjunction with the issuance of the SER-CP. The SER-CP conclusion was that 0.10 g and 0.05 g are adequate values for the peak horizontal accelerations for the safe shutdown earthquake (SSE) and OBE, respectively. The acceleration levels (0.10 g and 0.05 g) proposed for the SSE and OBE have been used to anchor an RG 1.60 response spectrum. Since the conclusion of the CP review, no new information has become available to change the staff's original conclusions.

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The requirements of Appendix A to 10 CFR 100 in determining the vibratory ground motion corresponding to the SSE have been met.

2.5.2.1 Seismicity

In FSAR Section 2.5.2.1, the applicant presents the results of a thorough study of historical seismicity for the Gulf Coast up to December 1978. The study shows that the Gulf Coast is an area of infrequent and low seismicity. This is substantiated by the work of Barstow et al. (1981), who have produced an earthquake frequency map of the United States east of the Rocky Mountains. This map shows that the area within 100 miles (160 km) of the River Bend site has 0 to 3 earthquakes (Modified Mercalli Intensity (MMI) greater than or equal to III or magnitude greater than or equal to 2.0), in 11,689 km^2. Details of the significant events utilized to evaluate the River Bend site are discussed in Section 2.5.2.3 of this report.

2.5.2.1.1 Seismicity Associated with Growth Faults

Recently, the staff became aware of detailed seismic monitoring in areas associated with the development of Gulf Coast geopressured-geothermal energy wells. These studies (Mauk, 1981; Statton et al., 1981) have documented small-magnitude earthquakes (less than M = 2.0) in areas where growth faults exist. Growth faults are faults in sedimentary rock that forms contemporaneously with deposition, with normal fault motion due to downslope mass buildup of sediments. As described in FSAR Section 2.5.1, growth faults are known to exist from about 11.5 to 35 km south of the site. Because of this, the staff requested that the applicant discuss the potential for geopressured-geothermal energy development in the near vicinity of the River Bend site, and also discuss the seismicity that has been observed in areas where these wells exist.

The applicant, in FSAR Sections 2.5.1 and 2.5.2, has presented a thorough discussion of geopressured-geothermal energy potential and observed seismicity in these areas. Presently, the closest area of observed seismicity and energy well development is approximately 100 km southwest of River Bend. Although the possibility exists that geopressured-geothermal energy development may become

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feasible within 35 km, presently there are no plans for this acitivity to be initiated.

Based on its analysis and its review of the applicant's analysis, the staff concludes that there is no concern that strong ground motion will be produced by movement along growth faults in the River Bend site vicinity.

2.5.2.2 Tectonic Province

In the SER-CP the staff concluded that the River Bend site is in the Gulf Coastal Plain physiographic province. Current staff practice (outlined in Appendix A of 10 CFR 100) locates the River Bend site in the Gulf Coastal Plain Tectonic Province, which is characterized by thousands of feet of unmetamorphosed sediments of the Gulf Coast geosyncline. This is consistent with other SERs for sites in the Gulf Coastal region (Allens Creek, NRC 1974; South Texas, NRC 1975; Blue Hills, NRC 1976; Yellow Creek, NRC 1977; Comanche Peak, NUREG-0797; Waterford, NUREG-0787; Grand Gulf, NUREG-0831). King (1969) defines the Atlantic and Gulf Coastal Plains as platform deposits (Mesozoic age and younger) that were laid over the deformed basement (Paleozoic and older rocks of the Appalachian and Ouachita orogenic belts). The platform deposits thicken and slope seaward from the exposed parts of these orogenic belts, with the basement descending beneath them. From New Jersey, south and then west to the Llano Uplift in central Texas, the landward [redacted] of the platform deposits (Coastal Plain) is drawn at the contact of the Cretaceous and/or Tertiary deposits with the underlying and overlapped basement rocks. The River Bend site is located in the Gulf Coastal Plain Tectonic Province, which is part of the Coastal Plain extending from Florida westward and southward into Mexico.

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The applicant has proposed tectonic provinces for the River Bend region based on crustal patterns, geologic history, and earthquake history. The staff has reviewed this proposal, and although the staff established somewhat different boundaries, the applicant's analysis of the maximum earthquake for the River Bend site is equivalent to the staff's, in accordance with Appendix A to 10 CFR 100.

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2.5.2.3 Maximum Earthquake Potential

The staff recognizes that within the Gulf Coastal Plain Tectonic Province different regions of this large province exhibit vastly different levels of seismicity. In particular, to arrive at the appropriate choice of SSE for the River Bend site, the staff recognizes three areas of seismicity within or intersecting this province: (1) the Mississippi Embayment (New Madrid) Earthquake Zone, (2) the Ouachita-Wichita belt of seismicity, and (3) seismicity in the remainder of the Gulf Coastal Plain Tectonic Province. Each of these zones of seismicity is described below.

The staff is also currently evaluating seismic reflection profile data, and, through LLNL, performing a probabilistic seismic hazard analysis for the River Bend site. The reflection profile data will aid the staff in determining if any geologic structures or faults exist in the near-site region. The probabilistic hazard analysis will aid the staff in assessing the conservatism of the SSE. Both of these topics will be discussed in more detail in the SER, although it is not anticipated that either topic will alter the staff's previous conclusion regarding the adequacy of the River Bend site.

2.5.2.3.1 Mississippi Embayment Earthquake Zone

The Missisippi Embayment Earthquake Zone is a region of much higher seismic activity than the remainder of the eastern United States. It has also been the source region of the largest earthquakes in the eastern United States, the 1811-1812 New Madrid earthquakes. The largest event in this series, on February 7, 1812, had an MMI of XI-XII (Nuttli, 1973), and a magnitude (M_b) of 7.3 (Nuttli and Brill, 1981). During the CP review of the River Bend site, the staff concluded that the closest approach of earthquakes similiar to the 1811-1812 series was near Memphis, Tennessee, about 310 miles (500 km) from the site.

As discussed in Section 2.5.1 of the Grand Gulf SER (NUREG-0831), recent geological and geophysical data suggest that the subsurface geologic structure in the New Madrid region extends southwest of Memphis, to the Ouachita orogenic belt and within about 230 miles (370 km), of the site. This is consistent

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with the information presented by Hamilton and Zietz (1982). The staff has not completed its review of the ongoing geophysical and seismological research regarding the New Madrid extension and its large magnitude earthquake potential as far as establishing a final staff position on this issue. For the purposes of this draft SER, the staff has conservatively assumed for the seismic design of River Bend that an 1811-1812-type earthquake could occur at a distance of 230 miles (370 km) from the site (as opposed to about 500 km established at the CP stage). As discussed in SER Section 2.5.2.4.1, the ground motion from this assumed event does not exceed the SSE. In addition, the following factors argue for terminating the New Madrid Seismic Zone near Memphis, 500 km from the River Bend site:

(1) As part of the Systematic Evaluation Program (SEP), a group of seismologists were asked to give their expert opinions on input parameters for a seismic hazard analysis (TERA, 1980). One of these parameters was the configuration of seismic source zones. Results compiled by TERA demonstrate that there are varying opinions among experts on earthquake zonation and correlation with structure in this region of the United States. Most experts (ibid, pages C-1 to C-14) terminated the New Madrid source zone at or near Memphis.

(2) Both historic and recent microearthquake activity has been confined to the region north of Memphis. Detailed microearthquake monitoring in the New Madrid region has revealed that the vast majority of seismicity is occurring north of Memphis (St. Louis, 1980).

(3) Seismic reflection profiling in the New Madrid region demonstrates that faulting cuts Cretaceous material (although no single fault has been found that could be associated with the 1811-1812 earthquake sequence) (McKeown, 1981). Seismic profiling just south of Memphis (done at the CP stage) shows no faulting of Cretaceous material, suggesting there has been no faulting in the last 65 million years. These reflection lines, however, show little information at deeper depths so no definitive statements can be made about deeper faulting at the present time.

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(4) Results given by Zoback et al. (1980), using reflection data in the more seismically active areas of north Memphis, demonstrate that faulting is more abundant in the area north of Ridgely, Tennessee, than it is to the southwest, toward Memphis.

(5) A recent report by Nuttli and Brill (1981) on earthquake source zones in the central United States determined from historical seismicity shows the New Madrid source zone terminating near Memphis.

2.5.2.3.2 Ouachita-Wichita Belt of Seismicity

The boundaries of the east-west Ouachita-Wichita belt of seismicity are subject to interpretation by various investigators. Nuttli and Brill (1981) show this east-west trending area overlapping the Nemaha Ridge Seismic Zone; the applicant's tectonic features map (FSAR Figure 2.5-36) shows the buried Appalachian front. Both maps show the Ouachita-Wichita belt of seismicity, and tectonic features intersect and overlap the Gulf Coastal Plain Tectonic Province. Although the boundaries are somewhat ambiguous, the largest historically recognized earthquake in the Ouachita-Wichita belt of seismicity is the October 22, 1882 MMI VII earthquake that occurred near Paris, Texas. Estimates of its intensity range from MMI VI-VII by Coffman and Von Hake (1973), who placed the epicenter near Fort Smith, Arkansas, to those of Docekal (1970), who estimated that it approached or possibly even reached MMI VIII and who, upon reevaluation of the data, relocated the epicenter near Paris, Texas. Based on the available data, it is the staff's view that the earthquake occurred near Paris, Texas, and that the maximum intensity was no higher than VII (Comanche Peak, SER-CP, NRC 1974). A recent evaluation of this event by the USGS (Stover et al., 1982) supports this conclusion. Because this event has not been associated with known geologic structure, the staff assumes its occurrence at the closest approach of the quachita-Wichita belt of seismicity (150 miles) to the River Bend site. Assuming the occurrence of an MMI VII earthquake 150 miles (240 km) from the site and using the intensity-attenuation relationship of Gupta and Nuttli (1976) results in an MMI IV event at the site. MMI IV is well below that of the SSE.

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2.5.2.3.3 Gulf Coastal Plain Seismicity

Seismicity in the remainder of the Gulf Coastal Plain Tectonic Province is relatively uniform. The largest historical intensity earthquake that has not been associated with known geologic structure is the MMI VI event that was centered near Donaldsonville, Louisiana, on October 19, 1930. The largest instrumentally recorded earthquake that has not been associated with known geologic structure is the M_b = 4.8 event that occurred off shore in the Gulf of Mexico on November 5, 1963. Based on the lack of structural association, the staff assumes that an earthquake similar to that which occurred in Donaldsonville or the Gulf of Mexico could occur near or at the plant site. Although the January 8, 1891 Rusk, Texas, earthquake is reported as MMI VII (Coffman and Von Hake, 1973), the felt area of this event is very small (felt only at Rusk). This had led Nuttli and Brill (1981) to interpret this event as being a very small magnitude, shallow earthquake. The staff concurs with Nuttli and Brill (1981) and has concluded that this event should be considered as a very shallow, small magnitude earthquake (NUREG-0787, NUREG-0797). Ground motion estimates from the above events are discussed in Section 2.5.2.4.2 of this report.

2.5.2.4 Safe Shutdown Earthquake

In determining the SSE, the staff has followed the Tectonic Province approach described in Appendix A to 10 CFR 100. The applicant's proposed SSE acceleration level of 0.10 g is used as the high frequency anchor point for an RG 1.60 response spectrum. As discussed below, this is a conservative representation of the SSE response spectra.

2.5.2.4.1 Ground Motion from New Madrid Type Earthquake

In the operating license review, the staff has estimated the ground motion of the postulated New Madrid-type earthquake at a distance of 230 miles (370 km) from the site. Because of the low attenuation of seismic energy in the central United States, the duration of motion is longer than is typical for earthquakes of equal size in the western United States at large distances. Nuttli (1973) believes that damage from larger earthquakes in the midcontinent of the United

States can be attributed to long-duration ground motion produced by surface waves in the frequency range 0.33 Hz to 1 Hz. The staff concurs that the long-duration ground motion is the result of attenuation differences and is a reasonable explanation for the phenomena observed at large distances from the New Madrid earthquakes. To account for this attenuation difference between the eastern and western United States, the staff has used a recent ground motion relationship developed by TERA-LLNL and the NRC staff for the SEP (Jackson, 1980) and relationships developed by Campbell (1981) and Nuttli and Herrmann (1981).

The staff places more confidence in the Nuttli and Herrmann (1981) relationships because they allow for specific use of regional Gulf Coastal attenuation factors (developed by Singh, 1981).

Ground motion estimates have been calculated assuming an M_b = 7.3 earthquake occurred 230 miles (370 km) from the site. In the higher frequency range (frequencies about 2 Hz) the staff estimates for peak acceleration range from 0.02 to 0.06 g (Nuttli and Herrman, 1981, value = 0.02 g). For lower frequencies, the staff estimates for peak velocity range from 5.2 to 9.8 cm/sec.

The staff has compared the River Bend SSE, at 5% damping, to the postulated New Madrid earthquake using the above estimated peak ground motion values and the spectral amplification factors (84th percentile) of Newmark and Hall (1978). The River Bend SSE spectra would exceed the New Madrid spectra at all frequencies.

2.5.2.4.2 Ground Motion from Gulf Coastal Plain Earthquake

As discussed in SER Section 2.5.2.3.3, the largest historical earthquake that has not been associated with a geologic structure within the Gulf Coastal Plain is MMI VI or M_b = 4.8. The applicant's proposed SSE acceleration level of 0.10 g is used as the high frequency input to an RG 1.60 response spectrum. An acceleration level of 0.10 g roughly corresponds to an MMI VI-VII earthquake occurring near the River Bend site using the intensity-acceleration relationship of Trifunac and Brady (1975). An MMI of VI-VII is more conservative than the largest historical intensity earthquake (MMI VI) in the Gulf Coastal Plain.

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The ground motion from the largest historical magnitude earthquake (M_b = 4.8) was estimated in the Waterford SER (NUREG-0787). In addition, the staff has also used recent ground relationships of Joyner and Boore (1981), Campbell (1981), and Nuttli and Herrman (1981). It should be noted that great care should be exercised when using the relationships of Joyner and Boore (1981) and Campbell (1981) for an M_b = 4.8 where little data exist. Based on its analysis and review of results discussed in the Waterford SER, the staff finds that an RG 1.60 response spectrum utilizing a peak acceleration of 0.10 g accommodates the ground motion for an M_b = 4.8 earthquake. Thus, the design response spectrum for the River Bend site would accommodate the largest earthquakes not associated with known geologic structures (MMI VI or M_b = 4.8) if events similar to these were to occur near the site.

2.5.2.5 Operating Basis Earthquake

The applicant has proposed 0.05 g for the acceleration level corresponding to the OBE. The design vibratory ground acceleration for the OBE is taken to be one-half of the design vibratory ground acceleration for the SSE, consistent with Appendix A to 10 CFR 100.

The applicant has also provided the staff with results from a limited seismic hazard analysis, completed to assess the probability of exceeding the OBE during the plant life. Input into this analysis included three circular source zones, each with one recurrence relationship and upper intensity cutoff. Also used were three attenuation relationships, each with an uncertainty term. Uncertainty in source zonation, recurrence relationship, and upper intensity cutoff were not included. An important part of completing a seismic hazard analysis involves the selection of an approach to incorporate the uncertainty of all input parameters into the analysis. Difficulty in accounting for this uncertainty is one of the reasons the staff has used probability studies in a limited sense. In the River Bend study, it is the staff's opinion that the results should be used only in a qualitative sense because of the rudimentary source zones and relatively low upper intensity cutoff. Results of the River Bend study show that the OBE has a return period on the order of 100 to 1000 years.

Considering the low seismicity of the Gulf Coast and the compliance with Appendix A to 10 CFR 100, the staff finds that the proposed acceleration value for the OBE is conservative for the River Bend site.

2.5.3 Surface Faulting

Post-CP site and regional subsurface information presented to date by the applicant reinforces the previous NRC staff opinion that there is no known evidence either at the River Bend site or within 5 miles (8 km) of the plant site to indicate surface faulting or that the potential exists for such faulting. The new information consists of (1) the applicant's site mapping efforts, (2) examination of site excavations by an NRC staff geologist, and (3) structural contour maps of deep geologic horizons prepared by an independent consulting firm.

The staff has asked the applicant to acquire, examine, and interpret post-CP geophysical and bore hole data within 5 miles (8 km) of the River Bend station. The applicant has acquired this information and is preparing an FSAR amendment addressing the interpretation of these data. The applicant has given the staff proprietary seismic reflection data, which the staff is evaluating. The results of this evaluation--as well as that of the applicant--will be addressed in the final SER.

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Plant Name: River Bend Station Units 1 and 2

Docket Numbers: 50-458 and 50-459

Subject: Draft Safety Evaluation Report, Geotechnical Engineering

Prepared By: Owen O. Thompson, Geotechnical Engineer, Structural and Geotechnical Engineering Branch, DE

The following sections summarize the staff's geotechnical engineering review of the River Bend Station as described in the Final Safety Analysis Report (FSAR) through Amendment Number 6, dated September 27, 1982. The stability of subsurface materials (FSAR Section 2.5.4) and the stability of slopes (FSAR Section 2.5.5) have been evaluated in accordance with the criteria outlined in Appendix A to 10 CFR Part 100, Regulatory Guide 1.70, Revision 3, and the current Standard Review Plans (SRP) NUREG-0800. There are no embankments or dams (FSAR Section 2.5.6) associated with the River Bend Station.

2.5.4 Stability of Subsurface Materials and Foundations

2.5.4.1 Site Conditions

2.5.4.1.1 General Station Description

River Bend Station is located on the east bank of the Mississippi River about 24 miles northwest of the Baton Rouge, Louisiana, at River Mile 262. The main plant structures are located about 2 miles east of the river on a plateau at an elevation of 95 ft to 110 ft above mean sea level (el +95 to el +110). The alluvial flood plain (about el +35) extends about 3000 ft east of the river.

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The seismic Category I, safety-related structures, systems and components that were reviewed for foundation stability are listed in FSAR Table 3.2-1. They include the reactor, auxiliary, fuel, control and diesel generator buildings and the ultimate heat sink (UHS) that includes the standby service water cooling tower, basin and pump house. Structures within the main plant area are supported on reinforced concrete mat foundations (FSAR Table 3.8-7) located as shown on FSAR Figure 1.2-2. There are no safety-related structures, pipelines or conduits outside the main plant area.

The UHS contains a 30-day supply of water for safe shutdown of the plant. The normal service water make-up system (two 150 gpm deep wells) and the normal service water cooling towers are not safety related.

The seismic Category I structures are supported on structural backfill placed on dense sands and gravels (channel deposits) that overlie hard, overconsolidated Tertiary clays (Pascagoula Clay). The bottom of the backfill extends down to about el +20; the structural foundations are located at el +55 to el +65; the final grade is el +95 at the main plant structures and el +105 at the cooling towers.

2.5.4.1.2 Site Investigations

About 400 borings were made at the site (see FSAR Figures 2.5-23 and 2.5-24 for locations) during February 1970 through March 1974; about 100 of these borings were in the main plant area. The field investigation of soils included standard penetration tests (SPT) and split-barrel

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sampling according to American Society for Testing and Materials Method D-1586 (ASTM D-1586), and undisturbed sampling with hydraulically pressed, 3-in diameter thin-walled tubes (ASTM D-1587). Attempts to recover undisturbed samples of cohesionless channel deposits using the Waterways Experiment Station method (see FSAR Appendix 2L) were not very successful because the high density of these granular soils prevented adequate sampler penetration. Undisturbed samples were obtained in the underlying Pascagoula clay (below el -40).

Groundwater levels were recorded in all the test borings; additionally, about 40 piezometer were installed and they have been monitored since 1972 as described in FSAR Section 2.4.13.2.3. Some of the piezometers were used to measure the effects of field pumping tests to investigate the hydraulic characteristics of the various aquifers.

The geophysical investigations included crosshole, downhole and uphole seismic wave velocity determinations. Seismic refraction profiling was limited because of the great depth to bedrock.

The staff has concluded that the investigations at the site are sufficiently extensive to adequately define the site subsurface conditions and that all significant geotechnical engineering characteristics of the foundation soils have been identified, as required by 10 CFR Part 100, Appendix A and subsection 2.5.4.1 of the SRP.

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2.5.4.1.3 Properties of Subsurface Materials

The near-surface stratigraphy at the site consists of five layers as shown on SER Figure 2.5-1 and described below:

Loess. The site is covered with a layer of loess about 10 ft thick. The loess is predominantly clayey silt and it was removed from the main plant area. The loess was not used as structural fill but it was used as general plant fill around the site.

Port Hickey Top-Stratum Silts and Clays. These terrace deposits are about 10 ft thick underlying the loess. They are located above the elevations of the foundations in the main plant area and were excavated during construction of the safety-related structures. The Port Hickey top-stratum silts and clays are generally stiff to hard consistency (SPT values in excess of 10 blows/ft). They were not used as structural fill.

Port Hickey and Citronelle Sands and Clayey Sands. These terrace deposits consist primarily of silty sands, stratified fine to medium sands and clayey sands with occasional lenses of silty clay and clayey silt. The coarse grained materials have a very variable density, ranging from loose (SPT values as low as 8 blows/ft) to very dense (SPT values in excess of 100 blows/ft). The thickness of this deposit is variable but its base is generally near El +50. Because of the potential for liquefaction (see Section 2.5.4.3.3 of this SER) these

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terrace deposits were removed from excavations for safety-related structures as described in Section 2.5.4.2.2 of this SER.

Citronelle Buried Channel Deposits. The channel deposits consist of materials that are coarser and more dense than the overlying sands and clayey sands, particularly below about el +20. Using conventional correlations, SPT values of greater than 40 blows/ft indicate that the in situ relative densities of the channel deposits are greater than 70 percent at depths below el +20. Localized areas having lower SPT values were investigated by additional borings. The applicant determined that the low SPT values (20 blows/ft) occurred in clayey layers and were localized (less than 5 to 10 ft radius). Because of the limited occurrence of these localized soils having low SPT values, and the great depth below the elevation of plant foundations (more than 70 ft) the applicant concluded that the buried channel deposits below el +20 are suitable foundation materials. The staff concurs in the applicant's conclusion.

Based on gradation test results and field pumping tests, the permeability of the channel deposits were determined to be 10^{-1} to 10^{-2} cm/sec. The geophysical surveys showed consistent seismic shear wave velocities of 1100 to 1200 ft/sec.

Pascagoula Clay. The top of the Pascagoula clay is about el -40 and it extends below el -200. This deposit is predominantly greenish-gray, silty clay with a little sand. The clay has been preconsolidated;

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overconsolidation ratios in excess of 1.3 were obtained from laboratory tests. These results are consistent with the geologic history which indicates that the ground surface had been as high as el +200; changes in sea level during recent geologic history also have contributed to the preconsolidation of the Pascagoula clay.

The strength of the Pascagoula clay was investigated by unconsolidated-undrained triaxial shear strength tests on undisturbed samples, and the applicant concluded that the undrained compressive strength of the clay is, conservatively, 5.0 to 6.5 Kg/sq cm (approximately 10 to 13 Ksf). The staff has also concluded that these adopted design strength values are acceptable and conservative, based on our review of the field and laboratory test results.

The seismic shear wave velocities in the Pascagoula clay were determined from the geophysical crosshole tests to be about 1200 to 1300 ft/sec, similar to the overlying Citronelle buried channel deposits. The average shear modulus (31 ksi) was calculated from the shear wave velocity values and the design shear modulus of 18 ksi (2600 ksf) was obtained by reducing the calculated value to account for anticipated higher shear strain levels during the SSE. The same design shear modulus values were also used for the Citronelle buried channel deposits because these soils gave geophysical test results that were not significantly different from results of tests in the Pascagoula clays (FSAR Tables 2.5-11 and 2.5-12).

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Resonant column tests were performed on undisturbed samples of the clay to obtain shear modulus and damping values. The laboratory values of shear modulus (summarized in FSAR FIgure 2.5-66) showed good correlation with values calculated from the shear wave velocities in the Pascagoula clay.

2.5.4.1.4 Groundwater Conditions

The applicant has identified the Alluvial Aquifer, the Upland Terrace Aquifer and three Tertiary Aquifers in the vicinity of the site (see Section 2.4 of the SER). The aquifer of primary significance to the geotechnical engineering aspects of the plant structures is the Upland Terrace Aquifer that is within the granular materials above the Pascagoula clay. The piezometric level in the Upland Terrace Aquifer was about el +50 to el +55 prior to construction, as shown by piezometer readings (FSAR Figure 2.4-42) and on the test boring logs (FSAR Appendix 2H).

During construction the piezometric level in the Upland Terrace Aquifer was lowered to about el zero (see Section 2.5.4.2.1 of this SER); the piezometric level was subsequently permitted to return to its natural level after the foundation walls were completed and backfill was placed against them.

The applicant has evaluated future decline of the piezometric levels resulting from regional and on-site groundwater useage (see FSAR Section 2.4.13). Projected off-site usage is predicted to increase only

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slightly in the next 10 years (FSAR Figure 2.4-38), on-site usage is expected to create a localized cone of depression of the piezometric level in the plant area, with a maximum piezometric level difference across the plant of less than 2 ft. The applicant concluded that future changes in groundwater levels will not affect plant foundations because the expected small changes in groundwater level will not cause any significant settlement of the preconsolidated foundation soils. (These soils have already experienced significant groundwater level fluctuations as discussed in Section 2.5.4.1.3 of this SER). Additionally, any changes in groundwater level will occur uniformly across the site and therefore will not cause differential settlements. The staff concurs in these conclusions.

The main plant structures have been designed for a maximum groundwater level of el -70. Parts of structures below this level have been waterproofed and therefore a permanent dewatering system is not provided.

2.5.4.2 Groundwater Control, Excavation and Backfill

2.5.4.2.1 Groundwater Control

Groundwater levels were controlled during construction by pumping from 44 wells, 12-inch diameter, located around the perimeter of the main plant area. The well tips were located at the top of the Pascagoula Clay Layer (about el -40) and redundant pumping capability was provided. The wells were operated from May 1976 to May 1977 and from May 1979 through mid 1981.

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The groundwater level was maintained at least 10 ft below the excavation or backfill surface throughout construction to ensure that the working surface was kept dry. The groundwater levels were monitored during construction using 46 monitoring wells (see FSAR Figure 2.5-73). The applicant's conclusion that future groundwater level monitoring will not be required is evaluated in the staff's discussion of groundwater, provided in SER Section 2.4.

2.5.4.2.2 Excavation and Backfill

The dewatered main plant area was excavated to el +20, using normal earthmoving equipment, to remove the [redacted] materials that were not suitable for foundation support. The dense, Citronelle channel deposits exposed at final excavation level were proof-rolled, geologically mapped and photographed.

The soil backfill materials used to establish foundation level (El +55 to el +65) were uniform clean sands, obtained from an off-site source (see SER Section 2.5.4.2.3 for a discussion of backfill soil properties). The compaction characteristics of the backfill soils were determined from test fills at the borrow pit. The staff and its consultants (U.S. Army Corps of Engineers) visited the site during this phase of construction and found the excavation and proposed backfill program to be acceptable.

The backfill soil was placed in 10-inch (loose) layers and compacted by an 8-ton vibratory roller to a nominal relative density of at least 70%

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(ASTM D2049) and a minimum acceptable value of 60%. The applicant's statistical analyses of field test results (FSAR Figure 2.5-77) show that the compaction criteria were met; the mean relative density for all tests was 94% with a standard deviation of 10%.

Laboratory tests (FSAR Figure 2.5-74) on the backfill materials show that the fill soil was within the design gradation limits.

Based on the applicant's summaries of field tests, the staff has concluded that the structural backfill beneath seismic Category 1 structures has been properly placed and meets the design criteria discussed below, as required by subsection 2.5.4.5 of the SRP and 10 CFR Part 50, Appendix A, General Design Criterion 2.

2.5.4.2.3 Backfill Design

The applicant investigated a number of potential backfill sources and selected the Lambert Sand and Gravel pit on the west bank of Bayou Saran as the sole source. The pit is located about 8 miles northwest of the site (see FSAR Figure 2.5-1 for location map).

Before the start of plant construction, a subsurface investigation was made in the Lambert pit area and soil samples were tested to determine their properties. The soils are uniform, clean sands having a maximum size of about 1/2 inch and less than 5% silt and clay fines (material finer than 0.074 mm). The backfill soil has a uniformity coefficient of about 1.7 to 2.0 and is classified as SP in accordance with the Unified

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Soil Classification System (ASTM D-2487). Consolidated-drained triaxial tests on samples with relative densities of 70% to 90% gave an angle of internal friction of 36 to 40 degrees and zero cohesion. For design, the angle of internal friction was conservatively taken as 36 degrees.

Laboratory resonant column tests were made on compacted backfill material to establish design values of shear modulus and damping at strain levels anticipated during the SSE. These tests were performed in lieu of field determinations of seismic shear wave velocities in the backfill (discussed in staff question 2.5.4.5 in the PSAR). The results show (FSAR Figure 2.5-74) that the shear modulus of the compacted backfill is similar to the shear modulus of the Citronelle channel deposit and the Pascagoula clay and within the range of shear modulus values selected for parametric studies (18 ksi ± 33%). The staff concurs in the applicant's determination of shear modulus values for the structural backfill.

The applicant performed consolidated, undrained, cyclic triaxial tests on 89 samples of laboratory compacted backfill material to determine their response to dynamic loading for use in liquefaction analyses. These tests and analyses show that backfill material compacted to at least 70% relative density will not liquefy when subjected to the motion of the SSE, as discussed in Section 2.5.4.3.3 of this SER. Therefore, the staff concurs in the applicant's selection of a minimum relative density of 70% for the compacted backfill material.

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2.5.4.3 Foundation Stability

2.5.4.3.1 Static Loading

The main plant area was excavated to about el +20 to remove unsuitable shallow soils, and the area was backfilled to foundation level with well-compacted, cohesionless backfill (see Section 2.5.4.2.2 of this SER). The plant foundations are supported on the backfill, generally at el +55 to el +65. The final ground surface around the plant is el +95. The design maximum groundwater level is el +70 and the normal groundwater level at the site is at about el +57 (see FSAR Section 2.4.13.5).

The static design criteria for the major structures are provided in FSAR Tables 2.5-17. The maximum foundation load is 8 ksf (reactor building), the maximum net foundations load is 2.8 ksf because the foundations are about 40 ft below original grade. The applicant has calculated the factor of safety against a bearing capacity failure of more than 40. The staff agrees that there is no potential for a bearing capacity failure.

The applicant used a finite element analysis method to make predictions of anticipated stresses and deformations associated with excavation, dewatering, backfilling and structural loading. Soil movements have been recorded since the start of construction (as discussed in SER Section 2.5.4.4) and the measured movements are compatible with the calculated values (see FSAR Figure 2.5-86 through 88). The maximum heave measured was about 0.6 ft compared to the estimate of about 0.9

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ft. Recompression during backfill construction was about 0.2 ft compared to the estimate of about 0.5 ft. Thus, the final movement of the foundation soil was very close to the estimated value determined from the finite element analyses. Additionally, the applicant predicted that the settlement of the building foundations would be small and that essentially all the settlement would occur during construction, because the net foundation loads are small, the soils are dense, and because settlements of permeable cohesionless soils occur rapidly. The records of foundation settlement show that the applicant's predictions have been reasonable through 1978. We require the applicant to update the FSAR to show time rates of foundation settlements that verify the attenuation assumptions for these settlements.

The influence of regional growth faulting, petroleum extraction and long-term groundwater extraction on the stability of the plant have been studied by the applicant. These phenomena are not expected to have adverse effects on the plant foundations. The staff's evaluations of the applicant's analyses of growth faulting and petroleum extraction are presented in SER Section 2.5.1 and of groundwater extraction in SER Section 2.4.

In response to staff ~~question~~ request 241.1, the applicant committed to monitor structure settlements until there is essentially no movement during a 3 month period. Subsequently one marker on each seismic Category 1 building will be monitored biennially for 10 years and following a seismic event equal to or greater than the OBE. This monitoring

requirement is to be made a Technical Specification. The staff concurs in this approach but we also require the following:

a) readings should be made on at least three markers per building to detect differential settlement within buildings and,

b) readings should be continued through the operating life of the plant because conditions that could produce settlements (e.g., groundwater withdrawal) will continue, and

c) settlement readings should be evaluated by a Registered Professional Engineer. If any unexpected conditions develop, corrective action should be proposed and submitted to NRC for review and approval.

2.5.4.3.2 Dynamic Loading

The applicant has adopted a horizontal ground surface acceleration of 0.10g for the SSE and 0.05g for the OBE, based on the minimum requirements of 10 CFR 100 (see Section 2.5.2 of this SER). The design response spectra are provided in FSAR Section 3.7A. The staff's evaluation of the earthquake design basis is discussed in Section 2.5.2 of this SER.

The response of the plant structures to earthquake loading was developed using the response spectra modal analysis method. The structures and supporting soil were modeled as masses and springs as described in FSAR Section 3.7.2. The analyses were made for SSE and OBE loadings using

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soil shear modulus values of 12 ksi, 18 ksi and 24 ksi, (shear modulus varying by ±33%). The shear modulus values used are considered reasonable by the staff (see Section 2.5.4.1.3 of this SER). The staff's evaluation of the dynamic analyses is presented in Section 3.7 of this SER.

Using the results of the dynamic analyses, the applicant calculated overturning and sliding factors of safety for the structures during seismic loading where the resistance to overturning is provided by the dead weight of the structures and resistance to sliding is provided by base friction between the structural basemat and the foundation soil. The base friction was taken as 90% of the soil internal friction angle. The soil internal friction angle for compacted backfill was taken as 36 degrees (no cohesion) as discussed in Section 2.5.4.2.3 of this SER. Resistance that will be provided by passive soil pressures and wall friction were neglected in the analyses. Factors of safety for the major structures (see FSAR Table 2.5-16) were greater than 1.1 against sliding and 2.6 against overturning, because conservative assumptions were made in the analyses, the staff concurs that the structures will be adequately supported during earthquake loading, as required by subsection 2.5.4.7 of the SRP and 10 CFR Part 50, Appendix A, General Design Criterion 2.

2.5.4.3.3 Liquefaction Potential

The shallow in situ soils above el +20 were excavated and replaced by well-compacted granular backfill beneath structures because in some

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areas these shallow soils were loose and had the potential for liquefaction during the SSE.

The applicant analyzed the liquefaction potential of the foundation soils by comparing the soil shear strengths under cyclic loading conditions to the dynamic shear stresses induced in the soils by the vibratory motion associated with the SSE. This comparison was made at various depths below the ground surface to obtain factors of safety for liquefaction.

The soil shear strengths under cyclic loading were determined from laboratory tests on the backfill material and from published data for the underlying channel deposits, as follows.

a) The backfill material cyclic shear strength was determined at relative densities ranging from 60% to 90%, at confining pressures ranging from 1.5 kgm/sq cm to 4.5 kgm/sq. cm. The cyclic shear strength was taken as a value corresponding to a double amplitude strain of 2.5% due to 10 cycles of loading. These criteria are conservative because the test results show that a 2.5% sample strain occurred before the samples liquified; also, the design SSE is expected to have a duration of less than 10 equivalent cycles of loading,

b) The underlying channel deposits could not be sampled for cyclic testing (see Section 2.5.4 of this SER) so their cyclic shear

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strength was determined from curves published by Seed and Idriss (1971). The staff agrees that the selected cyclic shear strengths are conservative because the presence of gravel, that increases cyclic shear strength, was neglected.

The dynamic shear stresses induced in the foundation soils by the vibratory motion associated with the SSE were determined by the method proposed by Seed and Idriss (1971). The calculated factors of safety against soil liquefaction were greater than 4.1 for the backfill at a relative density of 70%, greater than 2.6 for the backfill at a relative density of 60%, and greater than 3.0 for the underlying channel deposits. The staff finds the analyses of liquefaction acceptable and concurs in the applicant's conclusion that there is an adequate factor of safety against liquefaction of the soils that support safety related structures at the River Bend site, as required by 10 CFR Part 50, Appendix A, General Design Criterion 2 and subsection 2.3.4.8 of the SRP.

2.5.4.3.4 Lateral Loads

The lateral loads on subsurface walls were determined using an "at rest" coefficient of earth pressure equal to 0.45 at depths greater than 20 ft below final grade. At shallower depths the coefficient was linearly increased, to account for stresses induced by compaction, to a maximum value of 1.2 at a depth of 6 ft. and extending to the ground surface.

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Earthquake-induced dynamic lateral loads on subsurface walls were calculated by the method of Seed and Whitman (1970).

The methods used to calculate lateral loads on subsurface walls are acceptable and in accordance with the state-of-the-art methods required by subsection 2.5.4.7 of the SRP.

2.5.4.4 Instrumentation and Monitoring

Soil heave and recompression during construction were monitored using seven extensometers located in the excavation area and on the excavation slopes as shown on FSAR Figure 2.5-83. The extensometers were anchored at a depth of about 300 ft (el -200) in the Pascagoula clay. Sensors were located near the top of the clay (el -40), about the mid-point of the foundation sand layer (el -15), and 5 ft below the base of the excavation (el +15). The sensors were read electronically (accuracy 0.02 inches) during excavation and backfilling. Because some movement of the deeper clay was expected, telltale rods were attached to the basal anchors and monitored by first-order, level survey (accuracy 0.02 inches).

After building construction started, movements were recorded on about 70 settlement markers placed on buildings as shown on FSAR Figure 2.5-85.

The results of the program to monitor soil and structure movements and the proposed on-going settlement monitoring program are discussed in Section 2.5.4.3.1 of this SER.

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The groundwater monitoring program that started before construction and the proposed on-going monitoring program are discussed in 2.4 of this SER.

2.5.4.5 Conclusion

Based on the applicant's design criteria, construction reports, and on the results of the applicant's investigations, field tests and analyses, the staff has concluded that the applicant's submittals are in accordance with Section 2.5.4 of the SRP and the foundation engineering aspects of the River Bend Station are adequate to assure the safety of the safety-related plant structures, as required by 10 CFR Part 50, Appendix A, General Design Criteria and 10 CFR Part 100, Appendix A, Reactor Site Criteria.

2.5.5 Stability of Slopes

There are no slopes proximate to the plant whose failure could damage safety-related structures. The closest slope is a 30-ft high cut slope more than 480 ft north of the main plant structures. Slopes along Grants Bayou, which is the principal drainageway from the plant, were conservatively assumed to fail and produce failed slopes of 20:1 (horizontal:vertical). The applicant asserts that the assumed failure would not cause flooding of the plant. We agree that the applicant has adopted conservative assumptions of potential slope behavior. The staff's evaluation of the hydrologic implications of these assumptions is presented in Section 2.4 of this SER.

2.5.6 Embankments and Dams

There are no embankments or dams associated with the River Bend plant.

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Figure 2.5-1

Generalized Subsurface Profile

Approx el (ft)	Depth (ft)	Soil Description	Notes
95-105		Final grade, el +95 to +105	Original grade el +95 to el +11
	10	Loess, clayey silt	Removed from plant area SPT = 10+ blows/ft
85-95			
	10	Port Hickey, top stratum silts and clays	Removed from plant area, SPT = 10+ blows/ft
75-85			
	30±	Port Hickey and Citronelle sands and clayey sands (Foundations El+55 to +65)	Removed from plant area. SPT = 8 to 100+ blows/ft Fine sand with more than 10% clay
50±		(Groundwater el +50 to +55)	
	30±	Citronelle buried channel deposits (coarse sands)	Removed from plant area to El +20.
20±		(Bottom of Fill El +20)	SPT = 8 to 40 blows/ft
	60+		SPT = 40+ blows/ft (relative density = 70+%) Lenses of silty clay and clay balls below El -20 (SPT -10 blows/ft)
-40		Pascagoula clay	Overconsolidated, stiff clay, Unconfined compressive strength = 5+ Kgm/sq cm (70 ksf)

Fill

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3 DESIGN ~~CRITERIA FOR~~ of STRUCTURES, SYSTEMS, ~~AND~~ COMPONENTS, Equipment and

3.1 ~~General~~ Conformance with NRC General Design Criteria

3.2 Classification of Structures, Systems, and Components

3.2.1 Seismic Classification

The staff has reviewed FSAR Section 3.2.1 and finds it to be generally acceptable. A few items requiring clarification and further justification have been identified regarding the classification of certain systems associated with the diesel generators. These open items have been transmitted to the applicant. Upon resolution of these open items, the staff findings will be as described below.

Structures, systems, and components (excluding electrical features) that are important to safety and that are required to withstand the effects of an SSE and remain functional have been classified as seismic Category I items and have been identified in an acceptable manner in FSAR Table 3.2-1 and on system piping and instrumentation diagrams (P&IDs) in the FSAR. Other structures, systems, and components that may be required for operation of the facility (excluding electrical features) need not be designed to seismic Category I requirements. The structures, systems, and components not required to be designed to seismic Category I include those portions of Category I systems such as vent lines, drain lines, fill lines, and test lines on the downstream side of isolation valves and those portions of the systems that are not required to perform a safety function.

The staff concludes that the structures, systems, and components important to safety have been properly classified as seismic Category I items and meet the requirements of GDC 2, "Design Bases for Protection Against Natural Phenomena," and 10 CFR 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants." The applicant has met these requirements by having properly

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classified structures, systems, and components important to safety as seismic Category I items in accordance with the positions of RG 1.29, "Seismic Design Classification." The staff has concluded that the structures, systems, and components identified by the applicant are the plant features necessary to ensure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, and (3) the capability to prevent and mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10 CFR 100.

3.2.2 System Quality Group Classification

The staff has reviewed FSAR Section 3.2.2 and find it generally acceptable. The applicant has taken exception to certain sections of RG 1.26. Although some of these exceptions are acceptable, others require further clarification and justification before they can be accepted. These exceptions have been addressed in questions transmitted to the applicant. Upon resolution of these open issues, the staff's findings will be as described below.

Pressure-retaining components of fluid systems important to safety--such as pressure vessels, heat exchangers, storage tanks, pumps, piping, and valves--have been classified Quality Group A, B, C, or D and have been identified in an acceptable manner in FSAR Table 3.2-3 and on system P&IDs in the FSAR. These components have been constructed to quality standards commensurate with the importance of the safety function to be performed. The review of Quality Group A and B (ASME Code, Section III, Class 1 and 2) reactor coolant pressure boundary components is discussed in Section 5.2.1.1 of this report. Other Quality Group B components of systems identified in positions C.1.a through C.1.e of RG 1.26 are constructed to the requirements of ASME Code, Section III, Class 2. Components in systems identified in positions C.2.a through C.2.d of RG 1.26 are constructed to Quality Group C standards, ASME Code, Section III, Class 3. Components in systems identified in position C.3 of RG 1.26 are constructed to Quality Group D standards such as ASME Code, Section VIII and ANSI B31.1.

The staff concludes that pressure-retaining components of fluid systems important to safety have been properly classified as Quality Group A, B, C, or D items and meet the requirements of GDC 1, "Quality Standards and Records." The staff has reached this conclusion because (1) applicant has met the requirements of GDC 1 by having properly classified these pressure-retaining components important to safety as Quality Group A, B, C, or D in accordance with the positions of RG 1.26, and (2) the staff has determined that the identified pressure-retaining components are those necessary (a) to prevent or mitigate the consequences of accidents and malfunctions originating within the reactor coolant pressure boundary, (b) to permit shutdown of the reactor and maintain it in a safe shutdown condition, and (c) to contain radioactive materials.

~~3.3 Wind and Tornado Loadings~~

~~3.4 Water Level (Flood) Design~~

~~3.4.1 Flood Protection~~

To ensure conformance with the requirements of GDC 2 with respect to protection against flooding, the staff reviewed the overall plant flood protection design, including systems and components whose failure as a result of flooding could prevent safe shutdown of the plant or result in uncontrolled release of significant radioactivity.

The applicant has established plant grade (el 94 ft-6 in.) approximately 30 ft above the probable maximum flood (PMF) level of the Mississippi River (el 60 ft). Local stream flooding as a result of probable maximum precipitation (PMP) is shown to flood the plant yard to a depth of 7 in. above the plant grade (to el 95 ft-1 in.), thus establishing the design-basis flood level. The applicant reports that these calculations were made in accordance with the guidelines of RG 1.59, "Design Basis Floods for Nuclear Power Plants." However, the staff has not verified the applicant's calculations. (See Section 2.4 of this SER for further discussion on the PMF and PMP levels.)

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ENCLOSURE 2

RIVER BEND STATION - UNIT NO. 1
OPERATING LICENSE REVIEW
~~STRUCTURAL AND GEOTECHNICAL ENGINEERING BRANCH~~
DOCKET NO. 50-458
DRAFT SAFETY EVALUATION REPORT

3.3 Wind and Tornado Loadings

3.3.1 Wind ~~Design Criteria~~ Loadings

All Category I structures exposed to wind forces were designed to withstand the effects of the design wind. The design wind specified has a velocity of 100 mph based on a recurrence of 100 years.

The procedures that were used to transform the wind velocity into pressure loadings on structures and the associated vertical distribution of wind pressures and gust factors are in accordance with ANSI A58.1 and ASCE Paper 3269 as appropriate. This document is acceptable to the staff.

The staff concludes that the plant design is acceptable and meets the recommendations of Standard Review Plan Section 3.3.1 and the requirements of General Design Criterion 2. This conclusion is based on the following:

The applicant has met the requirements of GDC 2 with respect to the capability of the structures to withstand design wind loading so that their design reflects

1. appropriate consideration for the most severe wind recorded for the site with an appropriate margin;

2. appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena; and

3. the importance of the safety function to be performed.

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The applicant has met these requirements by using ANSI A58.1 and ASCE paper No. 3269, which the staff has reviewed and found acceptable, to transform the wind velocity into an effective pressure on structures and for selecting pressure coefficients corresponding to the structures geometry and physical configuration.

The applicant has designed the plant structures with sufficient margin to prevent structural damage during the most severe wind loadings that have been determined appropriate for the site so that the requirements of Item 1 listed above are met. In addition, the design of seismic Category 1 structures, as required by Item 2 listed above, has included in an acceptable manner, load combinations which occur as a result of the most severe wind load and the loads resulting from normal and accident conditions.

The procedures used to determine the loadings on structures induced by the design wind specified for the plant are acceptable since these procedures have been used in the design of conventional structures and proven to provide a conservative basis which together with other engineering design consideration assures that the structures will withstand such environmental forces. The use of these procedures provides reasonable assurance that in the event of design basis winds, the structural integrity of the plant structures that have to be designed for the design wind will not be impaired and, in consequence, safety-related systems and components located within these structures are adequately protected and will perform their intended safety functions if needed, thus satisfying the requirement of Item 3 listed above.

3.3.2 Tornado ~~Design Criteria~~ Loadings

All Category I structures exposed to tornado forces and needed for the safe shutdown of the plant were designed to resist a tornado of 290 mph tangential wind velocity and a 70 mph translational wind velocity. The simultaneous atmospheric pressure drop was assumed to be 3 psi in 2 seconds. Tornado missiles are also considered in the design as discussed in Section 3.5 of this report.

The staff concludes that the plant design is acceptable and meets the recommendations of Standard Review Plan 3.3.2 and the requirements of General Design Criterion 2. This conclusion is based on the following:

The applicant has met the recommendations of Standard Review Plan 3.3.2 and the requirements of GDC 2 with respect to the structure capability to withstand design tornado wind loading and tornado missiles so that their design reflects

1. appropriate consideration for the most severe tornado recorded for the site with an appropriate margin;

2. appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena; and

3. the importance of the safety function to be performed.

The applicant has met these requirements by using ANSI A58.1 and ASCE paper No. 3269, which the staff has reviewed and found acceptable, to transform the wind velocity generated by the tornado into an effective pressure on structures and for selecting pressure coefficients corresponding to the structures geometry and physical configuration.

The applicant has designed the plant structures with sufficient margin to prevent structural damage during the most severe tornado loadings that have been determined appropriate for the site so that the requirements of Item 1 listed above are met. In addition, the design of seismic Category I structures, as required by Item 2 listed above, has included in an acceptable manner load combinations which occur as a result of the most severe tornado wind load and the loads resulting from normal and accident conditions.

The procedures utilized to determine the loadings on structures induced by the design basis tornado specified for the plant are acceptable since these procedures have been used in the design of conventional structures and proven to provide a conservative basis which together with other engineering design considerations assures that the structures will withstand such environmental forces.

The use of these procedures provides reasonable assurance that in the event of design basis tornado, the structural integrity of the plant structures that have to be designed for tornadoes will not be impaired and, in consequence, safety-related systems and components located within these structures will be adequately protected and may be expected to perform necessary safety functions as required, thus satisfying the requirement of item 3 listed above.

3.5.2 Maximum Level (Flood) Design Procedures

The maximum postulated design basis flood level (DBFL) for the River Bend site is at elevation 95.1*. Plant grade is approximately at elevation 94.5. Plant structures are designed to a DBFL at elevation 70.0. Normal ground water is at elevation 57.0.

The applicant has stated that the DBFL to elevation 95.1 will only cause the groundwater to rise 13 feet to elevation 70.0. However such an explanation is not fully understood. The site is essentially sand which should offer little resistance to water penetration. In addition, the applicant has not furnished information about the potential effects of flotation, overturning and structural loadings in the event that the maximum postulated DBFL of 95.1 feet occurs.

The applicant is required to resolve these discrepancies and furnish objective evidence of his position that a flood to elevation 95.1 will result in hydrostatic forces on seismic Category I structures no higher than elevation 70. Alternatively, the applicant may show that flood water hydrostatic heads to elevation 95.1, in combination with other applicable loads, will not produce any adverse effects on seismic Category I structures.

3.5.3 Barrier Design Procedures

The plant Category I structures, systems and components are shielded from, or designed for, various postulated missiles. Missiles considered in the design of structures include tornado generated missiles and various containment internal missiles, such as those associated with a loss-of-coolant accident.

*All elevations are given in feet above mean sea-level (MSL).

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ENCLOSURE 1

DRAFT SAFETY EVALUATION REPORT

RIVER BEND STATION

AUXILIARY SYSTEMS BRANCH

3.4 Water Level (Flood) Design

3.4.1 ~~Water Level (Flood) Design~~ Flood Protection

The water level (flood) design was reviewed in accordance with Section 3.4.1 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed, according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the water level (flood) design with respect to the applicable regulations of 10 CFR 50.

In order to assure conformance with the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," with respect to protection against flooding, we reviewed the overall plant flood protection design, including systems and components whose failure due to flooding could prevent safe shutdown of the plant or result in uncontrolled release of significant radioactivity.

The applicant has established plant grade at elevation 94 ft. - 6 inches; this level is approximately 30 ft. above

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the probable maximum flood (PMF) level of the Mississippi River (elevation 60 ft.); this is a dry site with regard to the Mississippi River as defined by Regulatory Guide 1.59. "Design Basis Floods For Nuclear Power Plants," as discussed in Section 2.4 of this SER. The calculation of local stream flooding due to a probable maximum precipitation (PMP) is shown to flood the plant yard to a depth of 7 inches above the plant grade (to 95 ft. 1 inch), which is the Design Basis Flood Level. The applicant states that these calculations were made in accordance with the guidelines of Regulatory Guide 1.59.

Safety-related systems and components that must be protected against flooding have been identified and are located in safety-related seismic Category I structures. All penetrations in these structures are watertight to the Design Basis Flood Level. Further, the structures are provided with waterstops in all construction joints below the Design Basis Flood Level.

Within these structures, protection against internal flooding from failures in fluid piping systems as identified in the guidelines of Branch Technical Position ASB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," is provided by placing critical equipment in watertight cubicles. This matter is discussed in

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more detail in Sections 3.6.1 and 9.3.3 of this SER. However, the applicant has not considered the effect of a break in a moderate energy line within the control building. [The applicant must show that all safety related equipment in the control building is protected against flooding as a result of a moderate energy line within the control building.] ~~[see Section 9.4.1 of this SER.]~~

Based on our review of the design criteria and bases, and of the safety classification of safety-related structures, systems, and components necessary for a safe plant shutdown during flood conditions, we conclude that the design of the facility provides adequate protection against flooding as a result of the PMP which will attain a level of 95 ft. 1 inch. (Compliance with Regulatory Guides 1.59, "Design and 1.102, "Flood Protection for Nuclear Power Plants" Basis Floods For Nuclear Power Plants," is discussed in Section 2.4 of this SER). The plant is protected from flooding as a result of a high or moderate energy break within a structure except for the possibility of flooding within the control building as noted above. With that exception the plant complies with the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena." However because of this exception we cannot conclude that the applicant's submittal is acceptable since the water level (flood) design does not comply with the requirements of SRP Section 3.4.1.

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The use of these procedures provides reasonable assurance that in the event of design basis tornado, the structural integrity of the plant structures that have to be designed for tornadoes will not be impaired and, in consequence, safety-related systems and components located within these structures will be adequately protected and may be expected to perform necessary safety functions as required, thus satisfying the requirement of item 3 listed above.

3.4.2 Analytical and Test ~~Water Level (Flood) Design~~ Procedures

The maximum postulated design basis flood level (DBFL) for the River Bend site is at elevation 95.1*. Plant grade is approximately at elevation 94.5. Plant structures are designed to a DBFL at elevation 70.0. Normal ground water is at elevation 57.0.

The applicant has stated that the DBFL to elevation 95.1 will only cause the groundwater to rise 13 feet to elevation 70.0. However such an explanation is not fully understood. The site is essentially sand which should offer little resistance to water penetration. In addition, the applicant has not furnished information about the potential effects of flotation, overturning and structural loadings in the event that the maximum postulated DBFL of 95.1 feet occurs.

The applicant is required to resolve these discrepancies and furnish objective evidence of his position that a flood to elevation 95.1 will result in hydrostatic forces on seismic Category I structures no higher than elevation 70. Alternatively, the applicant may show that flood water hydrostatic heads to elevation 95.1, in combination with other applicable loads, will not produce any adverse effects on seismic Category I structures.

3.5.3 Barrier Design Procedures

The plant Category I structures, systems and components are shielded from, or designed for, various postulated missiles. Missiles considered in the design of structures include tornado generated missiles and various containment internal missiles, such as those associated with a loss-of-coolant accident.

*All elevations are given in feet above mean sea-level (MSL).

3.5 Missile Protection

3.5.1 Missile Selection and Description

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3.5.1.1 Internally Generated Missiles (Outside Containment)

Internally Generated Missiles (Outside Containment) were reviewed in accordance with Section 3.5.1.1 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed, according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of Internally Generated Missiles (Outside Containment) with respect to the applicable regulations of 10 CFR 50.

General Design Criterion 4, "Environmental and Missile Design Bases," requires that plant structures, systems, and components required for safe shutdown be protected against damage from postulated missiles associated with plant operation or when such ~~failure~~ damage could lead to inoperability of these systems and/or to offsite radiological consequences. The missiles outside containment considered in this evaluation include those missiles generated by rotating or pressurized (high energy fluid system) equipment. Protection is provided by any one or a combination of compartmentalization, barriers, separation, and equipment design. The primary means of providing protection to safety-related equipment from

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damage resulting from internally generated missiles is through the use of plant physical arrangement. Safety-related systems are physically separated from nonsafety-related systems and redundant components of safety-related systems are physically separated such that potential missiles could not damage both trains of safety-related equipment. Stored spent fuel in the fuel building is protected from damage by internal missiles which could result in radioactive release in accordance with the guidelines of Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," by the fuel pool walls and by not locating high energy piping systems or rotating machinery in the vicinity of new or spent fuel.

The applicant's evaluation of potential missile sources from rotating equipment and pressurized equipment is incomplete since only missiles from the main turbine were considered. The applicant did not provide sufficient justification for not taking into account such items as valve bonnets, thermowells, nuts, bolts, studs and valve stems as possible missiles. (The applicant should justify this position.) We are also concerned with possible failure of fans and the

possibility of fan blades becoming missiles in the heating, ventilating and air conditioning systems reviewed in Sections 9.4-1 through 9.4.5 of this SER. (The applicant should justify the position that parts of fans rotating at high speed will either not become missiles or that such missiles will not have any effect on plant safety.) (The applicant should evaluate the potential for gravitational missiles.) The submittal does not show that gravitational missiles were considered.

Protection of safety-related equipment and stored fuel from the effects of turbine missiles including compliance with the guidelines of Regulatory Guide 1.115, "Protection Against Low-Trajectory Turbine Missiles," is discussed in Section 3.5.1.3 of this SER.

We have reviewed the adequacy of the applicant's design to maintain the capability for a safe plant shutdown and prevent unacceptable radiological release in the event of internally generated missiles outside containment. We conclude that the design is in conformance with the guidelines of Regulatory Guide 1.113 as it relates to protection of spent fuel

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from internal missiles. However, until the question of potential missiles such as valve bonnets, thermowells, valve stems, and gravitational missiles parts of fans is resolved, we cannot find that the applicant's design is in conformance with the requirements of General Design Criterion 4. Therefore, we cannot conclude that the applicant's submittal is acceptable. The design of the plant for protection against internally generated missiles does not comply with the requirements of SRP Section 3.5.1.1.

3.5.1.2 Internally Generated Missiles (Inside Containment)

Internally Generated Missiles (Inside Containment) were reviewed in accordance with Section 3.5.1.2 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed, according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of Internally Generated Missiles (Inside Containment) with respect to the applicable regulations of 10 CFR 50.

All plant structures, systems, and components inside containment whose failure could lead to offsite radiological consequences or that are required for safe plant shutdown require protection against the effects of internally generated missiles in accordance with the requirements of General Design Criterion 4, "Environmental and Missile Design Bases." Potential missiles that could be generated inside containment are from failures of rotating components, pressurized components, and high energy fluid system failures.

(The applicant should evaluate the potential for gravitational missiles inside containment.) The applicant's submittal does not show that gravitational missiles were considered.

Spent fuel within the containment is stored in an area that is not located in the vicinity of any high energy lines or rotating machinery. The upper containment pool walls also provide protection from potential internally generated missiles. Thus, the guidelines of Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," are satisfied.

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The information provided by the applicant does not address, in detail, missiles resulting from the failure of pressurized equipment such as valve bonnets, thermowells, nuts, bolts, studs, and valve stems. [The applicant should justify the position that the items addressed above are not considered potential missiles or provide the results or analyses which show that such failures and the resulting missiles are not a problem.] (The applicant should also address the concern that failure of fans within containment could result in missiles which could endanger safety-related equipment.)

The applicant must provide further information to address the concerns of potential falling missiles, potential missiles generated by pressurized equipment and potential missiles generated by the failure of fans in order to comply fully with the requirements of General Design Criterion 4. Therefore, we cannot find the applicant's design regarding internally generated missiles inside containment to be acceptable nor that the design complies with the criteria of SRP Section 3.5.1.2.

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3.5.1.4 Missiles Generated by Natural Phenomena

The section of the FSAR regarding missiles generated by natural phenomena was reviewed in accordance with Section 3.5.1.4 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed, according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of missiles generated by natural phenomena with respect to the applicable regulations of 10 CFR 50.

General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," requires that structures, systems and components essential to safety be designed to withstand the effects of natural phenomena, and General Design Criterion 4, "Environmental and Missile Design Bases," requires that these same plant features be protected against missiles. The missiles generated by natural phenomena of concern are those resulting from tornadoes. The applicant has identified a spectrum of missiles for the tornado region in which the site is located (Region I) as identified in Regulatory Guide 1.76, "Design Basis Tornado For Nuclear Power Plants." The spectrum includes the weight, velocity and impact

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area. The missile spectrum chosen is identical to missile Spectrum A of Standard Review Plan 3.5.1.4 (NUREG-0800). We have evaluated this spectrum and conclude it is representative of missiles at the site and is, therefore, acceptable. A discussion of the protection afforded safety-related equipment from the identified tornado missiles including compliance with the guidelines of Regulatory Guide 1.117, "Tornado Design Classification," is provided in Section 3.5.2 of this SER. A discussion of the adequacy of barriers and structures designed to withstand the effects of the identified tornado missiles is provided in Section 3.5.3.

Based upon our review of the tornado missile spectrum, we conclude that the spectrum was properly selected and meets the requirements of General Design Criteria 2 and 4 with respect to protection against natural phenomena and missiles and the guidelines and Positions C.1 and C.2 of Regulatory Guides 1.76 and the guidelines of Regulatory Guide 1.117 with respect to identification of missiles generated by natural phenomena and is, therefore, acceptable. The tornado missile spectrum complies with the acceptance criteria of SRP Section 3.5.1.4.

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3.5.2 Structures, Systems, and Components To Be Protected From Externally Generated Missiles

Structures, systems and components to be protected from externally generated missiles was reviewed in accordance with Section 3.5.2 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed, according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the structures, systems and components to be protected from internally generated missiles with respect to the applicable regulations of 10 CFR 50.

General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," requires that all structures, systems and components essential to the safety of the plant be protected from the effects of natural phenomena, and General Design Criterion 4, "Environmental and Missile Design Bases," requires that all structures, systems and components essential to the safety of the plant be protected from the effects

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of externally generated missiles. The River Bend site is located in tornado Region I as identified in Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants." The tornado missile spectrum is discussed in Section 3.5.1.4 of this SER.

The applicant has identified all safety-related structures, systems and components requiring protection from externally generated missiles. All safety-related structures are designed to withstand postulated tornado-generated missiles without damage to safety-related equipment. All safety-related systems and components and stored fuel are located within tornado-missile-protected structures or are provided with tornado missile barriers. Buried safety-related systems such as piping and electrical circuits are adequately protected by the overlying earth. The ultimate heat sink consists of a water storage basin and standby cooling tower which is designed to seismic Category I and Safety Class 3 criteria and to withstand the effects of natural phenomena including tornadoes and tornado missiles. (Compliance of the design of the ultimate heat sink with the guidelines of Positions C.2 and C.3 of

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Regulatory Guide 1.27 is discussed in Section 9.2.5 of this SER.) Thus, the requirements of General Design Criteria 2 and 4 with respect to missile protection and the guidelines of Position C.2 of Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," Positions C.2 and C.3 of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants," and Positions C.1, C.2, C.3 and the Appendix of Regulatory Guide 1.117, "Tornado Design Clsssification," concerning tornado missile protection for safety-related structures, systems and components including stored fuel and the ultimate heat sink, are met. Protection from low trajectory turbine missiles, including compliance with Position C.1 of Regulatory Guide 1.115, "Protection Against Low Trajectory Turbine Missiles," is discussed in Section 3.5.1.3 of this SER.

Based on the above, we conclude that the applicant's identification of safety-related structures, systems and components to be protected from externally generated missiles and the protection provided in the plant design is in accordance with the requirements of General Design Criteria 2 and 4 with respect to missile

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and environmental effects by compliance with the guidelines of Regulatory Guides 1.13, 1.27, and 1.117 concerning protection of safety-related plant features including stored fuel and the ultimate heat sink from tornado missiles. The applicant has identified the structures, systems and components to be protected from externally generated missiles and the protection provided in the plant design ~~and therefor acceptable.~~ The design of the River Bend facility for providing protection from tornado generated missiles meets the acceptance criteria of SRP Section 3.5.2, and, therefore, is acceptable.

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The use of these procedures provides reasonable assurance that in the event of design basis tornado, the structural integrity of the plant structures that have to be designed for tornadoes will not be impaired and, in consequence, safety-related systems and components located within these structures will be adequately protected and may be expected to perform necessary safety functions as required, thus satisfying the requirement of item 3 listed above.

3.4.2 Analytical and Test Procedures

The maximum postulated design basis flood level (DBFL) for the River Bend site is at elevation 95.1*. Plant grade is approximately at elevation 94.5. Plant structures are designed to a DBFL at elevation 70.0. Normal ground water is at elevation 57.0.

The applicant has stated that the DBFL to elevation 95.1 will only cause the groundwater to rise 13 feet to elevation 70.0. However such an explanation is not fully understood. The site is essentially sand which should offer little resistance to water penetration. In addition, the applicant has not furnished information about the potential effects of flotation, overturning and structural loadings in the event that the maximum postulated DBFL of 95.1 feet occurs.

The applicant is required to resolve these discrepancies and furnish objective evidence of his position that a flood to elevation 95.1 will result in hydrostatic forces on seismic Category I structures no higher than elevation 70. Alternatively, the applicant may show that flood water hydrostatic heads to elevation 95.1, in combination with other applicable loads, will not produce any adverse effects on seismic Category I structures.

3.5.3 Barrier Design Procedures

The plant Category I structures, systems and components are shielded from, or designed for, various postulated missiles. Missiles considered in the design of structures include tornado generated missiles and various containment internal missiles, such as those associated with a loss-of-coolant accident.

*All elevations are given in feet above mean sea-level (MSL).

Information has been provided indicating that the procedures that were used in the design of the structures, shields and barriers to resist the effect of missiles are adequate. The analysis of structures, shields and barriers to determine the effects of missile impact was accomplished in two steps. In the first step, the potential damage that could be done by the missile in the immediate vicinity of impact was investigated. This was accomplished by estimating the depth of penetration of the missile into the impacted structure. Furthermore, secondary missiles are prevented by fixing the target thickness well above that determined for penetration. In the second step of the analysis, the overall structural response of the target when impacted by a missile is determined using established methods of impactive analysis. The equivalent loads of missile impact, whether the missile is environmentally generated or accidentally generated within the plant, are combined with other applicable loads as is discussed in Section 3.8 of this report.

The staff concludes that the barrier design is acceptable and meets the recommendations of Standard Review Plan 3.5.3 and the requirements of General Design Criteria 2 and 4 with respect to the capabilities of the structures, shields, and barriers to provide sufficient protection to equipment that must withstand the effects of natural phenomena (tornado missiles) and environmental effects including the effects of missiles, pipe whipping and discharging fluids. This conclusion is based on the following:

The procedures utilized to determine the effects and loading on seismic Category I structures and missile shields and barriers induced by design basis missiles selected for the plant are acceptable since these procedures provide a conservative basis for engineering design to assure that the structure or barriers are adequately resistant to and will withstand the effects of such forces.

The use of these procedures provides reasonable assurance that in the event of design basis missiles striking seismic Category I structures or other missile shields and barriers, the structural integrity of the structures, shields and barriers will not be impaired or degraded to an extent that will result in a loss of required protection. Seismic Category I systems and components protected

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by these structures are, therefore, adequately protected against the effects of missiles and will perform their intended safety function, if needed. Conformance with these procedures is an acceptable basis for satisfying in part the requirements of General Design Criteria 2 and 4.

3.7.1 Seismic Input

The input seismic design response spectra (OBE and SSE) applied in the design of seismic Category I structures, systems, and components comply with the recommendations of Regulatory Guide 1.60, "Design Response Spectra for Nuclear Power Plants." The specific percentage of critical damping values used in the seismic analysis of Category I structures, systems and components are in conformance with Regulatory Guide 1.61, "Damping Values for Seismic Analysis of Nuclear Power Plants."

The synthetic time history used for seismic design of Category I plant structures, systems and components is adjusted in amplitude and frequency content to obtain response spectra that envelope the response spectra specified for the site.

Conformance with the Regulatory Guides 1.60 and 1.61 requirements provides reasonable assurance that for an earthquake whose intensity is 0.05g for the OBE, and 0.1g for the SSE, the seismic inputs to Category I structures, systems, and components are adequately defined to assure a conservative basis for the design of such structures, systems and components to withstand the consequent seismic loadings.

The staff concludes that the seismic design parameters used in the plant structure design are acceptable and meet the recommendations of Standard Review Plan 3.7.1 and the requirements of General Design Criterion 2 and Appendix A to 10 CFR Part 100. This conclusion is based on the following:

The applicant has met the recommendations of Standard Review Plan 3.7.1 and the relevant requirements of GDC 2 and Appendix A to 10 CFR Part 100 by appropriate consideration for loads generated by the most severe earthquake recorded

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3.6 Protection Against Dynamic Effects Associated With The Postulated Rupture of Piping [redacted]

3.6.1 Postulated Piping Failures in Fluid Systems Outside of Containment

Protection against dynamic effects associated with the postulated rupture of piping (outside containment) was reviewed in accordance with Section 3.6.1 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed, according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of protection against dynamic effects associated with the postulated rupture of piping (outside containment) with respect to the applicable regulations of 10 CFR 50.

The staff's guidelines for meeting the requirements of General Design Criterion 4, "Environmental and Missile Design Bases," concerning protection against postulated piping failure in high-energy and moderate-energy fluid systems outside containment are contained in Branch Technical Position ASB 3-1, "Protection Against Postulated Failures in Fluid Systems Outside Containment." The applicant has identified all high

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and moderate-energy piping systems in accordance with these guidelines and has also identified those systems requiring protection from postulated piping failures. The plant design accommodates the effects of postulated pipe breaks in high energy fluid piping systems outside containment with respect to pipe whip, jet impingement, and other environmental effects, and the effects of postulated cracks in moderate-energy fluid systems outside containment with respect to jet impingement, flooding and other environmental effects. The means used to protect safety-related systems and components throughout the plant include physical separation, enclosure in suitably designed structures or compartments, drainage systems, pipe whip restraints, equipment shields, and necessary equipment and environmental qualification.

The applicant has not completed the analysis of the rupture of high energy piping systems, specifically, the high pressure core spray (HPCS), reactor core isolation cooling (RCIC), reactor water cleanup (RWCU), main steam, feedwater, and recirculation systems. (The applicant has stated that the complete analysis will be submitted in a future amendment.) For moderate

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energy systems, protection of safety-related systems from the jet, flooding and other environmental effects due to critical cracks is incorporated into the plant design. (The applicant has stated that the results of the compartment flooding analysis will be submitted in a future amendment.)

The main steam lines, including the outboard isolation valves, and the feedwater lines are all located in the common auxiliary building steam tunnel and have been classified as part of the break exclusion boundary. The applicant has performed a subcompartment analysis for the steam tunnel and the main steam lines in order to assure that the resulting jet impingement and environmental effects from a postulated nonmechanistic crack with an area equal to the pipe area in one of these lines or a feedwater line will not result in adverse consequences. The results of this analysis indicate that the steam tunnel structural integrity is not affected by the pressure increase from the resulting blowdown. The analysis also indicated that the MSIV closure is expected to terminate the blowdown from the reactor vessel at 5.5 seconds. (The applicant should provide detailed

information from this analysis for our review.) Qualification of equipment in the steam tunnel subject to the resultant environment is discussed in Section 3.11 of this SER.

Since the applicant's submittal is not complete, we cannot complete our evaluation of whether the plant design provides adequate protection for all systems required for safe plant shutdown following postulated events, including the combination of pipe failure and single active failure. Further, we cannot conclude that the plant's design meets the requirements of General Design Criterion 4 and the criteria set forth in Branch Technical Position ASB 3-1 with regard to the protection of all safety-related systems and components from a postulated high energy or moderate energy line failure. Therefore, we cannot conclude that the applicant's submittal relating to the plant design for the protection of safety-related equipment against dynamic effects associated with the postulated rupture of piping inside containment is acceptable. The plant provisions for protection against the dynamic effects associated with the postulated rupture of piping outside containment does not comply with the acceptance criteria of SRP Section 3.6.1.

cracks is incorporated into the plant design. The applicant has stated that the results of the compartment flooding analysis will be submitted in a future amendment.

The main steamlines, including the outboard isolation valves, and the feedwater lines are located in the common auxiliary building steam tunnel and have been classified as part of the break exclusion boundary. The applicant has performed a subcompartment analysis for the steam tunnel and the main steamlines to ensure that the resulting jet impingement and environmental effects from a postulated nonmechanistic crack with an area equal to the pipe area in one of these lines or a feedwater line will not result in adverse consequences. The results of this analysis indicate that the steam tunnel structural integrity is not affected by the pressure increase from the resulting blowdown. Main steam isolation valve (MSIV) functional capability is ensured by the environmental qualification of system components to the expected condition of 330°F for 1 hour. The analysis is also indicated that the MSIV closure will terminate the blowdown from the reactor vessel at 5.5 seconds.

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Because the applicant's submittal is not complete, the staff cannot find that the applicant has adequately designed and protected all of the systems required for safe plant shutdown following postulated events, including the combination of pipe failure and single active failure. Further, the staff cannot conclude that the plant's design meets the requirements of GDC 4 and the criteria set forth in BTP ASB 3-1 with regard to the protection of all safety-related systems and components from a postulated high-energy or moderate-energy line failure. Therefore, the staff concludes that the applicant's submittal relating to the plant design for the protection of safety-related equipment against dynamic effects associated with the postulated rupture of piping outside containment is not complete, as noted above, and, therefore, is not acceptable.

3.6.2 Determination of Rupture Locations and Dynamic Effects Associated With the Postulated Rupture of Piping

The staff has reviewed FSAR Section 3.6.2 and finds that, in general, it covers all topics requiring discussion. However, several areas need clarification and justification, and inconsistencies are present in a few areas.

Additional justification and clarification are required for the limits used in the criteria for the no-break zone, the pipe-break criteria, the pipe-thrust coefficients, jet-impingement analyses, and pipe-whip restraint design. Moreover, a very large amount of information will not be available until the new load program is complete. The staff review cannot be completed without this information. These open items have been transmitted to the applicant. When these open items are resolved, the staff will be able to conclude that the pipe rupture postulation and the associated effects are adequately considered in the plant design, meet the requirements of GDC 4, and are, therefore, acceptable.

The proposed pipe rupture locations have been adequately assumed, and the design of piping restraints and measures to deal with the subsequent dynamic effects of pipe whip and jet impingement provides adequate protection to the integrity and functionality of safety-related structures, systems, and components.

The provisions for protection against dynamic effects associated with pipe ruptures of the reactor coolant pressure boundary inside containment and the resulting discharging fluid provide adequate assurance that design-basis loss-of-coolant accidents will not be aggravated by sequential failures of safety-related piping, and that emergency core cooling system performance will not be degraded by these dynamic effects.

The proposed piping and restraint arrangement and applicable design considerations for high- and moderate-energy fluid systems inside and outside containment, including the reactor coolant pressure boundary, will provide adequate assurance that the structures, systems, and components important to safety that are in close proximity to the postulated pipe rupture will be protected. The design will mitigate the consequences of pipe ruptures so that the reactor can be safely shut down and maintained in a safe shutdown condition in the event of a rupture of a high- or moderate-energy piping system inside or outside of containment.

by these structures are, therefore, adequately protected against the effects of missiles and will perform their intended safety function, if needed. Conformance with these procedures is an acceptable basis for satisfying in part the requirements of General Design Criteria 2 and 4.

3.7 Seismic Design

3.7.1 Seismic Input

The input seismic design response spectra (OBE and SSE) applied in the design of seismic Category I structures, systems, and components comply with the recommendations of Regulatory Guide 1.60, "Design Response Spectra for Nuclear Power Plants." The specific percentage of critical damping values used in the seismic analysis of Category I structures, systems and components are in conformance with Regulatory Guide 1.61, "Damping Values for Seismic Analysis of Nuclear Power Plants."

The synthetic time history used for seismic design of Category I plant structures, systems and components is adjusted in amplitude and frequency content to obtain response spectra that envelope the response spectra specified for the site.

Conformance with the Regulatory Guides 1.60 and 1.61 requirements provides reasonable assurance that for an earthquake whose intensity is 0.05g for the OBE, and 0.1g for the SSE, the seismic inputs to Category I structures, systems, and components are adequately defined to assure a conservative basis for the design of such structures, systems and components to withstand the consequent seismic loadings.

The staff concludes that the seismic design parameters used in the plant structure design are acceptable and meet the recommendations of Standard Review Plan 3.7.1 and the requirements of General Design Criterion 2 and Appendix A to 10 CFR Part 100. This conclusion is based on the following:

The applicant has met the recommendations of Standard Review Plan 3.7.1 and the relevant requirements of GDC 2 and Appendix A to 10 CFR Part 100 by appropriate consideration for loads generated by the most severe earthquake recorded

for the site, as defined in FSAR Section 2.5, with an appropriate margin and considerations for two levels of earthquakes--the safe shutdown earthquake (SSE) and operating basis earthquake (OBE). The applicant has met these requirements by the use of the methods and procedures indicated below.

The seismic design response spectra (OBE and SSE) applied in the design of seismic Category I structures, systems, and components comply with the recommendations of Regulatory Guide 1.60, "Design Response Spectra for Nuclear Power Plants." The specific percentage of critical damping values used in the seismic analysis of Category I structures, systems, and components are in conformance with Regulatory Guide 1.61, "Damping Values for Seismic Analysis for Nuclear Power Plants." The artificial synthetic time history used for seismic design of Category I structures, systems, and components is adjusted in amplitude and frequency content to obtain response spectra that enevlop the design response spectra specified for the site. Conformance with the recommendations of Regulatory Guides 1.60 and 1.61 assures that the seismic inputs to Category I structures, systems, and components are adequately defined so as to form a conservative basis for the design of such structures, systems, and components to withstand seismic loadings.

3.7.2 Seismic System Analysis

close up

~~Combined with section 3.7.3~~

3.7.3 Seismic System and Subsystem Analysis

The scope of review of the Seismic System and Subsystem Analysis for the plant included the seismic analysis methods for all Category I structures, systems and components. It included review of procedures for modeling, seismic soil-structure interaction, development of floor response spectra, inclusion of torsional effects, seismic analysis of Category I dams, evaluation of Category I structure overturning, and determination of composite damping. The review has included design criteria and procedures for evaluation of interaction of non-Category I structures with Category I structures and effects of parameter variations on floor response spectra. The review has also included criteria

and seismic analysis procedures for reactor internals and Category I buried structures outside the containment.

The system and subsystem analyses were performed by the applicant on an elastic and linear basis. Modal response spectrum multidegree of freedom methods form the bases for the analyses of all major Category I structures, systems and components. In applying the modal response spectrum method, governing response parameters were combined in conformance with one position of Regulatory Guide 1.92. The absolute sum of the modal responses was used for modes with closely spaced frequencies. The square root of the sum of the squares of the maximum codirectional responses was used in accounting for three components of the earthquake motion. Floor spectra inputs used for design and test verifications of structures, systems, and components were generated from the time history method, taking into account variation of parameters by peak widening. A vertical seismic system dynamic analysis is employed for all structures, systems and components where analyses show significant structural amplification in the vertical direction. Torsional effects and stability against overturning are considered.

The inertial effects due to an earthquake upon buried systems and tunnels have been adequately accounted for in the analysis. The principles used, to account for the effects of static resistance of the surrounding soil on buried system deformations, was based on the theory of structures on elastic foundations and they are acceptable.

A coupled structure and soil model is used to evaluate soil-structure interaction effects upon seismic responses. The lumped parameter method is used for modeling of the structure. The elastic half-space method is used for modeling of the supporting soil. Translational, rocking, vertical and torsional soil spring constants are included to represent the subgrade. The staff concludes that the use of the seismic structural analysis procedures and criteria delineated above by the applicant provides an acceptable bases for the seismic design which are in conformance with the requirements of item 3 listed below.

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The staff concludes that the plant design is acceptable with exceptions as noted at the end of this section, and meets the recommendations of Standard Review Plan 3.7.3 and the requirements of General Design Criterion 2 and Appendix A to 10 CFR Part 100. This conclusion is based on the following: the applicant has met the recommendations of Standard Review Plan 3.7.3 and the requirements of GDC 2 and Appendix A to 10 CFR Part 100 with respect to the capability of the structures to withstand the effects of the earthquakes so that their design reflects:

(1) Appropriate consideration for the most severe earthquake recorded for the site with an appropriate margin (GDC 2). Consideration of two levels of earthquakes (Appendix A, 10 CFR Part 100),

(2) Appropriate combination of the effects of normal and accident conditions with the effect of the national phenomena, and

(3) The importance of the safety functions to be performed (GDC 2). The use of a suitable dynamic analysis or a suitable qualification test to demonstrate that structures, systems, and components can withstand the seismic and other concurrent loads, except where it can be demonstrated that the use of an equivalent static load method provides adequate consideration (Appendix A, 10 CFR Part 100).

The applicant has met the requirements of item 1 listed above by use of the acceptable seismic design parameters as per SRP Section 3.7.1. The combination of earthquake resulted loads with those resulting from normal and accident conditions in the design of Category I structures as specified in SRP Section 3.8.1 through 3.8.5 will be in conformance with item 2 listed above.

The following issues remain outstanding:

(1) Question 220.32 (3.7.2.8A) remains unanswered. According to amendment 7 of the FSAR, a response will be provided by June 1983. The question is: "Will the collapse of any non-Category 1 Structure impair the integrity

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of any Seismic Category 1 structure or component? If so, is the non-Category 1 Structure designed to Category 1 standards? Provide an explanation if necessary."

(2) Question 220.38 (3.7.2.16A) remains unanswered. According to amendment 7 of the FSAR, a response will be provided by June 1983. The question is: "Provide a detailed, step-by-step explanation of the method for applying earthquake input to the radwaste building; i.e., clarify and expand the last paragraph of this section."

(3) The applicant has referred to a particular paper by Gwinn and Goldstein, "Equivalent Static Loads from Amplified Response Curves," ASME Paper 74-NE-6, in his answers to questions 220.39 (3.7.3.1.1.1A) and 220.40 (3.7.3.1.1.1A). Question 220.39 requested justification for using a static coefficient of 1.3 times peak acceleration rather than 1.5 as the Standard Review Plan recommends. Question 220.40 asked for confirmation that a static analysis would always be conservative compared to dynamic analysis. The staff has not yet obtained and reviewed the referenced paper and therefore notes that the issue remains open at this time.

3.7.4 Seismic Instrumentation [redacted]

The type, number, location and utilization of strong motion accelerographs to record seismic events and to provide data on the frequency, amplitude and phase relationship of the seismic response of the containment structure comply with Regulatory Guide 1.12. Supporting instrumentation is being installed on Category I structures, systems and components in order to provide data for the verification of the seismic responses determined analytically for such Category I items.

The staff conludes that the seismic instrumentation system provided for the plant is acceptable, with the exceptions noted below, and meets the recommendations of Standard Review Plan 3.7.4 and the requirements of General Design Criterion 2, 10 CFR Part 100, Appendix A and 10 CFR Part 50, §50.55a except as noted below. This conclusion is based on the following:

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The applicant has met the recommendations of Standard Review Plan 3.7.4 except that a seismic instrumentation surveillance scheme has not yet been provided (see Question 220.44). The requirements of 10 CFR Part 100, Appendix A were met by providing the instrumentation that is capable of measuring the effects of an earthquake, which meets the requirements of GDC 2. The applicant has met the requirements of 10 CFR Part 50, §50.55a by providing the inservice inspection program that will verify operability by performing channel checks, calibrations, and functional tests at acceptable intervals. In addition, the installation of the specific seismic instrumentation on the reactor containment structure and at other Category I structures, systems, and components constitutes an acceptable program to record data on seismic ground motion as well as data on the frequency and amplitude relationship of the seismic response of major structures and systems. A prompt readout of pertinent data at the control room can be expected to yield sufficient information to guide the operator on a timely basis for the purpose of evaluating the seismic response in the event of an earthquake. Data obtained from such installed seismic instrumentation will be sufficient to determine that the seismic analysis assumptions and the analytical model used for the design of the plant are adequate and that allowable stresses are not exceeded under conditions where continuity of operation is intended. Provision of such seismic instrumentation complies with Regulatory Guide 1.12.

3.8 Design of Category I Structures

3.8.1 Concrete Containment

Not applicable to this facility.

3.8.2 Steel Containment

The containment consists of a free-standing steel shell located within a separate reinforced concrete reactor building. The containment was designed, fabricated, constructed and tested as a Class MC vessel in accordance with Subsection NE of the ASME Boiler and Pressure Vessel Code, Section III, Division 1. Loads include an appropriate combination of dead and live loads; thermal loads; seismic and loss-of-coolant accident-induced loads including pressure and jet forces.

The analysis of the containment was based on elastic thin shell theory. The allowable stress and strain limits are in accordance with those delineated in the applicable sections of Subsection NE of the ASME Code, Section III, Division 1, for the various loading conditions.

The staff concludes that the design of the steel containment is acceptable and meets the recommendations of Standard Review Plan 3.8.2 and the relevant requirements of 10 CFR Part 50, §50.55a, and General Design Criteria 1, 2, 4, 16, and 50 with exceptions as noted at the end of this section. This conclusion is based on the following:

1. The applicant has met the recommendations of Standard Review Plan 3.8.2 and the requirements of Section 50.55a and GDC 1 with respect to assuring that the steel containment is designed, fabricated, erected, constructed, tested and inspected to quality standards commensurate with its safety function to be performed by meeting the guidelines of regulatory guides and industry standards indicated below.

2. The applicant has met the requirements of GDC 2 by designing the steel containment to withstand the most severe earthquake that has been established for the site with sufficient margin and the combination of the effects of normal and accident condition with the effects of environmental loadings such as earthquakes and other natural phenomena.

3. The applicant has met the requirements of GDC 4 by assuring that the design of the steel containment is capable of withstanding the dynamic effects associated with missiles, pipe whipping, and discharging fluids.

4. The applicant has met the requirements of GDC 16 by designing the steel containment so that it is an essentially leaktight barrier to prevent the uncontrolled release of radioactive effluents to the environment.

5. The applicant has met the requirements of GDC 50 by designing the steel containment to accommodate, with sufficient margin, the design leakage rate, calculated pressure and temperature conditions resulting from

accident conditions, and by assuring that the design conditions are not exceeded during the full course of the accident condition. In meeting these design requirements, the applicant has used the recommendations of regulatory guides and industry standards indicated below.

6. At the staff's request, the applicant is performing an analysis to determine the containment ultimate capacity.

The criteria used in the analysis, design, and construction of the steel containment structure to account for anticipated loadings and postulated conditions that may be imposed upon the structure during its service lifetime are in conformance with established criteria, and with codes, standards, and guides acceptable to the Regulatory staff. These include meeting the position of Regulatory Guide 1.57, and industry standard ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NE.

The use of these criteria as defined by applicable codes, standards, and guides, the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control programs, and special construction techniques; and the testing and inservice surveillance requirements, provide reasonable assurance with exceptions as noted below that in the event of earthquakes and various postulated accidents occurring within and outside the containment, the structure will withstand the specified design conditions without impairment of structural integrity or safety function.

A Category I concrete shield building protects the steel containment from the effects of wind and tornadoes and various postulated accidents occurring outside the shield building. The shield building is covered in Section 3.8.4.

The following issues remain outstanding:

(1) Regarding the polar crane support, the staff is concerned about procedures to assure that distortation and/or lamination of the shell in the area where the polar crane support is welded to the shell are adequate. Such

procedures and the margins against distortation and lamination should be addressed in detail in the FSAR for all potential occurrences.

(2) The answer to the staff's question 220.49, which requested a comparison of the applicant's criteria for the containment factors-of-safety against buckling with the staff position, is inadequate. An adequate response is not contained in the referenced section of the FSAR noted in the applicant's response to question 220.49.

(3) The answer to the staff's question 220.51 (3.8.2.3.1), which asked for a numerical comparison of how much the guidance of the Regulatory Guide 1.57 was exceeded, is unresponsive. The specific data asked for was not provided.

(4) The answer to the staff question 220.52 (3.8.2.4.1), regarding methods of combining dynamic loads for the containment, is unresponsive. The referenced section of the FSAR does not contain the information which the staff requested.

(5) The tables in Appendix 6-A are incomplete. This item also applies to Section 3.8.3 regarding internal structures.

(6) From the Design Audit conducted in May, 1982 it was learned that not all of the computer codes used for the River Bend calculations were verified in accordance with methods outlined in S.R.P Section 3.8.2, II, 4, c. It has since been noted that the Wilson-GHOSH Code was in error. Potential consequences should be noted, if any exist.

3.8.3 Concrete and ~~Structural~~ Steel Internal Structures of Steel or Concrete Containments

The containment interior structures consist of reinforced concrete and steel framed walls, compartments and floors. The major code used in the design of concrete internal structures was ACI 318-71. The staff has asked the applicant to indicate the extent of this compliance with the requirements of ACI 349 as outlined in Reg. Guide 1.142 at both the acceptance review stage (question 220.8) and in the staff's FSAR review (question 220.58). In each case the

applicant has responded that he has complied with the requirements of ACI 318-71 as he committed to do in his construction permit application. The staff's position is that material deviations from the current SRP requirements to design applicable structures in accordance with ACI 349/Regulatory Guide 1.142 must be justified.

Steel internal structures were designed, fabricated, and inspected in accordance with the ASME "Boiler and Pressure Vessel Code, Section III, Subsections NE and NF" and the AISC, "Specifications for the Design, Fabrication, and Erection of Structural Steel for Buildings" and ANSI N45.2.5.

With the exception noted above the staff concludes that the design of the containment internal structures is acceptable and meets the relevant requirements of 10 CFR Part 50, §50.55a, and General Design Criteria 1, 2, 4, 5, and 50. This conclusion is based on the following:

1. The applicant has met the requirements of Section 50.55a, and GDC 1 with respect to assuring that the containment internal structures are designed, fabricated, erected, constructed, tested and inspected to quality standards commensurate with its safety function to be performed by meeting the guidelines of Regulatory Guides and industry standards indicated below.

2. The applicant has met the requirements of GDC 2 by designing the containment internal structure to withstand the most severe earthquake that has been established for the site with sufficient margin and the combinations of the effects of normal and accident conditions with the effects of environmental loadings such as earthquakes and other natural phenomena.

3. The applicant has met the requirements of GDC 4 by assuring that the design of the internal structures are capable of withstanding the dynamic effects associated with missiles, pipe whipping and discharging fluids.

4. The applicant has met the requirements of GDC 5 by demonstrating that structure systems and components are not shared between units or that if shared they have demonstrated that sharing will not impair their ability to perform their intended safety function.

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5. The applicant has met the requirements of GDC 50 by designing the containment internal structures to accommodate, with sufficient margin, the design leakage rate, calculated pressure and temperature conditions resulting from accident conditions and by assuring that the design conditions are not exceeded during the full course of the accident condition. In meeting these design requirements, the applicant has used the recommendations of Regulatory Guides and industry standards indicated below. The applicant has also performed appropriate analysis which demonstrate the ultimate capacity of the structures will not be exceeded and establishes the minimum margin of safety for the design.

The criteria used in the design, analysis, and construction of the containment internal structures to account for anticipated loadings and postulated conditions that may be imposed during the structures during their service lifetime are in conformance with established criteria, with codes, standards, and specifications acceptable to Regulatory staff. These include meeting the positions of Regulatory Guide 1.10, 1.15, 1.55, 1.57, 1.94 and 1.142 and industry standards ACI-349 (see exception above), ASME, "ASME Boiler and Pressure Vessel Code, Section III, Division 2, Code for Concrete Reactor Vessels and Containments," ASME, "Boiler Pressure Vessel Code, Section III, Subsections NE and NF," AISC, "Specifications for the Design, Fabrication, and Erection of Structural Steel for Buildings and ANSI N45.2.5.

The use of these criteria as defined by applicable codes, standards, and specifications, the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control programs, and special construction techniques; and the testing and in-service surveillance requirements provide reasonable assurance that, in the event of earthquakes and various postulated accidents occurring within the containment, the interior structures will withstand the specified design conditions without impairment of structural integrity or the performance of required safety functions.

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3.8.4 Other Seismic Category I Structures

These structures are the auxiliary building, shield building, fuel building, diesel generator building, standby service water cooling tower and basin, standby service water pump house, and piping and electrical tunnels housing safety related systems.

Category I structures other than containment and its interior structures are all of structural steel and/or concrete. The structural components consist of slabs, walls, beams and columns. The major code used in the design of concrete Category I structures was the ACI 318-71, "Building Code Requirements for Reinforced Concrete." For steel Category I structures, the AISC, "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," is used. The following issues remain outstanding with regard to Category I structures outside containment:

(1) Question 220.48 (3.8) regarding description of seal materials remains unanswered.

(2) Question 220.32 regarding the potential effects of collapse of non-Category I structures on Category I structures remains unanswered. This item is also referred to in SER Section 3.7.2/3.

(3) The analysis of the spent fuel pool liner for a heavy drop accident is not complete (see question 220.60).

(4) Drawings of the fuel building spent fuel racks have not been provided (see question 220.64).

(5) The applicant stated during the design audit of May, 1982 that the compressible material in the shake space between the buildings would be removed prior to completion of the plant. There is no statement to this effect in the FSAR.

(6) The staff's previously stated exception, as noted in Section 3.8.3 above, regarding the use of ACI 318-71 rather than ACI 349/Reg. Guide 1.142 applies.

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With the exception of the issues noted above the staff concludes that the design of safety-related structures other than containment or containment interior structures is acceptable and meets the relevant requirements of 10 CFR Part 50, §50.55a, and General Design Criteria 1, 2, 4, and 5. This conclusion is based on the following:

1. The applicant has met the requirements of 50.55a and GDC 1 with respect to assuring that the safety-related structures other than containment are designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with its safety function to be performed by meeting the guidelines of Regulatory Guides and industry standards indicated below.

2. The applicant has met the requirements of GDC 2 by designing the safety-related structures other than containment to withstand the most severe earthquake that has been established for the site with sufficient margin and the combinations of the effects of normal and accident conditions with the effects of environmental loadings such as earthquakes and other natural phenomena.

3. The applicant has met the requirements of GDC 4 by assuring that the design of the safety-related structures are capable of withstanding the dynamic effects associated with missiles, pipe whipping and discharging fluids.

4. The applicant has met the requirements of GDC 5 by demonstrating that structures, systems, and components are not shared between units or that if shared they have demonstrated that sharing will not impair their ability to perform their intended safety function.

5. The applicant has met the requirements of Appendix B because their quality assurance program provides adequate measures for implementing guidelines relating to structural design audits.

The criteria used in the analysis, design, and construction of all the plant Category I structures to account for anticipated loadings and

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postulated conditions that may be imposed upon each structure during its service lifetime are in conformance with established criteria, codes, standards, and specifications acceptable to the regulatory staff. These include meeting the positions of Regulatory Guide 1.10, 1.15, 1.55, 1.69, 1.91, 1.94, 1.115, 1.142, and 1.143 and industry standards ACI-349 (see exception above) and AISC, "Specifications for the Design, Fabrication, and Erection of Structural Steel for Buildings."

The use of these criteria as defined by applicable codes, standards, and specifications, the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control, and special construction techniques; and the testing and inservice surveillance requirements provide reasonable assurance that, in the event of winds, tornadoes, earthquakes, and various postulated accidents occurring within the structures, the structures will withstand the specified design conditions without impairment of structural integrity or the performance of required safety functions.

3.8.5 Foundations

Foundations of Category I structures are described in Section 3.8.5 of the SAR. Primarily, these foundations are reinforced concrete of the mat type. The major code used in the design of these concrete mat foundations was ACI 318-71. Our previously stated exceptions regarding the use of this document by the applicant are contained in SER Section 3.8.3 above.

In addition, the following issues, which were reported from the structural design audit conducted in May 1982 remain outstanding:

(1) The design range on the soil shear modulus was taken to be 18 ksi ± 1/3 (12 to 24 ksi). The test values showed the average soil shear modulus to be 24 ksi. This value should have spread ± 1/3 to account for scatter. Therefore the design soil shear modulus should have varied between 16 to 32 ksi, not 12 to 24 ksi.

(2) The values of Poission's Ratio for soil appeared to be inconsistent; that is, in some cases a value of 0.5 was used and in other instances a value of 0.3 was used.

(3) Dynamic soil pressures were not included in the mathematical model of the standby service water cooling tower. Thus these pressures were also not included in the detail design of the walls of the structure. Since the audit, Stone and Webster has recalculated the stresses in the walls using dynamic soil pressures. They found the walls to be satisfactory. This is because a large construction surcharge load has compensated for the required dynamic soil pressures. The staff's understanding of the resolution of this matter should be confirmed, in writing, by the applicant.

With the exceptions noted above the staff concludes that the design of the seismic Category I foundations are acceptable and meets the relevant requirements of 10 CFR Part 50, §50.55a, and General Design Criteria 1, 2, 4, and 5. This conclusion is based on the following:

1. The applicant has met the requirements of Section 50.55 and GDC 1 with respect to assuming that the seismic Category I foundations are designed, fabricated, erected, constructed, tested and inspected to quality standards commensurate with its safety function to be performed by meeting the guidelines of Regulatory Guides and industry standards indicated below.

2. The applicant has met the requirements of GCD 2 by designing the seismic Category I foundation to withstand the most severe earthquake that has been established for the site with sufficient margin and the combinations of the effects of normal and accident conditions with the effects of environmental loadings such as earthquakes and other natural phenomena.

3. The applicant has met the requirements of GDC 4 by assuring that the design of seismic Category I foundations are capable of withstanding the dynamic effects associated with missiles, pipe whipping and discharging fluids.

4. The applicant has met the requirements of GDC 5 by demonstrating that structure systems and components either are not shared between units or that if shared they have demonstrated that sharing will not impair their ability to perform their intended safety function.

The Criteria used in the analysis, design, and construction of all the plant seismic Category I foundations to account for anticipated loadings and postulated conditions that may be imposed upon each foundation during its service lifetime are in conformance with established criteria, codes, standards, and specifications acceptable to the Regulatory staff. These include meeting the positions of Regulatory Guide 1.142 and industry standards ACI-349 (see exception above) and AISC, "Specification for Design, Fabrication and Erection of Structual Steel for Building."

The use of these criteria as defined by applicable codes, standards, and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control, and special construction techniques; and the testing and in-service surveillance requirements provide reasonable assurance that, in the event of winds, tornadoes, earthquakes, and various postulated events, seismic Category I foundations will withstand the specified design conditions without impairment of structural integrity and stability or the performance of required safety functions.

~~3.7 Seismic Design~~

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~~3.8 Design of Seismic Category I Structures~~

3.9 Mechanical Systems and Components

3.9.1 Special Topics for Mechanical Components

The staff has reviewed FSAR Section 3.9.1 and finds it generally acceptable. However, more information is required on the verification of certain computer programs. The staff also has questions regarding the completeness of the design transients and the number of cycles. Additional clarification is needed regarding tests performed on a limited number of components. These open items have been transmitted to the applicant. When these open items are resolved the staff will be able to conclude that the design transients and resulting loads and load combinations--with appropriate specified design and service limits for mechanical components--are acceptable and meet the relevant requirements of GDC 1, 2, 14, and 15; 10 CFR 50, Appendix B; and 10 CFR 100, Appendix A.

The applicant has met the relevant requirements of GDC 14 and 15 by demonstrating that the design transients and resulting loads and combinations--with appropriate specified design and service limits that the applicant has used for designing ASME Code Class 1 and CS components and supports and reactor internals--provide a complete basis for design of the reactor coolant pressure boundary for all conditions and events expected over the service lifetime of the plant.

The applicant has met the relevant requirements of GDC 2 and 10 CFR 100, Appendix A by including seismic events in the design transients that serve as the design basis for withstanding the effects of natural phenomena.

The applicant has met the relevant requirements of 10 CFR 50, Appendix B, and GDC 1 by (1) having submitted information that demonstrates the applicability and validity of the design methods and computer programs used for the design and analysis of seismic Category I, ASME Code Class 1, 2, 3, and CS structures, and non-Code structures within the present state-of-the-art limits and (2) having design control measures that are acceptable to ensure the quality

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and analysis of seismic Category I, ASME Code Class 1, 2, 3, and CS structures and non-Code structures within the present state-of-the-art limits and (2) having design control measures that are acceptable to ensure the quality of the computer programs.

3.9.2 Dynamic Testing and Analysis of Systems, Components, and Equipment

The staff has reviewed FSAR Section 3.9.2 and finds it generally acceptable. However, the number of operating basis earthquake (OBE) stress cycles considered in the NSSS scope is not in compliance with the SRP. This is an open item. Additional information regarding the preoperational test program is required. In particular, more information is required on the locations of measurements and visual inspections, the limits for steady-state and transient vibration, and the limits and tolerances for thermal expansion. Compliance with NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking", is required. These open items have been transmitted to the applicant. After they are resolved, the staff will be able to conclude that the dynamic testing and analysis of systems, components, and equipment is acceptable and meets the relevant requirements of GDC 1, 2, 4, 14, and 15.

The applicant has met the relevant requirements of GDC 14 and 15 with respect to the design and testing of the reactor coolant pressure boundary to ensure that there is a low probability of rapidly propagating failure and of gross rupture and to ensure that design conditions are not exceeded during normal operation including anticipated operational occurrences. This is done by having an acceptable vibration, thermal expansion, and dynamic effects test program that will be conducted during startup and initial operation on specified high- and moderate-energy piping, and all associated systems, restraints, and supports. The tests provide adequate assurance that the piping and piping restraints of the system have been designed to withstand dynamic vibrational effects of valve closures, pump trips, and other operating modes associated with the design-basis flow conditions. In addition, the tests provide assurance that adequate clearances and free movement of snubbers exist for unrestrained thermal movement of piping and supports during normal system heatup and cooldown operations. The planned tests will develop loads similar to those experienced during reactor operation.

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The applicant has met the relevant requirements of GDC 2 with respect to demonstrating the design adequacy of all Category I systems, components, equipment and their supports to withstand earthquakes by meeting the regulatory positions of RGs 1.61 and 1.92 and by providing acceptable seismic systems analysis procedures and criteria. The scope of review of the seismic system analysis included the seismic analysis methods of all Category I systems, components, equipment, and their supports. It included review of procedures for modeling, inclusion of torsional effects, seismic analysis or Category I piping systems, seismic analysis of multiply supported equipment and components with distinct inputs, justification for the use of constant vertical static factors, and determination of composite damping. The review has included design criteria and procedures for evaluation of the interaction of non-Category I piping with Category I piping. The review has also included criteria and seismic analysis procedures for reactor internals and Category I buried piping outside containment.

The systems analyses are performed by the applicant on an elastic basis. Modal response spectrum multidegree of freedom and time history methods form the bases for the analyses of all major Category I systems, components, equipment, and their supports. When the modal response spectrum method is used, governing response parameters are combined by the square-root-of-the-sum-of-the-squares rule. However, the absolute sum of the modal responses is used for modes with closely spaced frequencies. The square root of the sum of the squares of the maximum codirectional responses is used in accounting for three components of the earthquake motion for both the time history and response spectrum methods. Floor spectra inputs to be used for design and test verifications of systems, components, equipment, and their supports are generated from the time history method, taking into account variation of parameters by peak widening. A vertical seismic system dynamic analysis has been employed for all systems, components, equipment, and their supports where analyses show significant structural amplification in the vertical direction.

The applicant has met the relevant requirements of GDC 1 and 4 with respect to the reactor internals being designed and tested to quality standards commensurate with the importance of the safety functions being performed and

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being appropriately protected against dynamic effects (1) by meeting the regulatory positions of RG 1.20 for the conduct of preoperational vibration tests and (2) by having a preoperational vibration program planned for the reactor internals that provides an acceptable basis for verifying the design adequacy of these internals under test loading conditions comparable to those that will be experienced during operation. The combination of tests, predictive analysis, and post-test inspection provides adequate assurance that the reactor internals will, during their service lifetime, withstand the flow-induced vibrations of reactor operation without loss of structural integrity. The integrity of the reactor internals in service is essential to ensure the proper positioning of reactor fuel assemblies and unimpaired operation of the control rod assemblies to permit safe reactor operation and shutdown.

The applicant has met the relevant requirments of GDC 2 and 4 with respect to the design of systems and components important to safety to withstand the effects of earthquakes and the appropriate combinations of the effects of normal and postulated accident conditions with the effects of the SSE by planning a dynamic system analysis to be performed that provides an acceptable basis for confirming the structural design adequacy of the reactor internals and unbroken piping loops to withstand the combined dynamic loads of postulated LOCAs and the SSE. The analysis provides adequate assurance that the combined stresses and strains in the components of the reactor coolant system and reactor internals will not exceed the allowable design stress and strain limits for the materials of construction, and that the resulting deflections of displacements at any structural elements of the reactor internals will not distort the reactor internals geometry to the extent that core cooling may be impaired. The methods used for component analysis have been found to be compatible with those used for the systems analysis. The proposed combinations of component and system analyses are, therefore, acceptable. The assurance of structural integrity of the reactor internals under LOCA conditions for the most adverse postulated loading event provides added confidence that the design will withstand a spectrum of lesser pipe breaks and seismic loading events.

By the proposed program to correlate the test measurements with the analysis results, the applicant has met the relevant requirements of GDC 1 with respect

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to systems and components being designed and tested to quality standards commensurate with the importance of the safety functions to be performed. The program constitutes an acceptable basis for demonstrating the compatibility of the results from tests and analyses, the consistency between mathematical models used for different loadings, and the validity of the interpretation of the test and analysis results.

3.9.3 ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures

The staff has reviewed FSAR Section 3.9.3 and finds it generally acceptable except in the areas of functional capability and design limits. Compliance with the SRP (NUREG-0800) must be shown. In addition, further clarification is required on certain aspects of support design. These open issues have been transmitted to the applicant. When they are resolved, the staff will be able to conclude that the applicant met the requirements of 10 CFR 50.55a and GDC 1, 2, and 4 with respect to the design and service load combinations and associated stress and deformation limits specified for ASME Code Class 1, 2, and 3 components by ensuring (1) that systems and components important to safety are designed to quality standards commensurate with their importance to safety and (2) that these systems can accommodate the effects of normal operation as well as postulated events such as LOCAs and the dynamic effects resulting from earthquakes. The specified design and service combinations of loading as applied to ASME Code Class 1, 2, and 3 pressure-retaining components in systems designed to meet seismic Category I standards provide assurance that in the event of an earthquake affecting the site or other loadings resulting from postulated events or operating transients, the resulting combined stresses imposed on system components will not exceed allowable stess and strain limits for the materials of construction. Limiting the stresses under such loading combinations provides a conservative basis for the design of system components to withstand the most adverse combination of loading events without loss of structural integrity.

The applicant has met the requirements of 10 CFR 50.55a and GDC 1, 2, and 4 with respect to the criteria used for design and installation of ASME Code Class 1, 2, and 3 overpressure relief devices by ensuring (1) that safety and

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relief valves and their installations are designed to standards that are commensurate with their safety functions, and (2) that they can accommodate the effects of discharge as a result of normal operation as well as postulated events such as LOCAs and the dynamic effects resulting from the SSE. The relevant requirements of GDC 14 and 15 are also met with respect to ensuring that the reactor coolant pressure boundary design limits for normal operation including anticipated operational occurrences are not exceeded. The criteria used by the applicant in the design and installation of ASME Class 1, 2, and 3 safety and relief valves provide adequate assurance that, under discharging conditions, the resulting stresses will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under the loading combinations associated with the actuation of these pressure relief devices provides a conservative basis for the design and installation of the devices to withstand these loads without loss of structural integrity or impairment of the overpressure protection function.

The applicant has met the requirements of 10 CFR 50.55a and GDC 1, 2, and 4 with respect to the design and service load combinations and associated stress and deformation limits specified for ASME Code Class 1, 2, and 3 component supports by ensuring that component supports important to safety are designed to quality standards commensurate with their importance to safety, and that these supports can accommodate the effects of normal operation as well as postulated events such as LOCAs and the dynamic effects resulting from the SSE. The combination of loadings (including system operating transients) considered for each component support within a system (including the designation of the appropriate service stress limit for each loading combination) has met the positions and criteria of RGs 1.124 and 1.130 and is in accordance with NUREG-0484 and NUREG-0609. The specified design and service loading combinations used for the design of ASME Code Class 1, 2, and 3 component supports in systems classified as seismic Category I provide assurance that in the event of an earthquake or other service loadings as a result of postulated events or operating transients, the resulting combined stresses imposed on system components will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under such loading combinations provides a conservative basis for the design of support components to

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withstand the most adverse combination of loading events without loss of structural integrity.

Class CS components are addressed in SRP 3.9.5 in connection with reactor internals.

3.9.4 Control Rod Drive Systems

The staff has reviewed FSAR Section 3.9.4 and finds it generally acceptable. If the prototype information is not available from the Kuo Sheng I reactor in a timely manner, the applicant may have to make alternate plans. This concern has been transmitted to the applicant. After receiving acceptable prototype information, the staff will be able to conclude that the design of the control rod drive system is acceptable and meets the requirements of GDC 1, 2, 14, 26, 27, and 29 and 10 CFR 50.55a.

The applicant has met the requirement of GDC 1 and 10 CFR 50.55a with respect to designing components important to safety to quality standards commensurate with the importance of the safety functions to be performed. The design procedures and criteria used for the control rod drive system are in conformance with the requirements of appropriate ANSI and ASME codes.

The applicant has met the requirements of GDC 2, 14, and 26 with respect to designing the control rod drive system to withstand effects of earthquakes and anticipated normal operation occurrences with adequate margins to ensure its reactivity control function and with extremely low probability of leakage or gross rupture of reactor coolant pressure boundary. The specified design transients, design and service loading, and combination of loads, and the limitation of the stresses and deformations under such loading combinations are in conformance with the requirements of appropriate ANSI and ASME Codes and acceptable regulatory positions specified in SRP 3.9.3.

The applicant has met the requirements of GDC 27 and 29 with respect to designing the control rod drive system to ensure its capability to control reactivity and cool the reactor core with appropriate margin, in conjunction with either the emergency core cooling system or the reactor protection

system. The operability assurance program is acceptable with respect to meeting system design requirements in observed performance as to wear, functioning times, latching, and overcoming a stuck rod.

3.9.5 Reactor Pressure Vessel Internals

The staff has reviewed FSAR Section 3.9.5 and finds it generally acceptable except that additional clarification and justification are required for the stress, deformation, and buckling limits given. This open issue has been transmitted to the applicant. After the open issue is resolved, the staff will be able to conclude that the design of reactor internals is acceptable and meets the requirements of GDC 1, 2, 4, and 10 and 10 CFR 50.55a.

The applicant has met the requirements of GDC 1 and 10 CFR 50.55a with respect to designing the reactor internals to quality standards commensurate with the importance of the safety functions to be performed. The design procedures and criteria used for the reactor internals are in conformance with the requirements of Subsection NG of the ASME Code, Section III.

The applicant has met the requirements of GDC 2, 4, and 10 with respect to designing components important to safety to withstand the effects of earthquake and the postulated LOCAs with sufficient margin to ensure that the capability to perform its safety functions is maintained and the specified acceptable fuel design limits are not exceeded.

The specified design transients, design and service loadings, and combination of loadings as applied to the design of the reactor internals structures and components provide reasonable assurance that in the event of an earthquake or of a system transient during normal plant operation, the resulting deflections and associated stresses imposed on these structures and components would not exceed allowable stresses and deformation limits for the materials of construction. Limiting the stresses and deformations under such loading combinations provides an acceptable basis for the design of these structures and components to withstand the most adverse loading events that have been postulated to occur during service lifetime without loss of structural integrity or impairment of function.

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3.9.6 Inservice Testing of Pumps and Valves

The staff has reviewed FSAR Section 3.9.6 and finds it incomplete. The specifics of the inservice testing program for pumps and valves are lacking. The criteria upon which the program will be based need clarification and amplification. These open items have been transmitted to the applicant. When they are resolved, the staff will be able to conclude that the applicant's pump and valve test program is acceptable and meets the requirements of GDC 37, 40, 43, 46, and 54 and 10 CFR 50.55a(g). This conclusion is based on the applicant's having provided a test program to ensure that safety-related pumps and valves will be in a state of operational readiness to perform necessary safety functions throughout the life of the plant. This program includes baseline preservice testing and periodic inservice testing. The program provides for both functional testing of the components in the operating state and for visual inspection for leaks and other signs of distress. The applicant has also formulated his inservice test program to include all safety-related ASME Code Class 1, 2, and 3 pumps and valves and to include those pumps and valves that are not Code Class 1, 2, and 3 but are considered to be safety related.

3.10 Seismic and Dynamic Qualification of Seismic Category I Mechanical and Electrical Equipment

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3.11 Environmental Qualification of Electrical Equipment Important to Safety and Safety-Related Mechanical Equipment

In December 1979, the staff issued guidance for the environmental qualification of safety-related electrical equipment in NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment." By letter dated February 5, 1980, the staff directed the applicant to provide a discussion of NUREG-0588 and NRC Memorandum and Order CLI-80-21. 10 CFR 50.49, which became effective on February 22, 1983 and supersedes the environmental qualification requirements set forth in NRC Memorandum and Order CLI-80-21, specifies the requirements to be met for demonstrating the environmental qualification of electrical equipment important to safety located in a harsh environment. In accordance with 10 CFR 50.49, equipment for ~~Clinton~~ Byron/Braidwood may be qualified in accordance with the acceptance criteria specified in Category I of NUREG-0588.

The NRC staff has reviewed the environmental qualification program in Section 3.11 of the Clinton FSAR; however, the applicant has indicated that additional information will be submitted. This submittal should include, for all electrical equipment important to safety located in a narsh environment, the information identified in Appendix E of NUREG-0588, with the exception of item 4 of that appendix. All electrical equipment important to safety, as defined in 10 CFR 50.49, should be included in the environmental qualification program.

In accordance with 10 CFR 50.49, all electrical equipment important to safety located in a harsh environment must be environmentally qualified or justifications for interim operation provided prior to granting of an operating license. The staff will determine at a later date, based on discussions with the applicant concerning the projected fuel load date, when all equipment must be demonstrated environmentally qualified and whether justification for operation for an interim period may be allowed, pending completion of qualification.

The applicant's qualification program should also include the qualification of safety-related mechanical equipment. To determine the extent to which safety-related mechanical equipment located in harsh environments is qualified, the staff review will concentrate on equipment containing materials that are sensitive to environmental effects, such as seals, gaskets, lubricants, fluids for hydraulic systems, and diaphragms. The suitability of materials used must be established, and replacement and maintenance schedules must be developed. The staff will provide guidance to the applicant concerning this subject by separate letter.

Mechanical and electrical equipment located in harsh or mild environments should be subjected to a maintenance and surveillance program (see RG 1.33, Revision 2 and RG 1.89, Revision 1) to ensure operability throughout the life of the plant.

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After receipt of the applicant's ~~submittal~~ additional information and a determination that it is acceptable, the staff will audit the applicant's equipment qualification files to determine the acceptability of the detailed documentation relied upon to demonstrate environmental qualification of the electrical equipment important to safety located in a harsh environment.

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4 REACTOR

4.1 [redacted] Summary Description

The reactor design for [redacted] River Bend Station is similar to that reviewed and found acceptable by the staff for other BWR/4/5/6 plants, such as LaSalle, Shoreham, Susquehanna, Zimmer, Fermi 2, Grand Gulf, Clinton, and WNP-2. The staff review of FSAR Chapter 4 is based on information supplied by the applicant, in the FSAR, and the referenced topical reports. In addition, the staff's [redacted] safety evaluation relies heavily on previous staff reviews, with appropriate modification as necessitated by the generation of new information. The review was performed in accordance with the requirements of the latest version of the Standard Review Plan, NUREG-0800 (July 1981).

4.2 Fuel System Design

The objectives of this fuel system safety review as described in NUREG-0800, Section 4.2 are to provide assurances that (1) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences, (2) fuel system damage is never so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures is not underestimated for postulated accidents, and (4) coolability is always maintained. A "not damaged" fuel system is defined as meaning that fuel rods do not fail, that fuel system dimensions remain within operational tolerances, and that functional capabilities are not reduced below those assumed in the safety analysis. Objective (1) implements GDC 10, and the design limits that accomplish this are called specified acceptable fuel design limits (SAFDLs). "Fuel rod failure" means that the fuel rod leaks and that the first fission product barrier (the cladding) has, therefore, been breached. Fuel rod failures must be accounted for in the dose analysis required by 10 CFR 100 for postulated accidents. "Coolability," which is sometimes termed "coolable geometry," means, in general, that the fuel assembly retains its rod-bundle geometrical configuration with adequate coolant channeling to permit removal of residual heat even after a severe accident. The general requirements to maintain control rod insertability and core coolability appear repeatedly in the GDC (such as, GDC 27 and 35). Specific coolability requirements for LOCAs are in 10 CFR 50.46.

To ensure that the above-stated objectives are met, the following areas are examined: (1) design bases, (2) description and design drawings, (3) design evaluation, and (4) testing, inspection, and surveillance plans. In assessing the adequacy of the design, operating experience, prototype testing, and analytical predictions are compared with acceptance criteria for fuel system damage, fuel rod failure, and fuel coolability.

Because the format of the FSAR and its support documents (NEDE-24011-P-A, NEDE-20944-P, and NEDE-20944-1P) does not correspond with the format of NUREG-0800, Section 4.2, the staff technical review focused to a large extent on clarifying the design bases and limits and determining the analytical, experimental, and operational support for those bases and limits. The review was

complicated by the fact that the FSAR itself contained little more than an outline that corresponds to the format used in a referenced Topical Report on BWR/4 and BWR/5 fuel designs (NEDE-20944-P and NEDE-20944-1P), whereas that report has been partially superseded as a reference document by the GE generic reload report (NEDE-24011-P-A). Therefore, as an aid in conducting the review, GE provided a guide or "roadmap" (Engel, August 11, 1981) that was used to cross reference the support documents with NUREG-0800 Section 4.2. This section will follow the format of NUREG-0800.

4.2.1 Design Bases

Design bases for the safety analysis address fuel system damage mechanisms and suggest limiting values for important parameters so that damage will be limited to acceptable levels. For convenience, the acceptance criteria for these design limits are grouped into three categories in NUREG-0800: (1) fuel system damage criteria, which are most applicable to normal operation, including anticipated operational occurrences (AOOs); (2) fuel rod failure criteria, which apply to normal operation, AOOs, and accidents; and (3) fuel coolability criteria, which apply to accidents.

4.2.1.1 Fuel System Damage Criteria

The following paragraphs discuss the staff's evaluation of the design bases and corresponding design limits for the damage mechanisms listed in the NUREG-0800. These design limits along with certain criteria that define failure (see Section 4.2.1.2) constitute the SAFDLs required by GDC 10. The design limits in this section should not be exceeded during normal operation including anticipated operational occurrences.

(1) Cladding Design Stress

As stated in Section 2.2 of NEDE-24011,* "the fuel assembly must be designed to ensure that possible fuel damage would not result in the release of radioactive materials in excess of applicable regulations." To satisfy that design basis, GE employs the strength theory, terminology, and stress categories presented in the ASME Code, Section III, as a guide for determining fuel rod cladding stress limits, which are presented in Table 2.6 of NEDE-24011. As indicated in NUREG-0800 Section 4.2.II.A.1, stress limits that are obtained by methods similar to those given in Section III of the ASME Code are acceptable.

(2) Strain Fatigue

As indicated in Table 2-13 and Table 2-13a of NEDE-24011, the strain fatigue criteria described in NUREG-0800, Section 4.2 (a safety factor of 2 on stress amplitude or a safety factor of 20 on the number of cycles) are applied. In addition, where stress rupture and fatigue cycling are both significant, GE applies the following limiting condition:

*Unless otherwise specified, reference to NEDE-24011 is intended to apply to the latest approved version of the report, NEDE-24011-P-A-2.

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$$\frac{\text{actual time at stress}}{\text{allowable time at stress}} - \frac{\text{actual number of cycles at stress}}{\text{allowable cycles at stress}} \leq 1.0$$

This design limit is more conservative than discussed in NUREG-0800 and is also acceptable to the staff.

(3) Fretting Wear

No design limits are given by GE for fretting wear, although NUREG-0800 states that wear limits should be stated. Because GE is able to show that fretting wear is negligible based on operating experience (see Section 4.2.3.1(3)), specific numerical limits are not necessary and this deviation from NUREG-0800 is justified.

(4) External Corrosion and Crud Buildup

No design limits are given by GE for corrosion or crud buildup although NUREG-0800 states that allowable levels should be discussed. Because corrosion and crud buildup can be assessed qualitatively from operating experience, as is addressed below in the staff's review (see Section 4.2.3.1(4)), this deviation from the quantitative requirements of NUREG-0800 is acceptable.

(5) Dimensional Changes

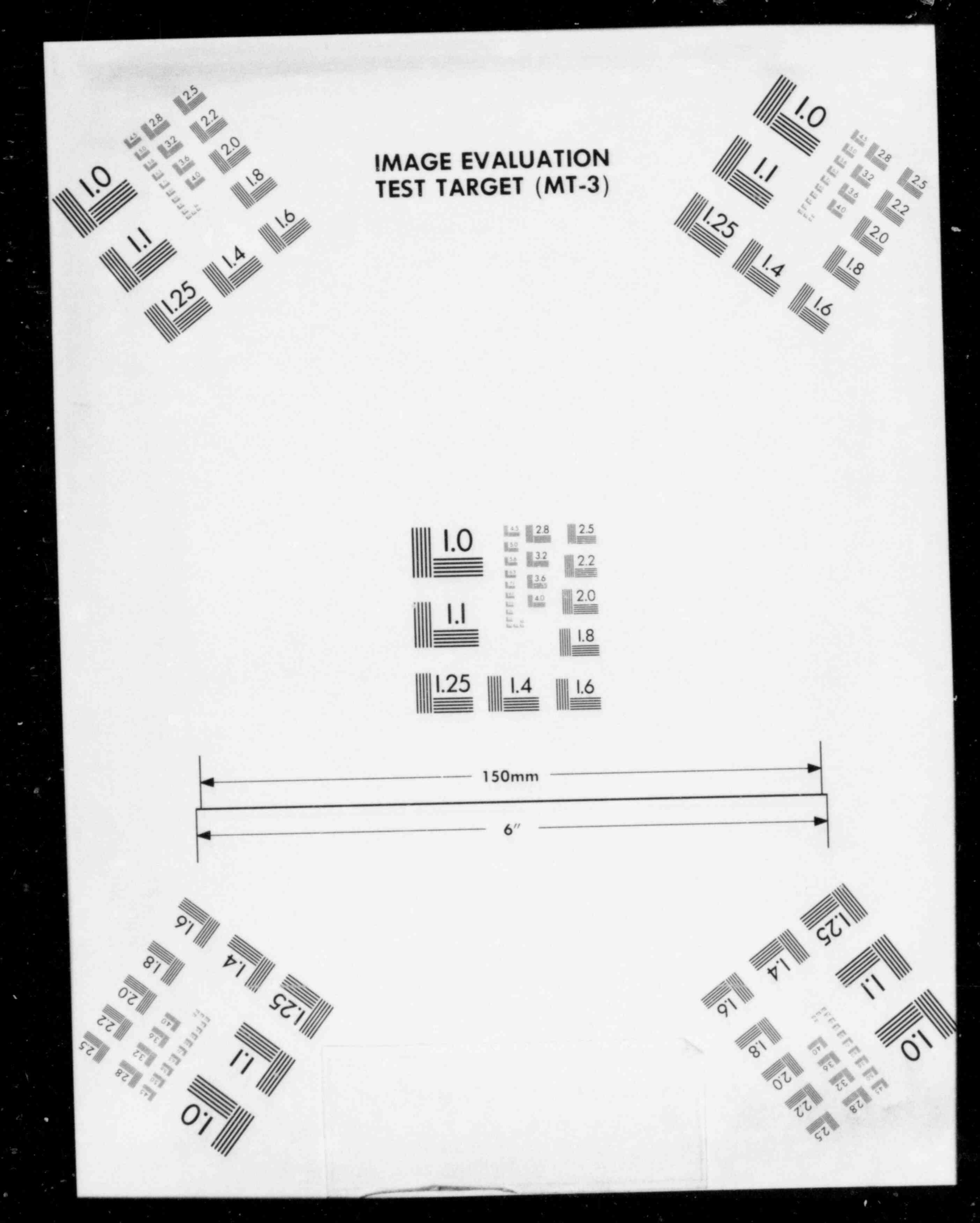
Fuel assembly components such as the fuel rods and channel boxes may undergo various types of dimensional changes such as fuel rod bowing, axial growth, and channel box deflection. Such phenomena are related to neutron fluence, fuel burnup, and assembly core residence time and must be accounted for in the fuel design analyses to establish operational tolerances and to ensure that all effects are accommodated in the thermal and mechanical design.

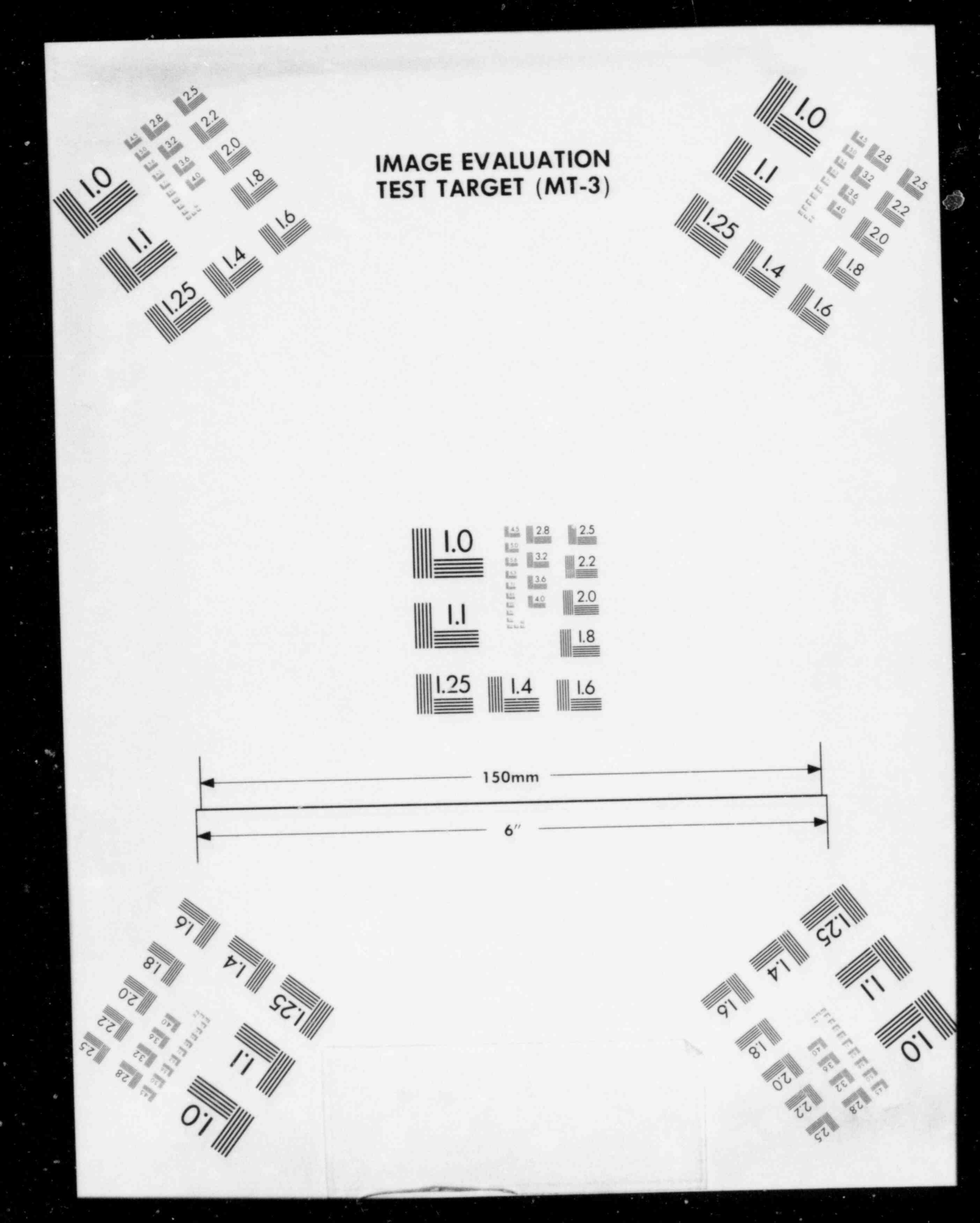
Fuel rod bowing is a phenomenon that alters the pitch dimensions between adjacent fuel rods and thus affects local nuclear power peaking and heat transfer to the coolant. GE has established (NEDE-24011-P-A, a "deflection" limit corresponding to about half the nominal spacing to ensure that the rod-to-rod and rod-to-channel clearances are sufficient to allow free passage of coolant water to all heat transfer surfaces. Although NUREG-0800 does not suggest specific numerical limits for rod bowing, the proposed deflection limit is consistent with the intent of NUREG-0800 and is, therefore, acceptable.

With regard to channel box deflection, GE has issued a generic report on channel design and deflection (NEDE-21354-P) and has responded to staff questions in two supplements. The GE report documents (1) the fuel channel description, (2) the design bases and analyses, and (3) the creep deflection phenomenon. In the design basis section of the generic report, the design basis for channel structural adequacy is discussed in terms of material properties such as yield strength, stress-rupture strength, and operational duty including normal maneuvers, transients, and accidents. The discussion meets the guidance of NUREG-0800 with regard to design bases and is, therefore, acceptable.

(6) Fuel and Poison Rod Pressures

Section 4.2 of NUREG-0800 identifies excessive fuel rod internal pressure as a potential fuel system damage mechanism. In this sense, damage is defined as an


IMAGE EVALUATION
TEST TARGET (MT-3)
1.0
1.1
1.25
1.4
1.6
1.8
2.0
2.2
2.5
2.8
3.2
3.6
4.0
150mm
6"


IMAGE EVALUATION
TEST TARGET (MT-3)
1.0
1.1
1.25
1.4
1.6
1.8
2.0
2.2
2.5
2.8
3.2
3.6
4.0
4.5
5.0
5.6
6.3
7.1
8.0
9.0
10.0
11.2
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increased potential for elevated temperatures within the rod as well as an increased potential for cladding failures. Because traditional analytical methods for fuel performance analysis do not adequately treat the effects of net outward stress on the cladding, and because these effects (such as, unstable high fuel temperatures and ballooning during departure from nucleate boiling (DNB) events) might be important, NUREG-0800 calls for rod pressures to remain below nominal system pressure during normal operation unless otherwise justified. Although the applicant does not present a rod pressure limit, it will be seen in Section 4.2.3.2(7) that the Perry fuel design meets the NUREG-0800 acceptance criterion.

(7) Fuel Assembly Liftoff

NUREG-0800 calls for the fuel assembly holddown capability (gravity and springs) to exceed worst case hydraulic loads for normal operation, which includes anticipated operational occurrences. Although the applicant does not discuss such a criterion, it will be seen in Section 4.2.3.1(8) of this report that the Perry fuel design meets this NUREG-0800 acceptance criterion.

(8) Control Material Leaching

NUREG-0800 and GDC require that control rod reactivity be maintained. Control rod reactivity can sometimes be lost by leaching of certain poison materials if the control rod cladding is breached, as has been observed (Eisenhut, October 22, 1979). GE has postulated (NEDE-24226-P) that breaching by cracking is caused by stress corrosion resulting from solidification (sintering) of the boron carbide (B_4C) particles in the rods followed by swelling (due to helium and lithium) of the sintered B_4C. The stress in the tubes caused by the B_4C is believed to accelerate the intergranular corrosion that proceeds from the outside surface of the rod cladding.

Before the discovery of the B_4C leaching loss mechanism, the previously defined life-limiting parameter for the control blades was loss of boron by depletion. The current criterion defining end of control blade life is a loss of total reactivity equal to 10% of initial control blade worth. If total boron-10 loss by depletion and leaching is considered, GE has determined (NEDE-24226-P) that a 10% reduction in worth occurs when the total boron-10 reduction, averaged over the top quarter of the blade, reaches 34%. Based on its review of the available data and discussions with GE personnel, the staff has determined that the 10% reduction-in-worth criterion is appropriate. The implementation of actions to ensure that the above criterion is satisfied is discussed in IE Bulletin 79-26. The applicant has committed to implement these actions. The staff finds this to be acceptable.

4.2.1.2 Fuel Rod Failure Criteria

The staff's evaluation of fuel rod failure thresholds for the failure mechanisms listed in NUREG-0800 is presented in the following paragraphs. When these failure thresholds are applied to normal or transient operation, they are used as limits (and hence SAFDLs), because fuel failures under those conditions should not occur (according to the traditional conservative interpretation of GDC 10). When these thresholds are applied to accident analyses, the number of fuel failures must be determined for input to the radiological dose calculations required by 10 CFR 100. The basis or reason for establishing these failure

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thresholds is thus predetermined, and only the threshold values are reviewed below.

(1) Internal Hydriding

To prevent internal hydriding, GE specifies a fabrication limit on moisture content. Based on the reported H_2O hydride formation threshold value, the H_2O specification limit for 8x8 fuel (set at a value less than half the hydride formation threshold value) was established on the basis of the previous limit for 7x7 fuel and information obtained from Joon, 1972. To include the total fuel moisture content, GE converts the total fuel rod moisture limit to a total fuel column hydrogen limit. The GE limits for moisture and hydrogen are well below those specified in NUREG-0800 Section 4.2, and the current ASTM specification (C776-76, Part 45) for a UO_2 pellet-equivalent limit of 2 ppm hydrogen from all sources is also satisfied. The GE limits for fuel moisture and hydrogen content to preclude hydriding failures are, therefore, acceptable.

(2) Cladding Collapse

If axial gaps in the fuel pellet column were to occur because of densification, the cladding would have the potential of collapsing into a gap (flattening). Because of the large local strains that would result from collapse, collapsed cladding is assumed to be failed. To define a collapse criterion to reflect the operational conditions of the reactor, GE has elected (NEDE-20606-P-A) to adopt a collapse criterion that is related to an assumed pressure increase during a turbine trip without bypass; that is, if the fuel rod can sustain, without collapse, an instantaneous increase in the hot system pressure of a given magnitude, it is considered safe against collapse during normal operation, including AOOs. The maximum ovality which precedes this collapse-safe transient is defined as the design limit ovality. NEDE-20606-P-A, which contains these limits and definitions, has been reviewed and approved (Butler, April 1975). The staff, therefore, concludes that the cladding collapse design basis and limit have been adequately addressed for the Perry fuel design.

(3) Overheating of Cladding

As indicated in NUREG-0800 Section 4.2.II.A.2, it has been traditional practice to assume that failures will not occur if the thermal margin criterion is satisfied. This is a conservative assumption for events that cause failures by high temperature cladding mechanisms. For BWR fuel, thermal margin is stated in terms of the minimum value of the critical power ratio (MCPR), which corresponds to the most limiting fuel assembly in the core. As indicated in Section 5 of NEDE-24011-P-A, GE ensures that adequate thermal margin is maintained by selecting an MCPR based on a statistical analysis as follows:

> Moderate frequency transients caused by a single operator error or equipment malfunction shall be limited such that, considering uncertainties in manufacturing and monitoring the core operating state, more than 99.9 percent of the fuel rods would be expected to avoid boiling transition.

Both the normal operation and transient thermal limits in terms of MCPR are derived from this basis, which is described fully in NEDE-10958-P-A and NEDO-10958-A. The design basis and associated limits are consistent with the

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thermal margin guidelines of NUREG-0800, Section 4.2.II.A.2, and are thus acceptable from the standpoint of the fuel mechanical design. The review of thermal/hydraulic design methods is described in Section 4.4 of this report.

(4) Overheating of Fuel Pellets

Although it is stated in NUREG-0800, Section 4.2.II.A.2.e, that it has been traditional practice to assume that failure will occur if fuel pellet centerline melting takes place, that conservative assumption has been strictly applied only to PWR fuel. For BWRs, a limited amount of calculated UO_2 melting has been permitted for certain events (such as rod withdrawal) as long as the melting is not sufficient to cause 1% cladding plastic strain. (See paragraph 4.2.1.2(6) below for a discussion of the 1% strain limit.) It should be noted that fuel melting is not expected to occur for normal steady-state full-power operation (see Section 2.4.2.5 of NEDE-24011).

(5) Excessive Fuel Enthalpy

For a severe reactivity initiated accident (RIA) in a BWR at zero or low power, fuel failure is assumed in NUREG-0800 to occur if the radially averaged fuel rod enthalpy is greater than 170 cal/gm at any location in the core. The 170 cal/gm enthalpy criterion, developed from SPERT tests (Bruno et al., August 1969), is primarily intended to address cladding overheating effects, but it also indirectly addresses pellet/cladding interactions of the type associated with severe RIAs. As indicated in the Engel letter of August 11, 1981 and in NEDO-10527, GE uses 170 cal/g as a cladding failure threshold and thus satisfies NUREG-0800.

(6) Pellet/Cladding Interaction

Fuel failures due to pellet/cladding interaction (PCI) have been encountered in operating BWR fuel (NEDE-24343-P). PCI generally occurs during a power increase as the fuel pellet expands and exerts stresses on the cladding. Although the exact mechanisms that contribute to PCI damage have not been established beyond a doubt, operating experience indicates that irradiated Zircaloy cannot readily accommodate stresses or strains of this kind, particularly when the Zircaloy has been exposed to certain embrittling (stress-corrosion) fission product species such as iodine or cadmium.

As indicated in NUREG-0800 Section 4.2.II.A.2.g, there are no generally applicable criteria for PCI failure, but two acceptance criteria of limited application are (1) 1% cladding strain and (2) centerline melting. The centerline melting criterion was discussed in paragraph 4.2.1.2(4) and found to be acceptable. GE employs a 1% plastic strain "safety" limit (design limit) for cladding strain based on data from fuel rods operated in BWRs (NEDO-10505). As noted in NEDE-24011, none of the data reported fell below the 1% plastic strain value, but a statistical distribution fit to the available data indicated that the 1% plastic strain value was approximately the 95% point in the total population. The distribution implied, therefore, that there was a small (less than 5%) probability that some cladding segments may have plastic elongations less than 1% at failure. Although a 1% plastic strain limit will not necessarily preclude all types of PCI failures (Tokar, November 14, 1979), that strain criterion

[illegible] with the only other licensing criterion for ECCS in NUREG-[illegible] and is, therefore, acceptable.

(7) Cladding Rupture

Directly, cladding will rupture (burst) under certain combinations of temperature, heating rate, and stress during a LOCA. Although Appendix K to 10 CFR 50 requires that the incidence of rupture during a LOCA not be underestimated, there are no design limits required for cladding rupture. A rupture-temperature correlation (NEDO-20566) is used in the LOCA ECCS analysis, and that correlation is evaluated in Section 4.2.3.2 of this report.

(8) Fuel Rod Mechanical Fracturing

The term "mechanical fracture" refers to a cladding defect that is caused by an externally applied force such as a hydraulic load or a load derived from core-plate motion. These loads are bounded by the loads of a LOCA and SSE, and the mechanical fracturing analysis is usually done as a part of the seismic-and-LOCA loads analysis (see Section 4.2.3.3(4) of this report). Because that analysis has not been completed for Perry, it is not clear what design limit will be used for the mechanical fracturing analysis. Fuel rod fracturing is considered to be a confirmatory item which the staff will report on in a supplement to this report.

4.2.1.3 Fuel Coolability Criteria

For major accidents in which severe fuel damage might occur, core coolability must be maintained as required by several GDC (GDC 27 and 35). The following paragraphs discuss the staff's evaluation of limits that will ensure that coolability is maintained for the severe damage mechanisms listed in Section 4.2 of NUREG-0800.

(1) Fragmentation of Embrittled Cladding

To meet the requirements of 10 CFR 50.46 as it relates to cladding embrittlement for a LOCA, acceptance criteria of 2200°F on peak cladding temperature and 17% on maximum cladding oxidation must be met. These criteria are used by the applicant (see NEDE-20566-A).

(2) Violent Expulsion of Fuel

In a severe RIA such as a BWR control rod drop, the large and rapid deposition of energy in the fuel can result in fuel melting, fragmentation, and violent dispersal of fuel droplets or fragments into the reactor coolant. The mechanical action associated with such fuel dispersal can be sufficient to destroy the cladding and rod-bundle geometry of the fuel and to produce pressure pulses in the reactor coolant system. To meet the guidelines of NUREG-0800 as it relates to the prevention of widespread fragmentation and dispersal of fuel and the avoidance of pressure pulse generation within the reactor vessel, a radially averaged enthalpy limit of 280 cal/g should be observed. As indicated in NEDO-10827 and Section 5.5 of NEDE-24011-P-A, GE employs this 280 cal/g criterion as a control rod drop accident design limit.

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(3) Cladding Ballooning

Zircaloy cladding will balloon (swell) under certain combinations of temperature, heating rate, and stress during a LOCA. Although Appendix K to 10 CFR 50 requires that the degree of swelling during a LOCA not be underestimated, there are no design limits required for cladding swelling. A burst-strain correlation (NEDO-20566) is used in the LOCA ECCS analysis, and that correlation is evaluated in Section 4.2.3.3 of this report.

(4) Fuel Assembly Structural Damage From External Forces

Earthquakes and postulated pipe breaks in the reactor coolant system would result in external forces on the fuel assembly (LRG-II generic issue 2-CPB). NUREG-0800, Section 4.2, and Appendix A to that section state that fuel system coolability should be maintained and that damage should not be so severe as to prevent control rod insertion when it is required during these low probability accidents. Because GE has not completed this analysis for [redacted], it is not clear what the exact design limits will be, but they must follow the guidelines of NUREG-0800 Section 4.2, Appendix A. This is considered a confirmatory item which the staff will report on in a supplement to this report.

River Bend Station

4.2.2 Description and Design Drawings

River Bend Station

The [redacted] fuel assembly design is described in FSAR Section 4.2.2 and Figures 4.2-1 and 4.2-6. Design specification details are provided in Table 2-1 of NEDE-24011. Although each parameter listed in NUREG-0800, Section 4.2, is not provided in the GE topical reports, enough information is provided in sufficient detail to provide a reasonably accurate representation of the [redacted] fuel design and thus satisfy the intent of NUREG-0800.

River Bend Station

4.2.3 Design Evaluation

Design bases and limits were presented and discussed in Section 4.2.1. In this section, the staff reviews GE methods of demonstrating that the [redacted] fuel design meets the design criteria that have been established. This section will, therefore, correspond to NUREG-0800, Section 4.2. The methods of demonstrating that the design criteria have been met include operating experience, prototype testing, and analytical predictions.

4.2.3.1 Fuel System Damage Evaluation

River Bend station

The following paragraphs discuss the staff's evaluation of the ability of the [redacted] fuel to meet the fuel system damage criteria described in NUREG-0800 Section 4.2. Those criteria apply only to normal operation and anticipated transients.

(1) Cladding Design Stress

A description of the fuel rod stress analysis is provided in subsection 2.5.3.1.2 of NEDE-24011. Detailed evaluations of fuel assembly component structural integrity are presented in NEDE-23542-P, along with descriptions of the methods used. As indicated in Section 2.3.1 of NEDE-20944-P, additional information

River Bend station

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regarding the fuel thermal and mechanical analyses is presented in NEDO-[illegible]. The results of these analyses indicate that the [redacted] fuel assembly design meets the stress criteria established by GE. The staff has not audited these results because problems with design stress during normal operation are not expected based on general LWR fuel operating experience.

(2) Strain Fatigue

According to information provided in Section 2.5.3.1.4 of NEDE-24011, GE's fuel design fatigue analysis utilizes the linear cumulative damage rule (Miner's hypothesis [Miner, 1945]). In the analysis it is assumed that the area most subject to fatigue damage is the intersection of a fuel rod tube and an end plug at the weld. That intersection is a circumferential notch. Following Peterson's fatigue theory (O'Donnell and Purdy, May 1964), fatigue failure at a sharp notch is assumed to be caused by the action of stress at a finite distance below the material surface. The method to calculate the stress at a small distance below the surface of a notch is provided in a reference (Winnie and Wundt, November 1958). Using that method, and taking the depth below the notch from a reference (O'Donnell and Langer, 1964), GE has calculated (NEDE-24011-P-A) that the cumulative fatigue damage is less than the allowable damage (see Section 4.2.1.2.(3) for fatigue limits). Therefore, based on the fact that well-established standard theory and methods were used to determine that the cumulative fatigue damage in GE prepressurized 8x8 retrofit (P8x8R) fuel will not exceed acceptable design limits, the staff concludes that the strain fatigue analysis for Perry fuel is acceptable and that there is reasonable assurance that strain fatigue will not be a problem at Perry.

(3) Fretting Wear

In a GE BWR P8x8R fuel design, individual rods in the fuel assembly are held in position by spacers located at intervals along the length of the fuel rod, and springs are provided in each spacer cell so that the fuel rod is restrained to avoid excessive vibration. Various inpile and out-of-pile tests are described [illegible] the results of the surveillance fuel examination program that has utilized nondestructive methods [illegible] to locate discontinuities in the cladding and detailed visual examinations to characterize the nature of defects. As reported in NEDE-24011, no significant fretting wear has been observed. GE concludes (NEDE-24011-P-A) that significant fretting wear is avoided through the use of an active spring force to eliminate any clearance that would otherwise exist between the spacer structure and the fuel rods. Based on the results of these fuel test and surveillance programs, the staff concludes that reasonable assurance has been provided that Perry fuel rods and spacers will perform adequately with respect to fretting wear.

In addition to the fuel rods and spacers, there is another fuel system component whose functionality must be ensured as an objective of the review of BWR fuel system fretting concerns--the fuel assembly channel box. The fuel "channel" that encloses the fuel bundle performs three functions: (1) the channel provides a barrier to separate two parallel flow paths (one to cool the fuel bundle and the other to cool the bypass region between channels); (2) the channel guides the control blade and provides a bearing surface for it; and (3) the channel provides rigidity for the fuel bundle. Thus, the potential for cracks or holes in a "channel" or channel "box" is of concern because it would allow part of

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the cooling water that normally flows through a fuel bundle to flow out of the cracks or holes and bypass the fuel rods. Such a change in flow pattern would reduce the safety margin for fuel thermal performance and could lead to fuel overheating and damage in the event of some anticipated operating transients and postulated accidents. Significant channel box cracking and wear could also adversely affect mechanical strength.

In the mid-1970s, channel box wear and cracking were observed. The wear was located adjacent to incore neutron monitor and startup source locations. It was postulated (NEDC-21084), and later confirmed by out-of-reactor testing, that the wear was caused by vibration of nearby incore tubes due primarily to a high-velocity jet of water flowing through the bypass flow holes in the lower core plate. To eliminate significant vibration of instrument and source tubes and the resultant wear on channel box corners, Perry will incorporate modifications similar to those described (NEDE-21156) for BWRs currently in operation. Those modifications involve the elimination of the bypass holes in the lower core plate and addition of two holes in the lower tie plate of each assembly to provide an alternate flow path. This design modification has been determined to have negligible adverse effects on the mechanical, thermal, and nuclear performance of the channel boxes, as is discussed in NRC's generic safety evaluation on this subject (Eisenhut, [illegible] 1, 1979). Because channel box wear has been observed (Engel, October 18, 1977) to have been significantly reduced in operating BWRs following the design modification, the staff concludes that there is reasonable assurance that channel box wear and cracking will not be a problem in Perry.

(4) External Corrosion and Crud Buildup

Revised

Waterside Corrosion--Corrosion problems associated with stainless steel cladding in BWRs, together with a desire to improve neutron economy, led to a change some years ago from stainless steel to Zircaloy cladding material, which has good resistance to the hot water and steam environment encountered under typical BWR operating conditions (Garzarolli et al., 1978). In at least two recent cases, however, cladding failures have been associated with external "waterside corrosion (GE Projects Div. Memo., April 6, 1979; Maury, November 10, 1981). Although the details of the most recent occurrence are still under investigation, and final conclusions regarding the exact nature (as to fundamental contributing factors) of the failures have not been drawn, the earlier occurrence has been characterized by GE as "crud-induced local corrosion failure" (NEDE-24[illegible]-P). The corrosion was reportedly associated with a variably high copper concentration in the core coolant water and a minor anomaly in the Zircaloy cladding metallurgy (Charnley, December 11, 1979; Engel, October 3, 1980; Smith, November 21, 1980; Smith, February 5, 1981; and DelGeorge, February 9, 1981), although both the water chemistry and the cladding metallurgy were within allowable specifications. The source of the copper contamination in the affected plant was judged to be the copper-bearing main condenser tubes (DelGeorge, February 9, 1981). The second affected plant also has copper alloy condenser tubing. It is notable that virtually all the BWR waterside corrosion failures to date have involved gadolinia burnable poison rods.

In plants such as Perry, which will have Type 304 stainless steel condenser tubes, waterside corrosion failure of this kind would not be expected because of the absence of the major source of copper contamination. Because the [illegible]

4 REACTOR

4.1 ~~Introduction~~ Summary Description

4.2 Fuel System Design

The fuel system design for River Bend is the same as that for WNP-2, and Section 4.2 of the River Bend FSAR is the same as Section 4.2 of the WNP-2 FSAR. Therefore, Section 4.2 of the staff's SER for WNP-2 (NUREG-0892) applies to River Bend with only a few exceptions. Those exceptions are discussed below.

4.2.1 Design Bases

See 4.2 above

4.2.2 Description and Design Drawings

See 4.2 above

4.2.3 Design Evaluation

See 4.2 above

4.2.3.1 Fuel System Damage Evaluation

Because of corrosion problems associated with stainless steel cladding in BWRs and because of the interest in improving neutron economy, some years ago, cladding material was changed from stainless steel to Zircaloy, which has good resistance to the hot water and steam environment under typical BWR operating conditions (Garzarolli et al., 1978). In at least two recent cases, however, cladding failures have been associated with external "waterside" corrosion (GE, 1979, and Maury, 1981). Although the details of the most recent occurrence are still under investigation and final conclusions regarding the exact nature

(4) External Corrosion and Crud Buildup

waterside corrosion

4.2.3.1 (4) cont)

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(as to fundamental contributing factors) of the failures have not been drawn, the earlier occurrence has been characterized by GE as "crud-induced local corrosion failure" (NEDE-24343-P). The corrosion was reportedly associated with a variably high copper concentration in the core coolant water and a minor anomaly in the Zircaloy cladding metallurgy (Charnley, 1979; Engel, 1980; Smith, 1980a; 1981; and DelGeorge, 1981), although both the water chemistry and the cladding metallurgy were within allowable specifications. The source of the copper contamination in the affected plant was judged to be the copper-bearing main condenser tubes (DelGeorge, 1981). The second plant affected also has copper alloy condenser tubing. It is notable that virtually all the BWR waterside corrosion failures to date involved gadolinia burnable poison rods.

Because there are a number of uncertainties regarding the relative importance of various factors (including the effects of gadolinia) involved in waterside corrosion of Zircaloy-2 cladding in BWRs, the staff is still investigating the occurrences and studying the problem. And, because the phenomenon appears to be rather complex, the staff has determined that ways to avoid the problem may vary, depending on specific plant component availability and operating conditions. For example, no further action has been required for plants with copper-bearing main condenser tubes, where all water that enters the feedwater system is processed by mixed-bed condensate polishing demineralizers, which limit the maximum effluent copper content to a level (typically <2 ppb) well below that at which copper deposition on the fuel rods will occur.

Therefore, in River Bend, which will have copper-bearing condenser tubes and mixed-bed demineralizers, waterside corrosion failure of this kind would not be expected because of the restriction on the maximum copper content of the effluent. Because the role of copper in BWR waterside corrosion processes is still somewhat conjectural, the staff is still studying this problem (including the effect of gadolinia) and believes that it is desirable to have post-irradiation surveillance to verify that the degree of waterside corrosion (or other visible fuel degradation) is not unacceptably high. Such surveillance is addressed in Section 4.2.4.3 of this report.

Because mixed-bed demineralizers are being used in the main condenser system and routine post-irradiation surveillance will be performed, the staff concludes that the waterside corrosion issue is adequately resolved for River Bend.

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of copper in BWR waterside corrosion processes is still somewhat conjectural, the staff is still studying this problem (including the effect of gadolinia) and believes that it is desirable to have postirradiation surveillance to verify that an unacceptably high degree of waterside corrosion (or other visible fuel degradation) has not occurred. Such surveillance is addressed in Section 4.2.4.

On the basis that copper tubes are not being used in the main condenser and that routine postirradiation surveillance will be performed, the staff concludes that the waterside corrosion issue (LRG-II generic issue 5-CPB) is adequately resolved for Perry.

Crud Buildup--The buildup of a corrosion film and a crud layer on the outer surface of a fuel rod during irradiation causes gradual flow reductions and impedes heat transfer to the coolant. The effects of crud buildup on flow are discussed in Section 4.4.5 of this report. As indicated in Section 2.4.2.2 of NEDE-24011-P-A, GE calculates the cladding surface temperature using the cladding surface heat flux at a given axial position of a fuel element in conjunction with an overall cladding-to-coolant film coefficient that is taken to represent the combined effects of crud and oxide resistances and a liquid film resistance based on the Jens-Lottes wall superheat equation (Jens-Lottes paper dated May 1951). The impact of high cladding temperature, such as decreased yield strength and reduced cladding thickness resulting from oxidation, was considered in GE's design evaluation (NEDE-24011-P-A). GE's methods for analyzing the effects of oxidation and crud on fuel cladding temperatures were reviewed and approved in connection with NEDE-24011-P-A, and the staff, therefore, finds that approach acceptable for Perry.

(5) Dimensional Changes

Fuel Rod Bowing -- To address the potential for bowing, the applicant referred to NEDE-24284, "Assessment of Fuel Rod Bowing in General Electric BWRs." We have recently completed our review of that report (Rubenstein, March 8, 1983). On the basis of (a) the reported GE data base and GE's statement that "To date, no significant fuel rod bowing has been detected in GE BWR fuel assemblies" [sic, excluding segmented-rod designs] and (b) our own calculations using other vendors' proprietary information, we conclude that significant fuel rod bowing in GE BWR fuel is not anticipated and no operational penalties on GE BWR fuel are warranted at this time. If adverse rod bowing behavior (gap closures greater than 50%) is observed in GE-supplied fuel in the future, NRC should be notified in order to ascertain the need for critical power ratio penalties.

Channel Box Deflection (LRG-II Generic Issue 3-CPB)--Fuel channel design and deflections under operating conditions are discussed in the BWR/4-5 fuel Topical Report (NEDE-20944-P), but that discussion is out of date and has been superseded by a generic report (NEDE-21354-P). The design bases and analyses portions of NEDE-21354 document the channel box development are, therefore, largely of historical interest because this design has been in use for some

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~~[illegible]~~

~~PERRY UNITS 1 AND 2, SSER NO. 2~~

~~4.2 Fuel System Design~~

~~B. Thermal Limits [illegible] Control Rods~~

Channel Box Deflection (LRG-II Issue 3-CPB)

BWR fuel channels provide structural stiffness for the fuel assemblies and distribute the coolant flow between the assemblies and channel bypass regions. The channels are subject to time-dependent, permanent dimensional changes (i.e., deflections) that result from irradiation, creep and stress-relaxation effects. The resultant bulge (resulting from long-term thermal creep) or bow (resulting from differential irradiation-induced axial growth) reduces the size of the gap available for control rod insertion. Channel box deflection is thus a potential life-limiting phenomenon.

With a letter (D. L. Holtzscher, Illinois Power Company, to Howard J. Faulkner, NRC) dated May 17, 1982, LRG-II submitted a position paper that applies to Perry on channel box deflection that incorporated several of the same features as the Zimmer proposal, viz:

(a) Records will be kept of channel locations and exposure for each operation cycle.

(b) Channels shall not reside in the outer row of the core for more than two operating cycles (because flux gradients are largest near the core periphery and, therefore, differential irradiation-induced growth and bowing will be greatest at those locations).

(c) At the beginning of each fuel cycle, the combined outer row residence time for any two channels in any control rod cell shall not exceed four peripheral cycles.

In addition, LRG-II stated that channels that reside in the periphery (outer row) for more than one cycle shall be situated each successive peripheral cycle in a location which rotates the channel so that a

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different side faces the core edge. We believe that this should help to reduce uni-directional irradiation-induced growth, and should thus lessen channel bowing, and that the other measures outlined above would also help to reduce the magnitude of channel deflection.

The LRG-II position statement also contains a description of a control rod drive friction test that would be performed for those core cells exceeding the above general guidelines or containing channels with exposures greater than 30,000 MWd/MTU (associated fuel bundle exposures). In the LRG-II position paper, the control rod drive settling friction test is described in considerable detail.

In a memorandum (L. S. Rubenstein to T. Novak) dated August 19, 1982, we accepted the LRG-II position that the proposed actions would preclude excessive channel bowing in the LRG-II plants. Therefore, we conclude that channel box deflection is not going to be a problem for ~~Grand~~ River Bend Station. However, we are continuing our review of this phenomenon (and the GE report), and should our review indicate that a modification to the proposed steps is necessary, the licensee will be so notified.

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time; however, the creep deflection phenomenon is relevant to ~~Perry~~ operation. Creep deflection results in the bulging out of the channel box in a way that reduces the size of the gap provided for control blade insertion. This dimensional change, if it were large, could interfere with control blade insertion, so channels are replaced before this deflection becomes excessive.

In NEDE-21354, GE describes a channel lifetime prediction method and a backup recommendation for periodic channel measurements that consist of settling friction tests. The settling friction tests would provide an exact profile of control rod drive friction versus position by measuring the hydraulic pressure under the drive piston as the drive "settles" to a latch position. The staff believes that the friction settling tests or an acceptable alternative (such as channel deflection measurements) should be performed.

Because channel box deflection is a function of service in reactor, there is no need to resolve this issue before a license is issued. However, resolution of this issue before startup of the second cycle of operation will be made a license condition. This issue has been resolved to the staff's satisfaction for Zimmer and other recent BWR operating license applications through acceptance of a channel box surveillance plan proposed by those applicants (Rubenstein, September 18, 1981). The applicant has been notified (Eisenhut, November 3, 1981) of the acceptable resolution of this issue on other plants and has been requested to commit to the acceptable plan for Perry.

(6) Fuel and Poison Rod Pressure

The applicant has indicated that the internal pressure is used in conjunction with other loads on the fuel rod cladding in the calculation of cladding stresses. The results of such calculations, as provided in NEDE-24011, show that the calculated cladding stresses can be accommodated. The staff has examined these calculations and agrees that the acceptance criterion for cladding stress (see NUREG-0800, Section 4.2.II.A.1.a) has been met, but that criterion is not the same as the one for rod internal pressure (see NUREG-0800, Section 4.2.II.A.1.f). Because the criterion for rod internal pressure involves more than the cladding mechanical limits, the staff considers the GE analysis described in NEDE-24011 to be insufficient for meeting the rod internal pressure criterion. The staff further notes that the rod internal pressures used in the GE cladding stress analysis are well in excess of the nominal coolant system pressure, but because the calculation of these rod internal pressures involves a number of conservative assumptions, the staff regards them as excessively conservative for this purpose.

In less conservative analyses such as those previously cited (Sherwood, December 22, 1976), fuel rod internal pressure was shown to remain below system pressure for rod peak burnups below 40,000 MWd/t. This conclusion remains unchanged for the newer prepressurized fuel design as well (NEDO-23786-1). Therefore, on the basis of these calculations, the staff concludes that the rod internal pressure criterion has been met for ~~Perry~~.

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(7) Fuel Assembly Liftoff

The potential for BWR fuel assembly liftoff is addressed in a GE letter (Gridley, July 11, 1977), which covers accident conditions as well as normal operation. Section 4.2.II.A.1.g of NUREG-0800 deals with normal operation only, however, and so in this report normal operation liftoff is separated from the combined seismic-and-LOCA liftoff concern, which is addressed in Section 4.2.3.3(4). As reported by GE in the letter of July 11, 1977, for normal operation and "abnormal transients" (of which a steamline break LOCA was considered bounding), the minimum net downward force on a fuel bundle, considering only the fluid lifting forces and bundle weight, is approximately 300 lb. The maximum upward frictional force that can be exerted by a control rod on a fuel bundle during end-of-life channel scram is approximately 100 lb. Based on this information, the NRC staff concludes that there is reasonable assurance that Perry fuel assemblies will have sufficient margin to preclude liftoff during normal operation and abnormal transients (up to a steamline break LOCA).

(8) Control Material Leaching

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The loss of boron carbide (B_4C) by leaching from cracked control blade tubing is addressed in IE Bulletin 79-26, Revision 1, which requires operating BWRs to perform various actions including, but not limited to, shutdown-margin tests. Because Perry will be subject to the IE Bulletin when the plant starts operation, assurance will be maintained that control blade reactivity will not be significantly degraded by boron carbide leaching. Subsequent to the issuance of IE Bulletin 79-26, GE performed analyses and postirradiation examinations (NEDE-24226-P) and developed a boron depletion model (NED-24325-P). This model supports GE's claim that the amount of boron loss can be accurately determined analytically and that potential control blade degradation resulting from this mechanism will not significantly affect plant operation. Further generic review of this matter may thus lead to a suspension or revision of IE Bulletin 79-26 requirements.

4.2.3.2 Fuel Rod Failure Evaluation

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The following paragraphs discuss the staff's evaluation of (1) the ability of the Perry fuel to operate without failure during normal operation and anticipated transients, and (2) the accounting for fuel rod failures in the applicant's accident analysis. The fuel rod failure criteria described in Section 4.2.1.2 were used for this evaluation. Subitems (1) through (6) of NUREG-0892 apply to Perry and are repeated here.

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(1) Internal Hydriding

GE employs a hot vacuum outgassing (drying) method for removing the moisture contamination of loaded fuel rods to ensure that the level of moisture is well below the limits discussed in Section 4.2.1.2(1). To prevent the introduction of other hydrogenous impurities such as oils, plastics, and the like, various procedural controls are utilized during manufacturing (NEDE-24011-P-A). If even in the face of those manufacturing controls, moisture or hydrogenous impurities are still inadvertently introduced into a fuel rod during manufacture, further assurance against chemical attack is provided by the use of a hydrogen getter material placed in the upper plenum of the fuel rod. The incorporation

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of a hydrogen getter in GE fuel rods is described in NEDO-20922. GE calculations, based on an assumed initial rod moisture content equal to the specification limit, indicate that no cladding damage will occur during the active hydrogen/moisture gettering period (NEDE-24011-P-A). Field experience (NEDE-24343-P) has confirmed that hydriding is not an active failure mechanism for fuel manufactured since mid-1972. On the basis of the information presented, the staff concludes that adequate assurance has been provided that fuel and burnable poison rod hydriding will not be a problem at [redacted]. River Bend Station

(2) Cladding Collapse

As stated in Section 2.5.3.1.1 of NEDE-24011, GE has performed a cladding (long-term creep) collapse analysis. Based on the results of that analysis, which was performed with an approved model (NEDE-20606-P-A), cladding collapse was not expected to occur for either 8x8 or 8x8R fuel rods. Consequently, P8x8R (pre-pressurized) fuel, the type to be used in Perry, will have an even greater margin against collapse. The staff, therefore, concludes that there is reasonable assurance that long-term creep collapse will not occur in [redacted] fuel during normal operation. River Bend Station

(3) Overheating of Cladding

The methods employed to show that the minimum critical power ratio (MCPR) design basis will be met are reviewed as part of a thermal-hydraulics review in Section 4.4 of this report. Analyses that show that the MCPR design basis is met for normal operation, and anticipated transients are discussed in Chapter 15 of the [redacted] FSAR. River Bend Station Other sections of FSAR Chapter 15 discuss accidents where the MCPR criterion is used to define failure in the accident analyses. All of the MCPR related analyses are discussed in Section 15.4 of this report.

(4) Overheating of Fuel Pellets

Fuel melting temperature is discussed in Section 2.4.2.5 of NEDE-24011 as a function of exposure (burnup) and gadolinia content (of burnable poison rods). GE states in that report that fuel melting is not expected to occur during normal operation, and that statement is based on fuel temperature calculations performed with a model described in the proprietary supplement to Amendment 14 of GESSAR. Although limited melting during certain events such as an uncontrolled control rod withdrawal is permissible, such melting is not predicted to occur.

The staff has reviewed the UO_2 properties (thermal conductivity and melting point) that are important in reaching this conclusion and agree that UO_2 melting will not be a problem at [redacted] during normal operation and anticipated transients. River Bend Station However, the effects of gadolinia concentration on thermal conductivity and melting temperature are addressed in a GE topical report on gadolinia fuel properties (NEDE-20943-P), and the review of this subject has not been completed. The staff is, therefore, unable to comment on that aspect of the fuel melting issue at this time. The absence of significant melting in gadolinia-bearing fuel pellets in [redacted] will be addressed in a supplement to this report and is considered to be a confirmatory issue. River Bend Station

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(5) Excessive Fuel Enthalpy

Large fuel enthalpies occur only in the postulated control rod drop accident. GE's analysis of the control rod drop accident is described in Section 5.5.1 of NEDE-24011 and in NEDO-10527. The analysis of this event is reviewed in Section 15.4.4 of this report.

(6) Pellet/Cladding Interaction (PCI)

The two PCI criteria in current use in licensing of BWR fuel (limited fuel melting and 1% cladding plastic strain) are easily satisfied for normal operation and anticipated transients. Fuel melting is addressed in Section 4.2.3.2(4) and, except for uncertainties with regard to gadolinia fuel, the fuel melting criterion is met. The model used by GE in the evaluation of the 1% plastic strain limit is described in detail in the proprietary supplement to Amendment 4 of GESSAR. As stated in NEDE-24011, dimensions used in conjunction with the model for that evaluation are the most limiting combination of tolerances. These results show that the Perry fuel meets the 1% strain limit. The staff has not audited these results because the traditional 1% strain limit is not believed to be precise.

Based on results of developmental investigations and feedback from production fuel experience, operating restrictions known as Preconditioning Interim Operating Management Recommendations (PCIOMRs) were issued by GE to the BWR operators (NEDS-10456-PC) and will be used at Perry. These restrictions are necessary to reduce the incidence of preconditioning interim (PCI) failures even though the above criteria are met. PCIOMRs have generally been effective in reducing PCI failures that result from operational power changes, but they would not prevent PCI failures during unexpected transients and accidents.

The staff is continuing to assess the potential for PCI failure during power-increasing transients and accidents, and new techniques are being developed to analyze the potential for PCI failures (NUREG/CR-1163) and determine whether or not further staff actions are required.

In conclusion, (1) the applicant has met the staff's licensing criteria for normal operation and anticipated transients, (2) the applicant will impose operating restrictions to reduce the potential for PCI, and (3) the staff is studying the need for new licensing requirements in this area. There are presently no other PCI licensing requirements that must be met for River Bend Station.

(7) Cladding Rupture (Bursting) (LRG-II Generic Issue 1-CPB)

The staff has been generically evaluating three fuel material models that are used in ECCS analysis. Those models predict cladding rupture temperature, cladding burst strain (ballooning), and fuel assembly flow blockage (used only in PWR analyses). The staff has (a) discussed its evaluation with vendors and other industry representatives (Denise, November 20, 1979), (b) published NUREG-0630, "Cladding Swelling and Rupture Models for LOCA Analysis," and (c) required licensees to confirm that their operating reactors would continue to be in conformance with 10 CFR 50.46 if the NUREG-0630 models were substituted for the present materials models in their ECCS evaluations and certain other compensatory model changes were allowed (Eisenhut, November 9, 1979; and Denton, November 26,

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Because mixed-bed demineralizers are being used in the main condenser system and routine post-irradiation surveillance will be performed, the staff concludes that the waterside corrosion issue is adequately resolved for River Bend.

4.2.3.2 Fuel Rod Failure Evaluation

The staff has been generically evaluating three fuel material models that are used in the analysis of the emergency core cooling system (ECCS). Those models predict cladding rupture temperature, cladding burst strain (ballooning), and fuel assembly flow blockage (used only in PWR analyses). The staff has (1) discussed its evaluation with vendors and other industry representatives (Denise, 1979); (2) published NUREG-0630, "Cladding Swelling and Rupture Models for LOCA Analysis"; and (3) required licensees to confirm that their operating reactors would continue to be in conformance with 10 CFR 50.46 if the NUREG-0630 correlations were substituted for the present materials models in their ECCS evaluations and certain other compensatory model changes were allowed (Eisenhut, 1979, and Denton, 1979a) to offset any penalties incurred as a result of the use of the NUREG-0630 correlations. This latter requirement was made a condition of approval of the GE ECCS evaluation model (Tedesco, 1981a).

A generic sensitivity study of fuel rod cladding ballooning and rupture phenomena during a LOCA was submitted (Buchholz, 1981) in response to this condition of approval. Later submittals provided clarification and additional information. The staff has completed (Rubenstein, 1982) its generic review of this issue, and the resolution is addressed in Section 4.2.3.3 below.

4.2.3.3 Fuel Coolability Evaluation

Revise subpar (1) of following to:

(1) Fragmentation of Embrittled Cladding

The primary degrading effect of cladding oxidation is embrittlement. Such embrittled cladding will have reduced ductility and resistance to fragmentation. The most severe manifestation of such embrittlement occurs during a LOCA. The overall effects of a LOCA on peak cladding temperature, oxidation, and embrittlement are analyzed in NEDE-20566 and are reviewed in Section 6.3.3 of this report.

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1979). This latter requirement was made a condition of approval of the GE ECCS evaluation model (Tedesco, February 4, 1981).

A generic sensitivity study of fuel rod cladding ballooning and rupture phenomena during a LOCA was submitted (Buchholz, May 15, 1981) in response to this condition of approval. Later submittals provided clarification and additional information. The staff has nearly completed its generic review of this issue and will describe the resolution of this issue in a supplement to this report.

(8) Fuel Rod Mechanical Fracturing (LRG-II Generic Issue 2-CPB)

The mechanical fracturing analysis is usually done as a part of the seismic-and-LOCA loads analyses (see Section 4.2.3.3(4) of this report). Because that analysis has not been completed for Perry, the information on mechanical fracturing is not available. This is considered to be a confirmatory item for [redacted]. The staff will report on this issue in a supplement to this report.

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4.2.3.3 [redacted] Fuel Coolability Evaluation

(1) Fragmentation of Embrittled Cladding (LRG-II Generic Issue 4-CPB)

The primary degrading effect of cladding oxidation is embrittlement. Such embrittled cladding will have reduced ductility and resistance to fragmentation. The most severe manifestation of such embrittlement occurs during a LOCA. The overall effects of a LOCA on peak cladding temperature, oxidation, and embrittlement are analyzed in NEDE-20566 and are evaluated in Section 6.3.3.

To establish the behavior of the cladding during the loss-of-coolant accident, it is necessary to first determine the initial fuel and cladding conditions at the onset of the LOCA. The initial conditions are provided by a steady-state fuel performance code, which is reviewed in Section 4.2. Although such codes provide analyses of a number of important conditions in Chapters 4 and 15 of the applicant's Safety Analysis Report, some of the most significant are the initial conditions for the LOCA analysis, where fragmentation of embrittled cladding is evaluated. The steady-state fuel performance code provides fuel pellet temperatures (stored energy) and fuel rod gas inventories for the ECCS evaluation model as prescribed by Appendix K to 10 CFR 50.46.

Initial conditions for the LOCA have been calculated for the Perry fuel with an approved version of the GE analytical model, GEGAP-III (NEDC-20181). The model incorporates time-independent fuel densification, time-dependent gap closure, and cladding creepdown for the calculation of gap conductance.

In 1976 the staff questioned (Ross, November 23, 1976) the validity of fission gas release calculations in most fuel performance codes including GEGAP-III for burnups greater than 20,000 MWd/tU. GE was informed of this concern and was provided with a method (NUREG-0418) of correcting gas release calculations for burnups greater than 20,000 MWd/tU. Although a reanalysis was not performed specifically for the Perry fuel, an 8 x 8 reanalysis (Sherwood, December 22, 1976) was performed for early reflooding plants and reportedly bounds the Perry case. Although the fuel rod internal pressure was shown to remain below system pressure for rod peak burnups below 40,000 MWd/tU, the generic reanalysis did

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Because mixed-bed demineralizers are being used in the main condenser system and routine post-irradiation surveillance will be performed, the staff concludes that the watersice corrosion issue is adequately resolved for River Bend.

4.2.3.2 Fuel Rod Failure Evaluation

The staff has been generically evaluating three fuel material models that are used in the analysis of the emergency core cooling system (ECCS). Those models predict cladding rupture temperature, cladding burst strain (ballooning), and fuel assembly flow blockage (used only in PWR analyses). The staff has (1) discussed its evaluation with vendors and other industry representatives (Denise, 1979); (2) published NUREG-0630, "Cladding Swelling and Rupture Models for LOCA Analysis"; and (3) required licensees to confirm that their operating reactors would continue to be in conformance with 10 CFR 50.46 if the NUREG-0630 correlations were substituted for the present materials models in their ECCS evaluations, and certain other compensatory model changes were allowed (Eisenhut, 1979, and Denton, 1979a) to offset any penalties incurred as a result of the use of the NUREG-0630 correlations. This latter requirement was made a condition of approval of the GE ECCS evaluation model (Tedesco, 1981a).

A generic sensitivity study of fuel rod cladding ballooning and rupture phenomena during a LOCA was submitted (Buchholz, 1981) in response to this condition of approval. Later submittals provided clarification and additional information. The staff has completed (Rubenstein, 1982) its generic review of this issue, and the resolution is addressed in Section 4.2.3.3 below.

4.2.3.3 Fuel Coolability Evaluation

(1) Fragmentation of Embrittled Cladding

The primary degrading effect of cladding oxidation is embrittlement. Such embrittled cladding will have reduced ductility and resistance to fragmentation. The most severe manifestation of such embrittlement occurs during a LOCA. The overall effects of a LOCA on peak cladding temperature, oxidation, and embrittlement are analyzed in NEDE-20566 and are reviewed in Section 6.3.3 of this report.

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To establish the behavior of the cladding during the LOCA, it is necessary to first determine the fuel and cladding conditions at the onset of the LOCA. The initial conditions are provided by a steady-state fuel performance code, which is discussed in Section 4.2. Such codes have been used to provide the analyses of a number of important conditions, as given in FSAR Chapters 4 and 15. Some of the most significant are the initial conditions for the LOCA analysis, where fragmentation of embrittled cladding is evaluated. The steady-state fuel performance code provides fuel pellet temperatures (stored energy) and fuel rod gas inventories for the ECCS evaluation model, as prescribed by Appendix K to 10 CFR 50.46.

Initial conditions for the LOCA have been calculated for the River Bend fuel with an approved version of the GE GEGAP III Code (NEDC-20181). The model incorporates time-dependent fuel densification, time-dependent gap closures, and cladding creepdown for the calculation of gap conductance.

In 1976, the NRC staff questioned (Ross, 1976) the validity of fission gas release calculations in most fuel performance codes including GEGAP III for burnups greater than 20,000 MWd/tU. GE was informed of this concern and was provided with a method (NUREG-0418) of correcting gas release calculations for burnups greater than 20,000 MWd/tU. Although a reanalysis was not performed specifically for the River Bend fuel, an 8x8 reanalysis (Sherwood, 1976) was performed for early reflooding plants and reportedly bounds the River Bend case. Although the fuel rod internal pressure was shown to remain below system pressure for rod peak burnups below 40,000 MWd/tU, the generic reanalysis did result in higher initial stored energy and rupture pressure in the LOCA analysis. Under LOCA conditions, the higher fission gas release results in a maximum increase of 85°F in a calculated peak cladding temperature (PCT) at end of life (<33,000 MWd/tU planar average exposure). This increase in PCT has not been accounted for in the River Bend ECCS analysis. In recent submittals (Engel, 1981a and b), GE requested that credit for calculated peak cladding temperature margin as well as credit for recently approved ECCS evaluation model changes be used to offset any operating penalties as a result of high burnup fission gas release.

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The staff found this proposal acceptable (Rubenstein, 1981). Therefore, the staff considers the issue of enhanced fission gas release at high burnup satisfactorily resolved for River Bend.

~~Rev. 2, subpart (3) of Section 4.2.3.3 to~~

Cladding Ballooning

As discussed in Section 4.2.3.2 of this report, GE submitted a generic sensitivity study (Buchholz, 1981 and Quirk, 1981) in response to a condition in the approval of the GE ECCS evaluation model. The staff has completed its generic review of this issue (Rubenstein, 1982).

The staff review concluded

(1) The proposed "adjusted" cladding rupture temperature model is acceptable because it does not underestimate the incidence of rupture based on applicable data including those data reported in NUREG-0630 and more recent data.

(2) The existing cladding models for prerupture strain, postruptūre strain, and strain used for oxidation calculations are acceptable because GE has shown that the calculated values of peak cladding temperature and percent oxidation are sufficiently insensitive to variations in cladding strain.

These conclusions generically resolve the issues related to NUREG-0630 and remove the condition in the staff approval of the ECCS evaluation model. Because these cladding models will be employed in the plant-specific River Bend ECCS analysis, the issue of cladding ballooning and rupture is resolved for the River Bend application.

4.2.4 Testing, Inspection, ~~and Surveillance Plans~~ and PLAN

4.2.4.1 see following

4.2.4.2 see following

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result in higher initial stored energy and rupture pressure in the LOCA analysis. Under LOCA conditions, the higher fission gas release results in a maximum increase of 85°F in calculated peak cladding temperature (PCT) at end of life (~33,000 MWd/tU planar average exposure). This increase in PCT has not been accounted for in the Perry ECCS analysis. GE requested (Engel, May 6 and May 28, 1981) that credit for calculated peak cladding temperature margin as well as credit for recently approved ECCS evaluation model changes be used to offset any operating penalties due to high burnup fission gas release.

The staff has found this GE proposal acceptable (Rubenstein, June 25, 1981) and therefore considers the issue of enhanced fission gas release of high burnup to be satisfactorily resolved for Perry.

(2) Violent Expulsion of Fuel

The analysis that demonstrates that the acceptance criteria are met is evaluated in Section 15.4.4.

(3) Cladding Ballooning (LRG-II Generic Issue 1-CPB)

As discussed previously in Section 4.2.3.2(7), GE has submitted a generic sensitivity study (Buchholz, May 15, 1981) in response to a condition in the approval of the General Electric ECCS evaluation model. The staff has completed its generic review of this issue (Rubenstein, April 5, 1982).

The staff review concluded the following:

(a) The proposed "adjusted" cladding rupture temperature model is acceptable because it does not underestimate the incidence of rupture based on applicable data including those reported in NUREG-0630 and more recent data.

(b) The existing cladding models for prerupture strain, postrupture strain, and strain used for oxidation calculations are acceptable because GE has shown that the calculated values of peak cladding temperature and percent oxidation are sufficiently insensitive to variations in cladding strain.

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These conclusions generically resolve the issues related to NUREG-0630 and remove the condition in the staff approval of the ECCS evaluation model. Because these cladding models will be used in the plant-specific Perry ECCS analysis, the issue of cladding ballooning and rupture is resolved for the Perry application.

(4) Fuel Assembly Structural Damage From External Forces (LRG-II Generic Issue 2-CPB)

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An analysis must be provided by the applicant that shows that the [redacted] fuel meets the structural requirements (including liftoff) of Appendix A to Section 4.2 of NUREG-0800. Because generic analytical methods presented in NEDE-21175 have been approved by the staff (Parr, May 17, 1979), the staff considers this issue to be confirmatory for [redacted], and will report on it in a supplement to this report.

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The staff found this proposal acceptable (Rubenstein, 1981). Therefore, the staff considers the issue of enhanced fission gas release at high burnup adequately resolved for River Bend.

Revise to form (2) of Section 4.2.3.3 to

Cladding Ballooning

As discussed in Section 4.2.3.2 of this report, GE submitted a generic sensitivity study (Buchholz, 1981 and Quirk, 1981) in response to a condition in the approval of the GE ECCS evaluation model. The staff has completed its generic review of this issue (Rubenstein, 1982).

The staff review concluded

(1) The proposed "adjusted" cladding rupture temperature model is acceptable because it does not underestimate the incidence of rupture based on applicable data including those data reported in NUREG-0630 and more recent data.

(2) The existing cladding models for prerupture strain, postrupture strain, and strain used for oxidation calculations are acceptable because GE has shown that the calculated values of peak cladding temperature and percent oxidation are sufficiently insensitive to variations in cladding strain.

These conclusions generically resolve the issues related to NUREG-0630 and remove the condition in the staff approval of the ECCS evaluation model. Because these cladding models will be employed in the plant-specific River Bend ECCS analysis, the issue of cladding ballooning and rupture is resolved for the River Bend application.

4.2.4 Testing, Inspection and Surveillance Plans

4.2.4.1 See following

4.2.4.2 See following

Revise

result in higher initial stored energy and rupture pressure in the LOCA analysis. Under LOCA conditions, the higher fission gas release results in a maximum increase of 85°F in calculated peak cladding temperature (PCT) at end of life (~33,000 MWd/tU planar average exposure). This increase in PCT has not been accounted for in the Perry ECCS analysis. GE requested (Engel, May 6 and May 28, 1981) that credit for calculated peak cladding temperature margin as well as credit for recently approved ECCS evaluation model changes be used to offset any operating penalties due to high burnup fission gas release.

The staff has found this GE proposal acceptable (Rubenstein, June 25, 1981) and therefore considers the issue of enhanced fission gas release of high burnup to be satisfactorily resolved for Perry.

(2) Violent Expulsion of Fuel

The analysis that demonstrates that the acceptance criteria are met is evaluated in Section 15.4.4.

(3) Cladding Ballooning (LRG-II Generic Issue 1-CPB)

As discussed previously in Section 4.2.3.2(7), GE has submitted a generic sensitivity study (Buchholz, May 15, 1981) in response to a condition in the approval of the General Electric ECCS evaluation model. The staff has completed its generic review of this issue (Rubenstein, April 5, 1982).

The staff review concluded the following:

(a) The proposed "adjusted" cladding rupture temperature model is acceptable because it does not underestimate the incidence of rupture based on applicable data including those reported in NUREG-0630 and more recent data.

(b) The existing cladding models for prerupture strain, postrupture strain, and strain used for oxidation calculations are acceptable because GE has shown that the calculated values of peak cladding temperature and percent oxidation are sufficiently insensitive to variations in cladding strain.

Revise

These conclusions generically resolve the issues related to NUREG-0630 and remove the condition in the staff approval of the ECCS evaluation model. Because these cladding models will be used in the plant-specific Perry ECCS analysis, the issue of cladding ballooning and rupture is resolved for the Perry application.

(4) Fuel Assembly Structural Damage From External Forces (LRG-II Generic Issue 2-CPB)

River Bend Station

An analysis must be provided by the applicant that shows that the [redacted] fuel meets the structural requirements (including liftoff) of Appendix A to Section 4.2 of NUREG-0800. Because generic analytical methods presented in NEDE-21175 have been approved by the staff (Parr, May 17, 1979), the staff considers this issue to be confirmatory for [redacted], and will report on it in a supplement to this report.

River Bend station

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4.2.4 Testing and Inspection Plans ~~and Surveillance Plans~~

4.2.4.1 Testing and Inspection

As described in Section 4.2 of NUREG-0800, fuel testing and inspection plans should include verification of significant fuel design parameters. Although details of the manufacturer's testing and inspection programs should be documented in quality control reports, the programs for onsite inspection of new fuel and control assemblies after they have been delivered to the plant should also be described in the FSAR or in supporting documents.

The applicant has stated that rigid quality control requirements are enforced at every stage of fuel manufacturing to ensure that the design specification are met. Fuel cladding is subjected to the dimensional and ultrasonic inspections. Destructive tests are performed on representative samples of each lot of tubing. The integral fuel rods are examined by metallographic and radiographic methods on a sample basis. Leak tests are performed to confirm hermeticity. In addition, an extensive program of testing and inspection is documented in Section 4.2.4 of the FSAR for fuel pellets, fuel bundles, end plugs, gadolinia concentration, and control rod assemblies. On arriving at the reactor site, inspections will be performed according to those outlined in Tables 4.2-7 and 4.2-8 of the FSAR to ensure that damage has not occurred during handling and shipping. The staff concludes, based on the information provided in Section 4.2.4 of the FSAR, that the new fuel testing and inspection program for the [redacted] fuel system is acceptable.

4.2.4.2 Online Fuel System Monitoring

BWRs such as [redacted] have two independent radiation detection systems that are directly capable of sensing fission product releases from failed fuel rods. The main steamline radiation (MSLR) monitors are used to detect high radiation levels resulting from gross fuel failure. At [redacted], four redundant MSLR monitors will be used, each of which provides a signal to trip the turbine and initiate reactor scram independent of operator action. Because the MSLR monitors are located relatively close to the rector core, they are capable of sensing gross fission product releases relatively quickly (that is, in a few seconds), and the off-gas system radiation (OGSR) monitors are capable of detecting low-level emissions of noble gases, which could indicate the occurrence of minor fuel damage, in 2 to 3 minutes, the time required for the activity to travel from core to the detectors. THE OGSR monitors are set to sound alarms that would initiate operator action (see Chapter 11 of the [redacted] FSAR for details). The NRC staff has reviewed the BWR activity monitors (NUREG-0401) and concludes that the combination of (1) both radiation detection systems and (2) implementation of applicable Technical Specifications on limiting safety system instrumentation setpoints and specific activity (Technical Specification, Section 2.2, and Technical Specification 3/4.4.5, respectively) provides reasonable assurance of the adequacy of the online fuel system monitoring at [redacted].

4.2.4.3 Postirradiation Surveillance

Revised

In a recent submittal (Davidson, April 21, 1982) the applicant has committed to establish a routine fuel inspection program to provide information on irradiated and discharged fuel following each refueling. The program will involve visual examination of selected assemblies. Typically, visual examination will be made

Design Followup 4.2.4.3 Tab

4.2.4.3 Post-Irradiation Surveillance

A general post-irradiation surveillance program for fuel assemblies that meets the guidelines of Paragraph II.D.3 of SRP 4.2 (NUREG-0800) should be provided. A plan similar to that proposed for WNP-2 would be acceptable. The staff will address this open issue in a supplement to this report.

4.2.5 Evaluation Findings

The River Bend fuel system design has been reviewed in accordance with SRP 4.2 (NUREG-0800). The staff concludes that, although most of the objectives of the fuel system safety review have been met, there are confirmatory and open issues to be resolved before the review can be completed. As identified in the WNP-2 SER (NUREG-0892), the confirmatory issues are the following: (1) fuel rod mechanical fracturing (SER Sections 4.2.1.2 and 4.2.3.2); (2) fuel assembly structural damage from external forces (SER Sections 4.2.1.3 and 4.2.3.3); (3) fuel rod bowing (SER Section 4.2.3.1); and (4) overheating of gadolinia fuel pellets (SER Section 4.2.3.2(d)). In the WNP-2 SER, the NRC staff also determined that a license condition is needed for the channel box deflection issue (SER Section 4.2.3.1). The open issue is the post-irradiation surveillance plan (SER Section 4.2.4.3).

When the confirmatory and open issues are resolved, the staff will be able to conclude that the River Bend fuel system has been designed so that (1) it will not be damaged as a result of normal operation and anticipated operational occurrences, (2) fuel damage during postulated accidents would not be severe enough to prevent control rod insertion when it is required, and (3) core coolability will always be maintained even after postulated accidents, thereby meeting the related requirements of 10 CFR 50.46; GDC 10, 27, and 35; and 10 CFR 50, Appendix K. This conclusion is based on the following two primary factors:

(1) With the exception of the remaining confirmatory and open issues, the applicant has provided sufficient evidence that the design objectives will be met based on operating experience, prototype testing, and analytical predictions. Those analytical predictions dealing with

control rod drop and fuel densification have been performed in accordance with RG 1.77, and with methods that the staff has reviewed and found to be acceptable alternatives to RG 1.126.

(2) The applicant has provided for testing and inspection of new fuel to ensure that it is within design tolerances at the time of core loading. Online fuel failure detection equipment will be in place to provide warning of cladding perforations during plant operation. The applicant is expected to make a commitment to perform post-irradiation surveillance to detect anomalies or confirm that the fuel has performed as expected.

The staff concludes that the applicant has described methods of adequately predicting fuel rod failures during postulated accidents so that radioactivity releases are not underestimated, thereby meeting the related requirements of 10 CFR 100. In meeting those requirements, the applicant has used the fission-product release assumptions of RGs 1.3 and 1.25.

On the basis of this review, the staff concludes that, with the above exceptions, all the requirements of the applicable regulations have been met and the applicable RGs and current regulatory positions have been followed for the River Bend fuel design.

4.3 Nuclear Design

The staff review of the nuclear design of the River Bend station was based on the FSAR, amendments thereto, and the referenced topical reports. The staff review was conducted within the guidelines provided by SRP 4.3.

4.3.1 Design Bases

The FSAR presents design bases that comply with the applicable GDC. Acceptable fuel design limits are specified (GDC 10), a negative prompt feedback coefficient is specified (GDC 11), and tendency toward divergent operation (power oscillation) is not permitted (GDC 12). Design bases are presented that require a control and monitoring system (GDC 13) that automatically initiates a rapid reactivity insertion to prevent the exceeding fuel design limits in

normal operation or anticipated transients (GDC 20). The control system must be designed so that a single malfunction or single operator error will not cause a violation of fuel design limits (GDC 25). A standby liquid control system is provided that is capable of bringing the reactor to cold shutdown conditions (GDC 26), and the control system, when combined with the engineered safety features, has the capability of reliably controlling reactivity changes to ensure that the capability to cool the core is maintained under postulated accident conditions (GDC 27). Postulated reactivity accident conditions must be limited so that no damage to the reactor coolant system boundary occurs (GDC 28).

The staff finds the design bases presented in the FSAR acceptable.

4.3.2 [redacted] Description

Descriptions of the first cycle enrichment distribution, burnable poison loading, plutonium buildup, delayed neutron fraction, neutron lifetime, and core burnup have been provided. The values presented for these parameters are consistent with the design bases and are acceptable.

Power Distribution

The staff has reviewed the methods used by GE to predict power distributions during core lifetime (see Section 4.3.3 below). These methods have been compared to measured power distributions in operating BWRs to demonstrate their acceptability. Power distributions are controlled during reactor operation by adherence to predetermined control rod sequences to limit the maximum heat generation rate and minimum critical power ratio to values specified in the Technical Specifications. Power distributions will be monitored during reactor operation by the incore detector system. This system, described in the GE Topical Report APED-5706, "Incore Monitoring System for General Electric Water Reactors," consists of a source range monitoring subsystem (up to 10^{-5} full power), an intermediate range monitoring subsystem (10^{-6} to 0.2 full power), and an average power range monitoring subsystem (~0.05 to 1.5 full power). In addition, a traversing incore probe subsystem is used to

calibrate the local power range monitors and to obtain detailed axial power distributions.

The intermediate range and average power range mc toring subsystems are each equipped with trips to scram the reactor if core power levels reach undesirable values. These systems satisfy GDC 13 with respect to neutron flux monitoring. A comparison of calculated and measured power distributions is given in NEDO-20946, "BWR Simulator Verification Methods" (see Section 4.3.3 below). This comparison demonstrates that GE design methods are capable of adequately representing reactor operating states.

The staff concludes that discussions of power distributions in Section 4.3 and in the other documents referenced above are acceptable. The staff further concludes that the information presented concerning monitoring of power distributions presented in the FSAR and in Topical Report NEDO-20340, "Process Computer Performance Evaluation Accuracy," is acceptable.

Reactivity Coefficients

The most significant reactivity coefficients with respect to the stability and dynamic behavior of the reactor are the void coefficient and the Doppler coefficient. Of less significance is the moderator temperature coefficient. The fuel temperature, or Doppler, coefficient of reactivity will be negative at all operating conditions and times in life. The moderator void coefficient will also be always negative. The presence of negative Doppler and void coefficients during all operating conditions satisfies GDC 11, which requires a negative prompt reactivity coefficient in the power operating range. The moderator temperature coefficient may become slightly positive for certain operating conditions (very low power at end of life), but its effect is overshadowed by that of the other coefficients. The GE has submitted a Topical Report, NEDO-20964, "Generation of Void and Doppler Reactivity Feedback for Application to BWR Design," which describes the methods used to obtain void and Doppler reactivity coefficients. This report has been approved (see Section 4.3.3 below), by the staff concluding that based on the review to date, the predictions of the various reactivity coefficients are suitably performed.

The calculated values of void and Doppler coefficients are multiplied by design conservatism factors and used in a point kinetics neutronics model to calculate the results of plant transients. Comparison of such calculations with experiments (for example, the Peach Bottom Unit 2 turbine trip tests) have shown them to be nonconservative for certain core-wide transients involving overpressurization and rapid void collapse. Accordingly, the staff requires that such transients be analyzed by the recently approved ODYN code, which is described in NEDO-24154, "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors," Volumes 1, 2, and 3. This code has been verified by comparison with the Peach Bottom Unit 2 tests and others. The staff concludes that the ODYN code provides an acceptable calculation of pressurization transients when used in the manner prescribed in the letter dated February 4, 1981 from R. Tedesco, NRC, to G. G. Sherwood, GE. The point kinetics code is acceptable for other transients when the design conservatism factors are used.

Control Requirements

To allow for changes in reactivity as a result of reactor heatup (fuel and moderator temperature rise and void formation), load following (transient xenon), equilibrium xenon and samarium, and fuel burnup with consequent fission product buildup, a significant amount of excess reactivity is built into the core at beginning of life. In BWRs this excess reactivity is accounted for by the control rods except for a portion of that needed to account for fuel burnup. That portion is accounted for by burnable poison located in the fuel assemblies. The burnable poison also functions to shape the radial and axial power.

The applicant has presented data to show that sufficient control exists to satisfy the above requirements with enough additional control to provide a cold xenon-free effective multiplication factor--$\leq$0.99--at the most reactive point in the core lifetime with the rod having the highest worth stuck out of the core. NEDO-29046 provides comparisons of calculated and measured cold critical states and provides a demonstration that calculation of shutdown margins is adequate. The staff concludes that suitably conservative assessments of reactivity control requirements have been made and that adequate reactivity

control has been provided to ensure shutdown capability, even with the highest worth rod stuck out of the core.

A standby liquid control system is provided that is completly independent of the control rod system and is capable of shutting down the reactor and maintaining it in the cold shutdown state at any time in core life. This satisfies the shutdown reactivity requirement of GDC 26.

Control Rod Patterns and Reactivity Worth

Startup and operation of the reactor will be performed by manipulation of control rods and control of recirculation flow. The control rods will be withdrawn in sequence according to predetermined patterns. These patterns are established in such a way that the following design criteria are met:

(1) Control rod worths shall be limited so there are acceptable consequences if a rod is dropped from the fully inserted to a withdrawn position (rod drop accident).

(2) Control rod withdrawal increments of rod worth shall be limited so that withdrawal of a control rod by one notch does not produce a period that cannot be handled by the operator.

In BWR-6 reactors, certain of the rods may be withdrawn in gangs of three or four rods at a time. Analyses have been done to show that the above design criteria are met for such withdrawals.

During operation, the rod patterns are monitored by the rod pattern control system function of the rod control and information system up to the preset power level (~20% power). Above this power level, rod worths are not sufficient to violate the above criteria. The rods are withdrawn in the banked position withdrawal sequence that is described in Topical Report NEDO-21231, "Banked Position Withdrawal Sequence." This report has been reviewed and accepted by the staff, and the staff concludes that this operating mode is acceptable for use in the River Bend reactor.

Stability

The stability of large BWRs to xenon oscillations has been discussed in the GE Topical Report APED-5640, "Xenon Considerations in Design of Large Boiling Waters Reactors." These studies show that a BWR will be stable to any xenon-induced power oscillation because of the damping effect of the large, negative, spatially varying void coefficient. In addition, attempts to induce undamped xenon oscillations in operating BWRs confirm the presence of a large damping effect.

The staff concurs with the conclusion that large BWRs will be stable to xenon-induced power oscillations.

This satisfies part of GDC 12, which requires that power oscillations that can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed. Thermal-hydraulic stability is discussed in Section 4.4 below.

Criticality of Fuel Assemblies

The staff has reviewed the information presented regarding criticality of fuel assemblies in storage and handling operations. Calculations are performed for higher-than-normal enrichment fuel assemblies, and no credit is taken for burnable poison in the fuel. The K_{eff} value for two bundles side by side in a pool is ~0.74. Four bundles in a square arrangement have a K_{eff} value of ~0.90. An array of 12 to 14 fuel bundles is required for criticality.

Vessel Irradiation

Neutron fluences at the inside surface of the pressure vessel have been calculated using a one-dimensional discrete ordinates code in an infinite cylinder geometry. Design radial and axial power distributions are used, and the calculation is performed with the axially peaked value of the flux. An azimuthal peaking factor is applied to the calculated flux to obtain the peak in the azimuthal direction. Operation at full power for 40 years with an 80% utilization factor was assumed. Anisotropic scattering effects were included

outside the core, and the resultant fluence was 8.7×10^{18} neutrons per cm^2 for neutrons having energies greater than 1 million electron-volts, including a factor of 2 for conservatism.

The staff concludes that the calculation method used to obtain the vessel fluence is state of the art and is acceptable. The staff further concludes that the core and externals have been properly modeled and that suitable values of vessel fluence have been obtained.

4.3.3 Analytical Methods

The staff has reviewed and evaluated the information presented on the analytical methods. The basic calculational procedures used by GE for generating neutron cross-sections are part of its so-called lattice physics model. In this model, the neutron spectrum is divided into energy regions, and a different technique is used in each region. The fast groups (fission source energy range) are treated by multigroup integral collision probabilities to account for geometrical effects in fast fission. Epithermal cross-sections are calculated by the B1 method of the widely used GAM code. Resonance energy cross-sections for this energy range are calculated by using the intermediate resonance approximation with energy- and position-dependent Dancoff factors included. The thermal cross-sections are computed by a THERMOS-type program. This program accounts for the spatially varying thermal spectrum throughout a fuel bundle. These calculations are performed for an extensive combination of parameters including fuel enrichment and distribution, fuel and moderator temperatures, burnup, voids, void history, the presence or absence of adjacent control rods, and gadolinia concentration and distribution in the fuel rods. As part of the lattice physics model, three-group two-dimensional XY diffusion calculations for one or four fuel bundles are performed. In this way, local fuel rod powers can be calculated as well as single-bundle or four-bundle (with or without a control rod present) average cross-sections.

GE has submitted a Topical Report, NEDE-20913-P, "Lattice Physics Methods," that describes in detail the procedures outlined above. The staff has reviewed this report and concludes that the discussion in it permits a knowledgeable person to conclude the methods employed are state of the art. The staff

further concludes that the methods satisfy the provisions of the SRP (NUREG-0800) for core physics methods and are acceptable.

The single- or four-bundle-average neutron cross-sections that are obtained from the lattice physics model are used in either two- or three-dimensional diffusion calculations. Two-dimensional XY calculations are usually performed in three groups at a given axial location to obtain gross power distribution, and reactivities. The three-dimensional diffusion calculations use one energy group and can couple neutronic and thermal-hydraulic phenomena. These three-dimensional calculations are performed using 24 axial nodes and 1 radial node per fuel bundle. This three-dimensional calculation provides power distributions, void distributions, control rod positions, reactivities, eigenvalues, and average cross-sections for use in the one-dimensional axial calculations. The three-dimensional calculations have been described in the GE Topical Report NEDO-20953, "Three Dimensional BWR Core Simulator." The staff has reviewed this report and has reached conclusions similar to those reached for the lattice physics methods report.

The one-dimensional calculation referred to above is a space-time diffusion calculation that is coupled to a single-channel thermal-hydraulic model. This axial calculation is used to generate the scram reactivity function for various core operating states. GE has compared results obtained using this one-dimensional space-time code with those obtained using the industry standard code, WIGLE, and shown to be conservative. A staff consultant, Brookhaven National Laboratories, has performed an extensive study of BWR scram reactivity behavior (BNL-NUREG-50584, "A Dynamic Analysis of BWR Scram Reactivity Characteristics") and has concluded that the end-of-cycle, all-rods-out configuration represents the limiting condition for BWR scram system effectiveness. Thus, the staff concludes that the method and assumptions used by GE to obtain the scram reactivity curve are acceptable. (See Section 4.3.2.2 above for application of the scram curve.)

The Doppler, moderator void, and moderator temperature reactivity coefficients are generated in a straightforward manner from data obtained from the lattice physics model. The effective delayed neutron fraction and the prompt mode neutron lifetime are computed using the one-dimensional space-time code. The

power coefficient is obtained by appropriately combining the moderator void, Doppler, and moderator temperature reactivity coefficients.

GE has submitted a Topical Report, NEDO-20964, "Generation of Void and Doppler Reactivity Feedback for Application to BWR Design" that the staff ~~is currently reviewing~~ has received and approved. ~~Based on the review to date, the staff does not anticipate any major changes to the methods~~. (See comment in Section 4.3.2 above.)

The effect of spatially-varying xenon concentrations on the stability of a BWR is specifically discussed in the GE Topical Report APED-5640. These studies show that a BWR will be stable to any xenon-induced power oscillations because of the damping effect of the large, negative, spatially-varying void coefficient.

Comparisions between calculated and measured local and gross power distributions have been presented by GE in Topical Reports NEDO-20939 and NEDO-20946. Local (intra-bundle) power distribution comparisons were made to data obtained from critical experiments and from gamma scans performed on operating plants. Gross radial and axial power distributions obtained from operating plants have been compared with values predicted by the BWR simulator code. These comparisons have yielded values for calculational uncertainties to be applied to power distributions. Comparisons have also been made of calculated values of cold, xenon-free reactivity and hot operating reactivity of a number of operating reactors as a function of cycle exposure. These comparisons have been used to establish shutdown reactivity requirements.

The staff has reviewed the two topical reports, NEDO-20939 and NEDO-20946, and found them acceptable for reference in licensing actions.

4.3.4 ~~Evaluation Findings~~ Changes

The applicant has described the computer programs and calculational techniques used to predict the nuclear characteristics of the reactor design and has provided examples to demonstrate the ability of these methods to predict experimental results. The staff concludes that the information presented adequately demonstrates the ability of these analyses to predict reactivity and physics characteristics of the River Bend plant.

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To allow for changes of reactivity as a result of reactor heatup, changes in operating conditions, fuel burnup, and fission product buildup, a signficant amount of excess reactivity is designed into the core. The applicant has provided substantial information relating to core reactivity requirements for the first cycle and has shown that means have been incorporated into the design to control excess reactivity at all times. The applicant has shown that sufficient control rod worth is available to shut down the reactor with at least a 1.0% $\Delta k/k$ subcritical margin in the cold condition at any time during the cycle with the highest worth control rod stuck in the fully withdrawn position.

On the basis of its review, the staff concludes that the applicant's assessment of reactivity control requirements over the first core cycle is suitably conservative, and that adequate negative worth has been provided by the control system to ensure shutdown capability. Reactivity control requirements will be reviewed for additional cycles as this information becomes available.

The staff concludes that the nuclear design is acceptable and meets the requirements of GDC 10, 11, 12, 13, 20, 25, 26, 27, and 28.

The applicant has met the requirements of GDC 11 with respect to prompt inherent nuclear feedback characteristics in the power operating range by (1) calculating a negative Doppler coefficient of reactivity, and (2) using calculational methods that have been found acceptable. The staff has reviewed the Doppler reactivity coefficients in this case and found them to be suitably conservative.

The applicant has met the requirements of GDC 12 with respect to power oscillations that could result in conditions exceeding specified acceptable fuel design limits by (1) showing that such power oscillations are not possible and/or can be easily detected and thereby remedied, and (2) using calculational methods that have been found acceptable. The staff has reviewed the analysis of these power oscillations in this case and found them to be suitably conservative.

The applicant has met the requirements of GDC 13 with respect to provision of instrumentation and controls to monitor variables and systems that can affect

the fission process by (1) providing instrumentation and systems to monitor the core power distribution, control rod positions and patterns, and other process variables such as water level and pressure, and (2) providing suitable alarms and/or control room indications for these monitored variables.

The applicant has met the requirements of GDC 26 with respect to provision of two independent reactivity control systems of different designs by (1) having a system than can reliably control anticipated operational occurrences, (2) having a system that can hold the core subcritical under cold conditions, and (3) having a system that can control planned, normal power changes.

The applicant has met the requirements of GDC 27 with respect to reactivity control systems that have a combined capability in conjunction with action (1) because the emergency core cooling system reliably controls reactivity changes under postulated accident conditions by providing a movable control rod system and a liquid poison system, and (2) by performing calculations to demonstrate that the core has sufficient shutdown margin with the highest worth stuck rod.

The applicant has met the requirements of GDC 28 with respect to postulated reactivity accidents by (1) meeting the fuel enthalpy limit of 280 cal/gm, (2) meeting the criteria on the capability to cool the core, and (3) using calculational methods that have been found acceptable for reactivity insertion accidents.

The applicant has met the requirements of GDC 10, 20, and 25 with respect to specified acceptable fuel design limits by providing analyses demonstrating (1) that fuel design criteria are met during normal operation, including the effects of anticipated operational occurrences; (2) that the automatic initiation of the reactivity control system ensures that fuel design criteria are not exceeded as a result of anticipated operational occurrences and ensures the automatic operation of systems and components important to safety under accident conditions; and (3) that no single malfunction of the reactivity control system causes violation of the fuel design limits.

4.4 Thermal and Hydraulic Design

The staff evaluation of the thermal-hydraulic design according to SRP 4.4 (NUREG-0800) included the design criteria, implementation of design criteria as presented by the final core design, and the analyses of core thermal-hydraulic performance.

4.4.1 Design Bases

The thermal-hydraulic safety design bases for River Bend can be summarized as follows:

(1) Fuel failure because cladding overheating should not occur during normal operation or anticipated operational occurrence. Specifically, the minimum critical power ratio (MCPR) operating limit is specified so that at least 99.9% of the fuel rods in the core are not expected to experience boiling transition during normal operation or anticipated operational occurrences.

(2) Core and fuel thermal-hydraulic design limits for steady-state operation (MCPR and linear heat generation rate (LHGR) limits) should be defined to provide margin between steady-state operating conditions and any operating condition that could result in fuel failure. Specifically, they should ensure that no fuel failure results during normal operation or anticipated operational occurrence even with uncertainties taken into account. SteA

(3) No undamped oscillations or other hydraulic instabilities should occur either for normal operation or for anticipated operational occurrences.

The applicant's thermal-hydraulic design bases are consistent with the SRP and are, therefore, acceptable.

4.4.2 ~~Critical Power Ratio Correlation~~ Description of Thermal and Hydraulic Design of the Reactor Core

The GEXL correlation is used to predict the (critical) quality at which boiling transition occurs as a function of the distance from the equilibrium boiling

boundary. The GEXL correlation and design application are described in the GE Topical Report, NEDO-10958A. The figure of merit chosen for reactor design and operation is the critical power ratio, that is, the ratio of the critical bundle power to the operating bundle power. This correlation and supporting topical report have been previously reviewed and approved by the staff (Butler, 1974).

Critical power tests have been run on prototypical 8x8 bundles with two water rods, and the results have been documented in NEDE-24196. Test data for cosine axial heat flux shapes indicate that the water rods do not affect the GEXL capability of predicting the bundle critical power performance for bundle radial peaking patterns typical of 8x8 retrofit fuel. The staff has previously found that the GEXL data base, which includes top and bottom peak axial heat flux distributions combined with the two-water rod data, demonstrates the adequacy of the GEXL correlation to predict critical power in both 8x8 and 8x8 retrofit bundles. The staff has previously concluded that the GEXL correlation is acceptable for both 8x8 and 8x8 retrofit fuel applications (Tedesco, 1981b).

4.4.3 Thermal-Hydraulic ~~Design Methods~~ Description of the Design of the Reactor Coolant System and

The MCPR limit originally proposed was based upon calculations using the REDY model described in NEDO-10802. The results from the tests performed at Peach Bottom Unit 2 revealed that in certain cases the results predicted by REDY are nonconservative. Therefore, the staff required the applicant to use ODYN methods to analyze the following transients: (1) feedwater controller failure at maximum demand with and without bypass, (2) generator load rejection, and (3) turbine trip with and without bypass. The staff reviewed the applicant's reanalysis results, ~~The results~~ which indicate that the most limiting transient with respect to the fuel clad thermal margin is the loss of feedwater heater ~~when in manual mode~~ because of the fast scram time. ~~This appears to be inconsistent with the GESSAR-238 NSSS and -251 NSSS findings whereby the most limiting event for the latter~~ Based on our review, we conclude that: is ~~generator load rejection without bypass. The staff requires the applicant to explain this difference. The staff will verify that the applicant has satisfactorily responded to the above concerns related to the steadystate operating limit for the MCPR and will document our evaluation results in the final SER.~~ (1) approved methods were used and (2) the results of analyses are consistent with the proposed operating limit MCPR to avoid violation of the safety limit MCPR for the design transients and are acceptable.

The staff also finds that the generator load rejection and turbine trip transients without bypass have been reclassified from moderate to infrequent event; the staff does not accept this reclassification. However, the staff has found that there is no impact on determination of the operating limit MCPR by reclassifying the transients.

4.4.4 ~~Thermal-Hydraulic Stability~~ Evaluation

Recent BWR fuel design changes that affect stability include decreasing the rod size and increasing the gap conductance because of pre-pressurization. As a consequence, the maximum decay ratio for most BWRs increases and becomes larger than 0.5, which is the original GE design criterion for BWR stability. Therefore, GE now proposes a decay ratio of 1.0 for its criterion. The staff has not agreed that the proposed criterion of a 1.0 decay ratio calculated by using the FABLE code is acceptable.

To further evaluate this criterion and other stability criteria, the staff is performing a generic study of the hydrodynamic stability characteristics of light water reactors under normal operation, anticipated transients, and accident conditions. The results of this study will be applied to the staff's review and acceptance of stability analyses and analytical methods now in use by the reactor vendors.

The stability analysis resulted in a maximum decay ratio of 0.98 for the end-of-life cycle, which is the limiting cycle with respect to stability. The staff has approved for the Susquehanna initial cycle operation ~~core design~~ (NUREG-0766), a core design which has a calculated maximum decay ratio value of 0.7. ~~for the initial cycle. Because the~~ Since the Susquehanna and River Bend plants have similar core design, the staff concludes that the River Bend core design for thermal hydraulic stability is acceptable for Cycle 1. ~~the initial cycle~~. However, to provide additional margin for stability, natural circulation operation will be prohibited until the staff's review of these conditions is completed. Any action resulting from the staff's study will be applied to River Bend.

The applicant's operating license will be conditioned to require that a new stability analysis be submitted and approved before second-cycle operation.

Also, because no analysis has been presented for MCPR limits or stability characteristics for single-loop operation, the staff will require by Technical Specifications that single-loop operation not be permitted until supporting analyses are provided and approved.

4.4.5 ~~Crud Deposition~~ Testing and Verification

Crud deposition causes gradual flow reductions in some light water reactor cores. However, measurement of core flow by jet pump pressure drop and core plate pressure drop will provide adequate indication of such flow reductions, if they should occur. Technical Specifications will require that the core flow be checked at least once every 24 hours to detect flow reduction.

For pressure drop considerations in design analyses, it is assumed that a conservative amount of crud is deposited on the fuel rods and the fuel rod spacers. This is reflected in decreased flow area, increased friction factors, and increased spacer loss coefficients. The effect of this crud is to increase the core pressure drop by approximately 1.7 psi. The staff concludes that the assumptions regarding crud deposition used in the design analysis in conjunction with the required flow monitoring are acceptable.

4.4.6 ~~Loose Parts Monitoring System~~ Instrumentation Requirements

No discussion of a loose parts monitoring system (LPMS) is given. The applicant is required to provide a description of the LPMS. The system should be installed to meet the operability requirements of RG 1.133, Revision 1 (May 1981). The LPMS must be operational and capable of recording vibration signals for signature analysis at the time of initial startup testing. The staff also requires that the applicant commit to evaluate the system to address conformance with RG 1.133, Revision 1. The conformance evaluation should emphasize the following:

(1) a description and evaluation of diagnostic procedures used to confirm the presence of a loose part

(2) a description of how the operators will be trained in the purpose and implementation of the system

(3) a report on system calibration, including signature analysis, evaluation of background noise, and alarm settings

4.4.7 NUREG-0737 Item II.F.2

A clarification of requirements for inadequate core cooling instrumentation, which is required to be installed and operational before fuel loading, was provided in a letter to all the licensees of operating nuclear power plants (Denton, 1979b) and in Item II.F.2 of NUREG-0737, "Clarification of TMI Action Plan Requirements." The applicant is required to provide a response to Item II.F.2 for review and approval before an operating license is issued.

4.4.8 Thermal-Hydraulic Comparison

The thermal-hydraulic parameters for River Bend are summarized in Table 4.1. A comparison with the parameters for the Hatch Unit 2 core design is given for reference. This core design was previously approved in the SER issued in June 1978 for Hatch Unit 2 (NUREG-0411), which is now an operating reactor. The primary difference in core design between River Bend and Hatch Unit 2 is size. Hatch Unit 2 is a BWR/4 core and River Bend is a BWR/6 core. Both use the improved 8x8R fuel assemblies.

Table 4.1 Thermal-hydraulic design comparison

Design Parameter	Hatch 2	River Bend
Design thermal output (MWt)	2,436	2,894
Final feedwater temerature (FFWT)(°F)	420	420
Steam flow rate at FFWT (10^6 lb/hr)	10.47	12.45
Core coolant flow rate (10^6 lb/hr)	77.0	84.5
Feedwater flow rate (10^6 lb/hr)	10.44	12.43
Steam pressure, nominal in steam dome (psia)	1,020	1,040
Steam pressure, nominal core design (psia)	1,035	1,055
Average power density (kW/liter)	49.15	52.4
Maximum linear thermal output (kW/ft)	13.4	13.4
Average linear thermal output (kW/ft)	5.38	5.7

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Table 4.1 (continued)

Design Parameter	Hatch 2	River Bend
Core total heat transfer area (ft^2)	54,879	61,161
Fuel type	P8x8R	P8x8R
Water rods per bundle	2	2
Core inlet enthalpy at FFWT (Btu/lb)	526.9	527.8
Core maximum exit void within assemblies (%)	76.3	76.0
Core average void, active coolant (%)	42.2	41.1
Active coolant flow area per assembly (in^2)	15.82	15.16
Core average inlet velocity (fps)	6.6	6.82
Total core pressure drop (psia)	23.9	25.26
Core support plate pressure drop (psia)	19.46	20.84
Average orifice pressure drop (psia)		
Central region	8.0	5.41
Peripheral region	16.52	17.95
Number of fuel rods per bundle	62	62
Rod outside diameter (in.)		
Fuel rod	0.483	0.483
Water rod	0.591	0.591
Active fuel length (in.)	150	150
Rod pitch (in.)	0.640	0.640

The comparability of River Bend design with Hatch Unit 2 supports the conclusion that the River Bend thermal-hydraulic design is acceptable.

4.4.9 Evaluation Findings

The River Bend thermal-hydraulic design has been reviewed according to SRP 4.4 (NUREG-0800). The scope of review included the design criteria, implementation of the design criteria as presented by the final core design, and the steady-state analysis of the core thermal-hydraulic performance. The review concentrated on the differences between the proposed core design (and criteria) and those designs and criteria that have been previously reviewed and found acceptable by the staff. It was found that all such differences were satisfactorily

justified by the applicant. The applicant's thermal-hydraulic analyses were performed using analytical methods and correlations that have been previously reviewed by the staff and found acceptable. However, the operating license should be restricted with the following conditions:

(1) Single loop operation is not permitted unless supporting analyses are provided and approved.

(2) Operation beyond the first cycle is not permitted until a stability analysis is provided and approved for the additional cycles of operation.

(3) The natural circulation operation mode is not permitted.

(4) The core flow should be checked at least once each day.

The above restrictions should be incorporated into the proposed Technical Specifications, except for Item (2), which should be incorporated as a license condition.

The applicant should submit the following documents for review and approval before an operating license is issued:

(1) a description of the LPMS and a schedule for an evaluation report of the LPMS to address conformance with RG 1.133, Revision 1

(2) a response to Item II.F.2 of the NUREG-0737 for review and approval

(3) an explanation for why the worst case with respect to the MCPR is loss of feedwater heater transient instead of generator load rejection without bypass transient as shown previously for the GESSAR-238 NSSS and -251 NSSS

With the exceptions noted above, the staff concludes that the thermal-hydraulic design of the core conforms to the requirements of GDC 10, RGS 1.68 and 1.133, and SRP 4.4 and is, therefore, acceptable.

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4.5 Reactor Materials

4.6 Functional Design of Reactivity Control Systems

The reactivity control systems were reviewed in accordance with Section 4.6 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed, according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the reactivity control systems with respect to the applicable regulations of 10 CFR 50.

4.6.1 Information for CRDS

The control rod drive system (CRDS) and recirculation flow control system (RFCS) are designed to control reactivity during power operation. The RFCS controls reactivity over a limited range by changing the coolant flow rate and concomitant coolant void content. The CRDS can affect reactivity by the insertion or removal of control rods interspersed throughout the core which contain a neutron absorber (poison). The CRDS is capable of controlling reactivity over a large range and serves to accommodate both the slow reactivity changes caused by core burnup and the rapid changes resulting from fast transients. The CRDS is the principal mechanism for shutting the reactor down

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safely and effectively. In the event of the inability of the CRDS to shut down the reactor, the standby liquid control system (SLCS) may be used to pump a solution of sodium pentaborate into the primary system in order to shut the reactor down. An evaluation of the SLCS is given in Section 9.3.5 of this SER.

4.6.2 Evaluation of the CRDS

The control rod drive hydraulic system is used to move control rods in and out of the core. Each control rod is moved by a separate hydraulic control unit (HCU). A supply pump provides each HCU with water from the condensate treatment system. In the event the condensate treatment system is not available, the condensate storage tank is used as an alternate water source for the CRDS. This water is used to cool the control rods and to move them in and out of the core. The pump also provides water to a scram accumulator in each hydraulic control unit to maintain the desired water inventory. When necessary, the accumulator forces water into the drive system to scram the control rod connected to that hydraulic control unit; at lower pressures the volume of water in the scram accumulator is sufficient to scram the rod. At higher pressures, most of the water to scram is provided from the reactor

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vessel. A single failure in a hydraulic control unit would result in the failure of only one rod. There is a spare pump on standby, in the event of failure of the operating pump. (The applicant must test the scram accumulators to show that the scram function capability will be preserved for at least 20 minutes in the event of loss of both CRD pumps in order to comply fully with General Design Criterion 29, "Protection Against Anticipated Operational Occurrences.")

4.6.3 Testing and Verification of the CRDS

The CRDS has been desiged to permit periodic functional testing during power operation with the capability to test individual scram channels and motion of individual control rods independently. The CRDS is designed so that failure of all electric power or loss of air to the scram inlet and outlet valves will cause the control rods to scram, thereby protecting the reactor. (The applicant must show that the scram function is maintained in the event of slowly decaying air pressure to the scram discharge inlet and outlet valves in order to comply with the requirements of General Design Criterion 23, "Protection System Failure Mode.")

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Preoperational tests of the control rod drive hydraulic system will be conducted to determine capability of the system. Startup tests will be conducted over the range of temperatures and pressures from shutdown to operating conditions in order to determine compliance with applicable technical specifications. Each rod that is partially or fully withdrawn during operation will be exercised one notch at least once each week. Control rods will be tested for compliance with scram time criteria, from the fully withdrawn position, after each refueling shutdown.

4.6.4 Information for Combined Performance of Reactivity Systems

A malfunction in the CRDS could result in a reactivity change. The applicant demonstrated in his safety analyses (Section 15 of the FSAR) that the CRDS limits postulated transients resulting from a single failure of the CRDS so that acceptable fuel design limits are not exceeded, as required by General Design Criterion 25, "Protection System Requirements for Reactivity Control Malfunction."

The CRDS is designed to provide reactivity control under normal operation and anticipated operational occurrences with an appropriate allowance for a stuck rod. This

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capability is demonstrated by the safety analyses discussed in Section 15 of the FSAR. However, the applicant must show scram capability upon loss of both CRD pumps in order to comply fully with the requirements of General Design Criterion 29. This system is also capable of accommodating reactivity changes during normal operating conditions (i.e., power changes and xenon burnout). The standby liquid control system is capable of bringing the reactor subcritical under cold shutdown conditions in the event the control rods cannot be inserted. These systems taken together, satisfy the requirements of General Design Criteria 26 "Reactivity Control System Redundance and Capability" and 27, "Combined Reactivity Control System Capability."

The CRDS is capable of providing reactivity control following postulated accidents with an appropriate margin for a stuck rod. This capability is demonstrated by the loss-of-coolant accident and rod dropout analyses presented by the applicant which, in turn, show that the consequences are acceptable and core cooling is maintained, as required by General Design Criterion 28, "Reactivity Limits."

4.2.5 Evaluations of Combined Performance

The applicant has committed to provide modifications to the scram discharge system to comply with the criteria contained in the Generic Safety Evaluation Report - BWR

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Scram Discharge Syscem of December 1, 1980, as indicated in Position 1 - ASB of LRG II. In addition, the plant will comply with the applicable recommendations regarding the control rod drive return line (CRDRL) specified in NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking," as follows:

(1) Equalizing valves will be installed between the cooling water header and exhaust water header,

(2) Flush ports will be installed on the exhaust header in the event the exhaust header is composed of carbon steel piping,

(3) The flow stabilizer loop will be composed of stainless steel and will be rerouted directly to the cooling water header.

Based on our review, we conclude that the functional design of the reactivity control systems meets the requirements of General Design Criteria 25, 26, 27, and 28. (However, the applicant must show that decaying air pressure will not adversely affect the scram function, in order to comply with the provisions of General Design Criterion 23.) (The applicant must also test the scram

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accumulators to assure scram capability for at least 20 minutes in the event of loss of both CRD pumps, in order to comply fully with General Design Criterion 29.)

Testing of the control rod drive hydraulic system to assure adequate flow into the reactor vessel (Recommendation 6, Section 8.1, NUREG-0619) is not required as stated in Position 5 - ASB of LRG II.

In view of the foregoing, we cannot find the applicant's submittal acceptable. The functional design of reactivity control systems does not comply with the acceptance criteria of SRP Section 4.6.

5 Reactor Coolant System

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5.1 Summary Description

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.1 Compliance with Codes and Code Cases



5.2.2 Overpressure Protection

The reactor coolant pressure boundary (RCPB) is provided with a pressure relief system to:

(1) prevent the pressure within the RCPB from rising beyond 110% of the design value, and

(2) provide automatic depressurization for small breaks in the nuclear system occurring together with failure of the high pressure core spray system so that the low pressure coolant injection and the low pressure core spray systems can operate to protect the fuel barrier.

The relief system must permit verification of its operability, and withstand adverse combinations of loadings and forces resulting from normal, upset, emergency, and faulted conditions.

Overpressurization protection at RBS is accomplished through the use of sixteen combination safety/relief valves of the Crosby type mounted on the four main steam lines. The following table indicates that the RBS pressure relief system design is similar to other BWR Class 4, 5, and 6 plants.

Plant	Class	Number of Pressure Relief Valves	Plant Rated Steam Flow (lb/hr)	Plant Rated Power (MWT)
RBS 1/2	6	16	1.24×10^7	2894
Perry 1/2	6	19	1.54×10^7	3579
Grand Gulf 1/2	6	20	1.65×10^7	3833
Clinton 1/2	6	16	1.25×10^7	2894
LaSalle 1/2	5	18	1.42×10^7	3293
Susquenanna 1/2	4	16	1.35×10^7	3293
Fermi 2	4	15	1.42×10^7	3293
Shorenam	4	11	1.05×10^7	2436

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All of the combination safety/relief valves discharge directly to the suppression pool. The valves are designed to meet seismic and quality standards consistent with the requirements of Regulatory Guide (RG) 1.26, "Quality Group Classifications and Standards for Water, Steam, and Radioactive Waste Containing Components of Nuclear Power Plants," and RG 1.29, "Seismic Design Classification," as discussed in Section 3.2* of this report.

The basis for overpressure protection in a nuclear reactor is Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), General Design Criterion (GDC) 15, "Reactor Coolant System Design." This criterion requires that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences. To satisfy this criterion, the overpressurization protection system for RBS was designed in compliance with the ASME Pressure Vessel Code - Section III which requires that the maximum pressure reached during the most severe pressure transient be less than 110 percent of the design pressure. For RBS this pressure limit is 1375 pounds per square inch gauge.

The nominal setpoints of the combination safety/relief valves are as follows:

	Setpoint, psig	
Mode of Operation	Minimum	Maximum
Relief (power actuated)	1103	1123
Safety (spring setpoint)	1165	1190

Their total capacity at their set pressure is approximately 109% of rated steam flow. Prior to installation the safety relief valve manufacturer tests the valves hydrostatically for valve response, set pressure, and seat leakage

*As an aid to DL in the preparation of the SER, references have been made in this draft of Sections 5.2.2, 5.4.6, 5.4.7, 6.3, 15 and 22 to other sections that are written by branches other than RSB. The LPM should verify that the listed references are consistent with his SER section numbering system.

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to certify that design and performance requirements have been met. During the preoperational test program, specified manual and automatic actuation is verified in compliance with Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Reactor Power Plants." In addition, 50% of the valves will be removed for maintenance and inspection, and tested each refueling outage in accordance with Section XI of the ASME Boiler and Pressure Vessel Code per plant Technical Specification requirements.

It is noted that the General Electric Company has agreed to work with the staff and their utility customers to maintain a surveillance program once new safety-relief valves become operational on any boiling water reactor (NUREG-0152). Information to be reported will include all abnormalities ranging from minor wear observed during normal inspection to complete failures, including failure to open or close and inadvertent operation. A safety/relief valve surveillance program is being developed by the BWR Owners Group (LRG II), the primary objective of which is to gather data to identify generic safety/relief valve problems. The applicant is participating in this program.

The applicant has analyzed a series of transients that would be expected to require pressure relief actuation to prevent overpressurization. These are tabulated below.

Pressurization Events Resulting In Pressure Relief Actuation

FSAR Subsection	Event
15.1.2	Feedwater Control Failure, Maximum Demand
15.1.3	Pressure Controller, Fail Open
15.2.2	Generator Load Rejection, Bypass-On
15.2.2	Generator Load Rejection, Bypass-Off
15.2.3	Turbine Trip, Bypass-On
15.2.3	Turbine Trip, Bypass-Off
15.2.4	Inadvertent MSIV Closure
15.2.5	Loss of Condenser Vacuum
15.2.6	Loss of Auxiliary Power Transformer
15.2.6	Loss of All Grid Connections

15.2.7	Loss of all Feedwater Flow
15.3.1	Trip of Both Recirculation Pump Motors
15.3.3	Seizure of One Recirculation Pump

The results of these analyses, using the nominal valve setpoints, demonstrate that the maximum vessel pressure will remain below the 1375 psig limit. The safety/relief valves are direct-acting devices and credit is assumed for all of the valves actuating in the analysis. The relief mode (power actuated) of the safety/relief valves provided at RBS is safety-grade and, therefore, half of the total stamped relieving capacity was assumed to be activated via the relief mode. The remainder of the required relieving capacity was assumed to be activated in the safety mode (spring setpoint). This position is consistent with the requirements of Article NB-7540 of Section III of the ASME Boiler and Pressure Vessel Code and we find this acceptable. For the severe transient of main steam isolation valve (MSIV) closure with a high neutron flux scram, the maximum vessel bottom pressure is estimated to be 1265 psig if all of the safety/relief valves operate as previously discussed. The analysis assumed the plant was operating at 105% of rated steam flow (13.08×10^6 lb/hr) and a vessel dome pressure of 1045 psig.

The analysis was performed using the computer-simulated model ODYN described in General Electric Topical Report NEDO-24154. The ODYN code has been reviewed by the staff and found acceptable. (Safety Evaluation for the General Electric Topical Report-Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors, NEDO-24154 and NEDE-24154-P Volumes I, II, III, June 1980).

Standard Review Plan Chapter 5.2.2., (Rev-2), Section II-2A(iii) states (for overpressurization analysis) "The reactor scram is initiated by the second safety grade signal from the reactor protection system." The applicant has based the sizing of the safety/relief valves on the initiation of a reactor scram by the high-neutron flux scram which is the second safety grade scram signal from the reactor protection system following MSIV closure. We believe that the qualification and redundancy of reactor protection system equipment coupled with the fact that the reactor vessel pressure is limited to 110 percent of design pressure provides adequate assurance that the reactor vessel

integrity will be maintained for the limiting transient event. Accordingly, we find use of the high-neutron flux scram to be acceptable for RBS.

A sensitivity study was performed for a BWR/3 to investigate the effects of a higher initial reactor pressure and recirculation pump trip on the results of the overpressure protection analysis. This analysis showed that increasing the initial operating pressure results in an increase in resultant peak system pressure which is less than half the initial pressure increase. For RBS, the proposed technical specification limit on the high reactor pressure scram is 1095 psig. Therefore, since the vessel dome pressure used in the overpressurization analysis was 1045 psig, the maximum increase in the initial pressure would be limited to 50 psi, and the maximum peak system pressure increase during the overpressure design transient would be less than 25 psi. These results indicate that considerable margin is available to RBS before reaching the code limit and that GDC 15 will be satisfied even if increased initial dome pressure and recirculation pump trip are considered. However, since RBS specific overpressure analysis (as well as all other Chapter 15 transient analyses) were performed assuming an initial dome pressure of 1045 pounds per square inch gauge, it is the staff position that the applicants technical specifications should include an operating pressure limit of 1045 psig for the power operation and startup modes.

The performance of essentially all types of safety/relief valves has been less than expected for a safety component. Because of reportable events involving malfunctions of these valves on operating BWRs, the staff is of the opinion that significantly better safety/relief valve performance should be required of new plants. We require a detailed description of improvements between the RBS plant and presently operating plants in the areas listed below. In addition, we require that the applicant explain why the noted differences will provide the required performance improvement.

(1) Valve and valve operator type and/or design
(2) Environmental specifications
(3) Testing

(4) Quality assurance
(5) Valve operability
(6) Valve inspection and overhaul

TMI Action Plan Item II.K.3.28 identified the need to assure that air or nitrogen accumulators for the ADS valves are provided with sufficient capacity to cycle the valves open a sufficient number of times at design pressures. Assurance must also be provided that the long-term air supply is designed to withstand a hostile environment and still perform its function 100 days after an accident. Finally, it should be verified that no single active failure can disable the long-term air supply (LRG II issue 8 RSB). The safety evaluation of Item II.K.3.28 is given in Section 22 of this SER.

Our review of Section 5.2.2 has been performed in compliance with the guidelines of the revised Standard Review Plan (SRP), NUREG-0800, with the following exceptions:

SRP Section 5.2.2 states that the acceptance criteria are, in part, based on General Design Criterion 31, as it relates to the fracture behavior of the reactor coolant pressure boundary. This review area is addressed in Section 5.3.2 of this report.

The SRP states that for relief valve sizing analyses, all components, instrumentation, and controls function normally. In actuality, some of the transients analyzed to confirm adequate relief valve sizing assume system failure, e.g., turbine trip and generator load rejection with failure of the turbine bypass system. In these cases, our review exceeds the SRP guidelines.

The SRP states that the proposed plant technical specifications are reviewed. We have not yet reviewed all of the plant technical specifications since they are not available. A review will be performed when the completed specifications are submitted by the applicant prior to fuel loading. A detailed review of all of the specifications will probably not be performed since it is expected that approved standard technical specifications will be used at RBS.

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Subject to the resolution of open items discussed above, we conclude that the pressure relief system in conjunction with the reactor protection system will provide adequate protection against overpressurization of the reactor coolant pressure boundary, is in conformance with the aforementioned Commission's regulations, applicable regulatory guides, and industry standards, and is therefore acceptable.

5.2.3 Reactor Vessel Materials
5.2.4 Reactor Coolant Pressure Boundary
Inservice Inspection and Testing

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5.2.5 Reactor Coolant Pressure Boundary (RCPB) Leakage Detection

The reactor coolant pressure boundary (RCPB) leakage detection systems were reviewed in accordance with Section 5.2.5 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed, according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the RCPB leakage detection system with respect to the applicable regulations of 10 CFR 50.

A limited amount of leakage is to be expected from components forming the reactor coolant pressure boundary. Means are provided for detecting and identifying this leakage in accordance with the requirements of General Design Criterion 30, "Quality of Reactor Coolant Pressure Boundary." Leakage is classified into two types - identified and unidentified. Components such as valve stem packing, pump shaft seals, and flanges are not completely leaktight; this leakage is expected. Unidentified leakage may be symptomatic of an unexpected

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failure of the reactor coolant pressure boundary, and therefore, is considered separately from leakage which is expected (identified) and is carefully measured and monitored. Both identified and unidentified leakage are kept separate by directing each to a different system designed to maintain this separation, in accordance with the guidelines of Position C.1 of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems." Sensitivity of detection methods for unidentified leakage (1 gpm) and leakage limits (5 gpm) are selected to detect a potential through-wall flaw (crack) in the reactor coolant pressure boundary before such a crack can grow sufficiently to threaten the safety of the plant and its environment.

Concern with reactor coolant system pressure boundary leakage involves three main areas; leakage within containment, intersystem leakage and leakage outside of containment which are discussed below. A fourth section, also discussed below, covers testing of leak detection methods.

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~~These are discussed below:~~

5.2.5.1

Leakage within containment

a. Identified leakage

Within the drywell, the drywell equipment drain sump collects leakage from several potential sources, as follows:

(1) Reactor recirculation pump seal drain line,

(2) Valve stem leakoff drain lines,

(3) Reactor head seal drain line.

Leakage from these sources is monitored by means of different parameters. The recirculation pump leakage is monitored by drain line flow rate, valve stem leakage is monitored by means of drain line temperatures, and the head seal leakage is monitored by means of drain line pressure. All of these parameters with the exception of recirculation pump seal leakage flow rate are monitored in the control room; all of these (including pump seal leakage) will activate an alarm in the control room upon detection of leakage from the components.

Outside of the drywell, but still within containment, identified leakage is collected by the containment equipment drain sump.

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Increases in the fillup and or drainout times for either the drywell equipment drain sump or the containment equipment drain sump will be used to detect an increase in identified leakage with a sensitivity of 50% of the normal background identified leakage. Each sump system is provided with an alarm in the control room which is designed to operate when the leakage rate reaches 25 gpm. In addition, the temperature in the discharge line from each safety/relief valve (SRV) to the suppression pool is monitored in the control room to detect SRV leakage; an alarm in the control room is activated upon detection of a leaking SRV. This meets the guidelines of Position C.7 of Regulatory Guide 1.45.

Leakage from the upper containment pool liner is monitored manually - by opening individual drain line valves and looking for leakage. If there is leakage it will be collected in the drywell equipment drain sump where it will be measured as part of the total.

Leakage from the upper pool bellows seals is monitored by a flow transmitter located on the upper pool drain line. Indication of flowrate is provided in the main

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control room with an alarm to indicate an excess flow rate.

b. Unidentified leakage

Unidentified leakage in the drywell is detected by the following:

(1) Drywell floor drain sump,

(2) Pedestal floor drain sump,

(3) Drywell coolers,

(4) Airborne gaseous and particulate radioactivity monitor.

The applicant stated that the unidentified leakage rate is detected by the rate at which either sump fills up, by the increase in flow rate of condensate from the drywell cooler, or by the increase in airborne gaseous and particulate radioactivity. The use of these methods (drywell floor drain sump, pedestal floor drain sump, drywell coolers, and airborne gaseous and particulate radioactivity monitor) complies with the guidelines of Position C.3 of Regulatory Guide 1.45 regarding number of type of leakage detection methods. In addition, these systems can reasonably be expected to

function following seismic events which do not shut the plant down. The airborne gaseous and particulate radiation monitor is powered from a 1E power source and is designed to seismic Category standards, in order to ~~ascertain~~ assure functioning following an SSE. This complies with the guidelines of Position C.6 of Regulatory Guide 1.45.

The FSAR states that "---the sensitivity of these primary detection methods for unidentified leakage within the drywell is 1 gpm within one hour---." The applicant also states that an alarm is activated in the control room when the unidentified leak rate reaches 5 gpm, as measured by the foregoing method(s), thus complying with the requirements of Position C.7 of Regulatory Guide 1.45.

Unidentified leakage within containment collected by the drywell and pedestal floor drain sumps includes leakage from the control rod drives (CRDs), from valve flanges, from the component cooling system, from the service water system and any leakage not connected to the equipment drain sump.

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A process line break within containment will be detected by a lowering of the water level inside the reactor vessel and/or indication of high drywell pressure; for a steam line break downstream of the steam flow measuring devices the break will be detected by the increase in steam flow rate. These breaks may also be detected by the temperature monitor in the drywell. These methods may be utilized for detection of gross leakage - other methods discussed above are used to detect small leaks. These measurements all have indicators or readouts in the control room with alarms or annunciators to indicate when the particular parameter measured is out of normal operating limits.

Unidentified leakage within containment but outside of the drywell will be detected by increases in the containment floor drain sump fillup and pump-out time. The FSAR states that the sensitivity of measurement is 50% of the background leak rate and that an alarm will be activated when the total leakage rate reaches 5 gpm. (The applicant must provide assurance that use of this sump will

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provide an accuracy of leak rate measurement of 1 gpm or better and the capability to detect a leakage rate of 1 gpm in less than one hour in accordance with the provisions of Positions C.2 and C.5 of Regulatory Guide 1.45.)

The applicant reported that leakage from thermally hot sources within the drywell is piped to a common header and then through the drywell equipment drain cooler to condense the steam contained in this fluid stream. The cooled fluid is then routed to the drywell equipment drain sump to be included in the measurement of unidentified leakage within the drywell.

The applicant noted that periodic testing will be conducted to detect blocked lines in the floor drain system as part of the preventative maintenance program.

5.2.5.2

Intersystem Leakage

Detection of leakage into systems connected to the reactor coolant system and damage to those systems

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could result from intersystem leakage. The systems of concern are:

a. High pressure core spray (HPCS) system,

b. Low pressure core spray (LPCS) system,

c. Low pressure coolant injection (LPCI) system (which also serves as the RHR system),

d. Reactor core isolation coolant system (both water and steam turbine sides),

e. Secondary side of reactor water cleanup (RWCU) system heat exchangers (Reactor plant component cooling water (RPCCW) system),

f. Secondary side of residual heat removal (RHR) heat exchangers (Service water system).

A radiation monitor in the service water effluent from the two RHR heat exchangers detects leakage from the RHR system. A radiation monitor in the common reactor plant component cooling water (RPCCW) return line detects leakage from the two RWCU heat exchangers, from the seal coolers for the RHR pumps and from the seal coolers for the recirculation pumps into the RPCCW. In addition, the applicant noted that leakage from the LPCS, HPCS,

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RCIC and RHR systems outside containment is detected by a combination of methods which include high area temperature, high area radiation, high sump level and water level in the reactor vessel. (The applicant must also show how leakage from the primary system into the HPCS, LPCS, RCIC, (both water and steam sides) and RHR systems is detected when these systems are supposedly isolated from the primary system. This assurance is needed to show that the design meets the guidelines of Position C.4 of Regulatory Guide 1.45.)

The plant has another means of detecting leakage from the RWCU system. The flow to the RWCU system from the reactor vessel can be compared with that returning to the reactor vessel from the RWCU system. A high differential flow is alarmed in the control room.

5.2.5.3 Leakage Outside of Containment

a. Identified Leakage

Steam flow to the RHR system (when in the steam condensing mode) and to the RCIC turbine is monitored. Abnormal flows exceeding preset

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values result in annunciation in the control room and isolation of the steam line(s) involved. Thermocouples are installed in equipment areas and in inlet and outlet ventilation ducts of equipment rooms for the RHR, RCIC and RWCU systems. High ambient temperatures and high differential temperatures are alarmed in the main control room. This is also true for the main steam tunnel, both inside and outside of containment, i.e., temperature monitors with high temperature and high differential temperature alarms are used to detect leakage. The high temperature/high differential temperatures also cause isolation valves in that particular area to close.

b. Unidentified Leakage

Sumps are provided in the rooms outside of containment in which equipment is located that is part of the reactor coolant pressure boundary. The sumps collect unidentified and identified leakage. Normal leakage to all sumps is identified during preoperational testing. Sump

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levels and sump pump operation are indicated in the control room and high sump level is alarmed. Operability is tested by continuously monitoring floor drain sump level.

The applicant stated that limiting conditions for leakage are 25 gpm for identified leakage and 5 gpm for unidentified leakage, thus complying partially with the guidelines of Position C.9 of Regulatory Guide 1.45. (In addition, the applicant must specify the availability of the leakage detection methods and the limiting conditions for operation based upon this availability, in order to comply fully with the guidelines for Position C.9.)

There are other sensors - temperature, pressure and radioactivity (gaseous and particulate) - within containment but outside of the drywell which appear to be intended for use or are capable of being used to detect leakage within containment outside of the drywell. No discussion of these sensors is contained in this

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section of the FSAR. (The applicant should revise the FSAR to include a discussion of these sensors in this section of the FSAR. The applicant should verify that all systems and measurements which are part of the reactor coolant pressure boundary leakage detection system are accounted for.)

5.2.5.4 Testing of Leak Detection Methods

The leak detection methods may be tested during operation for operability and calibration by the following methods:

a. Sump levels may be monitored and compared with flow rates of fluids into the sump,
b. Methods may be compared,
c. Channels may be compared when more than one channel is used for any one detection method,
d. Simulated signals may be used to trip monitors designed for that purpose.

These provisions for testing for operability and calibration comply with the guidelines of Position C.8 of Regulatory Guide 1.45.

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The Technical Specifications do not permit any operation with pressure boundary leakage. If this occurs, the plant will have to be in at least hot shutdown within 12 hours and in cold shutdown within the next 24 hours.

Leakage in excess of 1 gpm through an isolation valve with the plant at operating pressure requires that the plant isolate the affected portion of the high-pressure system from the low-pressure portion within 4 hours or that the plant be in at least hot shutdown within the next 12 hours and cold shutdown within the following 24 hours, as noted in the Standard Technical Specifications.

In the event of an increase in unidentified leakage of 2 gpm (or greater) within 4 hours, the operator must identify the leakage increase source as not service-sensitive type 304 or 316 austenitic stainless steel within 4 hours. Otherwise, the plant must proceed to at least hot shutdown within 12 hours and to cold shutdown within the following 24 hours, as noted in the Standard Technical Specifications.

Based on the above, the staff concludes that the leakage detection systems provided to detect leakage from components of the RCPB furnish reasonable assurance that structural degradation that may develop in pressure-retaining components of the RCPB and result in coolant leakage will be detected on a timely basis so that corrective actions can be taken before such degradation could become sufficiently severe to jeopardize the safety of the system, or before the leakage could increase to a level beyond the capability of the makeup system to replenish the loss.

The staff has noted that there are other sensors--temperature, pressure, and radioactivity (gaseous and particulate)--within containment but outside of the drywell that appear to be intended for use or capable of being used to detect leakage within containment outside of the drywell. Yet, no discussion of these sensors is contained in Section 5.2.5 of the FSAR. The applicant should correct the FSAR to include a discussion of these sensors in Section 5.2.5. The applicant should verify that all systems and measurements that are part of the RCPB leakage detection system are accounted for.

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5.2.5.5 Summary and Conclusions

The airborne gaseous and particulate radiation monitor is designed to seismic Category I standards and is powered from a 1E power source, thus is capable of operating following an SSE. This meets the guidelines of Position C.6 of Regulatory Guide 1.45 and Position C.1 of Regulatory Guide 1.29, "Seismic Design Criteria." Failure of any of the other systems used to detect RCPB leakage will not affect the airborne gaseous and particulate radiation moritor, thus satisfying the guidelines of Position C.2 of Regulatory Guide 1.29. Therefore, by meeting the guidelines of Positions C.1 and C.2 of Regulatory Guide 1.29, RCPB leakage detection complies with the requirements of General Design Criterion 2, "Design Basis for Protection Against Natural Phenomena." The methods used for RCPB leakage detection meet the guidelines of Positions C.1, C.3, C.6, C.7, and C.8 of Regulatory Guide 1.45; further information must be provided to show that the RCPB leakage detection methods comply with Positions C.2, C.4, C.5, and C.9 of Regulatory Guide 1.45. Thus, the system does not comply with the criteria of General Design Criteria 30, "Quality of Reactor Coolant Pressure Boundary."

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Without further information, as noted above, we cannot conclude that reactor coolant pressure boundary leakage detection is acceptable. The reactor coolant pressure boundary leakage detection does not comply with the acceptance criteria of SRP Section 5.2.5.

The SRP states that the proposed plant Technical Specifications are to be reviewed. The staff has not yet reviewed all of the Technical Specifications because they are not available. A review will be performed when the completed specifications are submitted by the applicant before fuel loading. However, a detailed review of all of the specifications will probably not be performed because the staff expects the applicant to use approved Standard Technical Specifications.

Subject to the resolution of open items discussed above, the staff concludes that the pressure relief system, in conjunction with the reactor protection system, will provide adequate protection against overpressurization of the RCPB and is in conformance with the aforementioned Commission regulations, applicable RGs and industry standards. It is therefore acceptable.

5.3 Reactor vessel

5.3.1 Reactor Vessel Materials

~~5.2.3 Reactor Coolant Pressure Boundary Materials~~

The fracture toughness of ferritic reactor vessel and RCPB materials, and the materials surveillance program for the reactor vessel beltline have been reviewed according to the acceptance criteria and references in Paragraphs II.5, II.6, and II.7 (Appendices G and H, 10 CFR 50) of SRP 5.3.1 (NUREG-0800).

GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," requires, in part, that the RCPB be designed with sufficient margin to ensure that, when stressed under operating, maintenance, and test conditions, the boundary behaves in a nonbrittle manner and the probability of rapidly propagating fracture is minimized. GDC 32, "Inspection of Reactor Coolant Pressure Boundary," requires, in part, that the RCPB be designed to permit an appropriate materials surveillance program for the reactor pressure vessel. The fracture toughness requirements for the ferritic materials of the RCPB are defined in Appendices G, "Fracture Toughness Requirements," and H, "Reactor Vessel Material Surveillance Requirements," of 10 CFR 50. The staff has reviewed the materials selection, toughness requirements, and extent of materials testing conducted by the applicant to provide assurance that the ferritic materials used for pressure-retaining components of the RCPB have adequate toughness under operating, maintenance, and testing conditions.

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Compliance with 10 CFR 50.55a

The River Bend Unit 1 reactor vessel, piping, pumps, valves, and steam generator were designed to meet the 1971 edition of the ASME Code, Section III, "Rules for Construction of Nuclear Power Plant Components," including Addenda through Winter 1972. Based on the plant's March 25, 1977 CP date, 10 CFR 50.55a requires that the ASME Code Edition and Addenda applied to the above reactor coolant system components be no earlier than those of the 1971 Edition, Summer 1972 Addenda. The design and construction of the River Bend RCPB components are, therefore, in compliance with the requirements of 10 CFR 50.55a.

Compliance with Appendix G, 10 CFR 50

The staff evaluation of the FSAR to determine the degree of compliance with the fracture toughness requirements of Appendix G, 10 CFR 50, indicates that the applicant meets all the requirements of this appendix with the exception of Paragraph IV.A.2c.

Paragraph IV.A.2.c of Appendix G requires, in part, that when the core is critical (other than for the purpose of low-level physics tests), the temperature of the reactor vessel shall be no less than the minimum permissible temperature for inservice system hydrostatic pressure test nor less than 40° F above the temperature required by Paragraph IV.A.2.a. The applicant has requested permission to operate the core when critical according to a proposed modification to Paragraph IV.A.2.c of Appendix G, 10 CFR 50. GE has proposed that 10 CFR 50 be amended to make this procedure acceptable in the regulation.

The proposed modification to Paragraph IV.A.2.c is described in GE Topical Report NEDO-21778-A. As noted in the November 13, 1978 letter from O. D. Parr, NRC, to Dr. G. G. Sherwood, GE, the staff has reviewed this report, has found it acceptable, and concurs that the proposed alternative to the criticality hydrostatic temperature limit is acceptable. The staff's previous evaluation and acceptance of the GE topical report provides sufficient information for the staff to conclude that the proposed alternate method is

adequate and that an exemption to Paragraph IV.A.2.c of Appendix G is justified.

Compliance with Appendix H, 10 CFR 50

The materials surveillance program at River Bend will be used to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region resulting from exposure to neutron irradiation and the thermal environment, as required by GDC 32. Under the River Bend surveillance program, fracture toughness data must be obtained from materials specimens that are representative of the limiting base, weld, and heat-affected zone (HAZ) materials in the beltline region. These data will permit the determination of the conditions under which the vessel can be operated with adequate margins of safety against fracture throughout its service life.

The fracture toughness properties of reactor vessel beltline materials must be monitored throughout service life by a materials surveillance program that meets the requirements of Appendix H of 10 CFR 50. The staff has evaluated the applicant's information for degree of compliance to these requirements. Based on its evaluation, the staff concludes that River Bend has met all the requirements of Appendix H, 10 CFR 50 with the exception of Paragraph II.B.

Paragraph II.B, Appendix H, to 10 CFR 50 requires that reactor vessels constructed of ferritic materials with a peak neutron fluence (E>1MeV) at the end of the design life of the vessel exceeding 10^{17} n/cm^2 shall have their beltline regions monitored by a surveillance program complying with the specifications of ASTM Standard E 185-73, except as modified by Appendix H. ASTM E 185 requires that material placed in the surveillance capsules represents the material that may limit operation of the reactor during its lifetime. The selection of the base metal, HAZ, and the weld metal surveillance specimens is based on consideration of the fracture toughness properties of all of the beltline materials (RT_{NDT} and upper shelf CVN energy) in the unirradiated condition, the chemical composition, and the neutron fluence to establish the limiting materials. The applicant has indicated that River Bend complies with all the requirements of Appendix H, 10 CFR 50,

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but has not submitted sufficient information for the staff to confirm that the reactor vessel materials surveillance program complies with the requirements of Appendix H, 10 CFR 50. The applicant must provide the following information:

(1) Identification of all materials in each surveillance capsule. For each base metal and HAZ specimen, identification of plate from which each sample was removed, and the heat treatment it received. For each weld metal surveillance specimen, the weld identification from which the sample was removed, the weld wire type and heat identification, flux type and lot identification, weld process and heat treatment used for fabrication of the weld sample.

(2) The lead factors for each capsule with respect to the vessel inner wall.

(3) A sketch that indicates the azimuthal location for each capsule relative to the reactor core and the location of each weld and base metal in the reactor vessel beltline region.

Without the data detailed above, the staff cannot complete its evaluation of Paragraph II.B, Appendix H, 10 CFR 50.

Conclusions for Compliance with Appendices G and H, 10 CFR 50

Based on its evaluation of compliance with Appendices G and H, 10 CFR 50, the staff concludes that the applicant has met all the fracture toughness requirements of these appendices except for Paragraph IV.A.2c of Appendix G and Paragraph II.B of Appendix H. Appendix G, "Protection Against Non-Ductile Failure," Section III of the ASME Code, will be used, together with the fracture toughness test results required by Appendices G and H, 10 CFR 50, to calculate the pressure-temperature limitations for the River Bend reactor vessel. The fracture toughness tests required by the ASME Code and by Appendix G of 10 CFR 50 provide reasonable assurance that adequate safety margins against the possibility of nonductile behavior or rapidly propagating fracture can be established for all pressure-retaining components of the RCPB.

The use of Appendix G, Section III of the ASME Code as a guide in establishing safe operating procedures and use of the results of the fracture toughness tests performed in accordance with the ASME Code and NRC regulations will provide adequate safety margins during operating, testing, and maintenance conditions. Compliance with these Code provisions and NRC regulations constitutes an acceptable basis for satisfying the requirements of GDC 31. The materials surveillance program required by Appendix H, 10 CFR 50 will provide information on material properties and the effects of irradiation of material properties so that changes in the fracture toughness of the material in the reactor vessel beltline caused by exposure to neutron radiation can be properly assessed, and adequate safety margins against the possibility of vessel failure can be provided. Compliance with Appendix H, 10 CFR 50 ensures that the surveillance program constitutes an acceptable basis for monitoring radiation-induced changes in the fracture toughness of the reactor vessel material and satisfies the requirements of GDC 32. This review was done with the assistance of the staff of EG&G, Idaho National Engineering Laboratory.

5.2.5 Reactor Coolant Pressure Boundary Leakage Detection

A limited amount of leakage is to be expected from components forming the RCPB. Means are provided for detecting and identifying this leakage in accordance with the requirements of GDC 30, "Quality of Reactor Coolant Pressure Boundary." Leakage is classified into two types, identified and unidentified. Components such as valve stem packing, pump shaft seals, and flanges are not completely leaktight; this leakage is expected. Unidentified leakage may be symptomatic of an unexpected failure of the RCPB. Therefore, it is considered separately from leakage that is expected (identified) and is carefully measured and monitored. Both identified and unidentified leakage are kept separate by directing each to a different system designed to maintain this separation, as much as practicable, in accordance with the guidelines of RG 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems." Some of the systems used to detect leakage also measure the leak rate. Sensitivity of detection methods for unidentified leakage (1 gpm) and leakage limits (5 gpm) are selected to detect a potential through-wall flaw (crack) in the RCPB

The applicant must provide assurance that radiation monitoring systems have a radioactive ("check") source built into the system to permit operability testing and calibration during operation, as required by SRP 5.2.5.

Until the applicant adequately resolves the issues raised above, the staff cannot find that the systems are in compliance with the guidance of RG 1.45 and in compliance with the requirements of GDC 2 and 30. The staff concludes, therefore, that the applicant's submittal regarding RCPB leakage detection is not acceptable.

5.3 Reactor Vessel

5.3.1 Reactor Vessel Materials

5.3.2 Pressure-Temperature Limits

The applicant's pressure-temperature limits for operation of the reactor vessel have been reviewed according to SRP 5.3.1 (NUREG-0800).

Appendices G and H of 10 CFR 50 describe the conditions that require pressure-temperature limits and provide the general bases for these limits. These appendices specifically require that pressure-temperature limits must provide safety margins at least as great as those recommended in the ASME Code, Section III, Appendix G. Appendix G, 10 CFR 50, requires additional safety margins whenever the reactor core is critical, except for low-level physics tests. The following pressure-temperature limits imposed on the RCPB during operation and tests are reviewed to ensure that they provide adequate safety margins against nonductile behavior or rapidly propagating failure of ferritic components, as required by GDC 31:

(1) preservice hydrostatic tests

(2) inservice leak and hydrostatic tests

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(3) heatup and cooldown operations

(4) core operation

The applicant has presented (FSAR Figure 5.3.4) heat up and cooldown pressure-temperature curves for 32 effective full power years (EFPY). The staff has evaluated the curves and found them acceptable for 32 EFPY.

The pressure-temperature limits to be imposed on the reactor coolant system for all operating and testing conditions to ensure adequate safety margins against nonductile or rapidly propagating failure must be in conformance with established criteria, codes, and standards. The use of operating limits based on these criteria, as defined by applicable regulations, codes, and standards, will provide reasonable assurance that nonductile or rapidly propagating failure will not occur, and will constitute an acceptable basis for satisfying the applicable requirements of GDC 31. This review was done with the assistance of EG&G, Idaho National Engineering Laboratory personnel.

5.3.3 Reactor Vessel Integrity

The staff has reviewed the FSAR sections related to reactor vessel integrity. Although most areas are reviewed separately in accordance with other review plans, reactor vessel integrity is of such importance that a special summary review of all factors relating to reactor vessel integrity is warranted.

The fracture toughness of ferritic reactor vessel and RCPB materials, the pressure-temperature limits for operation of the reactor vessels, and the materials surveillance program for the reactor vessel beltline have been reviewed according to Paragraphs II.2, II.6, and II.7 (Appendices G and H, 10 CFR 50) of SRP 5.3.3 (NUREG-0800).

The staff has reviewed the factors contributing to the structural integrity of the reactor vessel and concludes that the applicant has fully complied with the required regulations, codes, and standards except for Paragraph IV.A.2c of Appendix G and Paragraph II.B of Appendix H. The applicant has indicated that

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5.4 Component and Subsystem Design

5.4.1 Reactor Core Isolation Cooling System

The reactor core isolation cooling (RCIC) system is a high-pressure reactor coolant makeup system that will operate independently of alternating current power supply. The system provides sufficient water to the reactor vessel to cool the core and to maintain the reactor in a standby condition if the vessel becomes isolated from the main condenser and experiences a loss of feedwater flow. The system is also designed to permit complete plant shutdown under conditions of loss of normal feedwater flow by maintaining the necessary reactor water inventory until the vessel is depressurized to the point where the RHR system can function in the shutdown cooling mode.

The RCIC system consists of a steam-driven turbine-pump unit and associated valves and piping capable of delivering makeup water to the reactor through a separate nozzle at the top of the reactor vessel. Fluid removed from the reactor vessel following a shutdown from power operation is normally made up by the feedwater system and supplemented by inleakage from the control rod drive system. If the feedwater system is inoperable, the RCIC system starts automatically when the water level in the reactor vessel reaches the level two (L2) trip setpoint or is started by the operator from the control room. The system is capable of delivering rated flow within 30 seconds of initiation. Primary water supply for the RCIC system comes from the condensate storage tank with a secondary supply from the suppression pool.

The RCIC system was compared to designs and capacities of similar plant systems via comparison tables in Section 1.3 of the final safety analysis report (FSAR), and no unexplained departures from previously reviewed plants were determined.

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RCIC design operating parameters are consistent with expected operational modes as noted in Figures 5.4-9 "RCIC Process Diagram" of the FSAR and this complies with the requirements of GDC 34 regarding residual heat removal. Essential components of the RCIC system are designated seismic Category I in accordance with Regulatory Guide 1.29 and Quality Group B in accordance with Regulatory Guide 1.26 as discussed in Section 3.2 of this report. Preoperational and initial test program are discussed in Section 14 of this report.

The RCIC system is housed within the auxiliary building which provides protection against wind, tornadoes, floods, and other weather phenomena. Compliance with the requirements of Criterion 2 of the General Design Criteria in this regard is discussed in Section 3.8 of this report. Since the condensate storage tank, which is the normal source of water for this system, is not a seismic Category I structure an automatic safety-grade suction switchover to the suppression pool has been provided to ensure a water supply in the event of a safe shutdown earthquake and concomitant failure of the condensate storage tank.

In addition, the system is protected against pipe whip inside and outside containment as required by Criterion 4 of the General Design Criteria, as discussed in Section 3.6 of this report.

The high pressure core spray and reactor core isolation cooling systems are located in different corners of the auxiliary building for additional protection against common mode failures. They use different energy sources for pump motivation (steam turbine for reactor core isolation cooling pump, electric power for high pressure core spray pump) and different power systems for control power. This diversity conforms to the requirements of Section 5.4.6 of the Standard Review Plan.

To protect the RCIC pump from overheating, the reactor core isolation cooling system contains a miniflow line which discharges into the suppression pool when the line to the reactor vessel is isolated. When sufficient flow to the vessel is achieved, a valve in the miniflow line automatically closes thus directing all flow to the reactor.

The available net positive suction head for the RCIC pump should have adequate margin to prevent cavitation and assure pump operability in accordance with Regulatory Guide 1.1 "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps." The RCIC system meets the guidelines of Regulatory Guide 1.1.

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A high point vent is provided and the system will be checked at least once every 31 days to assure that the pump discharge lines are filled. The reactor core isolation cooling system includes a full flow test line with water return to the condensate storage tank for periodic testing. Technical specifications will include a flow test at least every 92 days and a system functional test at least every 18 months with simulated automatic actuation and verification of proper automatic valve position. In both tests verification is obtained that the reactor core isolation cooling pump will develop a minimum flow of 600 gallons per minute.

Isolation between the reactor coolant system and the reactor core isolation cooling system is provided by: (1) two check valves and a closed DC powered valve in the reactor core isolation cooling system discharge line, and (2) two normally open AC powered valves in the steam line to the reactor core isolation cooling steam turbine. We require that the motor operated valves be classified category A and the check valves category A/C in accordance with the provisions of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code and that these valves be leak tested periodically. Specific testing requirements for these valves are discussed in Section 3 of this report.

High ambient temperature in the equipment and pipe area will initiate automatic isolation of the system. Spurious ambient temperature signals from in and around the system should not prevent the system from operating when needed. We require information from the applicant to verify that the high temperature trip setpoints are properly set.

The suction piping of the RCIC system is designed for low pressure. A relief valve is, therefore, provided to protect against overpressurization of the

line. Rupture discs in the steam turbine exhaust line are used to protect the turbine casing from overpressurization. To protect the check valve in the turbine exhaust line, a sparger is installed in the line to suppress flow-induced oscillations due to steam bubble collapse in the suppression pool.

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Items II.K.3.13 and II.K.3.15 of the Action Plan (NUREG-0737) address restart capability of the RCIC on low water level, the potential for separating the RCIC and HPCS initiation levels, and the prevention of inadvertent RCIC isolation or trip due to spurious signals. The safety evaluation of these items is given in section 22 of this SER.

The reactor core isolation cooling system is capable of supplying coolant to the reactor following feedwater isolation and reactor shutdown under normal and accident conditions. The reactor core isolation cooling system conforms to the requirements of General Design Criterion 5 in that the RCIC System is not shared between units; Criterion 29 (in conjunction with the HPCS System) through quality controlled construction and periodic testing; Criterion 33 (in conjunction with HPCS) in that operation with only offsite or only onsite power is possible for protection against small breaks; Criterion 34 (in conjunction with HPCS) in that residual heat removal while still at high pressure can be accomplished assuming a single failure and with or without offsite power. Conformance with Criterion 54 with regard to leak detection and isolation capability on piping penetrating containment is discussed in Section 6.2. Our review of Section 5.4.6 has been performed in compliance with the Guidelines of revised SRP, NUREG-0800, with the following exception:

The SRP states that the proposed plant technical specifications are reviewed. We have not yet reviewed all of the plant technical specifications since they are not available. A review will be performed when the completed specifications are submitted by the applicant prior to fuel loading. A detailed review of all of the specifications will probably not be performed since it is expected that approved standard technical specifications will be used at RBS.

The drawings, component descriptions, and design criteria for the reactor core isolation cooling system have been reviewed. We conclude that the design of

the reactor core isolation cooling system conforms to the Commission's regulations and to the applicable regulatory guides, and is acceptable, subject to resolution of the open issues noted or referenced above.

5.4.2 Residual Heat Removal System

The residual heat removal (RHR) system comprises three independent loops; each loop contains its own motor-driven pump, piping, valves, instrumention, and controls. Loops A, B, and C have a suction source from the suppression pool and each is capable of discharging water to the reactor vessel via a separate nozzle or back to the suppression pool via a full flow test line. In addition, the A and B loops have heat exchangers which are cooled by the standby service water system. They can also take suction from the reactor recirculation system or fuel pool, and can discharge into the reactor via the feedwater line, or into the fuel pool. Loops A and B also have connections to reactor steam via the reactor core isolation cooling (RCIC) steam line and can discharge condensate to the RCIC pump suction or to the suppression pool.

The RHR system is used in conjunction with the main steam and feedwater systems (main condenser), or with the RCIC system in conjunction with the safety/relief valves to cool down the reactor coolant system following shutdown. Interlocks are provided on motor operated valves which interface between the reactor coolant system and the RHR suction piping for protection against overpressurization. In the event the RHR isolation valves are inoperative, an alternative shutdown method is used. In this method, water is pumped from the suppression pool through the RHR heat exchangers and into the reactor vessel. The vessel water is allowed to overflow the steam lines and discharge back to the suppression pool via the discharge lines of those safety relief valves which are part of Automatic Depressurization System.

The RHR system operates in four different modes:

(1) Shutdown cooling,

(2) Steam condensing,

(3) Suppression pool cooling, and

(4) Low pressure coolant injection.

RBS does not have a containment spray mode for RHR. Containment spray cooling criteria is satisfied by plant ventilation system.

All four modes of operation use the same hardware. Only the shutdown cooling and the steam condensing modes are covered in this section. Modes (3) and (4) are reviewed through other sections of the Standard Review Plan (6.2 and 6.3).

The normal operational mode of the residual heat removal system is the shutdown cooling mode, which is used to remove decay heat from the reactor core to achieve and maintain a cold shutdown condition. The steam condensing mode is used to condense steam while the reactor is isolated from the main condenser and vessel level is being maintained by the reactor core isolation cooling system. The heat removed in the RHR heat exhangers is transported to the ultimate heat sink by the standby service water system.

The RHR system was compared to designs and capacities of such systems in similar plants and no unexplained departures from previously reviewed plants were found. The RHR system is designed to operate, with or without offsite power, with a single active failure. Control of the RHR system is accomplished from the control room. Using the system process diagrams, P&IDs, system safety analyses, and component performance specifications, it was determined that the system provided at RBS has the capacity to bring the reactor to cold shutdown conditions in a reasonable period of time, assuming operation of only safety-grade equipment.

Isolation between the RHR suction line from the reactor coolant system recirculation loop is provided by an inside containment isolation valve and an outside containment isolation valve. Each valve is interlocked with a separate switch which prohibits opening of the associated valve if the recirculation

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loop pressure exceeds the shutdown range. These same interlocks initiate valve closure on increasing reactor pressure. A single operator error cannot open either valve at a pressure above the shutdown range. Isolation between the RHR shutdown return to the reactor coolant system feedwater line and the RHR connection to the RCIC head spray both have a check valve and globe valve at the pressure boundary. The globe valve has a pressure interlock which prevents opening of the valve due to operator error when the higher pressure system exceeds the shutdown range. This same interlock initiates valve closure on increasing reactor pressure. We require additional information from the applicant to confirm that RHR system complies with the isolation requirements given in Branch Technical Position RSB 5-1 of SRP 5.4.7.

The low pressure coolant injection mode pressure isolation provisions are discussed in Section 6.3. The containment isolation requirements of the residual heat removal system are discussed in Section 6.2 of this report.

We require that the motor-operated valves be classified Category A and the check valves Category A/C in accordance with Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code and that they be tested periodically. Specific testing requirements for these valves are discussed in Section 3 of this report.

The residual heat removal system is designed to the seismic Category I requirements of Regulatory Guide 1.29 as discussed in Section 3.2 of this report. It is housed in the auxiliary building for protection against the effects of flooding, tornadoes, hurricanes, and other natural phenomena. Compliance with Criterion 2 of the General Design Criteria in this regard is discussed in Section 3.8 of this report. Compliance with Regulatory Guide 1.26 regarding Quality Group classifications is discussed in Section 3.2 of this report. The containment isolation requirements of Criteria 55, 56, and 57 of the General Design Criteria are discussed in Section 6.2 of this report. Systems used for cooling the residual heat removal system conform to the requirements of Criteria 44, 45, and 46 of the General Design Criteria, as discussed in Section 9.2 of this report. Those portions of the residual heat removal system which are also part of the emergency core cooling system are designed to

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operate under both normal and accident conditions. The system is protected against missiles (discussed in Section 3._ of this report) and pipe whip (discussed in Section 3.6 of this report). In this way, the residual heat removal system complies with the requirements of Criterion 4 of the General Design Criteria.

There is only a single line from the recirculation system to the residual heat removal system for use in cooling the reactor in the shutdown mode. This line is vulnerable to a single failure of either of the isolation valves. The applicant has an alternate cooling path using the safety valves and suppression pool cooling in the event of a failure in the suction line which would preclude residual heat removal system operation. Both paths are provided with emergency power supplies. To assure the long term operability of the automatic depressurization system (ADS) valves for the alternate shutdown cooling mode, accumulators, and a safety grade air charging system (Penetration Valve Leakage Control System PVLCS) have been provided to recharge the ADS valve accumulators.

The accumulators and associated valves and all the interconnecting air system piping to the ADS valves are designed to requirements of ASME Section III, Class 2 and 3, as applicable, and are seismic Category 1. These alternate cooling provisions satisfy the single failure requirements of Criterion 34 of the General Design Criteria. For the alternate cooling mode to achieve cold shutdown requires that the safety/relief valves be qualified to pass water as well as steam. Branch technical position RSB-5-1 of the SRP 5.4.7 also requires testing of this alternate method. It is the staff position that the applicant must verify that the safety/relief valves are qualified to pass water (See Item II.D.1 in Chapter 22). We require additional information from the applicant to verify that sufficient flow capacity exists for all piping and valves required in this alternate shutdown path and also to show that the RHR pump head-flow requirements for the worst path resistance can be met.

The available NPSH for the RHR pumps should have adequate margin to prevent cavitation and assure pump operability in accordance with Regulatory Guide 1.1 "Net Positive Suction Head For Emergency Core Cooling and Containment Heat

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Removal System Pumps." We require additional information to confirm that adequate margin exists for RHR pumps in all modes of operation.

For the low pressure coolant injection mode, flow is diverted to a miniflow line when low flow is sensed in the injection line, with the water discharging to the suppression pool. The miniflow line is designed to prevent pump overheating when the valves in the injection line are closed because the reactor vessel pressure is too high to permit injection. The valves in the miniflow lines are closed automatically when flow in the injection line is sufficient, thus directing all flow to the reactor. Each train of the residual heat removal system is tested during normal plant operation by pumping water from the suppression pool back into the pool. We require that the low-pressure coolant injection mode operability be verified every 31 days; that every 92 days each pump be shown to start from the control room; and every 18 months that a system functional test be performed. This periodic testing is in conformance with the requirements of Criterion 37 of the General Design Criteria. The preoperational test program for the RHR system is discussed in Section 14 of this SER.

Our review of Section 5.4.7 has been performed in compliance with the guidelines of the revised Standard Review Plan (SRP), NUREG-0800, with the following exception:

The SRP states that the proposed plant technical specifications are reviewed. We have not yet reviewed all of the plant technical specifications since they are not available. A review will be performed when the completed specifications are submitted by the applicant prior to fuel loading. A detailed review of all of the specifications will probably not be performed since it is expected that approved standard technical specifications will be used at RBS.

The RHR system design has been compared with the functional, isolation, pressure relief, pump protection, and test requirements of Branch Technical Position RSB 5-1 "Design Requirements of the Residual Heat Removal System" and found to comply with the implementation criteria for RBS, subject to resolution of the open issues noted above.

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The drawings, component descriptions, and design criteria associated with the residual heat removal system have been reviewed. We conclude that the design of the residual heat removal system conforms to the Commission's regulations and to the applicable regulatory guides and is acceptable, subject to resolution of the open issues noted or referenced above.

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5.4.3 Reactor Water Cleanup System

The staff has reviewed the reactor water cleanup system in accordance with SRP 5.4.8 (NUREG-0800)

The reactor water cleanup system continuously removes solid and dissolved impurities from the reactor water through filter demineralizers. The system flow path is from the reactor vessel, through two regenerative and two non-regenerative heat exchangers, through two parallel filter demineralizers, and back to the reactor feedwater line. The limits of the conductivity, pH, and chloride concentration in the reactor coolant have been established in the Technical Specifications in accordance with the recommendations of RG 1.56, Revision 1. Conductivity will be continuously measured before startup and during power operation, hot standby, and cold shutdown to ensure that the limits will not be exceeded. High conductivity will be annunciated in the control room. Surveillance requirements and limiting conditions for operation are specified in the Technical Specifications for the chemistry of reactor coolant systems.

Spent resins are not regenerative and are sluiced from the filter demineralizer unit to a backwash receiving tank from which they are transferred to the radwaste system for processing and disposal. To prevent resins from entering the reactor recirculation system, a strainer is installed on each filter demineralizer unit. Each strainer and filter demineralizer vessel has a control room alarm that is energized by high differential pressure. In the event of low flow or loss of flow in the cleanup system, flow is maintained in each filter demineralizer by its own holding pump.

The suction line of the RCPB portion of the reactor water cleanup system contains two isolation valves that automatically close in response to signals from the reactor pressure vessel low water level leak detection system and actuation of the standby liquid control system. The outboard isolation valve will automatically close to prevent damage of the filter demineralizer resins if the outlet temperature of the nonregenerative heat exchanger is high. The nonregenerative heat exchanger is sized to maintain the process temperature required for the cleanup demineralizer resin when the cooling capacity of the regenerative heat exchanger is reduced when flow is partially bypassed to the main condenser.

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At the request of the staff, the applicant is reviewing the total capacity of new demineralizer resins; the method of determining the condition of the demineralizer units so that the ion exchange resin can be replaced before an unacceptable level of depletion is reached; the control room alarm setpoints of the conductivity meters that will indicate the noticeable breakthrough of one or more demineralizers, and corrective actions to be taken if the conductivity, pH, or chloride content exceed the established Technical Specification limits; and a summary description of the water chemistry monitoring program. The staff will evaluate these items in a supplement to the SER after the requested information is provided.

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6 ENGINEERED SAFETY FEATURES

6.1 Engineered Safety Features Materials

6.1.1 ~~Post-Accident Emergency Cooling Water Chemistry~~ Metallic Materials

6.1.2 Organic Materials

The organic coatings on mechanical equipments have been reviewed in accordance with SRP 6.1.2 (NUREG-0800).

This evaluation is conducted to verify that protective coatings applied inside containment meet the testing requirements of ANSI 101.2, "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities" (1972) and the quality assurance guidelines of RG 1.54. Compliance with these requirements provides assurance that the protective coatings will not fail under design-basis accident (DBA) conditions and generate quantities of solid debris or combustible gas sufficient to adversely affect the engineered safety features.

In FSAR Section 6.1.2, the applicant indicates that the coating systems used on exposed surfaces inside the containment have been qualified in accordance with American National Standard Institute (ANSI) Standard N101.2, (1972) and ANSI 5.12, "Protective Coatings (Paints) for the Nuclear Industry" (1974). The applicant also stated that the protective coating system for the containment are applied in accordance with RG 1.54.

At the request of the staff, the applicant is reviewing the total amount of unqualified protective coatings and organic materials used inside containment. The staff will provide an evaluation in the SER concerning the acceptability of the unqualified materials exposed to DBA conditions after the requested information is provided.

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6.2 Containment Systems

The containment systems for the River Bend station include a Mark III type containment structure as the primary containment and a secondary containment surrounding the primary containment. The secondary containment is designed to confine the leakage of airborne radioactive materials from the primary containment until it can be processed and released.

The staff has reviewed the applicant's design, design criteria, and design bases for the containment systems. The review guidelines and acceptance criteria used as the basis for the evaluation are in SRP 6.2.1, 6.2.2, 6.2.3, 6.2.4, 6.2.5, and 6.2.6 (NUREG-0800). These acceptance criteria include the applicable GDC, RGs, BTPs, and industry codes and standards, as specified in the appropriate sections of NUREG-0800. The results of the staff's review are discussed in this section.

In addition, the following documents have been used to assess the pool dynamic load specifications:

(1) Technical Evaluation Report--"Mark III LOCA - Related Hydrodynamic Load Definition, Generic Technical Activity B-10," to be published in the third quarter of 1983

(2) "Safety/Relief Valve Quencher Loads: Evaluation for BWR Mark II and III Containments," NUREG-0802, October 1982

(3) "Guidelines for Confirmatory In-Plant Tests of Safety-Relief Valve Discharges for BWR Plants," NUREG-0763, May 1981

(4) "Suppression Pool Temperature Limit for BWR Containments," NUREG-0783, November 1981.

Not addressed in this report are certain Mark III containment design issues that have recently come to the attention of the Commission. These new issues were communicated to the staff by a former employee of GE who was associated with the STRIDE (Standard Reactor Island Design) and GESSAR (GE Standard

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Safety Analysis Report) projects. The staff is assessing the impact of these design issues, and will report its findings in a supplement to this report.

6.2.1 Containment Functional Design

The Mark III containment design has three main features

(1) a drywell surrounding the reactor pressure vessel and a large part of the reactor coolant pressure boundary.

(2) a suppression pool that serves as a heat sink during normal operational transients and accident conditions.

(3) a containment structure to prevent the uncontrolled release of radioactivity to the environment. Figure 6.1 shows the principal features of the Mark III containment.

The drywell is designed as the temporary retention boundary that separates the reactor pressure vessel and the recirculation system from the suppression pool and the containment annulus. If a LOCA should occur, the drywell will channel the released steam around the weir wall and through the horizontal vents and into the suppression pool. The drywell has a net free volume of 236,196 ft^3 and is designed for an internal pressure differential of 25 psid and a temperature of 330°F.

The suppression pool is a 360° annular pool of water located at the bottom of the containment and retained between the containment wall and weir wall, with the drywell wall located between the two. The weir wall is a 360° reinforced concrete wall located inside the drywell, 30 in. from the drywell wall. At the proposed minimum water height, 19.5 ft, the volume of water in the containment suppression pool is 124,726 ft^3. The suppression pool serves both as a heat sink for postulated transients and accidents and as the source of cooling water for the emergency core cooling system (ECCS). In the case of transients that result in a loss of the main heat sink, energy would be transferred to the pool by the discharge piping from the reactor system's safety/relief valves (SRV). In the event of a LOCA caused by a line break within the drywell,

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the drywell atmosphere will be vented to the containment by way of horizontal vents in the drywell wall. There are 129 vents in the vertical section of the drywell wall below the suppression pool water level, each with a nominal diameter of 27.5 in. These vents are arranged in 43 circumferential columns each containing three vents. The three vent centerlines in each column are located 17.25 ft., 12.25 ft., and 7.25 ft. below the suppression pool water level when the suppression pool is at the low water level of 19.5 ft.

In the event of a LOCA, the pressure will rise in the drywell because of the release of reactor coolant, and force the level of water down in the weir annulus. When the water level has been depressed to the level of the first row of vents, the differential pressure will cause air, steam, and entrained water to flow from the drywell into the suppression pool. The steam will be condensed in the pool, and the air driven from the drywell will be transferred into the containment. The containment is a free-standing steel structure consisting of a vertical cylinder, a domed top, and a flat base with a net free volume of 1.19 million ft^3. The design pressure is 15 psig, and the design temperature is 185°F. To satisfy the design basis as a fission product leakage barrier, the containment is designed for a leakage rate of 0.26% of the net free volume per day at the calculated peak pressure of 6.3 psig.

6.2.1.1 Containment Analysis

The staff's review of the containment design included the temperature and pressure responses of the drywell and wetwell to a spectrum of LOCAs; the capability to withstand the effects of steam bypass from the drywell directly to the air region of the suppression pool; and the external pressure capability of the drywell and containment. In addition, the review considered the applicant's proposed design bases and design criteria for the containment and the analyses and test data in support of the criteria and bases.

The staff's review also considered the loads resulting from pool dynamic phenomena. Following a LOCA, an air-steam mixture will be forced from the drywell through the vent system into the suppression pool. The air component of the vent flow forms high-pressure bubbles in the pool. The air bubbles will create an upward acceleration of the pool surface that can impact internal

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containment structures. Additional containment loads result as the steam portion of the vent flow condenses in the pool. Actuation of relief valves also results in containment loads. Pressure waves are generated within the suppression pool when the relief valves discharge air and steam into the pool.

The containment is divided into subcompartments by internal structures. The staff's review of the subcompartment designs is discussed in Section 6.2.1.6.

6.2.1.2 Review of Boiling Water Reactor Containment Technology

There are two basic pressure suppression designs for boiling water reactors (BWRs) in the U.S. that have preceded the Mark III containment. These are the Mark I (or "light bulb-torus") and the Mark II (or the "overunder design").

A comparison of design parameters for the Mark I, II, and III containment types is provided in Figures 6.1, 6.2, and 6.3. The Mark I containment was the first widely used design for BWR pressure suppression containments. In the Mark I design, the drywell consists of an inverted lightbulb shaped vessel, and the suppression chamber is a torus-shaped steel vessel located below and encircling the drywell. The vent system consists of ducts, vent header, and downcomers that enter the suppression pool vertically instead of horizontally, as with the Mark III vent. The typical design pressure for both drywell and pressure suppression chamber is 56 psig. The Mark II containment, like the Mark I and Mark III, is a pressure suppression containment. The drywell is in the form of a truncated cone, closed by an elliptical steel dome. The drywell concrete floor serves as a barrier separating the drywell from the suppression chamber. The pressure suppression chamber is in the shape of a cylinder located below the drywell floor. The vent system consists of vertically straignt pipes (downcomers) that extend through the drywell floor to below the water level in the suppression pool. The typical design pressure for both drywell and pressure suppression chamber is 45 psig.

The Mark III type containment proposed for the River Bend station is different from the Mark I and Mark II types of containment in three basic ways. First, the BWR/6 type reactor system proposed for the River Bend station has larger

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(26 in. diameter) steamlines relative to those of previous BWR designs for the same power level. The effect of this difference is that the postulated LOCA involving ruptures in the main steamline results in very nearly equivalent peak drywell pressures when compared with recirculation line ruptures. Therefore, both of these postulated pipe breaks must be considered in determining the design-basis accident (DBA) for the Mark III containment pressure response.

Second, the wetwell and drywell of the Mark I and II designs are connected by a vent system that enters the suppression pool vertically at a constant submergence. The Mark III design utilizes a circumferential arrangement of horizontal vents at three different elevations, creating an additional functional dependence on vent-clearing and vent-flow phenomena when compared with the Mark I and II containments. In addition, because of the relatively large vent areas provided, the peak drywell differential pressure is controlled by vent clearing; that is, the highest differential pressure across the drywell wall occurs during vent clearing. This places added emphasis on the dynamics of vent clearing, but reduces the impact of vent-flow assumptions on the peak drywell pressure. For example, in both the Mark I and II design, the peak drywell pressure occurs in the range of approximately 10 to 50 seconds following onset of the postulated accident, after the vent-clearing process and during the vent-flow part of the transient. However, for the Mark III design, the peak drywell pressure occurs at about 1.8 second, which is during the vent-clearing process.

Third, because the volume of the Mark III containment is about four times that of the drywell, the compression of drywell air into the containment during vent flow results in only a small rise in containment pressure. This small effect leads to a long-term containment peak pressure that is not specifically related to the size of the reactor coolant system break or to the short-term pressure response.

6.2.1.3 Short-Term Pressure Response

The maximum drywell differential pressure occurs during the blowdown phase of a LOCA. The maximum containment pressure occurs in the long term and is discussed in Section 6.2.1.4.

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The applicant has performed analyses of various postulated primary system breaks, including a double-ended rupture of the recirculation line, a double-ended rupture of the main steamline, an intermediate-size rupture of a liquid line, and a small-size rupture of the steamline. Results of the analyses indicate that the main steamline break (MSLB) yields the limiting drywell differential pressure and peak pressure and is, therefore, the DBA for the drywell.

Following the postulated double-ended MSLB, the mass and energy released from the primary system pressurize the drywell. As the drywell pressure increases, water in the vent annulus is accelerated downward. At about 0.92 second, the first row of vents is cleared of water, and then a mixture of air, steam, and water flows into the suppression pool where the steam is condensed and the air is released to the containment air region above the pool.

The water behind the weir wall inside the drywell continues to accelerate downward, resulting in the clearing of the second row of vents at about 1.17 seconds and the third row at about 1.65 seconds. The peak drywell differential pressure occurs at 1.17 seconds, i.e., after the first row of vents has cleared. This results because sufficient vent area is uncovered to start reducing the pressure in the drywell.

The above process is called the vent-clearing transient, which occurs less than 2 seconds following onset of the postulated accident. A detailed discussion of pool dynamics including LOCA and SRV loads is in Section 6.2.1.8.

The containment is pressurized early in the transient by the carryover of noncondensible gases from the drywell. At the end of the blowdown, the drywell pressure stabilizes at a slightly higher pressure than the containment, the difference being equal to the hydrostatic head of vent submergence. The drywell and containment will remain in this equilibrium condition until the ECCS injection water floods the reactor vessel to the level of the steamline nozzles and the water cascades into the drywell. This results in condensation of the steam in the drywell and a rapid reduction in the drywell pressure. This is a conservative assumption for the drywell depressurization transient because the plant's emergency procedures guides (EPG) would direct the operator

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to control reactor water level to a point below the break elevation. This would keep ECCS flow from emerging from the break into the drywell and would reduce the rate of drywell depressurization. As soon as the drywell pressure drops below the containment pressure, noncondensible gases from the containment air volume will flow back into the drywell via the vents.

The applicant's calculated drywell peak differential pressure and temperature for the MSLB are 18.63 psid and 316°F. The calculated peak pressures and temperatures for the containment as a result of a recirculation line break are 6.31 psig and 141°F. The design differential pressure and temperature of the drywell are 25 psid and 330°F, which provides a margin of 34% above the peak calculated pressure and 14°F above the peak calculated temperature. The design differential pressure and temperature of the containment are 15 psig and 185°F, which provides a margin of 138% above the peak calculated pressure and about 44°F above the peak calculated temperature. The staff is analyzing ~~analyses of~~ the drywell and containment temperature and pressure response using the CONTEMPT-LT 28 computer code and will report its findings in a supplement to this SER.

6.2.1.4 Long-Term Response

Following the short-term blowdown phase of the accident, the suppression pool temperature and containment pressure continuously increase because of the input of decay heat and sensible energy into the containment. At the end of the blowdown phase of the accident the drywell pressure stabilizes at a sightly higher pressure than the containment, as discussed previously. At a later time, the drywell and containment chamber pressure equalize because of the return of air from the containment through the vents.

During this period, the ECCS pumps, which take suction from the suppression pool, reflood the reactor pressure vessel up to the level of the main steam nozzles. Subsequently, ECCS water flows out of the break and fills the drywell up to the top of the weir wall, establishing a recirculation flow path for the ECCS coolant. (This again is a conservative assumption for drywell depressurization transients because the EPG would limit flow out the main steamline break.)

Reduction of drywell pressure relative to containment pressure, as a result of steam condensation on the drywell's heat sinks and the subcooled ECCS flow spills from the break onto the drywell floor, will cause the level of the suppression pool in the containment to fall. The top row of vents becomes uncovered and reverse flow of air from containment to drywell occurs. Shortly thereafter, the drywell pressure decrease is halted at about -5.83 psig, and an external pressure differential of -9.07 psid is attained (-10.21 psid for recirculation line break). This external drywell pressure differential is within the design value of -20 psid.

After approximately 30 minutes, the residual heat removal (RHR) heat exchangers are activated manually to remove energy from the containment via recirculation cooling of the suppression pool with the RHR service water system.

In the long-term analysis, the applicant accounted for potential post-accident energy sources. These included decay heat, pump heat rate, sensible heat, and metal-water reaction energy. The applicant's long-term model also assumed that the containment atmosphere would be saturated and equal to the suppression pool temperature at any time. Therefore, the containment pressure is equal to the sum of the partial pressure of air and the saturation pressure of water corresponding to the pool temperature. It should be noted that this plant does not have upper containment pool dump.

Based on the above assumptions, the applicant calculated a peak suppression pool temperature of 167.5°F for the most limiting RHR system cooling mode; that is, only one operating RHR heat exchanger with only one RHR pump available and one containment unit cooler is functional. The calculated long-term secondary peak containment pressure is about 6.19 psig. The containment is designed for 15 psig and 185°F. The staff will analyze the containment pressure and temperature response using the CONTEMPT-LT 28 computer code to confirm the applicant's analysis.

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6.2.1.5 Reverse Pressurization

Transients that could result in negative pressure loads on the drywell and containment are described below. Vacuum relief devices to minimize external pressure loading effects are not provided for this plant.

Drywell depressurization following a LOCA is terminated by reverse vent clearing, which allows air to flow from the containment back into the drywell, as discussed in Section 6.2.1.4. Containment depressurization as a result of uncontrolled cooldown by the containment unit coolers (this plant does not use containment spray but instead uses air cooler systems) is terminated by differential pressure switches that automatically isolate the coolers when containment pressure is -0.43 psig.

6.2.1.5.1 Reverse Drywell Pressurization

As discussed in Section 6.2.1.4, following reactor blowdown, the initial drywell air inventory has been purged into the containment and the drywell atmosphere is filled with steam. As a result of condensation of the steam on the heat sinks and by the spilled ECCS subcooled water, drywell depressurization occurs, which is terminated when the top rows of vents become uncovered and allow air to return to the drywell. The applicant contends that because the drywell depressurization is slow, the dynamic forces on the suppression pool as a result of the reverse vent clearing are negligible.

However, to demonstrate the capability of the drywell to withstand the negative pressure, instead of assuming the slow depressurization described above, the applicant conservatively assumed (1) the steam in the drywell condenses instantaneously, (2) ECCS flow out of the reactor vessel is at suppression pool temperature (140°F), (3) and all air has been purged from the drywell. This would cause an instantaneous drywell depressurization. Using these conservative assumptions, the applicant calculated a maximum reverse drywell pressure of 19.4 psid, which is within the design value of -20 psid. The staff has not completed its assessment of (1) the applicant's position with respect to not having vacuum breaker devices to control the negative drywell pressure and

(2) the applicant's position regarding the reverse vent-clearing dynamic loads. The staff will report on these issues in supplements to this SER.

6.2.1.5.2 Containment External Pressure Differential

Inadvertent activation of the containment unit cooler under minimum heat load conditions was considered by the applicant as the event that results from maximum containment negative pressure differential. This event results from (1) the failure of one of the chilled water control valves to close and isolate the chilled water supply to one containment unit cooler on low containment temperature and (2) the failure of the operator to recognize the problem and turn off the containment cooler.

The applicant contends that one containment unit cooler continues to reduce the containment temperature and pressure until a pressure difference of -0.43 psig is sensed by differential pressure transmitters, which measure the pressure difference between the containment and the shield building annulus.

Isolation signals based on 2 out of 3 logic are developed by six differential pressure transmitters arranged into two divisions, each division powered from separate emergency sources. The signals isolate the chilled water to all coolers. There are three unit coolers, 1A, B and C; two of them--1A and 1B--are safety-related systems. In addition, the signal from division I will trip the fan on the 1A cooler, and the division II signal will trip the fan on the 1B unit cooler. The staff was unable to complete its evaluation of the applicant's single failure analyses because the chilled water isolation valves mentioned above are not listed in the containment isolation valves in FSAR Table 6.2.40. The staff will report its finding in a supplement to this SER.

6.2.1.6 Subcompartment Pressure Analysis

Within both the drywell and containment, internal structures form subcompartments or restricted volumes that are subjected to differential pressure after postulated pipe ruptures. In the drywell, there are two such volumes: the reactor pressure vessel annulus, which is the annular region formed by the reactor pressure vessel and the biological shield, and the drywell head, which

is a cavity surrounding the reactor pressure vessel head. In the containment, subcompartments contain various components of the reactor water cleanup (RWCU) system such as the heat exchanger room and the filter/demineralizer rooms.

The design of the containment subcompartments was based upon the worst case DBA postulated to occur in each subcompartment. For each containment subcompartment in which high-energy lines are routed, mass and energy release data corresponding to a postulated line break were calculated. The mass and energy release data and the subcompartment free volumes, vent path geometry, and vent loss coefficients were used as input into the applicant's analysis to obtain the pressure/temperature transient response for each subcompartment. The Stone and Webster THREED and the RELAP4 computer programs were used to perform these pressure/temperature transient analyses. All breaks were considered to be full, double-ended circumferential breaks. No credit was taken for pipe restraint locations in the containment subcompartments.

Vent loss coefficients were used in the analysis for all vent paths analyzed and were dependent upon the wall thickness/opening diameter ratio and pressure ratio across the opening. The values were calculated from formulas and data contained in "Handbook of Hydraulic Resistance - Coefficients of Local Resistance and Friction" by Idel'Chek. Blowdown flow rates were evaluated with the model described in NEDO-24548 using the Moody steady-slip flow model with subcooling.

Blowout panels are not used and no credit is taken for vent areas that become available after the pipe break occurs. The applicant utilizes flow diverters for the two recirculation water outlets to minimize the asymmetric loads resulting from the pressurization of the reactor pressure vessels (RPV) shield wall annulus. The feedwater line break results in the greatest forces on the RPV and the greatest pressure differentials across the primary shield wall. The staff will perform a confirmatory analysis of the various subcompartments to verify that the calculated differential pressure will not exceed the design values.

6.2.1.7 Steam Bypass of the Suppression Pool

The concept of the pressure suppression containment is that any steam released from the reactor coolant system will be condensed by the suppression pool and will not have an opportunity to produce a significant pressurization effect on the containment. This is accomplished by channeling the steam into the suppression pool through a vent system. Bypass leakage paths could exist between the drywell and the containment, which may produce pressurization of the containment. Potential sources of steam bypass include cracking of the drywell concrete structure, and penetrations through the drywell structure. To mitigate the consequences of any steam that may bypass the suppression pool, the containment unit coolers, which are engineered safety features, are initiated automatically on high drywell pressure signal (with an interlock to delay initiation until 10 minutes after high drywell pressure signal) to mitigate the consequences of steam bypass.

The allowable bypass leakage is defined as the amount of steam which could bypass the suppression pool without exceeding the containment design pressure.

The allowable value has been evaluated by the applicant for the complete spectrum of credible primary system pipe ruptures. It is expressed in terms of the parameter $(A/\sqrt{K})$, where

A = flow area of leakage path ($ft.^2$) and
K = geometric and friction loss coefficient

This parameter $(A/\sqrt{K})$ is dependent only on the geometry of the drywell leakage paths and is a convenient numerical definition of the overall drywell leakage capability.

An analysis was performed by the applicant that evaluated the bypass capability of the containment for small primary system breaks considering containment unit cooler and containment heat sinks as means of mitigating the effects of bypass leakage.

The suppression pool water was assumed to pass through the one available RHR heat exchanger and put in the suppression pool cooling mode 30 minutes after onset of the LOCA. One containment unit cooler was assumed to operate 10 minutes from the onset of the accident.

Containment heat sinks were considered with variable convective heat transfer coefficients based on local instantaneous air-steam ratio; i.e., Uchida heat transfer coefficient. The reactor vessel cooldown was assumed to initiate 10 minutes from onset of the accident at a rate 200°F per hour, and maximum design standby service water temperature of 95°F was used. The cooldown rate corresponds to the maximum rate that does not thermally cycle the reactor vessel. This analysis results in an allowable drywell leakage capability of $A/\sqrt{k}$ of 1.00 ft^2.

The staff will perform a confirmatory analysis to confirm the adequacy of the proposed steam bypass capability.

The applicant has proposed both high- and low-pressure leakage tests. The high pressure preoperational test will be performed once following the completion of the high pressure structural integrity test. The proposed low pressure drywell leak rate test will be conducted periodically (three every 10 years), which the staff finds unacceptable. The staff will require this test to be performed during each refueling outage.

The applicant has proposed that the acceptance criterion for the high-pressure leak tests be demonstrating that the drywell bypass leakage is less than 10% of the $A/\sqrt{k}$ value of 5.3 ft^2, which results from a main steamline double-ended rupture. This is not acceptable to the staff. As per SRP 6.2.1.1.6, the intention of both the preoperational and inservice steam bypass test is to demonstrate that the actual bypass is less than 10% of the capability of the containment to accommodate bypass leakage under the spectrum of postulated LOCA break sizes. To do this, the applicant must either perform the preoperational high-pressure test with an acceptance criterion of less than 10% of $A/\sqrt{k}$ of 1.00 ft^2 or perform a high-pressure test with an acceptance criterion of less than 10% of $A/\sqrt{k}$ of 5.3 ft^2 and a low-pressure preoperational test with 10% of $A/\sqrt{k}$ 1.00 ft^2. The staff will require the applicant to commit to the

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use of one of these two alternative preoperational test acceptance criteria. The applicant has proposed a low-pressure inservice test acceptance criterion of an allowable leakage less than 10% of $A/\sqrt{k}$ of 1.00 ft^2. The staff finds this acceptable; however, as stated earlier, the staff does not agree with the proposed frequency of conducting this test.

The staff will report on this issue in a supplement to the SER.

6.2.1.8 Pool Dynamics

The Mark III pool dynamic loads were reviewed at the preliminary design assessment (PDA) stage for GESSAR-II. The staff concluded at the time that the information available was sufficient to adequately define the pool dynamic loads for those nuclear plants under review for construction permits. Since the issuance of the GESSAR-II SER (December 1975), GE has conducted further tests and analyses to confirm and refine the original load definitions. To keep the NRC and Mark III applicants apprised of the current status of these tests, GE issued an Interim Containment Loads Report (22A4365) in April 1978 and revised this report several times before GESSAR-II was provided to the staff in March 1980. Appendix 3B of GESSAR-II provides the finalized pool dynamic load definition for Mark III containments and is the basic document used for review by the staff and its consultants.

The staff is currently reviewing GE's LOCA-related pool dynamic load definitions to arrive at a finalized LOCA-related hydrodynamic load definition that can be utilized by Mark III containment applicants for operating licenses. The pool dynamic loads are being reviewed under Task Action Plan (TAP) B-10, "Behavior of BWR Mark III Containments." The staff has completed its review of the Mark III containment safety relief valve hydrodynamic load definition presented in GESSAR-II as part of TAP A-39, "Determination of Safety Relief Valve (SRV) Pool Dynamic Loads and Temperature Limits for BWR Containments."

6.2.1.8.1 Description of Phenomena

LOCA Dynamics: Figure 6.4 shows the sequence of events occurring during a DBA and the potential loading conditions associated with these events. Following a

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postulated LOCA, the drywell pressure increases as a result of blowdown of the reactor system. Pressurization of the drywell causes the water initially in the vent system to be accelerated into the pool until the vents are cleared of water. During this clearing process, the water leaving the horizontal vents forms jets in the suppression pool and causes water jet impingement loads on the structures within the suppression pool and on the containment wall opposite the vents.

During the vent clearing transient, the drywell is also subjected to a pressure differential and the weir wall experiences a vent clearing reaction force.

Immediately following vent clearing, an air and steam bubble forms at the exit of the vents. The bubble pressure initially is assumed equal to the existing drywell differential pressure (the peak calculated is 21.8 psig). This bubble theoretically transmits a pressure wave through the suppression pool water and results in a loading on the suppression pool boundaries and on equipment located in the suppression pool.

As the air and steam flow from the drywell become established in the vent system, the initial vent exit bubble expands to the suppression pool hydrostatic pressure. GE's large-scale Pressure Suppression Test Facility (PSTF) tests show that the steam fraction of the flow is condensed, but continued injection of drywell air and expansion of the air bubble result in a rise of the suppression pool surface. During the early stages of this process, the pool swells in a bulk mode (i.e., a slug of solid water is accelerated upward by the air pressure). Structures close to the pool surface will experience loads as the rising pool surface impacts the lower surface of the structure. In addition to these initial impact loads, these same structures will experience drag loads as water flows past them. Equipment in the suppression pool will also experience drag loads.

Data from PSTF air tests indicate that after the pool surface has risen approximately 1.6 times the initial submergence of the top vent, which translates to 12 ft above the initial pool surface for the Mark III design, the thickness of the water ligament has decreased to 2 ft or less and the impact loads are signficantly reduced. This phase is referred to as "incipient breakthrough," i.e., the ligament begins to break up. To account for possible nonconservatisms

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in the test facility arrangement, the staff has determined that the breakthrough height should be set at 18 ft above the initial pool surface.

Ligament thickness continues to decrease until complete breakthrough is reached and the air bubble can vent to the containment free space. The breakthrough process results in formation of an air/water froth and, for load definition purposes, is defined to happen at 19 ft above the initial pool surface. For conservatism, the incipient breakthrough height and the height at which froth loads begin have been set higher than predictions from test results.

Continued injection of drywell air into the suppression pool results in a period of froth pool swell. This froth swell impinges on structures it encounters but the two-phase nature of the fluid results in loads that are much less than the impact loads associated with bulk pool swell.

When the froth reaches the elevation of the floors on which the hydraulic control units for the control rod drives are located (approximately 24 ft above pool level), the froth encounters a flow restriction; at this elevation, there is approximately 25% open area. The froth pool swell experiences a two-phase pressure drop as it is forced to flow through the available open areas. This pressure differential represents a load on both the floor structures themselves and on the adjacent containment and drywell. The result is a discontinuous pressure loading at this elevation.

As the drywell air flows through the horizontal vent system decreases and the air/water suppression pool mixture experiences gravity-induced phase separation, pool upward movement stops, and the fallback process starts. During this process, floors and other flat structures experience downward loading and the containment wall theoretically can be subjected to a small pressure increase. However, this pressure increase has not been observed experimentally.

The pool swell transient associated with drywell air venting to the pool typically lasts 3 to 5 seconds. Following this, there is a long period of high steam flow through the vent system; data indicate that this steam will be entirely condensed in a region located at the vent exits. For the DBA reactor blowdown, steam condensation lasts for a period of approximately

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1 minute. Potential structural loadings during the steam condensation phase of the accident have been observed, and are included in the containment loading specification.

As the reactor blowdown proceeds, the primary system is depleted of high-energy fluid inventory and the steam flow rate to the vent system decreases. This reduced steam flow rate leads to a reduction in the drywell/containment pressure differential which, in turn, results in a sequential recovering of the horizontal vents. Suppression pool recovery of a particular vent row occurs when the vent stagnation differential pressure corresponds to the suppression pool hydrostatic pressure at that row of vents.

Toward the end of the reactor blowdown, the top row of vents is capable of condensing the reduced blowdown flow and the two lower rows will be totally recovered. As the blowdown steam flow decreases to very low values, the water in the top row of vents starts to oscillate, causing what has become known as vent chugging. This action results in dynamic loads on the top vents and on the weir wall opposite the upper row of vents. In addition, an oscillatory pressure loading condition can occur on the drywell and containment. Because this phenomenon is dependent on steam mass flux (the chugging threshold appears to be in the range of 10 $lb/sec/ft^2$), it is present for all break sizes. For smaller breaks, it may be the only mode of condensation that the vent system will experience.

Shortly after onset of a DBA, the ECCS pumps automatically start up and pump condensate water or suppression pool water into the reactor pressure vessel. This water floods the reactor core and then, if the operator fails to follow the emergency procedure guidelines requiring ECCS flow to be throttled, starts to cascade into the drywell from the break. When this occurs depends upon break size and location. Because the drywell is full of steam at the time of vessel flooding, the sudden introduction of cool water causes rapid steam condensation and drywell depressurization. When the drywell pressure falls below the containment pressure, air from the containment redistributes between the drywell and containment via the vent system (reverse vent clearing). Eventually sufficient air returns to equalize the drywell and containment pressures; however, during this drywell depressurization transient, there is a

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period of negative pressure on the drywell structure. A conservative negative load condition of 20 psid is, therefore, specified for drywell design.

The DBA is the design-limiting case for the pool swell-related pool dynamic loads including jet, drag, impact and fallback loads. The intermediate-break accident (IBA) and small-break accident (SBA) cases have a much lower rate of drywell pressurization. Therefore, for these cases, the IBA and SBA pool swell loads are correspondingly lower. However, LOCA-related steam loads at design values can occur over a wider spectrum of breaks because the maximum condensation loads will occur at low vent mass flux. Condensation oscillation and chugging will occur over an extended period of time for small breaks as a result of the reduced reactor vessel depressurization rate compared to a DBA. For IBAs and SBAs, the range of parameters that control the load characteristics of condensation oscillation (CO) and chugging phenomena are within the parameter ranges used by GE to arrive at the load definitions for these loads. Therefore, the staff concludes that the magnitude of CO and chugging loads for IBAs and SBAs will be equal to or less than the magnitude of these loads following a DBA.

Also, the IBA and SBA cases must be considered in developing the design thermal stratification loads for the suppression pool. In the event of the postulated LOCA, the suppression pool water in the immediate vicinity of the vents will be heated as a result of the energy release during the steam condensation portion of the transient. For the Mark III horizontal vent configuration, the majority of the mass and energy will be transferred to the pool through the top vents. Consequently, the upper portion of the suppression pool will be heated more than the lower portion. The vertical temperature gradient resulting from this effect is defined as thermal stratification and the thermal stratification load definitions are in Appendix 3B of GESSAR-II.

6.2.1.8.2 SRV Actuation

In addition to loads on the valves and discharge piping, actuation of the SRVs causes pressure disturbances in the suppression pool water that result in dynamic loads on the containment adjacent to the pool. Structures in the pool also experience this loading. Relief valve actuation can be initiated either

automatically by a reactor pressure increase to the valve setpoints or by an active system such as the automatic depressurization system (ADS). The phenomena that cause these loads are described below.

Before actuation, the SRV discharge lines contain air at atmospheric pressure and in the submerged section, a column of water. Following SRV actuation, the pressure builds up inside the piping and expels the water column. The air follows the water through the holes in the quencher arms and forms a large number of small bubbles. Once in the pool, the bubbles expand, coalesce, and form four large bubbles. Each of the four bubbles expands, analogous to a spring, and accelerates the surrounding pool of water. The momentum of the accelerated water causes the bubble to over expand and the bubble pressure becomes negative. This negative pressure slows down and finally reverses the motion of the water, leading to compression of the bubble. This sequence of expansion and contraction is repeated with a mean frequency of about 8 Hz until the bubble reaches the pool surface.

The bubble oscillation causes oscillating pressures throughout the pool. The magnitude of the pressure amplitude decreases with time and with distance from the bubble. The duration of this load is less than 1 second. SRV steam condensation during valve discharge to the pool also occurs. This phenomenon results in low-amplitude pressure fluctuations.

6.2.1.8.3 Hydrodynamic Load Assessment

6.2.1.8.3.1 LOCA-Related Hydrodynamic Load Assessment

The following acceptance criteria were developed from the staff's review of Appendix 3B of GESSAR-II 238, Nuclear Island, and the supporting analytical and experimental programs as referenced therein. The staff has determined that the procedures described in GESSAR-II are acceptable for evaluation of the Mark III containment response to LOCA-related pool dynamic loads with the following exceptions, modifications, qualifications and/or clarifications discussed below.

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Pool Swell Loads

(1) Pool Swell Velocity

The water slug velocity, V, to be used for the determination of slug drag loads, as well as slug impact loads as prescribed in items (2), (3), and (5) below, shall be determined from the relation

$$V = 5\,H\,(2.6 - 0.506\,\sqrt{H}); \qquad H < 10 \text{ ft, and}$$
$$V = 50 \text{ fps} \qquad\qquad H > 10 \text{ ft}$$

In these relations, V is the slug velocity in feet per second (fps); H denotes the height (in feet) above the initial pool surface.

(2) Pool Swell Loads on Structures Attached to the Containment Walls

The GESSAR-II specification for these loads is given in FSAR Section 3B.6.1.5. It is based on steady drag at a flow velocity of 40 fps. The specification should be modified as follows to make it acceptable:

- If the local pool swell velocity, as specified in item (1) above, is greater than 40 fps, the drag pressures obtained by the GESSAR-II specification shall be multiplied by the factor (V/1600), where V is the local pool velocity in fps.

- If the frontal area of the structure is not immersed before pool swell, it will experience an impact force. This must be included in the specification.

For half-wedge protrusions the force history during impact shall be determined from Figure 6.5. The local velocity of impact (needed in Figure 6.5) shall be taken from the specification in item (1) above corresponding to the height where the wedge is first fully submerged. If the lower portion of the wedge is initially submerged, the same impact history is applied except the abscissa is replaced by Vt/h, where h is the "unsubmerged height" of the wedge.

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If the wedge angle is other than 45°, the following ratios can be used after obtaining the force history for a 45° wedge is obtained from Figure 6.5.

$$\frac{F_\beta}{F_{45}} = \left(\frac{\beta}{90-\beta}\right)^2 \cot \beta, \text{ and} \qquad (1a)$$

$$\frac{t_\beta}{t_{45}} = \cot \beta \qquad (1b)$$

Where b is the wedge angle.

For horizontal wedges the impact force shall be determined in the following manner:

- Note the force history which will have a triangular shape, as shown in Figure 6.6.
- Determine the hydrodynamic mass of impact (per unit area) for flat targets from Figure 6-8 of NEDE-13426. Use b (and not b/2) for target widths.
- Calculate the impulse using the equation

$$\frac{I}{P} = \frac{M_H}{A} V \times \frac{1}{(32.2)\ (144)} \qquad (2)$$

where:

I_p = impulse per unit area, psi-sec

$\frac{M_H}{A}$ = hydrodynamic mass per unit area, lbm/ft^2.

V = impact velocity, fps, determined according to item (1) above.

- Calculate the pulse duration from the equation

$$\tau = 0.02 \frac{H}{V} \frac{b}{2} \qquad (3)$$

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where:

τ = pulse duration, sec

H = height above the pool, ft

b/2 = width of the ledge, ft

V = impact velocity, fps, determined according to item (1) above.

- The value of P_{max} will be obtained using the following equation:

$$P_{max} = 2 I_p / \tau$$

where:

P_{max} = the peak pressure amplitude, psi.

(3) Bulk Impact Loads on Small Structures

"Small structures" are defined as beams and pipes with lateral dimensions no larger than 20 inches. For these structures the bulk impact specification, as described in Section 3B.10.11 of GESSAR-II, is acceptable with the specific limitations discussed in Section 3B0.3.2.33 of Attachment 0 (Question/Response 38.33).

These limitations are

- Targets must have combinations of widths and natural frequencies such that Figures 3B.33-1, 2, 3, and 4 indicate them to be in the "GESSAR conservative" region with respect to the V = 50 fps pool velocity curve.

- There are no structures smaller than 4 ft long.

- There are no structures closer than 6 ft above the pool.

For those structures that do not meet the second and third limitations, the pulse width and amplitude of the bulk impact specification described in Section 3B.10.11 of GESSAR-II should be modified as follows:

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o For structures smaller than 4 ft long, reduce the GE-specified pulse width (7 msec for radial oriented structures; 2 msec for circumferentially oriented structures) by $\frac{x}{4}$, where x is the actual length of the structure in question.

(2) For structures closer than 6 ft above the pool, reduce the GE-specified pulse width by $\frac{y}{6}$, where y is the actual elevation of the structure in question.

For structures less than 10 ft above the pool, the amplitude may be reduced by

$$\frac{P}{P_{max}} = \frac{V^2}{V^2_{max}} = \frac{H^2}{100} (2.6 - 1.6 \sqrt{H})^2$$

where P is the amplitude at a structure H feet above the pool surface and P_{max} is the GE-specified amplitude for bulk impact on small structures above the pool.

(4) Froth Impact Loads

The froth impact specification is applicable between an elevation of 19 ft above the pool and the hydraulic control unit (HCU) floor. Over open areas of the HCU floor, this specification extends to an elevation of 26 ft for flat structures and 28.5 ft for pipes. Between these elevations and 30 ft the froth drag specification in GESSAR-II, Section 3B.12 is applicable.

The forcing function for froth impact shall be an isoceles triangle with a maximum amplitude as given in Figure 6.7. The pulse duration shall be chosen such as to give a maximum dynamic load factor (DLF) with a triangular pulse. For elongated structures that span the annulus of the wetwell, pulse durations shorter than 50 msec need not be considered.

(5) Drag Loads

The GESSAR-II specification for drag loads on small structures, as described in Section 3B.10.2, is acceptable with the following limitation: if the

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local pool velocity, as specified in item (1) above, is greater than 40 fps, the drag forces obtained by the GESSAR-II specification shall be multiplied by a $(V/40)^2$, where V is the local velocity in fps.

If Figure 3B-75 in GESSAR-II is used for drag loads on plates and if the short side (b) is attached to a wall, the abscissa of the graph becomes 2a/b instead of a/b.

Loads on Submerged Structures

The procedures for the computation of loads on submerged structures as described in Sections 3.8.8 to 3.8.8.1.7 and Attachment L of GESSAR II are acceptable, subject to the limitations and/or modifications of Section 3.8.0.3.2.31 of Attachment O. In particular, the computation of acceleration loads on non-cylindrical structures and the elevation of standard drag for CO loads must be based on Mark I Acceptance Criteria (NUREG-0661) as described in the response to questions 3.8.31(b.ii), (c.ii), and (c.iii) in Attachment O.

Impact Loads on Structures Above the Weir Annulus

The staff finds acceptable the GESSAR-II impact specification on structures above the weir wall, with the exception of radial structures within 1 ft of the top of the weir wall. The stresses in these structures must be increased by a multiplier. This multiplier is 1.0 when the natural frequency of the structure is less than 200 Hz, and 1.8 when the natural frequency is greater than 500 Hz, with a linear ramp between these values.

For structures located between zero and 0.25 feet above the weir wall, flat pool impact should be considered (e.g., the acceptance criteria in Section 2.7 of Appendix A, NUREG-0661, are appropriate for flat pool impact). The velocity at impact for these structures shall be taken at 4 fps.

In response to question 480-05, the applicant indicated an intention to utilize the load definition presented in GESSAR-II; however, the applicant will review the NRC acceptance criteria after they are issued. In addition, the applicant stated that because the HCU in River Bend station is higher than the elevation

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of the HCU floor in GE's standard plant, the plans are to use reduced pool swell velocity at River Bend, considering deceleration due to gravity, to determine the impact load on the floor structure.

The staff cannot complete its review of the applicant's positions regarding LOCA-related pool dynamic loads. The staff will require the applicant to state his position regarding the modification to the GESSAR-II load specification, as described in the previous section. The staff will report its findings in a supplement to this SER.

6.2.1.8.3.2 SRV Hydrodynamic Load Assessment

The acceptance criteria for Mark III containment SRV hydrodynamic loads are in NUREG-0802, "Safety/Relief Valve Quencher Loads: Evaluation for BWR Mark II and III Containments," which includes in Appendix B, "Proposed NRC Acceptance Criteria for Mark III Safety/Relief Valve Related Pool Dynamic Loads." These acceptance criteria apply to SRV boundary and tie-down loads and were developed primarily on the basis of the work done by GE for the GESSAR-II Nuclear Island.

The applicant's response to question 480.05 indicates that the load definition presented in GESSAR-II will be used at River Bend.

The staff is currently reviewing the detailed information presented in Appendix 6A to the FSAR to determine its compliance with the staff acceptance criteria set forth in NUREG-0802. The staff will report its findings in a supplement to this SER.

6.2.1.8.3.3 Pool Temperature Limit and SRV Inplant Test

In NUREG-0783, "Suppression Pool Temperature Limits for BWR Containments," the staff states that the suppression pool local temperature shall not exceed 200°F; (alternatively, compliance with the acceptance criteria specified in Section 5.1 of NUREG-0783 would be acceptable).

The applicant has provided the results of the analyses for pool temperature responses to transients involving SRV operation.

The staff is unable to complete its review of the applicant's analyses, because the applicant has not listed the assumptions used in the analyses. The applicant must address the staff requirement set forth in Section 5.6 of NUREG-0783. In addition, the staff requires that the applicant provide for staff review proposed River Bend suppression pool temperature monitoring system and show how it complies with the staff guidelines set forth in NUREG-0487. The staff requires that the applicant demonstrate the adequacy of the number and distribution of the monitors in the proposed design. The staff will report its evaluation of its review in a supplement to this report.

The staff will require the applicant to either perform inplant SRV tests or demonstrate that the SRV discharge conditions in the plant are sufficiently similar to conditions previously tested. In doing so, the applicant should address the five criteria set forth in Section 4 of NUREG-0763. The staff will report its finding regarding these issues in a revision to this SER.

6.2.3 Secondary Containment Functional Design

The River Bend station's secondary containment is provided to collect fission products that may leak from the primary containment following a postulated LOCA. The secondary containment, which completely surrounds the primary containment and is maintained at negative pressure during normal plant operation, consists of the following:

(1) the annulus (shield) building, which is a reinforced concrete, cylindrical shell completely surrounds the primary containment.

(2) the auxiliary building, which houses RWCU, RCIC and ECCS pump room. The annulus building and the auxiliary building are maintained at negative pressure following a LOCA by the standby gas treatment system (SGTS).

(3) the fuel building, which is maintained at negative pressure following a LOCA and a high radiation condition by the fuel building charcoal filtration system.

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with either of above mentioned systems and the rationale for eliminating them as potential bypass leakage paths.

Before plant operation begins, the shield building will be tested to verify that the inleakage will not exceed 2000 cfm at a pressure of -3 inches wg.

The staff will require the applicant to identify all openings to the secondary containment and whether all these openings are equipped with position indicators which alarm in the control room. The staff also will require the applicant to commit to performing a periodic test to verify the inleakage and the drawdown time.

The staff will report on the resolution of the bypass leakage issue in a supplement to this report.

6.2.3 Containment Heat Removal System

The containment heat removal function for the River Bend station is performed by the suppression pool cooling mode (SPCM) of the residual heat removal (RHR) system and by the two safety related containment unit coolers of the containment ventilation system. The containment heat removal system includes the piping, valves, and mechanical components that will be used to remove energy from the containment to limit temperature and pressure in the drywell and containment following a postulated LOCA. The SPCM is an integral part of the RHR system and is manually aligned in this mode approximately 1800 seconds after the onset of a LOCA or after the reactor vessel water level has been recovered.

The RHR system consists of two complete loops including heat exchangers and main system pumps. Each loop is considered so that a failure in one loop cannot cause a failure in another. In addition, each of the loops and its associated equipment is located in a separate protected area of the reactor building to minimize the potential for a single failure (including loss of onsite or offsite power) causing the loss of function of the entire system. The system equipment, piping, and support structures are designed to seismic Category I criteria.

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Provisions have been made in the RHR system to permit inservice inspection of system components and functional testing of active components.

When operating in the containment cooling mode, the RHR pump takes suction from the suppression pool. Flow is then directed through the RHR heat exchangers to the reactor vessel and back to the suppression pool.

The RHR heat exchangers and the unit coolers are assumed to be in the fully fouled condition at the time of the postulated accident.

The location of suction and return lines in the suppression pool facilitates mixing of the return water with the total pool inventory before the return water becomes available to the suction lines.

Strainers are provided on the suction line inlets. The strainer assemblies will consist of 200% capacity corrosion resistant units that are designed to ASME Code Class 2 standards. They shall not exceed the maximum allowable pressure drop of 0.5 psi with 50% of the strainer openings plugged at a flow rate of 5050 gpm. The strainers will be located far enough above the suppression pool flow to prevent any accumulated debris from being picked up.

The strainer configuration precludes the possibility of one large piece of debris blocking the entire suction. The mesh openings will be the maximum size possible without allowing the passage of any object with a diameter greater than 0.094 inch.

The applicant stated that the strainers have the capability to accommodate hydrodynamic loads.

RG 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Sumps," recommends against reliance on the pressure or temperature transients expected during a LOCA for ensuring net positive suction head (NPSH). The River Bend station design assumes 0 psig containment pressure and the maximum expected fluid temperatures resulting from a LOCA and, therefore, is acceptable.

The River Bend station utilizes three containment unit coolers; two of these are safety related. The coolers are sized for normal operating conditions and are supplied with coolant from the chilled water system. Following onset of a LOCA, standby service water at a maximum of 95°F and at a flow rate of 54 gpm per cooler is used for cooling water. Each of the unit coolers is powered from a different division standby bus. The coolers are automatically initiated on high containment pressure with an interlock to delay initiation 10 minutes to minimize the short-term peak load on the standby diesel generators.

The unit coolers, along with the passive heat sinks, provide a reliable means for steam condensation if a direct path develops for steam bypass of the suppression pool. The unit coolers are not required to mitigate the effects of a LOCA.

Based on the analysis of the long-term pressure and temperature analysis and of the potential for suppression pool steam bypass, the staff concludes that the containment heat removal system can be operated to provide adequate cooling of the containment following onset of a LOCA and that the system conforms to GDC 38, 39, and 40. It is, therefore, acceptable.

6.2.4 Containment Isolation System

The containment isolation system includes the containment isolation valves and associated piping and penetrations necessary to isolate the primary containment in the event of a LOCA. The staff review of this system includes the number and location of isolation valves, the valve actuation signals and valve control features, the positions of the valves under various plant conditions, the protection afforded isolation valves from missiles and pipe whip, and the environmental design conditions specified for the design's components.

The design objective of the containment isolation system is to allow the normal or emergency passage of fluids through the containment boundary to prevent or limit the escape of fission products from a postulated LOCA. The applicant has specified design bases and design criteria as well as the isolation valve arrangement to be used for isolation of primary containment penetrations.

The staff is unable to complete its review of the applicant's analyses, because the applicant has not listed the assumptions used in the analyses. The applicant must address the staff requirement set forth in Section 5.6 of NUREG-0783. In addition, the staff requires that the applicant provide for staff review proposed River Bend suppression pool temperature monitoring system and show how it complies with the staff guidelines set forth in NUREG-0487. The staff requires that the applicant demonstrate the adequacy of the number and distribution of the monitors in the proposed design. The staff will report its evaluation of its review in a supplement to this report.

The staff will require the applicant to either perform inplant SRV tests or demonstrate that the SRV discharge conditions in the plant are sufficiently similar to conditions previously tested. In doing so, the applicant should address the five criteria set forth in Section 4 of NUREG-0763. The staff will report its finding regarding these issues in a revision to this SER.

6.2.3 Secondary Containment Functional Design

The River Bend station's secondary containment is provided to collect fission products that may leak from the primary containment following a postulated LOCA. The secondary containment, which completely surrounds the primary containment and is maintained at negative pressure during normal plant operation, consists of the following:

(1) the annulus (shield) building, which is a reinforced concrete, cylindrical shell completely surrounds the primary containment.

(2) the auxiliary building, which houses RWCU, RCIC and ECCS pump room. The annulus building and the auxiliary building are maintained at negative pressure following a LOCA by the standby gas treatment system (SGTS).

(3) the fuel building, which is maintained at negative pressure following a LOCA and a high radiation condition by the fuel building charcoal filtration system.

The annulus building is maintained at negative pressure during plant operation by the annulus pressure control system.

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The SGTS is also designed to process exhaust air from the containment/drywell purge system during normal operation for fast entry (this issue will be discussed later).

During accident and post-accident conditions, including a LOCA, the SGTS is initiated automatically and processes air from the shield annulus and the auxiliary building. In addition, the system can be started manually from the control room if required.

The annulus building will be maintained at -0.5 inch wg, and the fuel building will be kept at -0.25 inch wg, with respect to the outside environmental atmospheric pressure following a LOCA. After a postulated LOCA, the primary and secondary containment volumes will be automatically isolated from the normal exhaust air paths. The SGTS will be started automatically and exhaust air drawn through the SGTS to maintain negative pressure in the secondary containment. The SGTS reaches rated flow within 30 seconds after the LOCA, assuming loss of offsite power.

An annulus mixing system is provided for thorough mixing of any leakage in the volume between the free standing steel primary containment and the shield building. Startup of the SGTS is automatic in response to the following signals:

(1) LOCA (high drywell pressure, or low reactor water level signal, or both)

(2) high high radiation from monitors in the shield annulus exhaust

Upon receipt of a signal, the annulus exhaust, which is normally through the annulus pressure control system, will automatically line up to the SGTS. The exhaust from the shielded compartments in the auxiliary building is manually diverted through the SGTS filter trains on high radiation conditions. Radiation monitors are also provided in the fuel building exhaust duct, which, together with the LOCA signal, exhaust the fuel building through the charcoal

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filtration system. The pressure response for the shield building annulus indicates that the annulus is kept at a negative pressure of approximately -3 inches wg and maximum inleakage during normal plant operation is 200 cfm. Assuming that it takes 30 seconds for the SGTS fans to reach their rated flow, as indicated by the applicant the annulus region will reach a pressure of approximately 2 inches wg at 100 seconds with the building exhaust rate of 2500 cfm. The -0.25 inches wg is attained in 161 seconds, which indicates that for approximately 135 seconds the annulus pressure is greater than -0.25 inches wg.

The auxiliary building pressure response indicates that -0.25 inches wg is attained in 54 seconds if the auxiliary building exhaust rate is 10,000 cfm and the SGTS exhaust fan starts operating at 30 seconds. For the fuel building, the analysis indicates that the -0.25 inches wg will be attained in 14.5 seconds if the fuel building exhaust rate is 10,000 cfm and exhaust starts at 10 seconds.

Although the primary containment is enclosed by the secondary containment, there are systems that penetrate both the primary and secondary containment boundaries, creating potential bypass paths. This means that, in the event of a LOCA, radioactivity in the primary containment could bypass the leakage collection and filtration systems associated with the secondary containment. A number of these lines contain physical barriers or design provisions that can effectively eliminate leakage; for example, water seals, a closed seismic Category I piping system, or vent return lines to controlled regions. The criteria by which potential bypass leakage paths are determined has been set forth in BTP CSB-6-3, "Determination of Bypass Leakage Paths in Dual Containment Plants."

The applicant indicated that a penetration valve leakage control system (PVLCS) is provided to control and minimize the fission product leakage through the isolation valves of any process line that penetrates the containment and terminates in an untreated area. The staff is reviewing the design of the PVLCS. In addition, the main steam positive leakage control system controls and minimizes the fission products that could leak through the main steam isolation valves and the main steam drain lines. The staff will require the applicant to identify all other bypass leakage paths that are not equipped

with either of above mentioned systems and the rationale for eliminating them as potential bypass leakage paths.

Before plant operation begins, the shield building will be tested to verify that the inleakage will not exceed 2000 cfm at a pressure of -3 inches wg.

The staff will require the applicant to identify all openings to the secondary containment and whether all these openings are equipped with position indicators which alarm in the control room. The staff also will require the applicant to commit to performing a periodic test to verify the inleakage and the drawdown time.

The staff will report on the resolution of the bypass leakage issue in a supplement to this report.

6.2.3 Containment Heat Removal System

The containment heat removal function for the River Bend station is performed by the suppression pool cooling mode (SPCM) of the residual heat removal (RHR) system and by the two safety related containment unit coolers of the containment ventilation system. The containment heat removal system includes the piping, valves, and mechanical components that will be used to remove energy from the containment to limit temperature and pressure in the drywell and containment following a postulated LOCA. The SPCM is an integral part of the RHR system and is manually aligned in this mode approximately 1800 seconds after the onset of a LOCA or after the reactor vessel water level has been recovered.

The RHR system consists of two complete loops including heat exchangers and main system pumps. Each loop is considered so that a failure in one loop cannot cause a failure in another. In addition, each of the loops and its associated equipment is located in a separate protected area of the reactor building to minimize the potential for a single failure (including loss of onsite or offsite power) causing the loss of function of the entire system. The system equipment, piping, and support structures are designed to seismic Category I criteria.

The River Bend station utilizes three containment unit coolers; two of these are safety related. The coolers are sized for normal operating conditions and are supplied with coolant from the chilled water system. Following onset of a LOCA, standby service water at a maximum of 95°F and at a flow rate of 54 gpm per cooler is used for cooling water. Each of the unit coolers is powered from a different division standby bus. The coolers are automatically initiated on high containment pressure with an interlock to delay initiation 10 minutes to minimize the short-term peak load on the standby diesel generators.

The unit coolers, along with the passive heat sinks, provide a reliable means for steam condensation if a direct path develops for steam bypass of the suppression pool. The unit coolers are not required to mitigate the effects of a LOCA.

Based on the analysis of the long-term pressure and temperature analysis and of the potential for suppression pool steam bypass, the staff concludes that the containment heat removal system can be operated to provide adequate cooling of the containment following onset of a LOCA and that the system conforms to GDC 38, 39, and 40. It is, therefore, acceptable.

6.2.4 Containment Isolation System

The containment isolation system includes the containment isolation valves and associated piping and penetrations necessary to isolate the primary containment in the event of a LOCA. The staff review of this system includes the number and location of isolation valves, the valve actuation signals and valve control features, the positions of the valves under various plant conditions, the protection afforded isolation valves from missiles and pipe whip, and the environmental design conditions specified for the design's components.

The design objective of the containment isolation system is to allow the normal or emergency passage of fluids through the containment boundary to prevent or limit the escape of fission products from a postulated LOCA. The applicant has specified design bases and design criteria as well as the isolation valve arrangement to be used for isolation of primary containment penetrations.

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The containment isolation system is designed to automatically isolate the containment atmosphere from the outside environment under accident conditions. Double-barrier protection, in the form of two isolation valves in series or a closed system and isolation valve, are provided to ensure that no single active failure will result in the loss of containment integrity. The containment isolation system's components--including valves, controls, piping and penetrations--are protected from internally or externally generated missiles, water jets, and pipe whip.

The basis for the staff's acceptance has been the conformance of the containment isolation provisions to the Commission's regulations as set forth in the GDC and applicable RGs, staff technical positions, the acceptance criteria of the SRP (NUREG-0800), and industry codes and standards.

The containment isolation system is designed to ASME Code, Section III, Class 1 or 2, and is classified as a seismic Category I system.

The containment isolation provisions for the lines penetrating containment conform to the requirements of GDC 55, 56, or 57, as appropriate. As provided by GDC 55 and 56, there are containment penetrations whose isolation provisions do not have to satisfy the explicit requirements of the GDC, but can be accepted on some other defined basis.

Most of those penetrations not satisfying the explicit requirements of the GDC were found acceptable based on their meeting alternative criteria specified in SRP 6.2.4, Item II. These alternative acceptance criteria are

(1) Lines that must remain in service following an accident and lines that should remain in service during normal operation for safety reasons are provided with at least one isolation valve. A second isolation boundary is formed by a closed system outside the containment.

(2) Where a closed system outside the containment forms the second isolation boundary, each of the systems and all components that form its boundary are designed to Quality Group B and seismic Category I standards. Valves

that isolate the branch lines of these closed systems outside containment are normally closed and under strict administrative control.

(3) On some engineered safety features or related systems, remote manual valves are used in lieu of automatic valves because these lines must remain in service following an accident. Where remote manual valves are used, leakage detection capabilities are provided.

(4) On some penetrations the containment isolation provisions consist of two valves in series both of which are outside the containment. The location of a valve inside containment would subject it to more severe environmental conditions (including suppression pool dynamic loads), and the valve would not be easily accessible for inspection.

To preclude containment pressurization in the case of a pipe line rupture outside the drywell, along the section of the pipe between the drywell and the primary containment penetrations, some lines are equipped with a guard pipe that encases the pipe line and extends from the drywell up to and including the primary containment penetration.

Certain other lines penetrating the containment, which are described below, do not meet either the explicit requirements of the GDC or the alternative SRP acceptance bases, but still meet acceptable isolation criteria on other defined acceptance bases.

o Feedwater Lines

The two feedwater lines penetrate the drywell to connect with the reactor pressure vessel. Each line has four isolation valves and a guard pipe. The isolation valve inside the drywell is a simple check valve. Outside the primary containment is a testable (air-assisted to close) check valve. Farther away from the primary containment are two motor-operated gate valves. Should a break occur in the feedwater line, the check valves prevent any significant loss of reactor coolant inventory and offer prompt primary containment isolation. During the postulated LOCA, it is desirable to maintain reactor coolant makeup from all sources of

supply. For this reason, the outermost valve does not automatically isolate upon receipt of a signal from the isolation system. Instead of having a leakage detection system in conjunction with a remote manual valve, this valve will be procedurally controlled and remotely closed from the control room to provide long-term leakage protection.

The staff finds this acceptable provided the applicant commits to implementing a procedure for isolating the feedwater lines 20 minutes after onset of a postulated LOCA or sooner if the available information indicates insufficient water supply.

o High Pressure Core Spray

The high pressure core spray (HPCS) line penetrates both the primary containment and the drywell. Isolation is provided by a testable check valve inside the drywell and a motor-operated gate valve as close as possible to the shield building boundary. A guard pipe is not provided, because the HPCS operates following onset of a LOCA, when the liquid temperature is low and cannot overpressurize the containment if a line break occurs between the primary containment and drywell. The check valve inside the drywell protects against containment overpressurization by preventing high-energy reactor water from entering the primary containment. Based on its assessment of this design provision, the staff concludes that it is an acceptable alternative to GDC 55 and, therefore, acceptable.

o Control Rod Drive

The control rod drive system (CRDS), located between the drywell and the primary containment, has two types of influent lines: (1) the supply line that penetrates the primary containment, and (2) the CRD insert lines (145) that penetrate the drywell and connect to the CRD flange.

Isolation of the supply line is provided by a simple check valve on the line inside the primary containment and a remote manual block valve

(motor-operated valve) on the line outside the primary containment. The supply line is not a part of the RCPB.

The CRD insert and withdraw lines are not part of the RCPB. They are designed in accordance with ASME Code Section III, Class 2. The basis to which the CRD insert and withdraw lines are designed is commensurate with the safety importance of maintaining pressure integrity of these lines.

As a means of providing isolation, manual shutoff valves are used. In the event of a break in these lines, the manual valves may be closed to ensure isolation. In addition, a ball valve located in the CRD automatically seals the insert line in the event of a break. Containment overpressurization does not result from a line break in the primary containment because these lines connect to small volumes at low energy levels.

The staff finds that the system design represents a departure from the explicit requirements of the GDC. However, in accordance with GDC 55 (which permits departures from its explicit requirements) the staff finds that the CRD containment isolation provision described above is acceptable on the bases stated in NUREG-0803, "Safety Evaluation Report Regarding Integrity of BWR Scram Systems."

o Standby Liquid Control

This standby liquid control system is located inside the containment; however, because this is a nonflowing line, rupture of this line is highly unlikely. Should a break occur, the check valves, on both sides of the drywell, ensure isolation and containment overpressurization protection.

The staff concludes that the system isolation provision and the absence of a guard pipe are acceptable.

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- Traversing Incore Probe

  The traversing incore probe (TIP) system lines do not communicate through the primary containment because they penetrate only the drywell. The size of each TIP line is 0.28 inch; the lines are essentially equivalent to an instrument line with 0.25-inch orifice and, therefore, are not provided with containment isolation valves.

  The staff will report its findings regarding the isolation provision of these lines in a revision to this SER.

- Turbine Plant Miscellaneous Drains

  Note 55.11 to FSAR Table 6.2-36 indicates that two motor-operated valves in series, located outside the primary containment, are provided for containment isolation. Figure 6.2-63 indicates that one valve is located inside the containment and the other is outside the containment. The staff will require the applicant to provide a clarification of this discrepancy.

- RHR Relief Valve Vacuum Breaker

  The staff will require the applicant to provide justification for having a check valve outside containment on the RHR relief valve vacuum breaker line.

- Miscellaneous RHR Penetrations

  The applicant has not provided enough justification for having a single isolation valve on some RHR lines, such as: RHR heat exchanger vent, HPCS pump minimum flow bypass, HPCS test return, RHR loop test return, RHR pump minimum flow bypass, and reactor core isolation cooling turbine exhaust. The staff will require the applicant to provide justification for deviation from the provisions of the SRP.

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o LPCI, HPCS, and RHR Relief Lines

These lines are isolated by the relief valves themselves; the addition of block valves for isolation would defeat the purpose for which the relief valves are installed. The staff concludes that this deviation from the explicit requirements of the GDC is acceptable.

o Equipment and Floor Drains Within the Containment

These lines, which connect the reactor building equipment drains and floor drain headers to drain sumps in the drywell, do not penetrate the containment; therefore, isolation provided by simple check valves on both sides of the drywell is acceptable. Similarly, the recirculation pump seal, which does not penetrate the containment, is provided with check valves on each side of the drywell.

o Instrument Lines

The instrumentation lines isolation system is not in compliance with the provisions of RG 1.11. The staff will require the applicant to reconsider the design.

o Chilled Water

Note 56.1.1 to FSAR Table 6.2-38 does not match the information depicted on Figure 6.2-65. The staff will require the applicant to discuss the discrepancy for all lines listed in Table 6.2-38. In addition, the referenced figure indicates that the containment unit coolers use SC3 piping. Because these coolers are relied on to irrigate the consequency of suppression pool steam bypass, the staff will require that all system components be designed to SC2 criteria.

o Service Water System

The drywell coolers, which are designated as SC-3 system, do not appear to receive a containment isolation signal. The staff will require the

applicant to provide the rationale for not isolating the service water system to the drywell coolers after a LOCA.

o Main Steam SRV Air Supply Lines

These lines are not provided with a guard pipe to protect the containment from being over pressurized as a result of a line break between the drywell and the containment. The staff will require the applicant to provide the rationale for this design.

Based on its evaluation of the containment isolation system for the River Bend station, the staff finds that additional information is necessary for the review of the containment isolation system. In addition, the staff will require the applicant to revise FSAR Table 6.2-40 to provide all missing information and to ensure that all penetrations depicted on Figures 6.2-63 through 6.2-65 are listed in that table.

Item II.E.4.2. Containment Isolation Dependability

The applicant has not responded to the staff's request for additional information regarding this issue.

Containment Purge System

The design of the containment purge and ventilation system includes 36-inch supply and exhaust lines. The applicant proposes to use the system to continuously purge the containment atmosphere during normal operation and shutdown. Closure time for containment isolation valves in the purge lines is 3 seconds. It is the staff's view that purging should be minimized during reactor operation, because the plant is inherently safer with closed purge valves than with open valves requiring valve action to provide containment isolation. Therefore, the staff will require a detailed justification of the need to purge continuously. In addition, the applicant has indicated that his evaluations do not point to the need for debris screens to protect the purge valves. The staff disagrees with and, as stated in BTP CSB 6-4, requires debris screens to be provided to ensure that isolation valves will be able to close. The staff will require

the applicant to install debris screens and will report on these topics in a supplement to this report.

As a result of numerous reports on the unsatisfactory performance of the resilient seats for butterfly-type isolation valves in the containment purge supply and exhaust lines, the staff established Generic Issue B-20, "Containment Leakage Due to Seal Deterioration," and is investigating the problem. Excessive leakage past the resilient seats in these valves can be caused by severe environmental conditions or wear as a result of frequent use. The staff will require the applicant to test the containment isolation valves that have resilient seat materials every 3 months for those active systems in operation during operating modes 1 through 4. The staff will find a commitment in this regard acceptable. The staff will also verify that the valves in inactive systems (not used during modes 1 through 4) will be tested every 6 months and will report its findings in a supplement to this SER.

6.2.5 Combustible Gas Control in Containment

The combustible gas control system includes the piping, valves, components, and instrumentation necessary to detect the presence of combustible gases within the primary containment and to control the concentration of these gases.

The scope of review of the design and functional capability of the combustible gas control system for River Bend station included drawings of and descriptive information on the equipment to mix the containment atmosphere, monitor combustible gas concentration, and reduce combustible gas concentrations within the containment following onset of the design-basis accident. The review also included the applicant's proposed design bases for combustible gas control systems, and the analyses of the functional capability of the system provided to support the adequacy of the design bases.

The bases for the staff acceptance are conformance of system design and design bases with the Commission's regulations as set forth in 10 CFR 50.44 and the GDC, and with applicable RGs, BTPs, and industry codes and standards.

The containment atmospheric monitoring system (CAMS) design, including the hydrogen sampling points at 14 locations in the drywell and containment, will alert the operator to the need for initiating the hydrogen control equipment. An alarm in the main control room will indicate greater than 3.5 volume percent hydrogen at any one of the 14 sample location.

The applicant's combustible gas control system consists of three subsystems; they include: (1) a mixing system, (2) a hydrogen control system, and (3) a backup purge system. The major components of the mixing system are compressors, valves, piping, and the required instrumentation. The mixing systems, initiated 3 hours after onset of the accident and upon receipt of an alarm from any one of the four drywell sample points, will take suction from the containment and discharge into the drywell. The system consists of redundant components so that any single active failure will not prevent purging of the drywell volume. An interlock is provided to prevent opening of the outlet valves before the inlet valves are fully opened. The operator will then open the outlet valves; this starts the mixing fan when these valves are fully opened. An operator error can result in the opening of the 6-inch hydrogen mixing system inlet valve; however, the bypass capability $\frac{A}{\sqrt{K}} = 1.0$ ft^2 will not be exceeded.

The hydrogen control system consists of two 100% capacity hydrogen recombiners manufactured by Westinghouse. The staff has evaluated the Westinghouse recombiner design and qualification programs described in WCAP 7709-L and its supplement and found them acceptable. Each unit has a design flow rate of 100 scfm and is located inside the containment on opposite sides of the refueling floor. The backup purge system is also provided in accordance with RG 1.7. This system will purge the containment atmosphere, if necessary, through the containment filtration system's charcoal filter trains to reduce the activity released.

Following onset of a design-basis LOCA, hydrogen may accumulate within the containment as a result of metal-water reaction between the fuel cladding, corrosion of construction materials, and as a result of radiolytic decomposition of the post-accident emergency cooling water. The applicant analyzed the production and accumulation of hydrogen from the above sources. The guidelines regarding the metal-water reaction states that hydrogen production is five

times the maximum calculated reaction under 10 CFR 50.46 or the amount that would evolve from a corewide average depth of reaction into the original cladding of 0.23 mil, whichever is greater, in 2 minutes. The applicant's analysis of the combustible gas control system was based on the amount of hydrogen that would be evolved from a core-wide reaction of an average depth of 0.23 mil of the fuel cladding. The applicant considered the contribution to hydrogen generation from the corrosion of aluminum and zinc coated substances as well as galvanized steel in the drywell. Because the River Bend station does not utilize a spray system in the containment, corrosion of zinc and aluminum was neglected. However, for the drywell, the zinc corrosion rates used in the analyses were obtained from the interim zinc corrosion rate that was found acceptable by the staff; the aluminum corrosion rate in neutral water is small and hence was neglected.

Item II.B.8. Rulemaking Proceeedings on Degraded Core Accident

To reduce the possible consequences of a postulated degraded core accident at River Bend station, the staff requested that the applicant propose a program to improve the plant's hydrogen control capability. Specifically, the staff requested that the applicant provide a system to control hydrogen that would be generated from a postulated 75% metal-water reaction, which is well beyond the amount of hydrogen specified in 10 CFR 50.44(d).

With regard to the programs under way to improve the hydrogen control capability at River Bend for dealing with degraded core accidents, the applicant indicated that he is a member of the BWR/6 Mark III containmnt Hydrogen Control Owners' Group (HCOG), which will address the hydrogen control issue for degraded core accidents on a generic basis.

The applicant stated that the evaluation and studies performed under the auspices of the HCOG will be used to develop a River Bend-specific hydrogen control program, which will be submitted for the staff review 6 months before fuel loading. The staff informed the applicant that an acceptable interim system is needed before operation above 5% of full power.

The staff will provide its evaluation of the hydrogen production and control in a supplement to this SER.

Item II.E.4.1, Dedicated Hydrogen Penetrations

River Bend utilizes two recombiners that are located within the containment. A backup filtered containment purge through a dedicated seismic Category I penetration is also provided. Because the recombiners are located inside the containment, there are no piping penetrations associated with operation of the recombiners. Consequently, Item II.E.4.1 is not applicable to River Bend station.

6.2.6 Containment Leakage Testing

The staff has reviewed the applicant's containment leak testing program for compliance with the containment leakage testing requirement specified in Appendix J to 10 CFR 50. Such compliance provides adequate assurance that the containment leak-tight integrity can be verified throughout the service lifetime and that the leakage rates will be periodically checked during service on a timely basis to maintain such leakage within the specified limits. Maintaining containment within such limits provides reasonable assurance that, in the event of any radioactivity release within the containment, the loss of the containment atmosphere through potential leak paths will not be in excess of the limits specified for the site.

The staff has reviewed the containment leak testing program to ensure that the containment penetrations and system isolation valve arrangements are designed to satisfy the containment integrated leak rate testing requirements and the local leak testing requirements of Appendix J.

6.2.6.1 Type A Test

After completion of construction of the primary reactor containment and installation and local leak tests of all mechanical, fluid, electrical, and instrumentation systems penetrating the containment pressure boundary, preoperational integrated leakage rate Type A tests will be conducted in accordance with 10 CFR 50, Appendix J.

The objective of the preoperational integrated leak rate tests is to verify that the integrated leakage rate does not exceed the containment DBA leakage rate, L_a, which is 0.26% percent by weight of the contained atmosphere in 24 hours.

The preoperational integrated leak rate test will be performed at the accident pressure, P_a. After the preoperational integrated leak rate test, a set of three Type A tests will be performed at approximately equal intervals during each 10-year service period, with the third test of each set coinciding with the end of each 10-year major inservice inspection shutdown.

When the Type A containment integrated leak rate test is being conducted, certain systems that may operate under post-LOCA conditions need not be vented to the containment atmosphere if the following criteria are met: the systems must be protected against missiles and pipe whip designed to seismic Category I criteria and classified Safety Class 2, assuming a single active failure, and remain full of water for at least 30 days. The staff will require the applicant to identify those systems that will not be vented during the Type A test.

6.2.6.2 Type B Test

Containment penetrations whose designs incorporate resilient seals, gaskets or sealing compounds, piping penetrations fitted with expansion bellows serving as the containment boundary, air-lock door seals, equipment and access doors with resilient seals, and other testable penetrations will be leak tested at periodic intervals during the lifetime of the unit in accordance with Appendix J of 10 CFR 50. The leak test will be performed to ensure the penetrations continuing integrity.

Electrical penetrations will be provided with a leakage surveillance system that continuously pressurizes the penetration test chamber to pressures not less than P_a.

The personnel air lock design includes test connections to pressurize the annulus between the seals, the interior of the lock, and shafts and view ports in the penetration doors.

Tests shall be performed during each reactor shutdown for major fuel reloading, but in no case at intervals more than 2 years. Air locks that are opened frequently will be tested at selected intervals not to exceed 3 days, in accordance with Appendix J.

6.2.6.2
Type C Test

The applicant's proposed Type C tests for certain lines associated with the RHR system are not acceptable. The staff will report its findings regarding these issues in a supplement to this SER.

6.2.7 ~~6.3~~ Emergency Core Cooling System

The emergency core cooling system (ECCS) is designed to provide water to the reactor coolant system in the event of a break in the pressure boundary. The ECCS capability extends to failures as large as a double-ended rupture of the largest pipe carrying water or steam, as well as spurious SRV operation.

The basis for the design of the ECCS is to limit damage to the fuel cladding in accordance with 10 CFR 50.46. The system must be capable of performing its design function with or without offsite power and with a single failure, including loss of an emergency diesel generator.

6.3.1 System Description

The ECCS consists of the following systems:

(1) high pressure core spray system (HPCS)

(2) automatic depressurization system (ADS)

(3) low pressure core spray system (LPCS)

(4) low pressure coolant injection system (LPCI)

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6.3 Emergency Core Cooling System

The emergency core cooling system (ECCS) is designed to provide water to the reactor coolant system in the event of a break in the pressure boundary. The ECCS capability extends to failures as large as double-ended rupture of the largest pipe carrying water or steam, and spurious safety/relief valve operation.

6.3.1 Design Bases

The basis for the design of the ECCS is to limit damage to the fuel cladding in accordance with Title 10, Code of Federal Regulations, Part 50.46 (10 CFR 50.46). The system must be capable of performing its design function with or without offsite power and with a single failure, including loss of an emergency diesel generator.

6.3.2 System ~~Description~~ Design

The ECCS consists of the following systems:

(1) High Pressure Core Spray System (HPCS)

(2) Automatic Depressurization System (ADS)

(3) Low Pressure Core Spray System (LPCS)

(4) Low Pressure Coolant Injection System (LPCI)

The HPCS is provided to maintain the reactor vessel water level above the top of the active core in the event of small pipe breaks and to provide spray cooling in case the core is uncovered. Activation of the HPCS does not require the depressurization of the reactor vessel. The system includes a single motor-driven centrifugal pump [illegible] takes suction from the condensate storage tank or the primary containment suppression pool. An automatic switching

feature is provided. HPCS flow is dependent on the pressure differential that exists between the reactor system and the suction source. Pump characteristic curves indicate that the rated HPCS flow (5010 gpm) is attained above 200 psid, which is consistent with accident analysis assumptions. HPCS discharges water into the reactor via a spray sparger mounted on the reactor vessel internal wall above the core. The HPCS system is designed to operate from normal offsite auxiliary AC power or from its own diesel generator.

The ADS is provided to depressurize the reactor coolant system in the event a small pipe break occurs and the HPCS system can not maintain reactor vessel water level or fails to start. The ADS employs 7 of the 16 safety/relief valves to reduce system pressure so that the low pressure systems may inject water to cool the core.

The LPCS system is provided to replace reactor vessel water inventory and to supply spray cooling following large pipe breaks in which the core may uncover. The system includes a motor-driven centrifugal pump that takes suction from the suppression pool and discharges water to the reactor vessel via a spray sparger mounted on the reactor vessel internal wall above the core. Pump characteristic curves indicate that rated LPCS flow (5010 gpm) is attained at a pressure of approximately (later) psid. The LPCS sparger is separate from the HPCS sparger. The LPCS system is designed to operate from normal auxiliary AC power or from the standby AC power system. The LPCS pump and RHR pump A are on the same electrical bus and are supplied emergency power by the same diesel generator.

The LPCI system is provided to replace reactor vessel water inventory following large pipe breaks. The system is an operating mode of the Residual Heat Removal (RHR) system which consists of three independent loops (A, B, and C). Each loop has a motor-driven pump (5050 gpm) which takes suction from the suppression pool and supplies water to the reactor vessel via a separate nozzle through the reactor vessel wall. In addition, loops A and B can also take suction from the reactor recirculation system suction or fuel pool, and can discharge into the reactor via a feedwater line or into the fuel pool cooling and clean-up system. RHR loops A and B have heat exchangers which are cooled by the standby service water system and are used to transfer the decay heat

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from the reactor core to the ultimate heat sink. The three LPCI (RHR) pumps are powered from AC power buses having standby power source backup supplies. RHR pumps B and C are on the same electrical bus and receive emergency power from the same diesel generator. RHR pump A is on the same electrical bus as the LPCS pump.

6.3.3 Performance Evaluation

~~6.3.3 Evaluation~~

~~Single Failures~~

Single Failures

We reviewed the system description and piping and instrumentation drawings to assure that abundant core cooling will be provided during the injection phase with and without offsite power and assuming a single failure, as required by Criterion 35 of the General Design Criteria. A low reactor vessel water level and/or high containment pressure signal is required to start pumps and open discharge valves.

The applicants provided in Section 6.3.3 of the Final Safety Analysis Report an analysis to demonstrate that the most limiting break size, break location, and single failure had been considered for RBS. The most limiting combinations are tabulated below.

Break size	Break location	Single failure	Systems remaining
Small	Recirculation Suction Line	HPCS	ADS, LPCS, and all LPCI
Intermediate and Large	Recirculation Suction Line	LPCI diesel Generator	HPCS, ADS, LPCS, one LPCI

The applicant has analyzed main steam breaks inside and outside containment, HPCS line breaks, and feedwater line break locations. The analyses have shown these break locations, assuming the worst single failure, are not limiting. These breaks all occur at higher elevations which result in faster depressurization and earlier actuation of the emergency core cooling system. In addition, the reduced loss of inventory results in lower peak clad temperatures.

Qualification of the Emergency Core Cooling System

The emergency core cooling system is designed to meet seismic Category I requirements in compliance with Regulatory Guide 1.29, "Seismic Design Classification," as discussed in Section 3.2 of this report. It is housed in structures designed for seismic events, tornadoes, floods, and other phenomena in accordance with the requirements of Criterion 2 of the General Design Criteria as discussed in Section 3.8 of this report. Emergency core cooling system equipment is designed in compliance with Regulatory Guide 1.26, "Quality Group Classification and Standards for Water, Steam, and Radioactive Waste Containing Components of Nuclear Power Plants," as discussed in Section 3.2 of this report.

Protection of the emergency core cooling system against pipe whip and against discharging fluids in compliance with the requirements of Criterion 4 of the General Design Criteria and Regulatory Guide 1.46, "Protection Against Pipe Whip Inside Containment," is discussed in Section 3.6 of this report. Evaluation of the instrumentation and controls for the emergency core cooling system is discussed in Section 7.3 of the report. Compliance with the inservice inspection requirements of Criterion 36 of the General Design Criteria is discussed in Section 6.6 of this report. Environmental qualification of the emergency core cooling system equipment for operation under normal and accident conditions as required by Criterion 4 of the General Design Criterion is discussed in Section 3.11 of this report.

Functional Design

The available net positive suction head for the pumps in the emergency core cooling system should have adequate margin to prevent cavitation and assure pump operability in accordance with Regulatory Guide 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps." We have requested additional information from the applicant to verify adequate NPSH for the ECCS pumps.

The HPCS incorporates relief valves to protect the components and piping from inadvertent overpressure conditions resulting from either thermal expansion or

backpressure leakage into the low pressure portions of the system. The LPCS and LPCI systems are not designed to withstand normal reactor operating pressure. Relief valves are provided in these lines to protect against leakage from the reactor coolant system. Each of the low pressure lines that interface with the reactor coolant system has a testable check valve inside primary containment, backed up by a normally closed motor-operated gate valve outside of containment. We have requested the applicant to verify this valve is interlocked so that it does not open until the reactor coolant pressure is below the system design pressure. We will report on this item in a supplement to this report.

Containment isolation in accordance with the requirements of Criterion 55 of the General Design Criteria is discussed in Section 6.2 of this report. The periodic testing and leak rate criteria for these valves that isolate the reactor coolant system from the emergency core cooling system are discussed in Section 3.9.6 of this report. The detection of leaks from those portions of the emergency core cooling system within primary containment is discussed in Sections 5.2.5 and 9.3.3. Leak detection for portions of the emergency core cooling system outside of primary containment is provided primarily by the reactor building floor and equipment drain sump leak detection system and by the equipment area high temperature detection system.

All the emergency core cooling systems have miniflow lines to permit a limited amount of flow in the event an isolation valve between the reactor coolant system and emergency core cooling system is closed, for any reason, in order to protect the pumps from overheating. When sufficient flow passes through the injection lines, valves in the miniflow lines automatically shut, diverting all flow to the pressure vessel. The lines from the suppression pool to the suctions of the low pressure coolant injection and low pressure core spray pumps each have an open motor-operated valve outside of containment, with controls arranged so that a key is required to unlock a lever to close the valve. The suction of the high pressure core spray from the suppression pool contains a closed motor-operated valve outside the containment designed to open so that

the system automatically pumps water from the suppression pool instead of the condensate storage tank when the condensate storage tank water is exhausted.

As a backup to the high pressure core spray system, the automatic depressurization system can be used to depressurize the system and allow the functioning of the low pressure cooling systems in the event of a small break. The air supply to the automatic depressurization system valves is provided in accident conditions by seismically qualified accumulators and a safety-grade air charging system.

One of the design requirements of the emergency core cooling system is that cooling water flow be provided rapidly following the initiation signal. By always keeping the emergency core cooling system pump discharge lines full, the lag time between the signal for pump start and the initiation of flow into the reactor pressure vessel can be minimized. In addition, full discharge lines will prevent potentially damaging waterhammer occurrences on system startup. In RBS a fill system consisting of a jockey pump in each of the five ECCS loops is provided. Maintenance of the filled status of the system is ensured by continuous indication of pump operation and pump discharge pressure. In accordance with monthly surveillance procedures, the uppermost vent lines in the filled systems are opened and checked for flow to reduce the possibility of the formation of air pockets. Pressure instrumentation provided on the jockey pump discharge line initiates an alarm in the main control room when pressure in the discharge line is less than the hydrostatic head required to maintain the line full of water up to the injection valves.

An electrical interlock is incorporated into the HPCS circuitry that prevents the injection valve from closing automatically upon receipt of the high reactor water level (L8) signals if a high drywell pressure signal still exists. The interlock was added as a result of the NRC staff review of GESSAR-238 which indicated that the interlock was needed to assure diversity of HPCS initiation signals and to prevent premature HPCS termination. However, flooding of the steam lines could result in damage to the safety/relief valves and primary system piping unnecessarily, since the interlock tends to keep the HPCS in operation past the point of reflooding the core and does not significantly add to the overall safety. We requested clarifications from the

applicant regarding this item. We will report on this item in a supplement to this SER.

Safety/relief valve operability will be demonstrated during the power ascension phase of the plant startup test program by manually actuating each safety/relief valve (including the ADS valves) one at a time to measure discharge capacity and to demonstrate that no blockage exists in the valve discharge line. After commercial turnover, all of the safety relief valves will be tested in accordance with Section XI, Article IWV of the ASME Boiler and Pressure Vessel Code. Safety relief valve position indication provisions are discussed under Item II.D.3 in Chapter 22. Environmental qualification of the safety relief valves is discussed in Section 3.11.

6.3.3 Performance Evaluation

We reviewed the loss-of-coolant accident analyses presented by the applicant in Section 6.3.3 of the Final Safety Analysis Report. Calculations were conducted in accordance with the methods described in General Electric Topical Report NEDO-20566, "General Electric Company Analytical Model for Loss-of-Coolant Analyses in Accordance with 10 CFR Part 50, Appendix K," dated August 1974 and "General Electric Refill Reflood Calculation," transmitted December 20, 1974. During 1977, the General Electric Company proposed several changes to its emergency core cooling system evaluation model. These changes have been approved by us and are described in our report "Safety Evaluation for General Electric Emergency Core Cooling System Evaluation Model Modifications." These methods constitute an evaluation model that conforms to the requirements of Appendix K to 10 CFR Part 50.

The five major acceptance criteria for the emergency core cooling system as specified in 10 CFR 50.46 are:

(1) The calculated maximum peak cladding temperature shall not exceed 2200°F.

(2) The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

(3) The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(4) Calculated changes in core geometry shall be such that the core remains amenable to cooling.

(5) The calculated core temperature shall be maintained at an acceptable low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core after any calculated successful initial operation of the ECCS.

At the time of this review, the FSAR LOCA analysis results were obtained for a lead plant representative of RBS. The licensee has committed to supply plant-specific LOCA analyses in a later FSAR amendment. We will update the results as necessary after our receipt and review of the LOCA analyses.

The lead plant results for the first three items are:

	Maximum Values From Break Analyses	Allowable
Peak Cladding Temperature (PCT)	2062°F	2200°F
Maximum Cladding Oxidation	1.60%	17%
Maximum Total Hydrogen Generation	0.11%	1%

A coolable geometry is demonstrated by the compliance with the criteria for the PCT and the maximum cladding oxidation as discussed in NEDO-20566.

Long-term cooling is assured by the use of redundant systems which have adequate water sources available to remove the decay heat generated within the reactor core and transfer the heat to the ultimate heat sink. No single failure was identified that would prevent the ECCS from meeting this criterion.

The systems are designed to reflood the reactor core to at least the 2/3 core level and maintain this level even under the most adverse circumstances. The major equipment for each system, other than ADS, is located in separate water-tight rooms outside primary containment.

The applicant has indicated that some operator actions are assumed in the loss-of-coolant accident analyses 10 minutes after accident initiation. Section 6.3 of the Standard Review Plan states that no credit for operator actions should be taken prior to 20 minutes. We requested additional information from the applicant to confirm that no operator action is required until 20 minutes after the LOCA. We will report on this item in a supplement to this SER.

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The applicant has not discussed the simultaneous closure of a recirculation flow control valve during a loss-of-coolant accident. We requested the applicant to address this concern. We will report our findings in a supplement to the SER.

The core spray spargers for both the high and low pressure core spray systems each consist of two semicircular segments which form an essentially complete circular sparger. Water is sprayed radially onto the tops of the fuel assemblies by short elbow nozzles spaced around the sparger. Tests of this type of spray system were performed in a full-scale test in which air at atmospheric pressure simulated the post loss-of-coolant accident steam environment and indicated adequate cooling was delivered to each fuel assembly. However, recent tests conducted on a single nozzle indicate that the acual steam environment may adversely affect the distribution of flow from certain types of core spray nozzles. As discussed in NUREG-0410, "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants," this problem is being studied by us under Task Action A-16 entitled, "Steam Effects on BWR Core Spray Distribution." Preliminary analyses and measurements have been made which support the existence of a significant safety margin between the amount of spray flow provided to each fuel assembly in the post loss-of-coolant accident steam environment and that are used to calculate the spray cooling coefficients assumed in the loss-of-coolant analyses. Tests have recently been conducted by General Electric to confirm spray flow margins used in the emergency core cooling system loss-of-coolant accident analyses. We have reviewed the results of these tests and they have been found acceptable for BWR/6 plants. Our evaluation was forwarded to General Electric by letter dated January 30, 1981 from D. G. Eisenhut (NRC) to G. G. Sherwood (General Electric), "Acceptance for Referencing Topical Report NEDO-24712 Core Spray Design Methodology Confirmation Tests."

6.3.4 ~~testing~~ Tests and Inspections

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The applicant states that operability of the emergency core cooling systems will be demonstrated by preoperational and periodic testing as required by Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Reactor Power Plants," and Criterion 37 of the General Design Criteria.

6.3.4.1 Preoperational Tests

Preoperational tests will assure proper functioning of controls, instrumentation, pumps, piping, and valves. Pressure differential and flow rates will be measured for later use in determining acceptable performance in periodic tests. The applicant has committed to meet the guidelines of Regulatory Guide 1.68 mentioned above for preoperational and initial startup testing of the emergency core cooling system as noted in Section 14 of this report.

6.3.4.2 Periodic Component Tests

We will require the applicant to test the subsystems comprising the emergency core cooling system every 92 days to show that specified flow rates are attained. Also, we will require every 18 months that a test be performed in which all subsystems are actuated through the emergency operating sequence. These tests comply with Criterion 37 of the General Design Criteria.

Conclusions

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Our review of Section 6.3 has been performed in compliance with the guidelines of the revised Standard Review Plan (SRP), NUREG-0800, with the following exceptions:

The SRP states that the proposed plant technical specifications are reviewed. We have not yet reviewed all of the plant technical specifications since they are not available. A review will be performed when the complete specifications are submitted by the applicant prior to fuel loading. A detailed review of all of the specifications will probably not be performed since it is expected that approved standard technical specifications will be used at RBS.

6.3.5 Instrumentation Requirements

We reviewed piping and instrumentation drawings and the description of the emergency core cooling system presented in the Final Safety Analysis Report. We find the design of the system acceptable because it conforms to the pertinent Regulatory Guides, Standard Review Plan and General Design Criteria. In addition, based on the discussion above, we find the performance of the emergency core cooling system acceptable because it conforms to the requirements of 10 CFR 50.46 pending resolution of the issues noted above.

6.3.6 Conclusions

6.4 Control Room Habitability

6.5 Fission Product Removal and Control Systems

6.5.1 Engineered Safety Feature (ESF) Filter Systems

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~~6.5 Engineered Safety Feature (ESF) Filter Systems~~

~~6.5.1 Introduction~~

Section 6.5 of the Final Safety Analysis Report (FSAR) contains information pertaining to engineered safety feature (ESF) filter systems, their design bases, and applicable acceptance criteria.

~~6.5.2 [illegible]~~

The staff has reviewed the applicant's design, design criteria, and design bases for the ESF filter systems for the River Bend Nuclear Generating Station. The acceptance criteria used as the basis for our evaluation are set forth in the Standard Review Plan (SRP) NUREG-0800, in Section II of SRP 6.5.1. These acceptance criteria include the applicable General Design Criteria (Appendix A to 10 CFR Part 50), ANSI Standard N509-1980, "Nuclear Power Plant Air Cleaning Units and Components", and ANSI Standard N510-1980, "Testing of Nuclear Air Cleaning Systems". Guidelines for implementation of the requirements of the acceptance criteria are provided in the ANSI Standards, Regulatory Guides, and other documents identified in Section II of the SRP. Conformance to the acceptance criteria provides the bases for concluding that the ESF filter systems meet the requirements of 10 CFR Part 50.

~~6.5.3 Method of Review~~

The River Bend Station has three ESF filter systems, the Control Room Emergency Fresh Air Filter System, the Fuel Building Charcoal Filtration System, and the Standby Gas Treatment System (SGTS). Each of these systems was reviewed in accordance with the SRP. The results of these reviews are discussed below.

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~~6.5.1 Review Discussion~~

6.5.1.1 ~~Design Bases~~ Control Room Emergency Air Filter System (CREAFS)

The function of the control room emergency fresh air filtration system is to supply non-radioactive air to the control room after a DBA and to pressurize the control room. This system will permit operating personnel to remain in the control room following a DBA. The CREAFS is a redundant system; each train processes air at a flow rate of 4000 cfm. Each train contains the following components: demister, prefilter, HEPA filter, carbon adsorber, HEPA filter and fan. Heating coils are also provided for relative humidity controls and a water spray system is provided in the event of high temperature or fire in the carbon beds. The equipment and components are designed to seismic Category I powered by a class 1E buses and are located in a seismic Category I structure. The system will be automatically activated by an emergency signal and can also be activated manually from the control room.

From the system description in the FSAR, we determined that the air cleanup system of the CREAFS is designed consistent with GDC's 19, 61, and 64, as referenced in the SRP. In our evaluation of the system design efficiencies for removal of elemental iodine and organic iodine, we assigned the system decontamination efficiencies of 90% for normal and 99% for accident elemental and organic iodines for the CREAFS charcoal adsorbers (one 4 inch deep bed) and 99% for particulates for the CREAFS HEPA filters, in accordance with Regulatory Guides 1.140 and 1.52.

Except for some parameters not provided with recording instruments, provisions for instrumentation, readout, and alarm were determined to be consistent with Table 6.5.1-1 of the SRP. While flow and pressure drop is not recorded, adequate indication and alarms are provided in the control room to assure proper monitoring of systems performance.

Based on the above determinations, we find that the CREAFS is adequately designed to control the pressure and the concentration of radioactive materials in control room air in accordance with applicable regulations following a postulated DBA.

6.5.1.2

Standby Gas Treatment System

The function of the Standby Gas Treatment System (SGTS) is to collect and process the leakage of radioactive materials from the drywell into the secondary containment and into the annulus spaces following a design basis accident (DBA). The SGTS is a seismic Category 1 design with redundant physically separated trains. Each train has a design capacity of 12,500 cfm and includes the following components: demister, heating coil, prefilter, high efficiency particulate air (HEPA) filter, one 4-inch deep bank of carbon adsorbers, HEPA filter, and fan. Deluge-type water spray systems are provided for flooding the charcoal in case of fire or overheating. The equipment and components are designed to seismic Category I, and are located in a seismic Category I structure. While the trains are idle, a 100 cfm fan circulates air through the charcoal to remove decay heat.

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From the system description in the FSAR, we determined that the SGTS is designed consistent with GDC's 41, 42, 43, 61, and 64, as referenced in the SRP. In our evaluation of the system design efficiencies for removal of elemental iodine and organic iodines, we assigned the system decontamination efficiencies of 90% for normal and 99% for accident and organic iodines for the SGTS carbon adsorbers (one 4-inch deep bed), in accordance with Regulatory Guides 1.140 and 1.52 and 99% for particulates for the SGTS HEPA filters. Except for some parameters not provided with recording instruments, provisions for instrumentation, readout, and alarm were determined to be consistent with Table 6.5.1-1 of the SRP. While flow and pressure drop is not recorded as suggested in Regulatory Guide 1.52, adequate indication and alarms are provided in the control room to assure proper monitoring of system performance.

6.5.1.3

Fuel Building Ventilation System

The function of the Fuel Building Ventilation system is to provide supply air to the fuel handling building and also to collect and process the leakage of radioactive materials from accidents which could occur during the handling of irradiated reactor fuel in the Fuel Handling Building.

The exhaust system for the building ventilation collects radioactive materials during accident conditions. The exhaust system consist of two 100% capacity trains with a design flow of 10000 cfm. The following components that make up each train are: a moisture separator and heater

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element for humidity control, high efficiency particulate air (HEPA) filter, one 2-inch deep bank of carbon adsorber, HEPA filter, and fan. Deluge-type water spray systems are provided for flooding the carbon in case of fire or overheating. When a filter train is idle, decay heat is removed by a fan circulating 100 cfm through the charcoal. The equipment and components are designed to seismic Category I, and are located in a seismic Category I structure. During normal operation, these charcoal filter trains are bypassed; they are called into service only on high radiation in the exhaust plenum or on a LOCA signal.

From the system description in the FSAR, we determined that the Fuel Handling Building Ventilation System is designed consistent with GDC's 41, 42, 43, 61, and 64, as referenced in the SRP. In our evaluation of the system post accident design efficiencies for removal of elemental iodine and organic iodines, we assigned the system decontamination efficiencies of 95% for elemental and organic iodines for the carbon adsorbers (one 2-inch deep bed) in accordance with Table 2 of Regulatory Guide 1.52 and 99% for particulates for the HEPA filters. Except for some parameters not provided with recording instruments, provisions for instrumentation, readout, and alarm were determined to be consistent with Table 6.5.1-1 of the SRP. While Flow and pressure drop is not recorded as suggested in Regulatory Guide 1.52, adequate indication and alarms are provided in the control room to assure proper monitoring of system performance.

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6.5.1.4

Deviations from Standard Review Plan

Our review of the above systems has found agreement with the guidelines of the revised Standard Review Plan (SRP), NUREG-0800, with the following exceptions:

Table 6.5.1-1 of SRP 6.5.1 calls for recorded indication of inlet or outlet flow rate of ESF filter system trains, for deluge valve hand switch status indication on a continuously manned control panel, and for indication of total system pressure drop on a continuously-manned control panel.

None of the above design features are incorporated in the River Bend, Unit 1, ESF filter system design. All of these features are design criteria changes incorporated in versions of ANSI N509 and Regulatory Guide 1.52, published subsequent to the staff's construction permit (CP) stage of review. In the implementation of Regulatory Guide 1.52, it has been the staff's policy that design criteria changes which do not materially affect the safety of the plant need not be imposed as "backfits" to the design of plants which have been reviewed and approved for construction under previously existing criteria. The above design features come within the category of features which do not materially affect the safety of the plant and, therefore, the absence of these features from the River Bend, Unit No. 1, design does not constitute

a technical deviation from the criteria of the SRP, since said criteria were promulgated subsequent to staff approval of the design at the CP stage of review.

6.5.1.5

Evaluation Findings

The staff concludes that the design of the ESF atmosphere cleanup systems, including the equipment and instrumentation to control the release of radioactive materials in gaseous effluents following a postulated DBA, are acceptable. This conclusion is based on the applicant having met the requirements of GDC's 19, 41 and 61 by providing ESF atmosphere cleanup systems on the control room habitability, containment and associated systems. The applicant has met the requirements of GDC's 42, 43, and 64 by providing for inspecting and testing the ESF atmosphere cleanup systems and monitoring for radioactive materials in effluents from these systems. In meeting these regulations, the applicant has demonstrated that the design of the ESF atmosphere cleanup systems meet the guidelines of Regulatory Guide 1.52 and the ANSI N509 and N510 industry standards, as referenced in the SRP. We have reviewed the applicant's system descriptions and design criteria for the ESF atmosphere cleanup systems. Based on our evaluation with respect to the SRP criteria, we find the proposed ESF atmosphere cleanup systems are acceptable. The filter efficiencies given in Table 2 of Regulatory Guide 1.52 are appropriate for use in accident analyses.

6.5.2 Containment Spray Systems

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6.5.3. Fission Product Control Systems

Three engineered safety features (ESF) have been designed to control the release of radioactivity to the environment in the event of an accident. Two of these, the standby gas treatment system (SGTS) and the fuel building charcoal filtration system (FBCFS), are filtered exhausts which maintain the secondary containment at pressures below atmospheric pressure. In addition, the main steam positive leakage control system (MS-PLCS) is designed to pressurize isolated sections of the main steam lines. For accidents in which the reactor pressure vessel is depressurized, this latter system prevents leakage through the main steam line isolation valves from escaping into the turbine building. In order to determine the effectiveness of these systems to be used in evaluating accident consequences, their designs were reviewed using Standard Review Plan section 6.5.3.

The River Bend Station secondary containment consists of the auxiliary building, the fuel building, and the annular volume between the steel shell of the primary containment and the shield building which encloses it. Except for the concrete foundation upon which both the primary containment and shield building are supported, the secondary containment completely surrounds the primary containment, with all penetrations passing through the annular volume and either the auxiliary building or the fuel building. Systems of fans, dampers, ductwork, and associated instruments and controls provide for maintaining the pressure

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differential between the annular volume and the remainder of the secondary containment, and for a recirculation flow to assure uniform mixing with the annular volume. Upon actuation, these systems are isolated from the normal plant ventilation, and become part of the SGTS. The ability of these ESF systems to perform as intended has been reviewed earlier in this section.

During normal plant operations the secondary containment is kept at reduced pressure by the normal plant ventilation system. Upon automatic or manual initiation, the SGTS and FBCFS isolate the normal plant ventilation, and continue to maintain a reduced pressure in the secondary containment sufficient to prevent uncontrolled diffusive loss to the environment.

The four main steam lines are the only direct pathways from the primary containment through which leakage to outside the secondary containment can pass. All other pathways either terminate within the secondary containment or are normally isolated. The MS-PLCS, described in earlier in this chapter, acts to prevent leakage via these pathways. As a result, there are no identifiable leakage paths which bypass the secondary containment. We conclude that the design of the ESF systems, as described above, is appropriate for the inclusion of their effective operation in the models used to compute dose consequences in section 15 of this SER.

6.6 Inservice Inspection of Class 2 and 3 Components

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6.7 Main Steam Positive Leakage Control System

The main steam positive leakage control system (MS-PLCS) was reviewed in accordance with Section 6.7 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the bases for our evaluation of the MS-PLCS with respect to the applicable regulations of 10 CFR 50.

6.7.1 Design Bases

The main steam positive leakage control system (MS-PLCS) is designed to control and minimize the release of radioactivity which could leak through the closed MSIVs after a loss-of-coolant accident. The system consists of two separate and redundant subsystems. One subsystem functions to maintain the steam lines between the outboard MSIVs and the inboard MSIVs and the corresponding drain lines at a modest positive air pressure following system actuation. The other subsystem functions to maintain the steam lines between the outboard MSIVs and the main steam line shut-off valves, and the outboard MSIV's drain lines and valve steam packing glands at a modest positive air pressure with respect to reactor vessel pressure following system actuation. Each subsystem receives power from a separate division of the emergency power supplies. Both subsystems are actuated manually and simultaneously.

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The MS-PLCS is supplied with compressed air by two separate and redundant seismic Category I compressed air supply systems which are integral components of the penetration valve leakage control system. These compressed air supply systems are described in Section 9.3.6 of this SER. Each compressor receives power from a separate 1E onsite power supply.

The main steam positive leakage control system is designed to be manually initiated twenty minutes after a postulated design basis loss-of-coolant accident in accordance with Position C.7 of Regulatory Guide 1.96, "Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants." This actuation time period is consistent with loading requirements on the emergency electrical buses, with reasonable times for operator information gathering, decision making and action, and is consistent with the time required for main steam line pressure decay following a postulated loss-of-coolant accident.

6.7.2 System Description

The system is located in the reactor building and auxiliary building which are seismic Category I flood- and tornado-protected buildings. The MS-PLCS itself is designed in accordance with sesimic Category I (Regulatory Guide 1.29, Position C.1) and Quality Group B criteria with the exception that the portion of the MS-PLCS piping that connects to the

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main steam line between the inner and outer containment isolation valves is designed in accordance with seismic Category I and Quality Group A criteria in accordance with the criteria of Position C.1 of Regulatory Guide 1.96. Thus, the MS-PLCS meets the requirements of General Design Criterion 2 since the system is designed in accordance with the requirements of Regulatory Guide 1.29, Position C.1 and Regulatory Guide 1.96, Position C.1. The system would be called on to function in the event of a loss-of-coolant accident (LOCA) and is designed to be capable of performing its safety function under the expected LOCA environmental conditions appropriate to the system equipment location (refer to Section 3.11 of this SER) and in the event of a single active failure, thus complying with Position C.3 of Regulatory Guide 1.96. The components of each subsystem are protected by separation and barriers against potential missiles generated inside and outside of containment, and dynamic effects associated with pipe breaks so that their function would not be impaired under postulated LOCA conditions, thus complying with the criteria of Position C.2 of Regulatory Guide 1.96. In addition, no single active failure can affect the integrity or operability of the main steam lines or main steam isolation valves, thus complying with the criteria of Position C.4 of Regulatory Guide 1.96.

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In this way the MS-PLCS complies with the requirements of General Design Criterion 4, "Environmental and Missile Design Bases."

6.7.3 System Evaluation

The MS-PLCS is capable of performing its function following loss of all offsite power coincident with a design basis LOCA, thus complying with the criteria of Position C.5 of Regulatory Guide 1.96.

The MS-PLCS is designed to control and minimize leakage through the main steam isolation valves for up to 30 days. [However, the applicant has not submitted an acceptable MSIV allowable leak rate.] Therefore, we cannot be sure that enough compressed air can be provided to maintain the desired pressure in the MS-PLCS system. Consequently, we cannot determine that the MS-PLCS complies with Position C.6 of Regulatory Guide 1.96.

6.7.4 Instrumentation Requirements

The instrumentation and circuits necessary for proper operation of the MS-PLCS are designed to comply with standards applicable to engineered safeguards components (Class 1E). In this way the MS-PLCS complies with the requirements of Position C.8 of Regulatory Guide 1.96.

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The MS-PLCS isolation valves do not open in the event the main steam line pressure is higher than the pressure interlock setpoint, thus preventing the MS-PLCS from being damaged by high pressure. A pressure control valve maintains the required pressure differential between the reactor vessel and main steam line. (The applicant must establish this differential pressure.) In the event of high flow or low pressure differential, an alarm is sounded. Further flow increase or drop in pressure differential will result in automatic isolation of that part of the MS-PLCS. This, together with the fact that there are two MS-PLCS systems, prevents direct access of the containment atmosphere to the MS-PLCS. However, we cannot determine that compliance with Position C.9 of Regulatory Guide 1.96 is complete until the applicant determines the required differential pressure.

6.7.5 Inservice Testing

The applicant states that complete functional testing is performed during extended plant shutdown or during refueling. This complies with Position C.10 of Regulatory Guide 1.96.

The MS-PLCS complies with Position C.11 of Regulatory Guide 1.96 since the use of compressed air as the fluid medium does not result in stresses adversely affecting the integrity or operability of the main steam lines or main steam isolation valves nor induce leakage in the main steam isolation valves greater than the capacity of the MS-PLCS.

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The MS-PLCS is designed to control stem leakage from the outboard main steam isolation valves by connecting the MS-PLCS process line to the socket weld provided on the inlet side of the outboard main steam isolation valve, thus complying with Position C.12 of Regulatory Guide 1.96.

The MS-PLCS complies with the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," by meeting the guidelines of Position C.1 of Regulatory Guide 1.29. The MS-PLCS complies with the criteria of General Design Criterion 4, "Environmental and Missile Design Bases," by meeting the guidelines of Positions C.2 and C.4 of Regulatory Guide 1.96. However, we cannot determine whether the MS-PLCS meets the guidelines of Positions C.6 and C.9 of Regulatory Guide 1.96; therefore, we cannot determine whether the MS-PLCS complies fully with the requirements of General Design Criteria 54, "Piping Systems Penetrating Containment." Therefore, the system does not comply with the requirements of SRP Section 6.7.

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Figure 6.1 Reactor building, Mark III containment shield building

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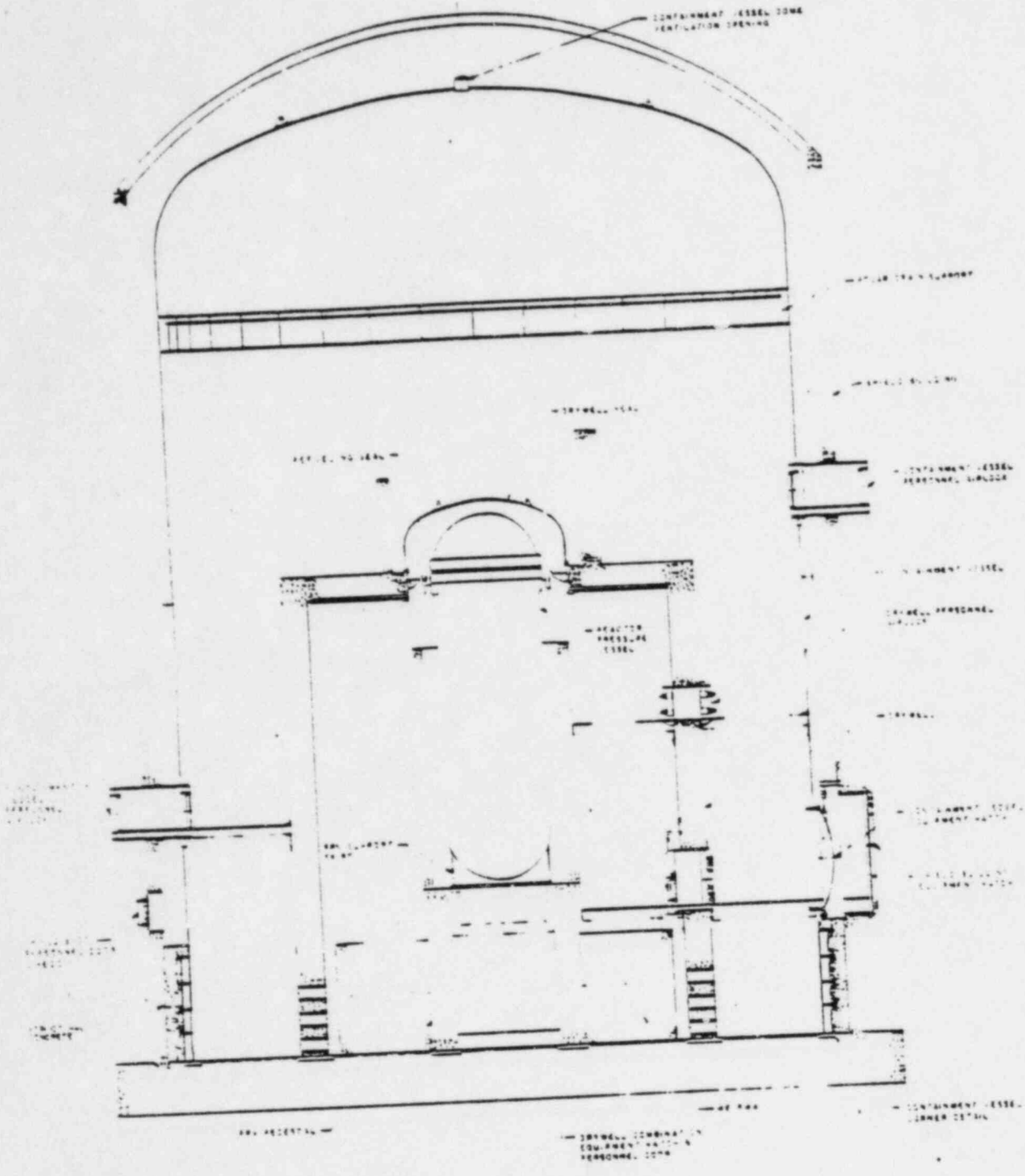


Figure 6.1 Reactor Building (Mark III Containment and Shield Building)

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Figure 6.2 Mark II containment

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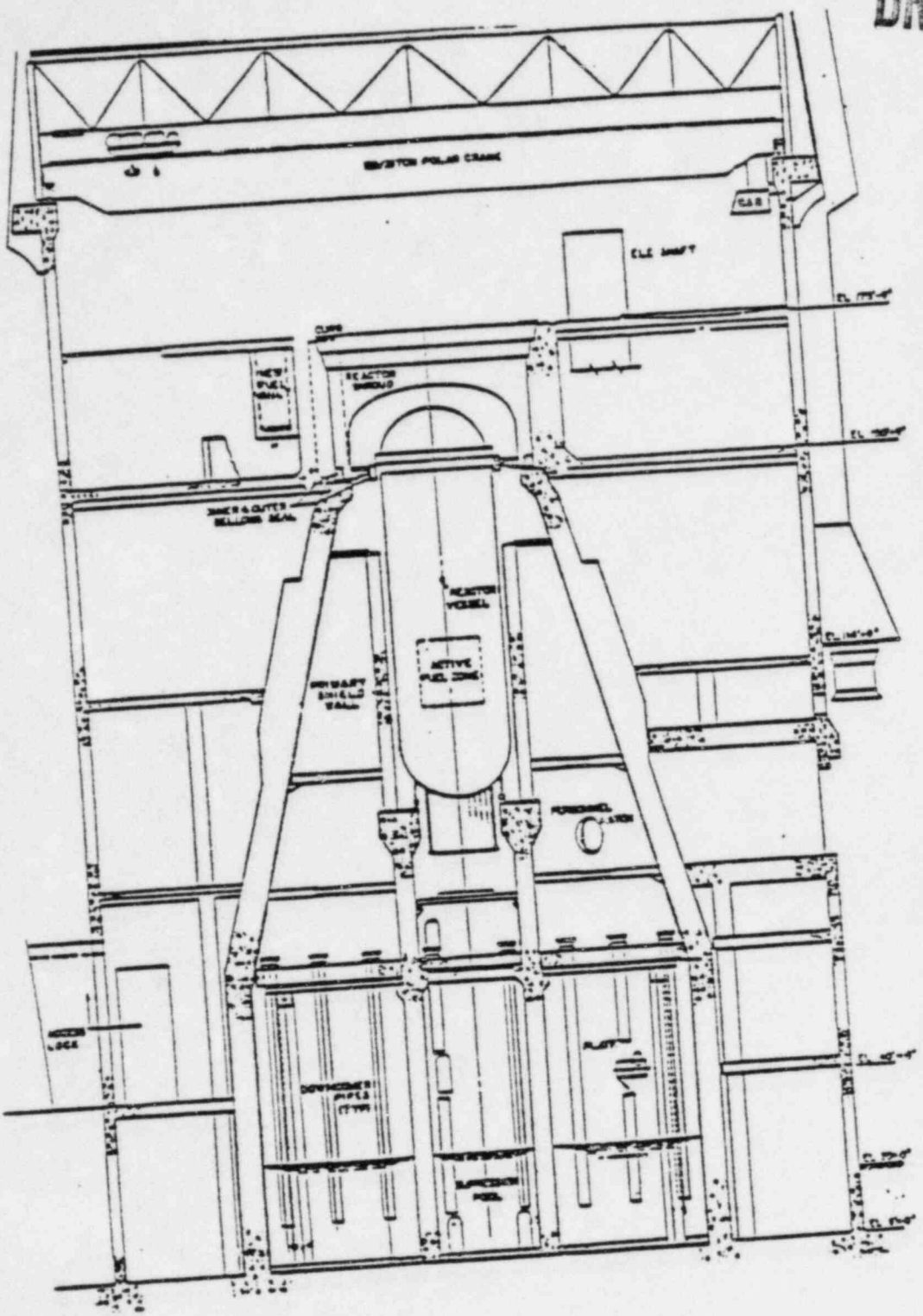


Figure 6-~~12~~ 62 Mark II Containment

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Figure 6.3 Primary and secondary containment systems

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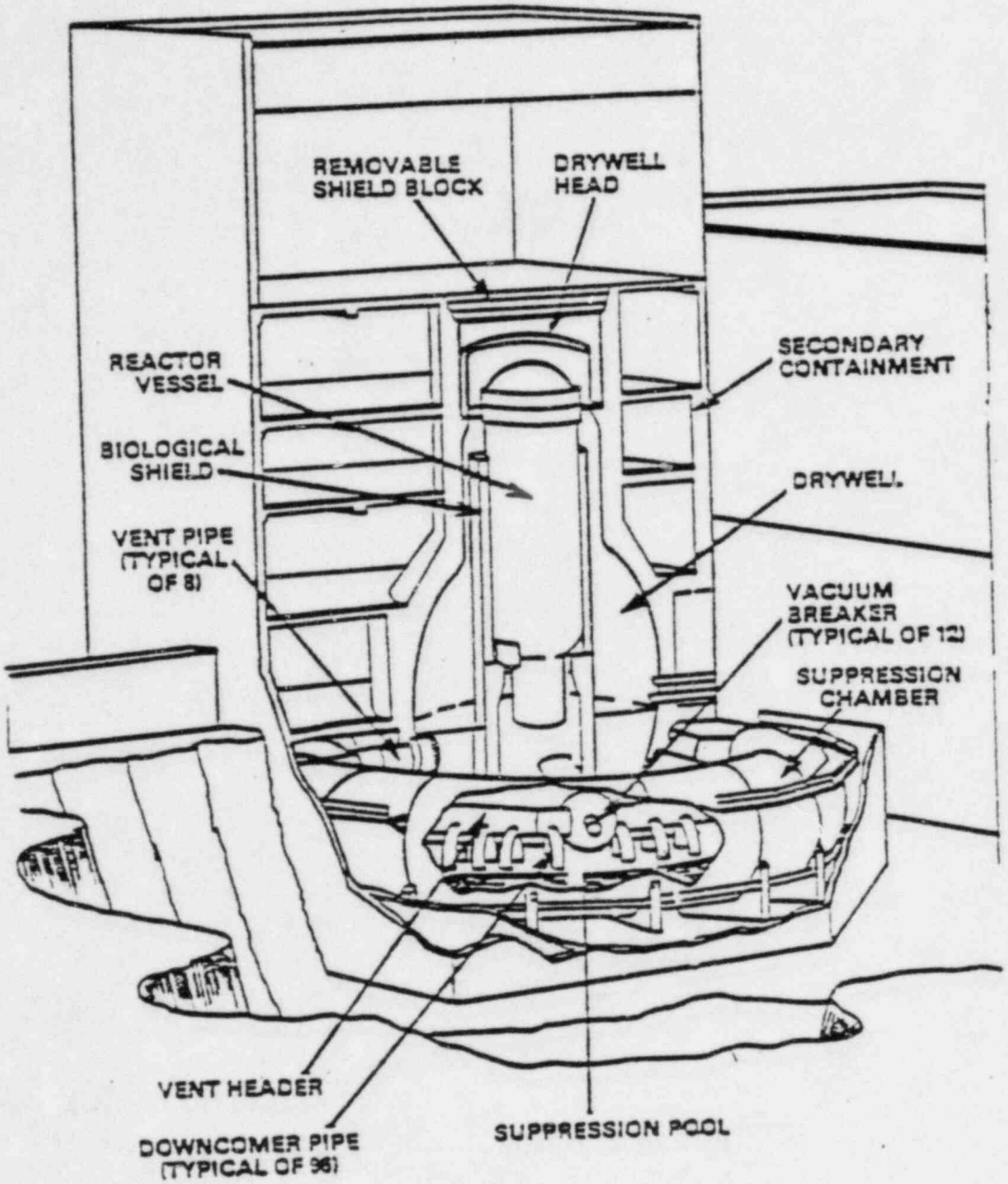


Figure 6.2-1c Primary and Secondary Containment Systems

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Figure 6.4 LOCA chronology (DBA)

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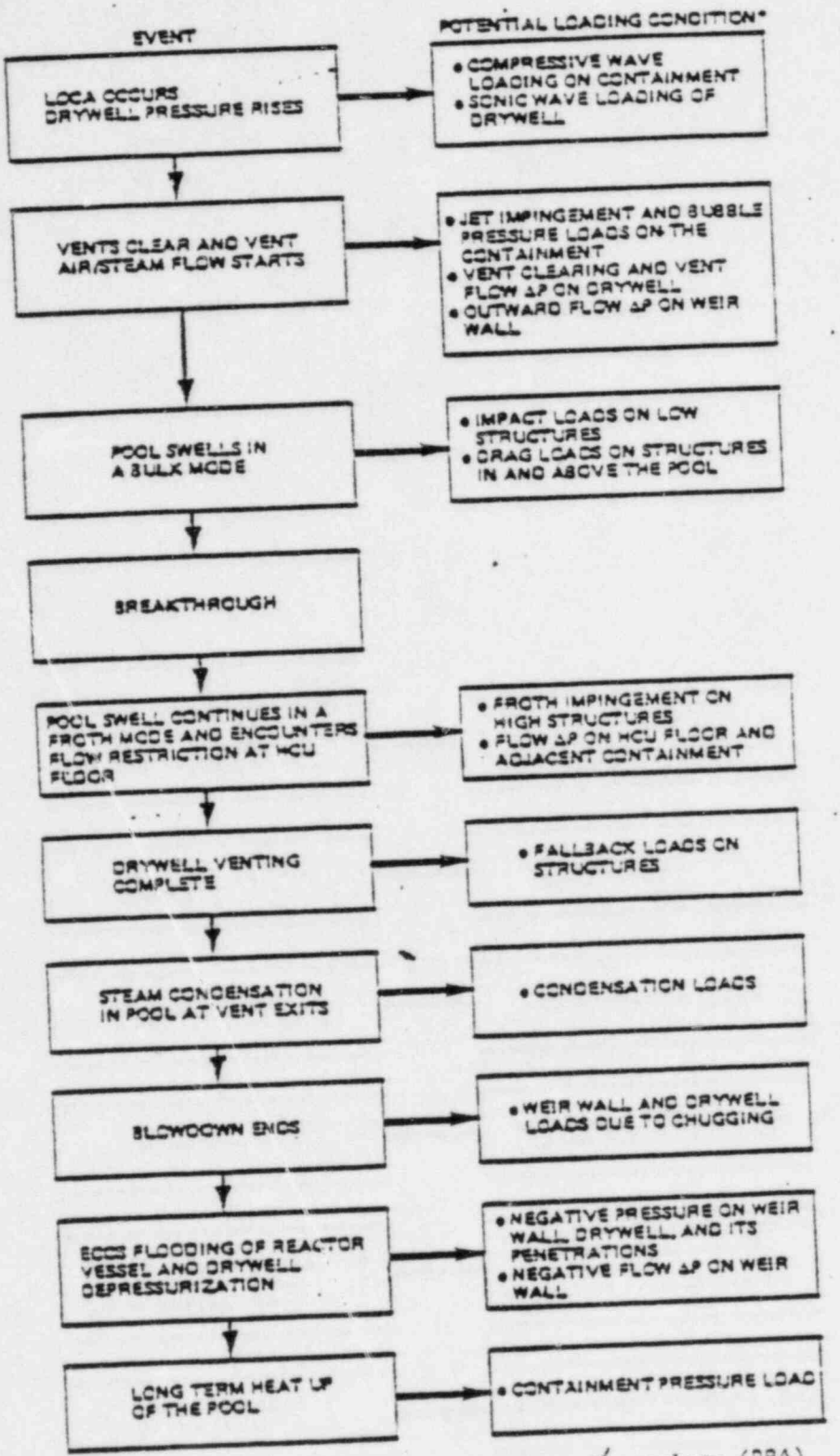


Figure 6-~~7~~ 4 ~~Loss-of-Coolant Accident~~ Chronology (DBA)
LOCA

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Figure 6.5 Impact Force (per unit length) on Wedge-Shaped Protrusions from the Containment Wall

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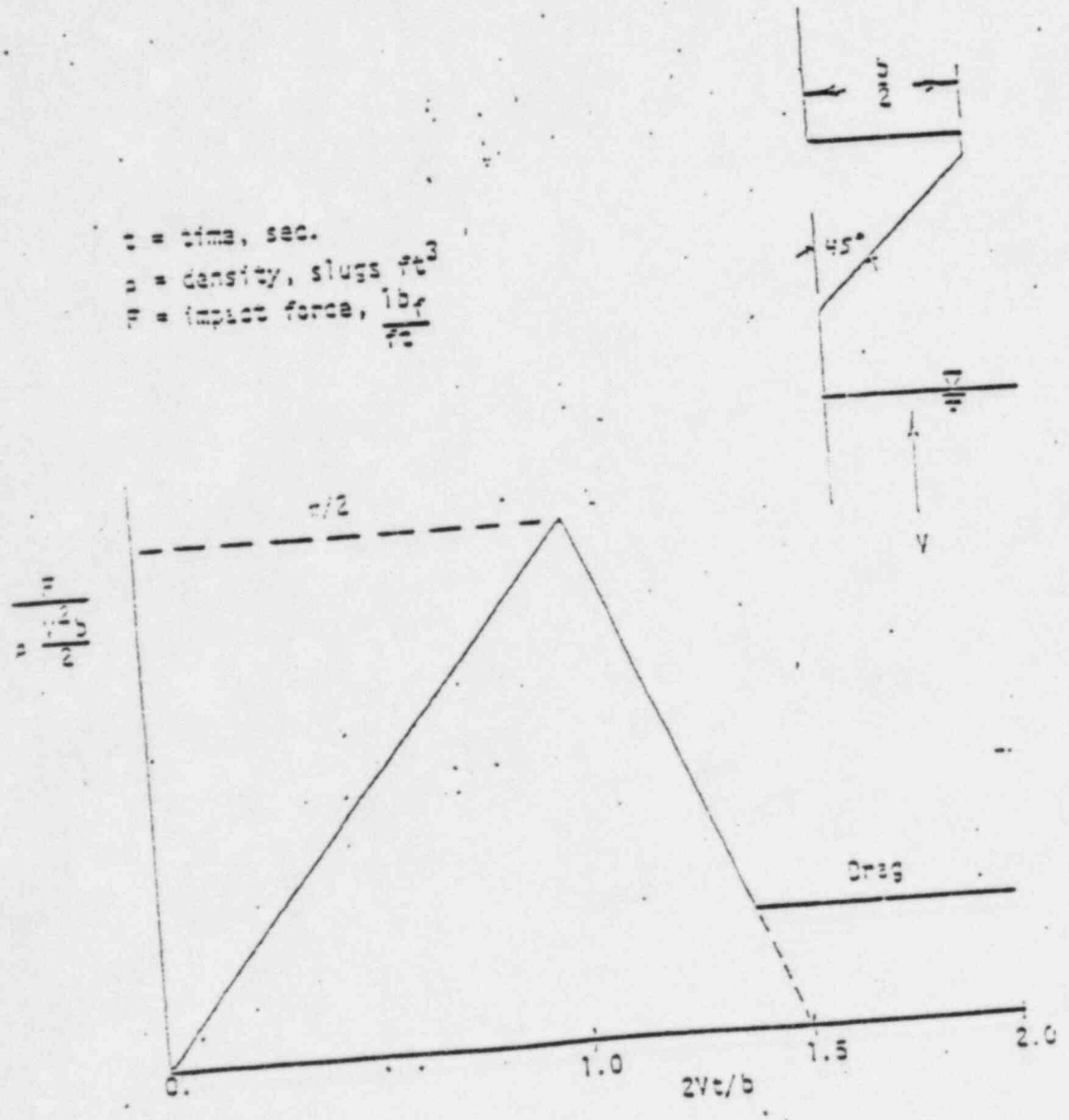


Figure 6.5 Impact Force (per unit length) on Wedge-Shaped Protrusions from the Containment Wall

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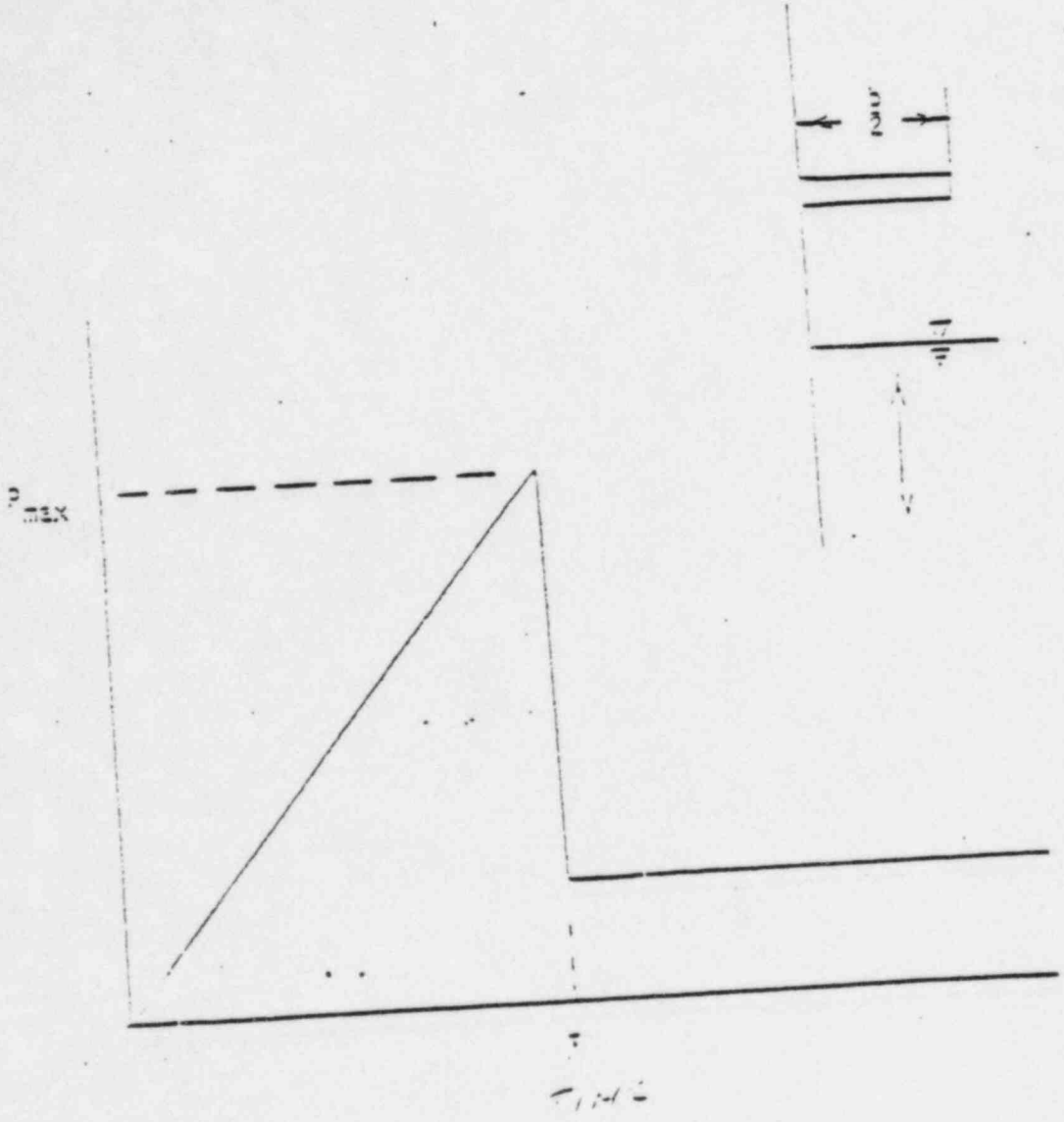


Figure 6.6 Impact Pressure on Horizontal Ledges Attached to Containment Wall

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Figure 6.6 Impact Pressure on Horizontal Ledges Attached to Containment Wall

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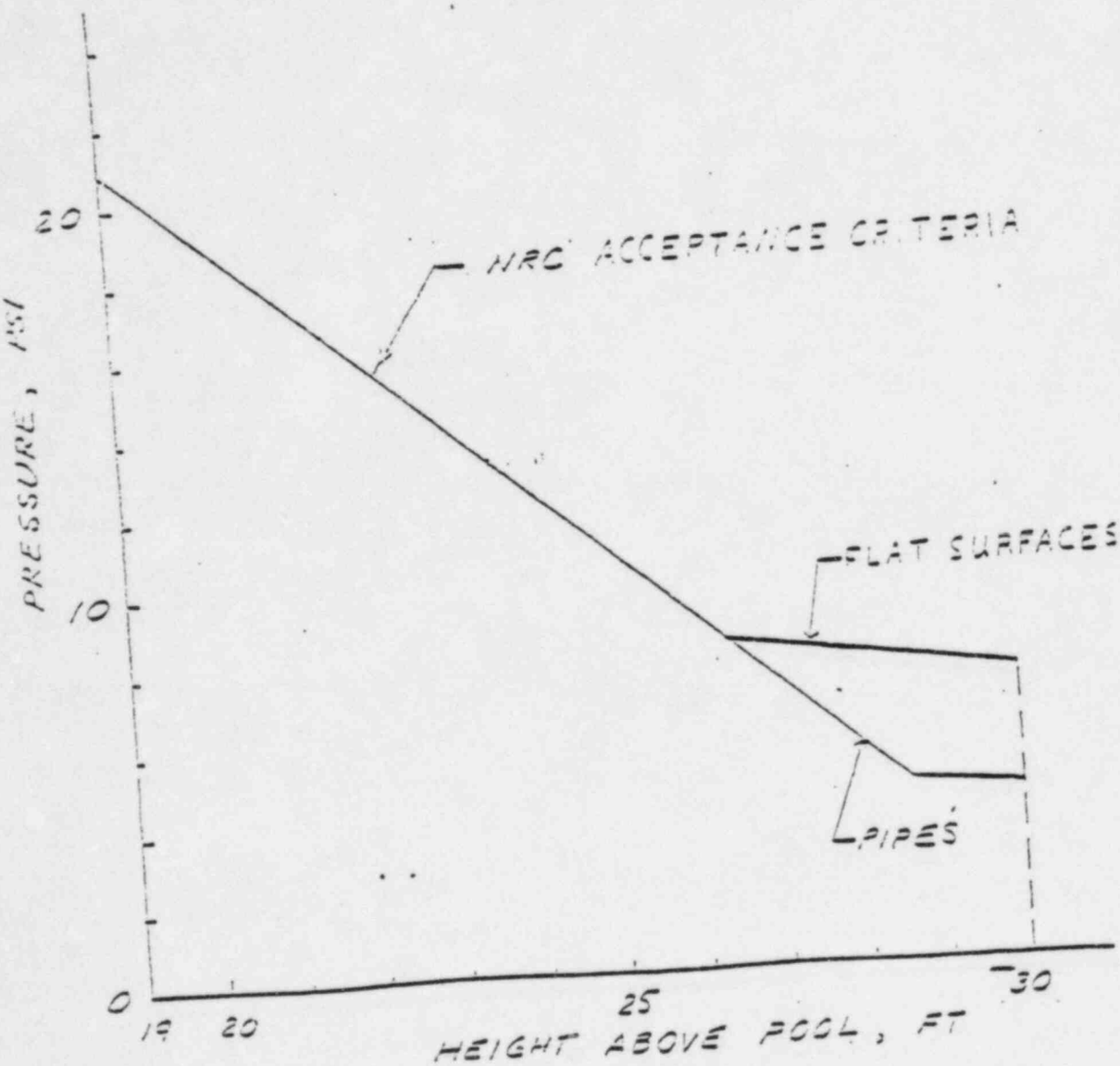


Figure 6.2-1 NRC Acceptance Criteria for Forth Impact: Peak Amplitude of Pressure Pulse

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Figure 6.7 NRC Acceptance Criteria for Tornado Impact: Peak Amplitude of Pressure Pulse

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Table 6.1 Most limiting combinations of break size and location and single failure

Break size	Break location	Single failure	Systems remaining
Small	Recirculation suction line	HPCS	ADS, LPCS, and all LPCI
Intermediate and large	Recirculation suction line	LPCI diesel generator	HPCS, ADS, LPCS, and one LPCI

just T6.2 line

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Table 6.2 Lead plant LOCA analysis results

Parameter	Maximum values from break analyses	Allowable
Peak cladding temperature	2062°F	2200°F
Maximum cladding oxidation	1.60%	17%
Maximum total hydrogen generation	0.11%	1%

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9 Auxiliary Systems

We have reviewed auxiliary systems in the River Bend Station necessary to assure the safety of the stored fuel, auxiliary systems needed for transient or accident mitigation or for safe plant shutdown and auxiliary systems not required for transient or accident mitigation or for safe plant shutdown. The latter have been reviewed to assure that their failure will not prevent safe shutdown or the mitigation of transients and accidents.

The systems reviewed to assure the safety of the stored fuel included both new and spent fuel storage, fuel pool cooling and cleanup, light load and overhead heavy load handling systems.

Systems required to function in the event of an accident or to aid in plant shutdown include the following systems:

(1) Reactor plant component cooling water*,
(2) Ultimate heat sink,
(3) Standby service water,
(4) Control building chilled water,

*A portion of these systems are not required for safe shutdown or mitigation of transients or accidents; these portions are reviewed to assure that their failure will not adversely affect proper functioning of systems required for such purpose.

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(5) Compressed gas systems,*

(6) Standby liquid control system,

(7) Control room heating, ventilating and air conditioning,*

(8) Spent fuel pool heat ventilating and air conditioning,*

(9) Auxiliary and radwaste area ventilation,*

(10) Engineered safety features ventilation

Those systems which have been reviewed to assure that their failure would not affect systems required for safe shutdown and/or accident or transient mitigation include:**

(1) Normal service water,

(2) Makeup water treatment,

(3) Domestic water and sanitary drains and disposal,

(4) Condensate storage facilities,

(5) Turbine plant component cooling water,

(6) Ventilation chilled water,

*A portion of these systems are not required for safe shutdown or mitigation of transients or accidents; these portions are reviewed to assure that their failure will not adversely affect proper functioning of systems required for such purposes.

**Some of these systems have interfaces with safety systems and/or containment penetrations which are required to be safety-related to provide suitable isolation.

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(7) Cooling tower makeup water,

(8) Equipment and floor drainage,

(9) Turbine building ventilation,

(10) Miscellaneous buildings heating, ventilating and air conditioning.

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9.1 Fuel Storage ~~facility~~ and Handling

9.1.1 New Fuel Storage

The new fuel storage facility was reviewed in accordance with Section 9.1.1 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the plans for new fuel storage with respect to the applicable regulations of 10 CFR 50.

The new fuel storage facility consists of a new fuel storage vault located in the fuel building of each unit. Each storage vault provides dry storage for a maximum of 220 fuel assemblies (30% of a core load) and includes the new fuel assembly storage racks and the concrete storage vault that contains the storage racks. There is no sharing of storage space. Therefore, the requirements of General Design Criterion 5, "Sharing of Structures, Systems, and Components" are not applicable.

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The fuel building which houses the facility, the storage racks and vault is designed to seismic Category I criteria. This building is also designed against flooding and tornado missiles (refer to Sections 3.4.1 and 3.5.2 of this SER). Thus, the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification," are satisfied.

The vault housing the new fuel storage racks is not located in the vicinity of any moderate or high energy lines or rotating machinery. This separation from such potential missile sources protects the new fuel from internally generated missiles and the effects of pipe breaks (refer to Section 3.5.1.1, 3.5.1.2 and 3.6.1 of this SER).

The facility is designed to store unirradiated, low emission fuel assemblies. Accidental damage to the fuel would release relatively minor amounts of radioactivity that would be accommodated by the spent fuel area ventilation system. Thus, the requirements of General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control," are satisfied.

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The racks are designed to preclude the inadvertent placement of a new fuel assembly in other than the prescribed spacing - a center-to-center spacing of 12.25 inches between rows, 7.00 inches within rows. This spacing permits the stored new fuel to be maintained with K_{eff} equal to or less than 0.95 when dry or completely flooded with water. The applicant noted, however, that the new fuel racks were not designed for optimum moderation; for example, foam or mist. As a result, a K_{eff} less than or equal to 0.98 cannot be assured with a completely filled array of fuel assemblies in the racks with the highest anticipated reactivity. Therefore, the applicant has proposed solid, noncombustible [redacted] covers over the fuel and administrative controls to preclude the entrance of foam or mist which [redacted] could result in a K_{eff} greater than 0.98. [The applicant must provide assurance that K_{eff} in excess of 0.98 will not be attained in the new fuel storage racks. The nature of the administrative controls and the times and conditions under which the new fuel covers will be removed should be addressed.]

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Based on our review, we conclude that the new fuel storage facility is in conformance with the requirements of General Design Criteria 2, 5, 61, as they relate to new fuel protection against natural phenomena, missiles, shared functions, and radiation protection and the guidelines of Regulatory Guide 1.29 relating to seismic classification. The applicant must provide further information regarding controls used to prevent conditions of optimum moderation in order to show compliance with General Design Criterion 62, "Prevention of Criticality in Fuel Storage and Handling."

Therefore, we cannot find that the applicant's submittal regarding new fuel storage is acceptable. The design for new fuel storage does not comply with the acceptance criteria of SRP Section 9.1.1.

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9.1.2 Spent Fuel Storage

The spent fuel storage facility was reviewed in accordance with Section 9.1.2 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed, according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the spent fuel storage with respect to the applicable regulations of 10 CFR 50.

The acceptance criteria for the fuel storage facility includes meeting the guidelines of ANS 57.2, "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations." The guidelines contained in the "Review Procedures" were used in lieu of ANS 57.2. Additionally, the acceptance criteria includes Regulatory Guide 1.115, Protection Against Low-Trajectory Turbine Missiles;" turbine missiles are evaluated separately in Section 3.5.1.3.

Each unit of the plant has its own spent fuel storage facility. Low density storage racks located in the dryer storage pool (one of the four containment pools) of each unit have a capacity of 200 fuel assemblies (30% of a full core). The high density storage racks in the fuel building spent fuel storage pool have a capacity of 3,288 fuel assemblies (525% of a full core);

~~the criteria of Paragraph 5.1.15 of ANS 57.2, "Design Objectives for Light Water Spent Fuel Storage Facilities at Nuclear Power Plants."~~

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The structures housing the fuel building and containment are designed to seismic Category I criteria as are the storage racks and storage pools, pool liners, and gates. These structures are also designed to withstand flooding and tornadoes (refer to Sections 3.4.1 and 3.5.2 of this SER). We conclude that the spent fuel storage satisfies the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," by compliance with the guidelines of Regulatory Guides 1.13, "Spent Fuel Storage Facility Design Basis," and 1.29, "Seismic Design Classification."

The spent fuel pools are not located in the vicinity of any high-energy lines or rotating machinery. Therefore, physical protection by means of separation is utilized to protect the spent fuel from internally generated missiles and the effects of pipe breaks (refer to Sections 3.5.1.1 and 3.6.1 of this SER). The spent fuel building provides protection against tornado missiles (see Section 3.5.2 of this SER). Thus, the requirements of General Design Criteria 4, "Environmental and Missile Design Bases," and the guidelines of Regulatory Guide 1.17 are satisfied.

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There is no sharing of storage facilities. Therefore, the requirements of General Design Criterion 5, "Sharing of Structures, Systems, and Components," are not applicable.

The applicant stated that both the low density storage area (The containment pool) and the high density (the fuel building pool) storage area of the facility are each designed to store the fuel assemblies in an array which limits K_{eff} to 0.95 or less, in order to comply with the requirements of General Design Criterion 62, "Prevention of Criticality in Fuel Storage and Handling." [The applicant has not provided a criticality analysis to confirm the criticality limits to be attained in the spent fuel storage facility.] The low density storage racks are aluminum with a fuel assembly nominal center-to-center storage spacing of 7 inches. The high density racks are stainless steel with a neutron poison material between storage spaces and provide a fuel assembly minimum center-to-center storage spacing of 6 1/4 inches. The racks are designed to preclude the inadvertent placement of a fuel assembly in other than the prescribed spacing. The racks can withstand the impact of a dropped fuel assembly without unacceptable damage to the stored fuel and can withstand the maximum uplift forces exerted by the fuel handling machine. Thus, the requirements of General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control," and the guidelines of Regulatory Guide 1.13 concerning fuel storage facility design are satis-

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fied. The design of the storage pools includes leakage detection systems for indication of pool liner leakage, water level monitoring systems, and radiation monitoring systems. These features satisfy the requirements of General Design Criterion 63, "Monitoring Fuel and Waste Storage." Further discussion of compliance with Position C.3 of Regulatory Guide 1.13 regarding crane interlocks is contained in Section 9.1.4 of this SER.

Based on our review, we conclude that the spent fuel storage facility is in conformance with the requirements of General Design Criteria 2, 4, 5, 61 and 63 with respect to protection of spent fuel from the effects of natural phenomena, missiles, environmental effects, shared functions, prevention of unacceptable radioactivity releases, and monitoring of the facility, and the guidelines of Regulatory Guides 1.13, 1.29 and 1.117 relating to the facility's design, seismic classification and protection against tornado missiles. [However, until the applicant provides a criticality analysis we cannot determine whether the spent fuel storage is in compliance with General Design Criterion 62.]

In view of the foregoing, we cannot find that the applicant's submittal regarding spent fuel storage is acceptable nor that the design for spent fuel storage complies with the criteria of SRP Section 9.1.2.

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9.1.3 Spent Fuel Pool Cooling and Cleanup System

The fuel pool cooling and cleanup system was reviewed in accordance with Section 9.1.3 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed, according to the guidelines provided in the "Review Procedures" portion of the SRP Section. Conformance with the acceptance criteria formed the basis for our evaluation of the fuel pool cooling and cleanup system with respect to the applicable regulations of 10 CFR 50.

The fuel pool cooling and cleanup system is designed to maintain water quality and clarity and remove decay heat generated by spent fuel assemblies in the pools. The system includes all components and piping from inlet to exit from the storage pools, piping used for fuel pool makeup, and the cleanup/filter/demineralizers to the point of discharge to the radwaste system. The fuel pool cooling system consists of two 100%-capacity fully redundant fuel pool cooling pump/heat exchanger trains while the cleanup system consists of one cleanup system train

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containing two 50% capacity pumps, two 50% capacity filters and one 100% capacity mixed bed demineralizer with post-demineralizer strainer.

The fuel pool cooling system is housed in the seismic Cagegory I, flood and tornado protected fuel building and containment (refer to Sections 3.4.1 and 3.5.2 of this SER). The cooling system trains are completely separate from the cleanup train and are designed to Quality Group C and seismic Category I requirements. The cleanup train is designed to Quality Group D and nonseismic requirements. It is located to assure that failure in any portion of the train would have no adverse effect on safety-related equipment. This design satisfies the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," by compliance with the guidelines of Regulatory Guide 1.13 (Positions C.1 and C.2), "Spent Fuel Storage Facility Design Bases," 1.26 (Position C.2), "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," and 1.29 (Positions C.1 and C.2), "Seismic Design Classification."

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The spare train will be operated periodically in accordance with plant Technical Specifications. Thus, the requirements of General Design Criteria 45, "Inspection of Cooling Water System," and 46, "Testing of Cooling Water System," are satisfied.

The nonsafety-related reactor plant component cooling water system provides cooling water to the fuel pool heat exchanger under normal conditions. The maximum normal heat load is based on storage of 4.25 spent cores. Space for an emergency core off-load is available. Under normal refueling conditions, a refueling batch will be placed in the spent fuel pool 150 hours after reactor shutdown. Using the reactor plant component cooling water system, the pool temperature would be maintained at or below 129°F with one pump and one fuel pool cooler in operation which is acceptable since it is less than our acceptance criterion of a maximum temperature of 140°F for a "normal heat load."

Under abnormal heat load conditions, the reactor plant component cooling water (RPCCW) system provides cooling

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water to the fuel pool heat exchangers. The maximum abnormal heat load is based on a full core off-load 10 days after the last normal refueling outage and a maximum storage load of 4.25 cores (maximum storage conditions). Under these conditions, the cooling system will maintain the temperature of the water at or below 156°F. If the reactor plant component cooling water system is not available, the safety-related standby service water system may be used to cool the heat exchangers through the portion of the RPCCW piping to the standby service water system which is designed to seismic Category I and Quality Group C standards for this purpose. While the FSAR states that Branch Technical Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling," was used to calculate the heat loads, this cannot be verified because the size of the refueling batch and frequency were not given by the applicant. (The applicant should provide the number of fuel assemblies and frequency assumed for a "normal" refueling so that the heat load calculations for the spent fuel pool may be verified.) (The applicant should also show that spent fuel stored in

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the dryer storage pool can be cooled in the event of DBA or loss of offsite power combined with the worst single active failure.)

The residual heat removal (RHR) system can be utilized to augment the fuel pool cooling system. (However, the applicant has not shown the interconnection of the RHR system with the fuel pool cooling system.) We, therefore, cannot verify that the spent fuel pool cooling systems meet the requirements of General Design Criterion 44, "Cooling Water," nor whether the interconnecting piping is of the proper seismic category, in accordance with the criteria of Regulatory Guide 1.29, "Seismic Design Classification."

All lines that connect to the pool and extend below the safe level of the pool water (10 ft. above the top of the fuel) are equipped with syphon breakers, check valves or other means to prevent inadvertent pool drainage. (The applicant must show that the syphon breaker design precludes failure so as to cause a reduction in the level of the spent fuel pool in the fuel building below 10 ft. above the top of the fuel.)

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[The applicant should explain why syphon breakers are not required for the dryer storage pool.] Both of these concerns must be addressed adequately in order to show compliance with Position C.6 of Regulatory Guide 1.13. Normal makeup to the pool is provided by the nonsafety-related condensate and refueling water storage and transfer system to replace losses due to leakage through the liner and evaporation. Emergency makeup is supplied by the redundant loops of the safety-related standby service water system.

The nonsafety-related spent fuel pool cleanup subsystem is designed to maintain water quality in both spent fuel storage pools by filtration and demineralization; the system can filter, simultaneously, the water in both pools at a rate sufficient to change the fuel building coolant volume twice per day and the containment pools water once per day. The cleanup subsystem is designed to remove radionuclides to maintain a safe working level in the associated fuel building areas while retaining the radioactive material in the demineralizer and filter so as to comply with Positions C.2.f(2) and (3) of Regulatory Guide 8.8.

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Thus, the requirements of General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control," are met by compliance with the guidelines of Regulatory Guide 1.13.

The system incorporated control room alarmed pool water level, water temperature, and building radiation level monitoring systems, thus satisfying the requirements of General Design Criterion 63, "Monitoring Fuel and Waste Storage."

Based on our review, we conclude that the spent fuel pool cooling and cleanup system is in conformance with the requirements of General Design Criteria 2, 4, 45, 46, 61, and 63 and the guidelines of Regulatory Guides 1.13 and 1.26 and BTP ASB 9-2 with respect to protection against natural phenomena, missiles, pipe break effects, inservice inspection, functional testing, radiation protection and performance monitoring, and quality group classification. The applicant has not presented sufficient information to enable us to determine whether the system is in conformance with the requirements of General Design

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Criteria 44 and 61 and the guidelines of Regulatory Guide 1.29 with respect to its ability to remove decay heat and its seismic classification. Therefore, we cannot conclude that the fuel pool cooling and cleanup system complies with the acceptance criteria of SRP Section 9.1.3.

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9.1.4 Fuel Handling System ~~(Sections Related to Standard Review Plan Section 9.1.4, "Light Load Handling System")~~

The fuel handling system was reviewed in accordance with Section 9.1.4 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed, according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the fuel handling system with respect to the applicable regulations of 10 CFR 50. Note that a light load is considered any load of less weight than a fuel assembly and its handling tool.

The fuel handling system provides the means of transporting, handling, and storing fuel (both new and spent fuel) in the fuel buildings and containment. The fuel handling system consists of equipment necessary to facilitate the periodic refueling of the reactor. The transfer of new fuel assemblies between the uncrating area and the new fuel storage vault is accomplished using the 15-ton fuel building crane. The auxiliary hoist on the fuel handling platform is used with an

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auxiliary fuel grapple to transfer the new fuel from the storage vault to the new fuel storage pool. From there the fuel is handled by telescoping grapples on the fuel handling platform (which is provided for fuel building fuel movement and servicing) or on the refueling platform (which is provided for containment building fuel movement and servicing). The inclined fuel transport system is used to move new or spent fuel, control rods and other small components between containment and the fuel building.

The transfer operation (when utilizing the inclined fuel transfer system) is an automatic sequencing function with capability for manual override. Interlocks assure the correct sequencing of the transfer operation in the automatic or manual mode. Additional interlocks prevent the refueling platform and the fuel handling platform from moving in the transfer area during operations of the transfer system which would be adversely affected by the presence of either platform. Compliance of the portion of the transfer tube serving for containment isolation with the guidelines of Regulatory Guide 1.29 is discussed in Section 6.2 of this SER. The

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refueling platform is a gantry crane used to move fuel and reactor components to and from containment fuel storage and the reactor vessel. The fuel handling platform is a gantry crane used to move fuel within the fuel building storage pool. The applicant stated that the fuel grapple hoist on the refueling platform and both the main fuel hoist and monorail auxiliary hoist on the fuel handling platform are designed so that no single component failure can result in dropping a fuel bundle. During transfer of fuel the grapple on each platform provides a minimum water shielding of 8 ft. 6 inches over the active fuel when in the fully retracted position. Limit switches on the end trucks minimize the possibility of having the fuel handling platform run into pool obstacles.

A fuel building bridge crane is provided to handle the transport of new fuel between the receiving areas and the new fuel inspection stand and/or new fuel storage vault. The refueling and fuel building platforms are designed to seismic Category I and Quality Group B standards. The fuel building bridge crane, however, is not designed to seismic Category I standards but does have seismic restraints which are designed to

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prevent the bridge and trolley from leaving the rails during a seismic event. During construction the rails extend over the spent fuel pool for placement of the high density fuel racks; after construction the fuel building bridge crane will be prevented from moving over the spent fuel by physical blocks. [Nevertheless, because of the proximity of the crane to the spent fuel pool, the applicant should show that damage to the crane is sufficiently limited in the event of an SSE so that either the crane retains its integrity or failing that, that any parts of the crane becoming dislodged would not result in damage to the spent fuel, the spent fuel pool, or the spent fuel cooling system in order to meet the criteria of Position C.2 of Regulatory Guide 1.29.] The fuel handling equipment is not required to function after an SSE. The jib crane which is mounted along the edge of the containment fuel pool is used to aid in fuel movement in the containment by serving in the fuel preparation area, thus leaving the refueling platform free for use in fuel shuffling. It is designed in accordance with seismic Category I criteria as is the stand for the inspection of new fuel.

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Each unit of the plant has its own fuel handling system and there is no sharing of facilities. Thus, the requirements of General Design Criterion 5, "Sharing of Structures, Systems and Components," are not applicable.

The entire system is housed within the fuel building and the containment which are seismic Category I flood and tornado protected structures (refer to Sections 3.4.1 and 3.5.2 of this SER). This meets the guidelines of Position C.1 of Regulatory Guide 1.29, "Seismic Design Classification," relating to protection of safety-related equipment and spent fuel from the effects of an earthquake. Failure of this equipment will not affect spent fuel or safety-related equipment, with the exception of the concern regarding failure of the fuel building bridge crane. Compliance of the design of the spent fuel storage facility with seismic Category I criteria as required by Position C.1 of Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Bases" is discussed in Section 9.1.3 of this SER. In view of our concern that a failure of the building bridge crane during an SSE could result in

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damage to spent fuel, to the spent fuel pool or to the spent fuel cooling system, we cannot conclude that the light load handling system complies with the provisions of Position C.2 of Regulatory Guide 1.29 nor the criteria of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

[The applicant should discuss what provisions are made for cooling the maximum number of spent fuel assemblies in the fuel transfer tube during normal transfer operations and in the event subassemblies become lodged in the tube. The possibility of coolant drainage from the tube, resulting in a dry tube should also be discussed, in order to show that this meets the guidelines of Section 6.2.4.1.13 of ANSI 57.1 and Position C.3 of Regulatory Guide 1.13, in compliance with the provisions of General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control."]

The applicant notes (in FSAR Section 9.1.2) that the design of the containment spent fuel storage facility provides a subcritical multiplication factor ($K_{eff} \geq$ 0.95) for abnormal storage which includes dropping of equipment; however, for the new fuel building spent

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storage facility, the applicant has not yet provided a criticality analysis (see Section 9.1.2 for this concern).

The applicant has responded to our concern regarding elimination of the possibility of handling a load with the light load handling system lighter than a fuel assembly with its handling tool but raised to a greater height so that its kinetic energy would exceed that of the fuel assembly and handling tool. The applicant stated that control would be by administrative procedures and training. [In addition, however, the applicant must provide an analysis to show that the maximum kinetic energy resulting from a fall of any object handled by the light load handling system when over spent fuel in either containment or fuel building storage facilities will not exceed that obtained in the fall of a fuel assembly and its handling tool.]

In view of the foregoing, we conclude that the fuel handling system is in conformance with the requirements of General Design Criterion 5. However, we cannot conclude that the system is in conformance with the

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requirements of General Design Criterion 2, 61, and 62. Therefore, we cannot find the applicant's submittal to be acceptable. The portion of the fuel handling system related to handling of light loads does not comply with the acceptance criteria of SRP Section 9.1.4.

Overhead Heavy Load

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9.1.5 ~~Fuel Handling System (Sections Related to Standard Review Plan 9.1.5, Overhead Heavy Load Handling System)~~

The fuel handling systems (parts related to overhead handling of heavy loads) was reviewed in accordance with Section 9.1.5 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the bases for our evaluation of the fuel handling system (parts related to overhead handling of heavy loads) with respect to the applicable regulations of 10 CFR 50.

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The acceptance criteria for the overhead heavy load handling system include meeting the guidelines of ANS 57.1 and 57.2. The guidelines in the SRP review procedures and NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," were used in lieu of ANS 57.1 and 57.2.

NUREG-0612 was transmitted to the applicant for action by generic NRC letters dated December 22, 1980 and February 3, 1981. NUREG-0612 resolved Generic Task A-36 and provides guidelines for necessary changes to ensure safe handling of heavy loads once a plant becomes operational. Enclosure 2 of the December 22, 1980 generic letter identified a number of interim measures dealing with safe load paths, procedures,

operator training, and crane inspections, testing, and maintenance. The applicant committed to comply with these interim actions. [The staff will require that the applicant implement the ~~final~~ guidelines contained in NUREG-0612 before any operating license can be issued.]

The applicant has provided a discussion only of the portion of the heavy load handling system dedicated to the handling of fuel; thus is reviewed below.

Each unit of the station has its own polar crane and spent fuel cask trolley and there are no shared components. Thus, the requirements of General Design Criterion 5, "Sharing of Structures, Systems, and Components," are not applicable.

The spent cask fuel trolley and polar crane are housed within the fuel building and containment, respectively, both of which are seismic Category I, flood and tornado protected structures (refer to Sections 3.4.1 and 3.5.2 of this SER). The 125 ton spent fuel cask trolley is used for handling the spent fuel shipping cask, and is designed to seismic Category I requirements. The containment polar crane is used to move the portable refueling shield, drywell head, reactor vessel head, steam separator, and steam dryer, and is designed to seismic Category I requirements. Therefore, the design meets the guidelines of Position C.1 of Regulatory Guide 1.29, "Seismic ~~Design~~ Design Classification."

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The portion of the handling system dedicated to handling heavy loads consists of equipment necessary for the safe handling of the spent fuel cask and for safe disassembly and reassembly of the reactor vessel head and internals during refueling operations. The containment polar crane is used for handling of heavy loads in containment and the spent fuel cask trolley is used for handling of heavy loads in the fuel building.

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The spent fuel cask pool is separated from the fuel storage pool by a common wall with a gate. The spent fuel cask trolley rails do not extend over any portion of the spent fuel storage pool or transfer pool, thereby preventing cask transportation over spent fuel. A dropped cask cannot, therefore, result in damage to spent fuel stored in the fuel pool. The trolley coverage area does not include any area over safety-related equipment. Procedures and design limitations prevent the cask from being lifted more than 30 ft. [The applicant is currently performing a spent fuel cask drop analysis ~~and polar crane load drop analysis~~ to verify that a cask drop ~~or load drop from the polar crane~~ does not result in unacceptable damage to the spent fuel storage facility or to safety-related equipment. Therefore, the cask trolley meets the guidelines of Position C.3 of Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis." (However, the applicant should verify that the polar crane will not be moved over stored fuel when refueling is not in progress, to meet the guidelines of Position C.3). (In addition, the applicant should show that the polar crane meets the guidelines of Position C.5 of Regulatory Guide 1.13 with regard to the effect of a load drop on spent fuel in the containment pool in order to comply with with requirements of General Design Criterion 4, "Environmental and Missile Design Bases." In this analysis the applicant should also consider the effect of a polar crane load drop on the level of water in the containment

spent fuel storage pool in order to meet the guidelines of Position C.6 of Regulatory Guide 1.13 so as to comply with the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."] Note that the polar crane must meet the guidelines of Positions C.3 and C.5 of Regulatory Guide 1.13 in order to comply with the criteria of General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control."

The applicant has made calculations to show the results of a drop of the reactor vessel head onto the vessel head flange; the calculations show that there is no release of radioactivity or production of vessel leaks. [The applicant should show the results of calculations of other polar crane load drops (where the loads could drop into the vessel) to determine their effect on spent fuel in the core.)

Based on the foregoing, we conclude that the portion of the heavy loads handling system dedicated to fuel handling complies with the requirements of General Design Criterion 2 with regard to protection against seismic events. The applicant has not shown that the system complies with the requirements of General Design Criteria 2 (in maintaining containment pool water level in the event of a polar crane load drop, and 4 (with respect to dropped polar crane loads acting as missiles), 61 (relating to release of ~~release of~~ radioactivity). In addition, the applicant must implement the ~~final~~ guidelines of NUREG-0612 before an operating

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license can be issued. Therefore, we cannot find the applicant's submittal to be acceptable. The overhead heavy loads handling system does not comply with the acceptance criteria of SRP Section 9.1.5.

9.2 Water Systems

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9.2.1 Normal Service Water System (SWS)

The normal service water system was reviewed in accordance with Section 9.2.1 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section relating to the nonsafety-related system was performed, according to the guidelines provided in the "Review Procedures" portion of the SRP section relating to nonsafety-related system. Conformance with the acceptance criteria formed the basis for our evaluation of the normal service water system with respect to the applicable regulations of 10 CFR 50.

The normal service water system (SWS) is used to provide cooling water for turbine plant and reactor plant auxiliary systems and components during all phases of normal plant operation. The standby service water system (discussed in Section 9.2.7) provides water to safety related equipment during emergencies. Power for the SWS is provided by a safety related, noninterruptable diesel generator so that the service water system is operable upon loss of offsite power. The SWS may also be used to supply cooling water to the residual heat removal heat exchangers when the standby service water system is not in use.

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The normal service water system consists of three pumps, each with a pumping capacity of 31,500 gpm. Two pumps operate under normal conditions with the third used as a spare in the event of inoperability of one of the operating pumps. Water is pumped from the cooling water basin of the circulating water system into a common header and then to the systems requiring cooling water. Check valves and automatic block valves in the normal service water return lines are used to isolate safety-related lines from the normal service water system; these valves are designed in accordance with the criteria for seismic Category I (in conformance with Position C.3 of Regulatory Guide 1.29) and Quality Group C criteria (except for containment isolation valves and associated piping which are designed to seismic Category I, Quality Group B criteria). Piping and valves common to both the normal and standby service water systems are designed to seismic Category I, Quality Group C criteria. Valves used to isolate the normal service water system from the standby service water system are designed to the same criteria: seismic Category I, Quality Group C in conformance with Positions C.2 and C.3 of Regulatory Guide 1.29. The remainder of the normal service water system is designed in accordance with applicable industry codes and standards. This meets the requirements of General Design Criteria 2, "Design Bases for Natural

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Phenomena" by compliance with the guidelines of Positions C.2 and C.3 of Regulatory Guide 1.29, "Seismic Design Classification.

A separate normal service water system is supplied for each unit however, the system need not comply with the requirements of General Design Criterion 5, "Sharing of Structures, Systems and Components," since the SWS is not safety-related.

The normal service water system has no safety function, therefore, the requirements of General Design Criterion 44, "Cooling Water," 45, "Inspection of Cooling Water Systems," 46, "Testing of Cooling Water System" do not apply.

We ~~found~~ found that the normal service water system has been designed in accordance with the applicable criteria of Section 9.2.1 of the Standard Review Plan, "Station Service Water System" and that this system is, therefore, acceptable.

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9.2.2 Reactor Plant Component Cooling Water System (Reactor Auxiliary Cooling Water System)

The reactor plant component cooling water (RPCCW) system was reviewed in accordance with Section 9.2.2 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed, according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the RPCCW system with respect to the applicable regulations of 10 CFR 50.

The reactor plant component cooling water (RPCCW) system includes pumps, heat exchangers, valves and piping, expansion tank, makeup piping and the points of connection with other systems. The RPCCW provides cooling water to the spent fuel pool heat exchangers, reactor recirculating pump seals, bearing and winding coolers, reactor water cleanup pump coolers, reactor water cleanup heat exchangers, process sampler coolers, control rod drive pump oil coolers, drywell sump coolers and pump seal coolers during normal operation.

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The reactor plant component cooling water system is mainly a nonseismic, nonsafety-related system with no functions necessary to mitigate transients or accidents and none necessary to achieve safe shutdown conditions. However, there is some piping in the RPCCW system which is used to convey water from the safety-related standby service water system (described in Section 9.2.7 of this SER) to the spent fuel coolers and to the residual heat removal pump seals and bearings when the normal service water system is not available. These portions of the piping, together with the valves used to isolate the standby service water system from the normal service water, are safety-related and are designed to seismic Category I and Quality Group C criteria, in accordance with Positions C.1 and C.2 of Regulatory Guide 1.29, "Seismic Design Criteria," thus complying with General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena." The RPCCW system has no other heat loads that are related to safe shutdown or accident prevention or mitigation. Thus, the requirements of General Design Criterion 44, "Cooling Water," 45, "Inspection of Cooling Water System," Insert are not applicable.

Insert and 46, "Testing of Cooling Water System,"

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When operation of the standby service water system is initiated in the event of an accident, the safety-related portion of the RPCCW system is automatically isolated from the nonessential portion of the RPCCW system by redundant seismic Category I, Quality Group C isolation valves. Redundant seismic Category I, Quality Group B isolation valves are provided at the RPCCW system piping containment penetrations. Thus, the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," as related to seismic design in accordance with the criteria of Positions C.1, C.2 and C.3 of Regulatory Guide 1.29, "Seismic Design Classification," are met.

There are redundant lines in the reactor plant component cooling water system used to convey water to the RHR pump seals and bearings and to the spent fuel pool coolers and to return water to the standby service water pumps. Therefore, the standby service water system will be able to cool the pump in one RHR train and one fuel pool cooler in the event of a single failure, thus providing the minimum cooling needed in the event of an accident or loss of offsite power.

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The safety-related portion of the RPCCW system is located in buildings which are protected against earthquakes (as noted in Section 3.7 of this SER) and tornadoes (as noted in Section 3.3 of this SER). Therefore, the safety-related portion of the RPCCW system is in compliance with the criteria of General Design Criteria 2, "Design Bases for Protection Against Natural Phenomena."

The applicant stated that the safety-related portions of the RPCCW system are protected against environmental conditions in the surrounding areas including those resulting from moderate energy pipe cracks and flooding resulting from such cracks. [The applicant must show that the safety-related portion of the RPCCW will not fail as a result of high energy pipe breaks.]

[In addition, the applicant must show that the safety-related portion of the RPCCW is protected against internally generated missiles because the FSAR Section relating to internally generated missiles (Section 3.5.1) does not list the RPCCW as one of the systems which need to be protected against internally generated missiles.] Therefore, we cannot conclude that the

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safety-related portion of the RPCCW conforms with the requirements of General Design Criterion 4 - "Environmental and Missile Design Bases."

The RPCCW system contains a radiation monitor in the pump suction header to detect in-leakage from radioactive systems. Each unit of the plant has its own RPCCW system. Thus, the requirements of General Design Criterion 5, "Sharing of Structures, Systems and Components," are not applicable.

Based on the above, we conclude that the RPCCW system meets the requirements of General Design Criteria 2, and the guidelines of Regulatory Guide 1.29 as it relates to seismic events and with respect to General Design Criterion 5 in regards to shared functions between units and that General Design Criteria 44, 45, and 46 do not apply. ~~However, the applicant must provide further information to verify the statement that "---Interruption of coolant to the recirculation pump does not result in unacceptable consequences" (which is contained in Section 9.2.2.1 of the FSAR) in order to show compliance with Section II.K.3.25 of the TMI Task Action Plan regarding the loss of coolant~~

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~~inventory due to pump seal failure as a result of loss of offsite power.~~ The applicant must also show conformance with the requirements of General Design Criterion 4 regarding possible damage as a result of high energy line breaks. Therefore, we cannot find the reactor plant component cooling water system to be acceptable. The reactor plant component cooling water system does not comply with the criteria of SRP Section 9.2.2.

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9.2.3 Makeup Water Treatment System (Demineralized Water Makeup System

The makeup water treatment system was reviewed in accordance with Section 9.2.3 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed, according to the guidelines provided in the "Review Prcoedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the makeup water treatment system with respect to the applicable regulations of 10 CFR 50.

The nonsafety-related makeup water treatment system includes all components and piping associated with the system from the plant makeup water source (deep wells) to the points of discharge to other systems. The system has no safety-related function. The system is capable of fulfilling the normal operating requirements of the facility for acceptable redundancy. The applicant states that entry of potentially radioactive water into the system is precluded by assuring a greater pressure for the makeup water treatment system than in the potentially radioactive sources to which it discharges. [The applicant should show whether this greater pressure is maintained under all conditions, including a design basis accident or loss of offsite power. If

makeup water with the necessary component

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not, the applicant must explain how, as the result of a seismic event, adjacent or continguous safety-related systems (including the control building chilled water system) and control room personnel are protected against the intrusion of radioactive water which has passed into the makeup water system, in order to comply fully with Position C.2 of Regulatory Guide 1.29, "Seismic Design Classification.] Conductivity of the water leaving each demineralizer exchange unit is monitored continuously; an alarm is sounded in the auxiliary control room to alert the operators when the effluent has a high conductivity denoting an abnormal condition. The applicant stated that makeup water is supplied only to one safety-related system the control building chilled water system and only during normal operating conditions; the safety-related standby service water systems takes over this function during accidents. Safety-related valves designed to seismic Category I, Quality Group C standards operate automatically during a LOCA to isolate the makeup water treatment system from the safety-related systems; the valves may be operated manually at other times. This meets the requirements of General Design Criterion 2 with regard to seismic events by being in compliance with Position C.3 of Regulatory Guide 1.29, "Seismic Design Classification."

Each unit has a separate makeup water treatment system. However, the system has no safety-related function, and

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therefore, the requirements of General Design Criterion 5, "Sharing of Structures, Systems and Components" do not apply. For the same reason, (no safety-related function) General Design Criteria 44, 45 and 46, entitled respectively, "Cooling Water," "Inspection of Cooling Water System," and "Testing of Cooling Water System." are not applicable.

Based on the above, we conclude that the makeup water treatment does not fully comply with the requirements of General Design Criterion 2 with respect to the need for protection of other systems which are safety-related in the event of failure of this system as a result of a seismic event because it does not meet the guidelines of Position C.2 of Regulatory 1.29. Therefore, we cannot find that the applicant's submittal with regard to the makeup water system is acceptable. The makeup water treatment system does not comply with the acceptance criteria of SRP Section 9.2.3.

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9.2.4 Domestic Water and Sanitary Drains and Disposal Systems (Potable and Sanitary Water System)

The domestic water and sanitary drains and disposal systems were reviewed in accordance with Section 9.2.4 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed, according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria ~~as noted below~~ formed the basis for our evaluation of the domestic water and sanitary drains and disposal systems with respect to the applicable regulations of 10 CFR 50.

The nonsafety-related (Quality Group D, nonseismic-Category I) domestic water and sanitary drains and disposal systems provide water for human consumption and sanitary waste water treatment. There are no cross-connections between these systems and any systems which contain radioactive material or which have the potential for carrying such material. One of these systems, the domestic water supply system, provides potable water to all installed plumbing fixtures.

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The water is pumped from deep wells by two operating pumps; the system contains three pumps with the third pump maintained as a spare. The pumps alternate in operation so all three pumps get equal wear. A controlled amount of sodium hypochlorite is added to the water in order to disinfect the well water. The disinfected, potable water is then transferred to the plant buildings through an underground piping system.

The sanitary drains and disposal system conveys the sanitary waste from the appropriate buildings to a waste treatment plant by means of two sewage pumps. The waste treatment plant treats the sewage by aeration after which the sewage is separated into sludge and clarified liquid. The sludge is trucked offsite for disposal while the clarified liquid is chlorinated in a dry tablet chlorination unit and then discharged into the storm drainage system.

The systems have no safety-related function. Since these systems have no contact with systems which contain radioactive products or which could carry radioactive products, the requirements of General Design Criterion 60, "Control of Releases of Radioactive Materials to the Environment," are met.

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Based upon our review, we conclude that the domestic water and sanitary drains and disposal systems meet the requirements of General Design Criterion 60 with respect to prevention of release of potentially radioactive water, and are, therefore, acceptable. The domestic water and sanitary drains and disposal systems comply with the criteria of SRP Section 9.2.4.

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9.2.5 Ultimate Heat Sink

The ultimate heat sink was reviewed in accordance with Section 9.2.5 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the bases for our evaluation of the ultimate heat sink with respect to the applicable regulations of 10 CFR 50.

The ultimate heat sink (UHS) provides cooling water to the standby service water systems for both units only during accident conditions and/or loss of off-site power for cooling of essential plant auxiliary components. The UHS for each unit consists of a separate mechanical draft standby service water cooling tower and its associated water basin. Each tower has four independent fan cells with five vaneaxial fans in each cell. Two operating fan cells are required for safe shutdown. The fans dissipate heat to the atmosphere from the warm standby service water returned to the tower after cooling plant

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components. The UHS is designed so that each cooling tower can provide adequate heat removal capability to safely shut down its own unit. The fans in each tower are powered from redundant essential power supplies with each providing power to two fan cells. The standby service water system pumps take suction from the cooling tower basins. The standby service water system is described in Section 9.2.7 of this SER.

The concrete cooling towers and basins are designed to seismic-Category I requirements and are designed to withstand the effects of floods and tornadoes, thus complying with the guideline contained in Position C.2.a of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants." The UHS can withstand reasonably probable combinations of less severe events, thus complying with the guideline contained in Position C.2.c of Regulatory Guide 1.27 and with credible single failure of manmade structural features, thus complying with the guideline contained in Position C.2.d of Regulatory Guide 1.27.

Compliance with Position C.2.b of Regulatory Guide 1.27 is discussed in Section 2.2.3 of this SER.

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The cooling tower fans are seismic Category I, Quality Group C. The UHS has four standby service water lines withdrawing water from the cooling tower basin and two lines returning water, thus meeting the guidelines of Position C.3 of Regulatory Guide 1.27. Thus, the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guides 1.29 (Position C.1), "Seismic Design Classification" and 1.27, "Ultimate Heat Sink for Nuclear Power Plants," related to protection against natural phenomena are satisfied.

There are no high-energy systems or other sources of potential internally generated missiles in the vicinity of the UHS. The tower walls are designed against the effects of externally (tornado) generated missiles. The cooling tower plenum openings are equipped with concrete baffles to protect the fans from possible damage. The fan motors are enclosed in the missile-protected concrete structure. Thus, the requirements of General Design Criterion 4, "Environmental and Missile Design Bases," and the guidelines of Regulatory Guide 1.117, "Tornado Design Classification," are satisfied.

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There is no sharing of UHS facilities between units. Therefore, the requirements of General Design Criterion 5, "Sharing of Structures, Systems and Components," are not applicable.

The applicant has used Branch Technical Position ASB 9-2, "Residual Decay Energy for Light-Water Reactors for Long Term Cooling," to establish the heat input to the UHS due to fission product and heavy element decay. The applicant performed analyses to verify the performance capability of the UHS assuming worst one-day and 30-day site meteorology. [We are reviewing these analyses in order to ascertain whether the UHS provides adequate heat transfer capability to remove decay heat without makeup water for a 30-day period in accordance with the requirements of General Design Criterion 44, "Cooling Water."] Primary makeup can be provided from the nonsafety-related makeup water system (see this SER, Section 9.2.3) if this system is available. Other sources include the Mississippi River by means of a temporary pump and piping into the storage basin or by the use of tank trucks to convey the river

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water from the barge slip, and the use of a new drilled well which, the applicant stated, could be put into operation within a few days. Protection of the system against freezing is discussed in Section 2.4.5 of this SER.

Based on the above, we conclude that UHS meets the requirements of General Design Criteria 2, 4, 5, 45, and 46 with respect to protection against natural phenomena, missiles, environmental effects, shared systems function, inservice inspection and functional testing, and the guidelines of Regulatory Guides 1.27, 1.29, 1.102, and 1.117 and Branch Technical Position ASB 9-2 with respect to seismic classification, ultimate heat sink design capability and flood and tornado missile protection. [However, we are reviewing the capability of the plant to provide the required heat removal with the available water supply in order to show compliance with General Design Criterion 44.] Therefore, we cannot conclude that the applicant's submittal relating to the UHS is acceptable. The ultimate heat sink does not comply with the acceptance criteria of SRP Section 9.2.5.

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9.2.6 Condensate Storage Facilities

The condensate storage facilities were reviewed in accordance with Section 9.2.6 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed, according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the condensate storage facilities with respect to the applicable regulations of 10 CFR 50.

The nonsafety-related (Quality Group D, nonseismic Category I) condensate storage and transfer system includes all components and piping associated with the system from the storage tanks to the points of connection or interfaces with other systems. The primary functions of the condensate storage system are to provide makeup to the condenser hotwell and to provide a dedicated water supply for the reactor core isolation cooling (RCIC) and the high pressure core spray (HPCS) systems and as an alternative source for the control rod drive system (the primary source is the condensate water treatment system). The alternative water supply for the RCIC and HPCS systems for safe shutdown is the safety-related suppression pool. Additionally, the 620,000-gallon condensate storage tank (including a total of 125,000 gallons reserved for use by both the RCIC and

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HPCS) provides demineralized water for the fuel pool cooling and cleanup system, the refueling water system and other miscellaneous uses.

A separate condensate storage and transfer system is provided for each unit of the plant. Thus, the requirements of General Design Criterion 5, "Sharing of Structures, Systems, and Components," are not applicable.

The valves at the suction of the HPCS and RCIC pumps operate automatically on receipt of a condensate storage tank (CST) low water level signal to isolate the CST from the HPCS and RCIC pumps and to open the valves leading from the safety-related suppression pool to the suction sides of the HPCS and RCIC pumps thereby protecting these pumps in the event of failure of either the CST or piping from the CST to the valves which isolate the CST from the HPCS and RCIC. The CRD hydraulic system only uses the CST an an alternate source of water. In the event of loss of both water supply sources, the scram accumulators provide for insertion of control rods. The other safety-related systems which use the CST as a source of water, are the spent fuel pool cooling system, the residual heat removal system, and the standby liquid control system. All of these systems are capable of performing their safety-related functions without the condensate storage facilities. Therefore, the

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criteria of General Design Criteria 44, "Cooling Water," 45, "Inspection of Ccoling Water Systems" and 46, "Testing of Cooling Water Systems," do not apply to the condensate storage facilities.

The level instruments which provide for shift of RCIC and HPCS pump suction are safety-related, as are their power supply transmitters and other auxiliary equipment. The level instruments are attached to that portion of the piping leading to the RCIC and HPCS pumps which is safety-related. Containment isolation boundaries between the condensate storage facilities and safety-related systems are designed in accordance with the guideline of Position C.1 of Regulatory Guide 1.29, "Seismic Design Classification," and are in buildings designed to resist tornadoes and the safe shutdown earthquake. The condensate storage facilities are designed to comply with the guidelines of Positions C.2 and C.3 of Regulatory Guide 1.29 in that valves serving as isolation boundaries between the condensate storage facilities and safety-related systems are designed in accordance with the criteria for seismic Category I components and that failure of the condensate storage facilities as a result of a seismic event would not result in damage to any safety-related system. This complies with the guidelines of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

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The applicant has stated in Section 15.7.3 of the FSAR that condensate storage tank failure would not affect the NSSS. (In order to verify this statement, the applicant should show that a catastrophic failure of the CST would not cause flooding of any safety-related systems, including electrical power supplies and controls, in the vicinity of the CST.) (The applicant must also show that a crack in the piping from the CST to the HPCS/RCIC would not adversely affect the operation of these or other safety-related systems.)

We conclude that the condensate storage facilities comply with the guidelines of Positions C.1, C.2, and C.3 of Regulatory Guide 1.29, thus complying with the requirements of General Design Criterion 2. (However, the applicant must show that failure of the CST and a crack in the piping does not prevent satisfactory operation of the HPCS and RCIC pumps nor does the CST failure or piping crack constitute a danger to these other safety-related equipment in order to comply with General Design Criterion 4, "Environmental and Missile Design Bases.)" Therefore, we cannot conclude that the design of the condensate storage facilities is acceptable nor that the design complies with the acceptance criteria of SRP Section 9.2.6.

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9.2.7 Standby Service Water System

The standby service water system (SSW) was reviewed in accordance with Section 9.2.1 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed, according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the standby service water system with respect to the applicable regulations of 10 CFR 50.

The standby service water system (SSWS) supplies cooling water to the plant from the basin of the ultimate heat sink cooling tower as discussed in Section 9.2.5 of this SER. The system ~~operated~~ operates during accident and/or loss of offsite power conditions. Under these conditions, it provides cooling to the following essential plant components: standby diesel-generator coolers, HPCS diesel-generator cooler, control building A/C chillers, RHR heat exchangers, RHR pump seal coolers, fuel pool cooling heat exchangers, penetration valve leakage control compressors, and unit cooler coils in the containment and the auxiliary building. In addition,

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the SSWS provides emergency makeup to the fuel pool. The SSWS coolant leaving the residual heat removal heat exchangers is monitored for radiation and can be isolated on a high-radiation alarm. A separate redundant SSWS is provided for each of the two units. Thus, the requirements of General Design Criterion 5, "Sharing of Structures, Systems and Components," are not applicable.

The SSWS consists of two independent piping loops per unit labeled "A" and "B", one of which is required to assure safe shutdown. The "A" and "B" loops serve redundant heat exchanger equipment. Diesel generator heat exchangers are served by the loop they power. The HPCS diesel generator heat exchanger and room cooler are served by either loop. Each loop is provided with two 50-percent capacity pumps located in a separate cubicle in the standby service water pumphouse. The SSWS pumps circulate water in a closed cycle from the ultimate heat sink cooling tower basin through the components to be cooled and back to the cooling tower. Each division is powered from its associated diesel generator emergency bus.

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The system is housed in seismic Category I, flood and tornado protected structures (refer to Sections 3.4.1 and 3.5.2 of this SER). It is designed to seismic Category I, Quality Group C requirements. Underground piping of the SSWS, which connects the parts of the system in these structures, is also protected from these natural phenomena.

The SSWS shares piping and valves with the nonsafety-related normal service water system (described in Section 9.2.1 of this SER). In the event of a design basis accident (DBA) automatic block valves and check valves act to isolate the safety-related SSWS from the nonsafety-related service water system. These shared components, as well as the isolation valves, are designed in accordance with the guidelines of Quality Group C and seismic Category I. Any valves which serve as containment isolation valves, however, are designed in accordance with the guidelines of Quality Group B and seismic Category I. This complies with the guidelines of Position C.1 and C.2 of Regulatory Guide 1.29, "Seismic Design Classification." Thus, the requirements of General Design Criterion 2,

"Design Bases for Protection Against Natural Phenomena," are satisfied.

The SSWS is protected against internally and externally generated missiles (see Section 3.5.1 of this SER) and against the effects of pipe whip and jet impingement resulting from high and moderate energy line breaks (as discussed in Section 3.6.1 of this SER). Therefore, the SSWS complies with the guidelines of General Design Criterion 4, "Environmental and Missile Design Bases."

In the event of a DBA two standby service water pumps, each powered by a different onsite essential (Class 1E) power source are started. Only one is required in the initial phases of a DBA. In the event one of these two pumps fail when two are required, i.e., later in the DBA, when the residual heat exchangers are needed to cool the plant, another pump can be started manually.

The applicant has provided calculations ~~calculations~~ to show that the SSWS is capable of transferring heat loads from safety-related components to the ultimate heat sink under ~~all~~ emergency conditions ~~modes of operation~~ in order to show compliance with

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General Design Criterion 44, "Cooling Water." We are reviewing these calculations and will report on the results of that review.]

The availability of the SSWS pumps is assured by periodic functional tests and inspections as delineated in the plant Technical Specifications. The system design also incorporates provisions for accessibility to permit in-service inspection as required. Thus, we conclude that the requirements of General Design Criterion 45, "Inspection of Cooling Water System," and 46, "Testing of Cooling Water System," are satisfied.

Based on the above, we conclude that the SSWS meets the requirements of General Design Criteria 2, 4, 45 and 46 with respect to the system's protection against natural phenomena, environment and missiles, inservice inspection and functional testing. [We are reviewing the capability of the system to transfer the required heat loads in accordance with the requirements of General Design Criterion 44.] Therefore, we cannot find the applicant's submittal acceptable with regard to the standby service water system. The system does not comply with the acceptance criteria of SRP Section 9.2.7.

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9.2.8 Turbine Plant Component Cooling Water (TPCCW) Syst DRAFT

The nonsafety-related turbine plant component cooling water (TPCCW) system was reviewed in accordance with applicable portions of Section 9.2.2 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of SRP section was performed, according to the guidelines provided in the "Review ~~Procedires~~ Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the bases for our evaluation of the turbine plant component cooling water system with respect to the applicable regulations of 10 CFR 50.

The TPCCW system is designed to remove heat from designated heat exchangers in the turbine and radwaste buildings. The TPCCW system is not a safety-related system; it is not required for operation during accidents nor for safe shutdown of the plant. It is designed in accordance with Quality Group D standards; it has no interfaces with any seismic Category I systems and is not designed in accordance with seismic Category I standards; therefore, Regulatory Guide 1.29, "Seismic Design Classification," is not applicable except for Position C.2. [In order to meet the guidelines of Position C.2 thus complying with the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," the applicant must demonstrate that all safety-related systems are protected against flooding in the event of a pipeline break in the TPCCW system.]

Since the TPCCW system has no safety function nor interfaces with a safety system, General Design Criterion 44, "Cooling Water," 45, "Inspection of Cooling Water" and 46, "Testing of Cooling Water" and 46, "Testing of Cooling Water System," do not apply.

The TPCCW system has no safety function and thus, the requirements of General Design Criterion 5, "Sharing of Structures, Systems and Components" are not applicable.

[The applicant must demonstrate that all safety-related systems are protected against flooding in the event of a break in the TPCCW system in order to show compliance with General Design Criterion 2.]

Therefore, we cannot find the applicant's submittal acceptable regarding the turbine plant component cooling water system. The TPCCW does not comply with the acceptance criteria of SRP Section 9.2.2.

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9.2.9 Ventilation Chilled Water System

The nonsafety-related ventilation chilled water system (VCWS) was reviewed in accordance with applicable portions of Section 9.2.2 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed, according to the guidelines provided in the "Review Procedures" portion of the SRP Section. Conformance with the acceptance criteria formed the basis for our evaluation of the ventilation chilled water system with respect to the applicable regulations of 10 CFR 50.

The ventilation chilled water system consists of two subsystems, one located in the turbine building, the other in the radwaste building. The subsystem in the turbine building serves the turbine building, turbine building sample room, condensate demineralizer offgas building and the containment. The second subsystem serves the radwaste and fuel buildings.

The FSAR states that the subsystems are not designed in accordance with seismic Category I or Quality Group A, B, or C standards because they are not required to operate during accident conditions or for safe shutdown. However, piping and valves required for containment and drywell isolation are designed to seismic Category I and Quality Group C standards, thus meeting the guidelines of Position C.3 of Regulatory Guide 1.29, "Seismic Design Classification." The applicant must show that the VCWS meets the

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guidelines of Position C.2 of Regulatory Guide 1.29, in order to comply fully with the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena." [In order to meet the guidelines of Position C.2 of Regulatory Guide 1.29, the applicant must demonstrate that all safety-related systems are protected against flooding in the event of a pipe break in the VCWS.]

The requirements of General Design Criterion 44, "Cooling Water," 45, "Inspection of Cooling Water System," and 46, "Testing of Cooling Water System" do not apply since the system is not required to mitigate any accidents nor is it required for safe shutdown of the plant.

Since the ventilation chilled water system is nonsafety-related, the requirements of General Design Criteria 5, "Sharing of Structures, Systems and Components" are not applicable.

[The applicant must demonstrate compliance with General Design Criterion 2.] Therefore, we cannot find the applicant's ~~applicant's~~ submittal acceptable with regard to the ventilation chilled water system. The VCWS does not comply with the acceptance criteria of SRP Section 9.2.2.

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9.2.10 Control Building Chilled Water System (CBCWS)

The control building chilled water system (CBCWS) was reviewed in accordance with Section 9.2.2 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed, according to the guidelines provided in the "Review Prcoedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the control building chilled water system with respect to the applicable regulations of 10 CFR 50.

This system is designed to remove heat from the main control room, standby switchgear room and chiller equipment room. It is designed in accordance with the criteria of seismic Category I and Quality Group C. The CBCWS will not be discussed further in this section as it is discussed fully in Section 9.4.1, including its compliance with the criteria of Section 9.2.2.

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9.2.11 Cooling Tower Makeup Water System (CTMWS)

The nonsafety-related cooling tower makeup water system was reviewed in accordance with the applicable portions of Section 9.2.2 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed, according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the cooling tower makeup water system with respect to the applicable regulations of 10 CFR 50.

The cooling tower makeup water system provides water from the Mississippi River as makeup to the normal service water system and to the circulating water system; it supplies water to replenish blowdown of 220 gpm from each River Bend Station unit. The CTMWS uses clarifiers to remove the solids suspended in the water supplied to each unit. The system contains three pumps to supply water; one for each unit with one on standby.

The system is nonsafety-related, it is not required to mitigate accidents nor for plant safe shutdown. It is designed to nonseismic Category I standards, and Quality D standards in accordance with the requirements of Regulatory Guide 1.26, "Quality Group Classifications and

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Standards for Water-, Steam-, and Radioactive Waste Containing Components of Nuclear Power Plants." The CTMW system has no interface with any safety system; therefore Regulatory Guide 1.29, "Seismic Design Classification," is not applicable except for Position C.2. [In order to meet the guidelines of Position C.2, ~~this~~ thus complying with the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," the applicant must demonstrate that all safety-related systems are protected against flooding in the event of a pipeline break in the TPCCW system.] Since the system is not safety-related, ~~related,~~ General Design Criterion 4, "Environmental and Missile Design Bases," is not applicable.

General Design Criterion 5, "Sharing of ~~Structures,~~ Structures Systems and Components," does not apply because the system has no safety function. Since the CTMW system has no safety function General Design Criterion 44, "Cooling Water," 45, "Inspection of Cooling Water" and 46, "Testing of Cooling Water System" are not applicable.

[The applicant must demonstrate that all safety-related systems are protected against flooding resulting from a break or rupture of the CTMW system in order to comply with the requirements of General Design Criterion 2.] Therefore, we cannot find the applicant's submittal acceptable with regard to the cooling tower makeup water system. The system does not comply with the acceptance criteria of SRP Section 9.2.2. ~~-110-~~ 9-70

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9.5 Process Auxiliaries

9.5.1 Compressed Air Systems

The compressed air system was reviewed in accordance with Section 9.3.1 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed, according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the compressed air system with respect to the applicable regulations of 10 CFR 50.

The compressed air system contains two compressed air subsystems. One is safety-related and supplies compressed air to the safety relief valves (SRVs), to the main steam positive leakage control system (MS-PLCS), and to the penetration valve leakage control system (PVLCS); the other is nonsafety-related and provides compressed air for three subsystems composed of the instrument air system, the service air system and the breathing air system.

The safety-related compressed air system for each unit consists of two redundant trains each containing an inlet filter, air compressor aftercooler, moisture separator, and accumulator tank. The compressors continuously maintain a set pressure in the accumulator tanks which in turn supply the ADS accumulators, the MS-PLCS and the PVLCS. Each unit

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of the plant is provided with a separate safety-related compressed air system. Therefore, the requirements of General Design Criterion 5, "Sharing of Structures, Systems, and Components," are not applicable.

The safety-related compressed air system is an integral part of the PVLCS. The compressors supplying air to the safety-related compressed air system are seismic Category I, Safety Class 2 as are the components and piping in the PVLCS. However, the seismic category and quality group of the piping and components that interface with the safety/relief valves cannot be determined because the figures provided by the applicant (Figures 5.2-14a, 5.2-14b, and 9.3-13a) do not show their safety class. (Therefore, we cannot determine whether the interface complies with the appropriate guidelines of Positions C.1 and C.3 of Regulatory Guide 1.29, "Seismic Design Classification.")

The safety-related compressed air system is located within seismic Category I, flood and tornado-protected structures (refer to Sections 3.4.1 and 3.5.2 of this SER). The system is also protected from the effects of missiles and pipe breaks (refer to Sections 3.5.1.1, 3.5.1.2, and 3.6.1 of this SER). Thus, the system satisfies the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena" with the exception of compliance of the interface between the compressed air system and the safety/relief valves with seismic criteria as contained in Regulatory Guide 1.29.

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The applicant stated that the MS-PLCS and PVLCS do not require dry air because the systems are wet when the compressed air is provided to them. The applicant noted that the inlet air to the compressors was filtered through a 5 micron filter. Regarding the safety/relief valves the FSAR Section 5.2.2 indicates that the air supplied to these valves from the PVLCS is dried to a dewpoint of 0 F and filtered to a maximum particle size of 3 micron. However, the 3 micron filter does not appear in the P&ID of the PVLCS (Figures 9.3.13a and b). (The applicant should show how the compressed air supply to the safety/relief valves complies with the requirements of ANSI MC 11.1-1976 (ISA-S7.3), "Quality Standard for Instrument Air," with regard to dew point, oil or hydrocarbon content, and freedom from corrosive or hazardous contaminants in order to comply with the criteria of General Design Criterion 1, "Quality Standards and Records.") ~~The applicant should correct the narrative in the FSAR in Sections 9.3.1 and 9.3.6, wherein both the nonsafety-related and safety-related compressed air systems appear to take credit for providing compressed air to the safety/relief valves.)~~ It is our position that a periodic check, at least yearly, be made to assure that air being provided to the safety/relief valves complies with the standards of ANSI MC 11.1-1976. (The applicant should commit to this periodic testing.)

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The nonsafety-related compressed air systems for each unit consists of three 50% capacity trains, each containing an air compressor, inlet filter, coolers, and receiver tank, all feeding a common manifold. The nonsafety-related service air system takes air from the manifold and distributes it throughout the unit. Air from the manifold is fed to two 50% capacity package dessicant air dryers and filters to produce clean, dry, oil-free air for the nonsafety-related instrument air system in accordance with ANSI Standard MC 11.1-1976 (ISA-S7.3), thus complying with the requirements of General Design Criterion 1. This air is distributed throughout the unit by a nonseismic Category I, Quality Group D piping system.

The nonsafety-related instrument and service air systems have no function necessary for achieving safe reactor shutdown conditions or for accident prevention or mitigation. Instruments, controls and services required for safe shutdown of the plant such as those controlling MSIVs and certain ventilation system dampers are provided with seismic Category I passive air accumulators to assure their proper function in case of a loss of instrument air. All other air-operated valves including the scram discharge inlet and outlet valves and other devices are designed for a fail-safe mode upon loss of instrument air and do not require a continuous air supply under emergency

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or abnormal conditions. Additionally, all instrument and service air system containment penetrations are provided with redundant seismic Category I, Quality Group B isolation valves, thus complying with the guidelines of Position C.1 of Regulatory Guide 1.29. Since a failure of the instrument and service air systems will not prevent safe reactor shutdown, we conclude the requirements of General Design Criteria 2 and 5 are not applicable.

The review of preoperational testing of the compressed air systems and compliance with Regulatory Guide 1.68.3, "Preoperational Testing of Instrument Air Systems," is contained in Section 14.0 of this SER.

Based on the above, we conclude that the safety-related compressed air system meets the requirements of General Design Criteria 2 regarding protection against natural phenomena and the guidelines of Regulatory Guide 1.29 concerning the seismic classification with the exception noted above regarding the interface between the safety/relief valves and PVLCS. Further, we cannot conclude that the safety-related compressed air system meets the criteria of General Design Criterion 1 and the guidelines of ANSI MC 11.1 regarding instrument air quality. In view of the foregoing, we cannot

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conclude that the applicant's submittal concerning the safety-related compressed air system is acceptable. However, we conclude that the applicant's submittal regarding tne nonsafety-related compressed air system is acceptable. Also, the applicant must commit to periodic testing of the air quality of the safety-related compressed air system and must correct the FSAR as noted above. The nonsafety-related compressed air system meets applicable acceptance criteria of SRP Section 9.3.1; the safety-related compressed air system does not comply with the applicable acceptance criteria of SRP Section 9.3.1.

the interface with the ADS system. The staff cannot conclude that the system meets the requirements of GDC 1 and ANSI/MC 11.1 (ISA-S7.3) regarding instrument air quality and thus cannot make a conclusion regarding the system's acceptability. Therefore, the staff concludes that the applicant's submittal concerning the safety-related air system is unacceptable.

9.3.2 Process Sampling System

The process sampling system has been reviewed in accordance with the SRP 9.3.2 (NUREG-0800).

The process sampling system monitors station operation and equipment performance and provides information so the operator can make corrections or adjustments to process system operations. Representative liquid, gas, and steam samples are obtained through the process sampling system either by online or laboratory analysis. Online continuous radiation monitoring of gaseous and liquid samples is also provided. The sampling system is designed to minimize contamination of samples, minimize radiation exposure at the sampling station, and reduce sample line plate-out as much as possible.

At the request of the staff, the applicant is reviewing the sample line purge flows and duration times to determine if they are sufficient to flush out stagnant lines to ensure that a representative sample is obtained. In addition, the applicant has been requested to describe how radiation exposure will be kept ALARA in the event of a rupture of the sample line containing contaminated primary coolant, in accordance with position C.2.i(6) of RG 8.8, Revision 3 (June 1978). The applicant has also been requested to provide passive flow restrictions or environmentally qualified, remotely operated isolation valves to limit reactor coolant loss from a rupture of the sample line and to verify the capability to sample the containment sump and standby liquid control system. The staff will provide an evaluation in the SER after it receives this information from the applicant.

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9.3.3 Equipment and Floor Drainage Systems

The equipment and floor drainage systems were reviewed in accordance with Section 9.3.3 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed, according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the equipment and floor drainage systems with respect to the applicable regulations of 10 CFR 50.

The equipment and floor drainage systems are nonsafety-related and are designed to transfer both effluents which are radioactive or potentially radioactive and those which are not radioactive or potentially radioactive. Systems which convey the radioactive effluents are separated from systems which convey the nonradioactive effluents except in the case of the equipment drain sumps in the turbine building, fuel building and reactor building where the sump discharges are monitored by the plant computer. When the conductivity of any of these three equipment drain sump discharges

is low, the effluent is directed to the condenser; when high, (indicating high radiation), to the radwaste system. Extreme high sump discharge conductivity or high or low sump water level activates a control room equipment drain sump alarm to alert plant operators. With these exceptions, radioactive effluents from various parts of the plant are conveyed in drainage subsystems which are separated from those conveying nonradioactive effluents. Each drainage sump receiving radioactive effluents is provided with a nonporous liner to prevent transport of its contents through the sump wall. Radioactive drainage is collected in floor and equipment drain sumps in each building and discharged to the radwaste processing system. Drainage from nonradioactive sources such as plumbing fixtures and roof drains is discharged to the sanitary waste treatment systems and discharge basin, respectively.

We cannot determine that there is no potential for inadvertent transfer of radioactive drainage to nonradioactive collection areas, since the description and P&IDs for floor drains in several areas of the plant have not yet been submitted by the applicant.

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[The applicant should provide drawings and narrative to complete the description of the equipment and floor drainage systems.]

Containment penetrations for the equipment and floor drainage system are designed to seismic Category I and Quality Group B requirements. Each unit has its own equipment and floor drainage systems; therefore, the guidelines of General Design Criterion 5, "Sharing of Structures, Systems and Components," are not applicable.

The LPCS, HPCS, RHR-A (Train A), RHR-B (Train B), RHR-C (Train C) and RCIC pumps are located in individual watertight cubicles at the lowest elevation in the auxiliary building. Each cubicle contains a sump with duplex sump pumps. Collected leakage in the cubicle is automatically pumped to the radwaste system. Back-flooding of the ECCS rooms is prevented by two check valves on the piping penetrating the cubicle wall with one inside, one outside the cubicle; the valves and intervening piping are designed in accordance with seismic Category I and Quality Group B criteria. Each

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cubicle sump contains a wall-mounted level transmitter (designed to seismic Category I, Quality Group B criteria) which sounds an alarm and provides level indication in the control room in case of rising water level. We therefore conclude that this portion of the system meets the requirements of General Design Criteria 2, "Design Bases for Protection Against Natural Phenomena," 4, "Environmental and Missile Design Bases," and 60, "Control of Releases of Radioactive Materials to the Environment," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification."

Section 9.2.6, "Condensate Storage Facilities," indicates that a sump collects leakage from the condensate storage tank and pumps it to the radwaste system. This sump is not shown on the P&IDs. Further, there does not appear to be a seismic Category I dike around this large outside storage tank. [Therefore, the applicant must explain how, in the event of rupture of the nonseismic Category I condensate storage tank, the radioactive contents of the tank are prevented from entering the environment and how flooding of safety-related equipment is prevented.]

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Since the description of the equipment and floor drainage system is incomplete, we cannot conclude that the systems meet the guidelines of Positions C.1 and C.3 of Regulatory Guide 1.29 (indicating seismic Category I design for both containment isolation and interfaces with safety-related systems); thus we cannot conclude that the systems comply with the requirements of General Design Criterion 2. Again, because of the incomplete submittal and lack of information regarding catastrophic failure of the condensate storage tank, we cannot conclude that the plant is protected against flooding, in compliance with the requirements of General Design Criterion 4 nor can we conclude that the systems comply with the requirements of General Design Criterion 60 with regard to release of radioactive material to the environment. Therefore, we cannot conclude that the applicant's submittal is acceptable. The equipment and floor drainage systems do not comply with the acceptance criteria of SRP Section 9.3.3.

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sounds an alarm in the control room in case of flooding. The staff, therefore, concludes that this portion of the system meets the requirements of GDC 2, 4, and 60, and the guidelines of RG 1.29.

FSAR Section 9.2.6 indicates that a sump collects leakage from the condensate storage tank and pumps it to the radwaste system. The staff cannot locate this sump on the P&IDs. Further, there does not appear to be a seismic Category I dike around this large outside storage tank. Therefore, the staff has asked the applicant to explain how, in the event of rupture of the nonseismic Category I tank, the radioactive contents of the tank are prevented from entering the environment and how flooding of safety-related equipment is prevented.

The capacities of sumps, drain tanks, and sump pumps have not been presented, and the FSAR does not have complete descriptions of and P&IDs for several areas of the plant. The applicant should review his descriptions of these systems to ensure that all capacities are listed.

The staff cannot conclude that the equipment and floor drainage systems meet the requirements of GDC 2, 4, and 60 and the guidelines of RG 1.29 with respect to protection against internal plant flooding as a result of postulated piping failures, releases of radioactive material to the environment, and seismic classification. Therefore, the applicant's submittal is unacceptable.

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9.3.4 Standby Liquid Control System

The standby liquid control system was reviewed in accordance with Section 9.2.5 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed, according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the standby liquid control system with respect to the applicable regulations of 10 CFR 50.

The standby liquid control system (SLCS) is a reactivity control system. Its purpose is to inject sodium pentaborate solution into the primary system to provide an independent means for shutting down the reactor. The SLCS can bring the reactor from rated power to cold shutdown any time during core life should the normal reactivity control system become inoperable. Thus, it (together with the control rod system) satisfies the requirements of General Design Criterion 26, "Reactivity Control System Redundancy and Capability." (Refer to Section 4.6 of this SER for a discussion of reactivity control.)

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The system consists of a storage tank, a test tank, and two trains, each containing a positive displacement pump, and an explosive-actuated valve at each pump discharge, a motor operated valve at each pump suction together with piping and controls all located within the containment. The maximum temperature at which the solid material would precipitate from solution 59°F while the room in which the equipment containing the borated solution is maintained at a temperature of 70 to 100°F. An electrical resistance heating system maintains the solution in the storage tank at a temperature between 75 and 85 degrees Fahrenheit to prevent precipitation of the sodium pentaborate from solution during storage. Both the high and low tank liquid level and temperature are alarmed in the control room. The two explosive-actuated valves provide assurance that they will open when needed, and the design of the valve assures that boron will not leak into the reactor during SLCS pump testing. The two pumps in parallel trains draw the solution from the storage tank via a common suction line and discharge it into the reactor vessel via a common injection line. The discharge from each pump is provided with a check valve (to prevent backflow), and a crossover line.

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Similarly, the piping at the pump suctions is also connected by a crossover.

Each pump and its associated valves are powered from separate emergency AC power supplies. They are arranged so that failure of a single pump or valve will not prevent adequate amounts of sodium pentaborate solution from entering the reactor vessel to effect shutdown.

System initiation is accomplished by manual actuation of either of two key-locked switches on the control room panel. Changing either switch status to "run" starts an injection pump, actuates an explosive valve on the pump discharge, opens a pump suction valve (which is also the tank outlet valve) and closes the reactor cleanup system isolation valves to prevent loss or dilution of boron. Should the instrumentation provided indicate that the solution is not entering the reactor vessel, the operator can turn the other key-operated switch to the "run" position to actuate the alternate train.

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The SLCS is located in a compartment within the seismic Category I, flood- and tornado-protected containment building outside of the drywell and below the refueling floor. All portions of the SLCS necessary for injection of sodium pentaborate solution into the reactor are seismic Category I, Quality Group B (or Quality Group A if they are part of the reactor coolant pressure boundary). Thus, the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," are complied with by meeting the guidelines of Regulatory Guide 1.29, "Seismic Design Classification."

The containment compartment in which the system is located provides protection against external or internally generated missiles. The SLCS is separated from nonseismic system components and from the effects of breaks in other high- and moderate-energy piping systems. (Refer to Section 3.5.1.2 and 3.6.1 of this SER). Thus, the SLCS complies with the criteria of General Design Criterion 4, "Environmental and Missile Design Bases."

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The SLCS is redundant so that no single active failure will compromise its functional capability. The injection portion of the system can be functionally tested by injecting demineralized water from a test tank into the reactor. In addition, the reactivity control systems have the capability of reliably controlling reactivity changes resulting from postulated accident conditions with allowance for the most reactive control rod withdrawn so that core cooling is maintained, thus satisfying the requirements of General Design Criterion 27, "Combined Reactivity Control Systems Capability" (See Section 4.6, of this SER for further details).

Based on the above, we conclude that the standby liquid control system is in conformance with the requirements of General Design Criteria 2, 4, 26, and 27 as they relate to protected against natural phenomena, system function and system redundancy and testability. The SLCS is, therefore, acceptable. The standby liquid control system complies with the criteria of SRP Section 9.3.5.

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9.4 Air Conditioning, Heating, Cooling and Ventilation Systems

9.4.1 Control Building Ventilation System (Control Room Area Ventilation System

The control building ventilation system was reviewed in accordance with Section 9.4.1 (Section 9.2.2 for the portion of the system known as the Control Building Chilled Water System) of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria except as noted below formed the basis for our evaluation of the control building ventilation system with respect to the applicable regulations of 10 CFR 50.

The control building ventilation system is composed of three separate ventilating systems and a common chilled water system. These systems are, as follows:

(1) Main Control Room Air-Conditioning System,
(2) Standby Switchgear Rooms Air-Conditioning System,
(3) Chiller Equipment Room Air-Conditioning System,
(4) Control Building Chilled Water System (CBCWS).

The main control room air conditioning (MCRAC) system covers the control room and the extended term habitability area of the control building. (Refer to

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Section 6.4 of this SER for further discussion of control room habitability.)

The control building in each unit is served by its own ventilation system and there is no sharing between units. Therefore, the requirements of General Design Criterion 5, ~~"Sharing of General Design Criterion 5,~~ "Sharing of Structures, Systems and Components," are not applicable.

Each of the systems comprising the control building ventilation system is described below, as is each system's compliance with the applicable acceptance criteria found in the SRP.

1) Main Control Room Air Conditioning System (MCRAC)

The main control room air conditioning system consists of two fully redundant trains, each containing filters, chilled water cooling coils, a fan, an electrical heating coil, the ductwork associated with each train and common ductwork, isolation dampers and tornado-missile protected inlet and exhaust louvers. There is also a parallel emergency recirculation system containing charcoal filtration trains for the removal of radioactive particles and noxious gases in the event

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of an accident or high radiation signal. Nonredundant ductwork serves the areas to be cooled. Each train is supplied with power from a separate ESF bus. Chilled water to the cooling coils is provided by a safety-related chilled water system described later in this section. An indication of low flow from the operating train automatically starts the redundant train, repositions the pertinent dampers and sounds an alarm in the control room.

The main control room air conditioning system (MCRAC) is designed to maintain the control room within the environmental limits required for operation of plant cont ols and for uninterrupted safe occupancy during all operating modes including LOCA conditions. The system is designed to maintain the control room under positive pressure, except in the event of a chlorine release in which case the control room is isolated.

The main control room air is normally exhausted through the kitchen and lavatory; isolation of this exhaust is provided by redundant isolation dampers which are designed to actuate in the event of a LOCA, high outdoor radioactivity or chlorine signal (concentration

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of 5 ppm or greater). The outside air supply is diverted automatically, through one of the two charcoal filtration trains in the event of a LOCA (as a precaution) or high radiation signal and is closed, to completely isolate the main control room area, on a high chlorine signal. The main control room air conditioning system also provides cooling for the rooms containing the standby ~~S~~ switchgear rooms air conditioning subsystem.

The emergency recirculation system in the MCRAC system consists of two parallel charcoal filtration trains. Each train contains a charcoal filter unit containing a demister, electric heating coil, HEPA filter, charcoal filter and HEPA filter, a main fan (capacity 4000 cfm) and a decay heat fan (capacity 100 cfm) which exhausts to the MCRAC exhaust duct. The ductwork for both filtration units is joined before the charcoal filter units and after the main (~~100~~ 4000 cfm) fan so that either MCRAC system train may use either filtration unit. Each charcoal filter unit contains a deluge water spray system to extinguish fires in the unit should they occur. A thermistor detection system is installed in the charcoal beds set to alarm in the main control room.

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The MCRAC system is seismic Category I in accordance with the guidelines of Position C.1 of Regulatory Guide 1.29, "Seismic Design Classification." The MCRAC system is located in a seismic Category I building, the control building, thus, is protected against tornadoes, floods and other natural phenomena.

Smoke is removed from the control room by a separate subsystem which is sesimically supported but is not designed to operate after a design basis accident; this meets ~~with~~ Position C.2 of Regulatory Guide 1.29, "Seismic Design Classification." Therefore, the main control room air-conditioning system complies with the criteria of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena by virtue of its location within the control building and by meeting Positions C.1 and C.2 of Regulatory Guide 1.29.

By virtue of its location in the control building, the MCRAC system is protected against damage resulting from high energy pipe breaks. The air intakes and exhaust are protected against damage resulting from high energy pipe breaks. The air intakes and exhaust are protected against damage resulting from high energy breaks. The air intakes and exhaust are protected against tornado missiles. Thus, the main control air conditioning

subsystem complies with the requirements of General Design Criterion 4, "Environmental and Missile Design Bases."

The applicant has considered the accidental release of hazardous chemicals stored onsite in excess of 100 lbs in the design of the MCRAC, thus complying with the requirements of Position C.3 (in regard to reviewing the toxic chemicals stored onsite) of Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release (a detailed discussion of this position is provided in Section 6.4 of this SER).

The applicant has also provided redundant isolation systems, air supply equipment and filtration equipment; this complies with the applicable portion of Position C.14 of Regulatory Guide 1.78. However, the detection instrumentation has been provided solely for chlorine and radioactivity. The need (or lack thereof) for detection of concentrations of other toxic or hazardous chemicals is discussed in Section 6.4 above of this SER. (The applicant must also show that the worst single failure of the radiation detection system will not allow entrance of radioactive material into the

control room, in order to satisfy the single failure criterion in compliance with Section C.14 of Regulatory Guide 1.78).

It is not clear whether the recirculating charcoal filter or equivalent is used when the control room is isolated in the event of a chlorine release as recommended in the guidelines of Section C.4.a of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against An Accidental Chlorine Release." (The applicant should justify this position.)

Compliance with the guidelines of Section C.4.d of Regulatory Guide 1.95 is discussed in Section 6.4, above, of this SER.

Therefore, we cannot find that the MCRAC system complies with the requirements of GDC 19, "Control Room," because of lack of information showing compliance with Positions C.7 and C.14 of Regulatory Guide 1.78 and Position C.4.a of Regulatory Guide 1.95.

The applicant notes that the charcoal ~~filtration~~ filtration trains in the MCRAC systems complies with the guidelines of Positions ~~C.2.a through C.2.a~~ C.2.a through C.2.f and C.2.k of Regulatory Guide 1.52, "Design, Testing, and Maintenance Criteria for

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Post Accident Engineered Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants." [The applicant must provide a detailed response to show compliance with the guidelines of Positions C.2.a through C.2.f and C.2.k of Regulatory Guide 1.52.]

The applicant notes, with regard to compliance with Regulatory Guide 1.52, that an abnormal pressure drop across all critical components of each charcoal filtration train in the MCRAC system is alarmed in the control room and that flow through the charcoal filter train is indicated in the control room but have provided no facilities to record these readings. [The applicant must show how these readings will be recorded, in order to comply with the guidelines of Position C.2.g.]

[The applicant should justify use of associated Air Balance Council tests (as noted in Table 6.5.1 of the FSAR) in determining duct leakage in lieu of testing in accordance with Section 6 of ANSI N510-1975 in order to show compliance with the guidelines of Position C.2.1 of Regulatory Guide 1.52.]

A discussion of compliance with the guidelines of Position C.2.h of Regulatory Guide 1.52 is found in Sections 7 and 8 of this SER; compliance with the guidelines of Position C.2.j is discussed in Section 12 of this SER.

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The charcoal filtration trains of the MRCAC system do not operate during normal operation; therefore the guidelines of Regulatory Guide 1.140, "Design, Testing, And Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration And Absorption Units of Light-Water-Cooled Nuclear Power Plants" are not applicable.

In view of the foregoing, i.e., incomplete compliance with Regulatory Guide 1.52, the MRCAC system fails to comply with the requirements of Regulatory Guide 60, "Control of Releases of Radioactive Materials to the Environment."

2. Standby Switchgear Rooms Air Conditioning (SSRAC) System

The standby switchgear rooms air conditioning system consists of two fully redundant 100% trains, each containing an air filter, electric heating coil, air cooling coils, fan, duct-work associated with each train, and redundant isolation dam-pers. Each of the three battery rooms served by this system contains two 100% exhaust fans. Nonredundant ductwork serves the areas to be cooled.

Both trains are powered from ESF buses so that emergency power is available from the diesel generator if offsite power is lost. Chilled water for the cooling coils is supplied by the safety-related control building chilled water system

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described later in this section. An indication of low flow from the operating train automatically starts the redundant train, repositions the pertinent dampers and sounds an alarm in the control room. This design prevents the failure of both trains due to a single active failure. Purging of smoke, carbon dioxide, or other contaminants from the areas served by the standby switchgear rooms air conditioning system may be accomplished by using a separate subsystem consisting of a fan and associated ductwork. The system serving to remove smoke from areas served by the SSRAC also serves to remove smoke from the main control room; it is seismically supported but it is not intended for use after a DBA, thus complying with Position C.2 of Regulatory ~~Guide~~ Guide 1.29.

The SSRAC system is designed to remove the heat generated in the areas served and maintain the environmental conditions within the limitation of the equipment involved during all operating modes including LOCA conditions. All essential portions of the system are located in the control building which is a seismic Category I, flood and tornado protected structure. The system itself is designed to seismic Category I, Quality Group C requirements. Therefore, the system complies with the guidelines of Position C.1 of Regulatory Guide 1.29. The SSRAC system thus complies with the criteria of General Design Criterion 2 by meeting Positions C.1 and C.2 of Regulatory Guide 1.29 and by virtue of its location in the control building.

[The applicant noted that the standby switchgear equipment is contained in three separate room (each of which includes an associated, walled-off battery room) but did not state where the standby switchgear room AC trains were situated; the FSAR narrative and P&ID should be corrected to show their locations.) [The applicant should also explain whether there is any significance to having two feed lines from the SSRAC system to battery room 1C while the other two rooms have only one.]

The system air intakes and exhausts are provided with tornado missile barriers. Thus, the applicant has shown that the standby switchgear rooms air conditioning system complies with the requirements of General Design Criterion 4, "Environmental and Missile Design Bases."

The equipment in the motor control center, switchgear and miscellaneous electrical areas is not required for control of releases of radioactive materials to the environment, and thus the requirements of General Design Criterion 60, "Control of Releases of Radioactive Materials to the Environment," and the guidelines of Regulatory Guides 1.52, "Design, Testing, and Maintenance Criteria for Post Accident Engineered Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants," and 1.140, "Design, Testing and Maintenance Criteria

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for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light Water Cooled Nuclear Power Plants," are not applicable.

The SSRAC system contains an exhaust subsystem which includes redundant full capacity exhaust fans to serve the battery rooms. One fan in each redundant set is powered from a separate ESF bus. An indication of low flow from the operating fan automatically starts the redundant fans, repositions the pertinent dampers and sounds an alarm in the control room. Thus, each battery room is protected against inoperability due to a single active failure of the SSRAC system. The air passing through each battery room is not recirculated; it passes through the room once, carrying with it any gas generated by the batteries (usually hydrogen) to prevent a dangerous concentration from accumulating.

The equipment in the battery rooms is not required for control of releases of radioactive material to the environment; thus, the requirements of General Design Criterion 60, as reflected in the guidelines of Regulatory Guides 1.52 and 1.140 are not applicable.

This portion of the SSRAC system, that provided ventilation for the three battery rooms meets the guidelines of Position C.1 of Regulatory Guide 1.29, "Seismic Design Classification,"

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thus complying with the requirements of General Design Criteria 2, "Design Bases for Protection Against Natural Phenomena," and 4, "Environmental and Missile Design Bases."

3. Chiller Equipment Room Air Conditioning System

The chiller equipment room air conditioning system consists of an outside air supply system and room exhaust system, constituting the ventilation system. In addition redundant full capacity unit coolers are supplied to cool the room. Both unit coolers are powered from ESF buses so that emergency power is available from the diesel generators if offsite power is lost. Chilled water for the cooling coils is supplied by a safety-related chilled water system described later in this section. An indication of low flow from the operating cooler automatically starts the redundant cooler and sounds an alarm in the control room. The unit coolers are safety-related and designed to maintain conditions so that equipment in the chiller equipment room may be kept operable and is designed to operate during normal conditions, shutdown, loss of offsite power and DBA conditions. The ventilation equipment, however, is not safety-related and is not designed to operate upon loss of offsite power but is seismically supported thus complying with Position C.2 of Regulatory Guide 1.29.

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The chiller equipment room air conditioning system is designed to remove the heat generated in the areas served and to maintain the environmental conditions within the limitations of the equipment in the areas involved during all operating modes including LOCA conditions. All essential portions of the system are located in the control building which is a seismic Category I flood and tornado protected structure. The coolers are designed to seismic Category I, Quality Group C requirements. The outside supply and room exhaust components are nonseismic, Quality Group D and are not powered from ESF buses. However, their loss during normal operating, accident or loss of offsite power conditions does not affect the cooling capability of the systems because the unit coolers continue to cool the rooms, thus meeting ~~with the~~ guidelines of Positions C.1 and C.2 of Regulatory Guide 1.29. The unit coolers have no interfaces with other systems and, therefore, the guidelines of Position C.3 of Regulatory Guide 1.29 do not apply. Thus the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," are complied with by meeting the guidelines of Positions C.1 and C.2 of Regulatory Guide 1.29.

The chiller room air conditioning system is protected against tornado missiles by virtue of its ~~tication~~ location in the control building. This, the system design complies with the criteria of General Design Criterion 4, "Environmental and Missile Design Bases."

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4. Control Building Chilled Water System (CBCWS)

The control building chilled water system consists of two redundant, closed-loop chilled water trains. Each train is capable of meeting the total chilled water needs of the building thus complying with the criteria of General Design Criterion 44, "Cooling Water." Each train contains two 50 percent capacity electric motor driven centrifugal liquid chillers, two 100% capacity chilled water recirculation pumps, two 100% capacity condenser cooling water pumps and one chilled water compression tank. Both trains are powered from ESF buses so that emergency power is available from the diesel generators if offsite power is lost. Operating parameters are indicated in the control room with off normal conditions alarmed. Switchover from faulted to standby equipment is manual. Cooling water for the chiller condensers is supplied by the normal service water system for normal operation and by the safety-related standby service water system for operation during accident or loss of offsite power conditions.

Because of crossties, it is possible to circulate standby service water through this system to provide emergency cooling should the chillers and/or chilled water recirculating pumps in both trains fail. However, neither this arrangement nor the condenser cooling water pumps are shown on the P&IDs. [The applicant must provide this information so that we may determine if the arrangement is acceptable.]

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The system is designed to remove the heat generated in the control building and maintain the environmental conditions within the limitations of the equipment involved during all operating modes including LOCA conditions. All essential portions of the system are located in the control building which is a seismic Category I, flood and tornado protected structure. All parts of the system are designed to seismic Category I, Quality Group C standards. This meets the guidelines of Position C.1 of Regulatory Guide 1.29. Therefore, we conclude that the system complies with the requirements of General Design Criteria 2, "Design Bases for Protection Against Natural Phenomena," and ~~General Design Criterion~~ 4, "Environmental and Missile Design Bases."

Controls are provided to permit periodic testing of the performance of individual system components and the performance of the entire system during normal operation and during shutdown. Water flow rates may be balanced during testing and set to design conditions. These provisions for testing comply with the requirements of General Design Criterion 46, "Testing of Cooling Water System." Instrumentation has been provided to determine proper system operation during performance testing. In addition, periodic inspections of equipment are scheduled to assure proper operation, thus complying with the requirements of General Design Criterion 45, "Inspection of Cooling Water System."

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CONCLUSION

Based on the foregoing, we conclude that the main control room air conditioning system, the standby switchgear rooms air conditioning system, the chiller equipment room air conditioning system, and the control building chilled water system of the Control Building Ventilation System comply with the requirements of General Design Criteria 2, 4 and 5.

The control building chilled water system complies with the criteria of General Design Criteria 44, 45 and 46 as required. The main control room air conditioning system (the only one of the four systems comprising the control building ventilation system required to comply with the criteria of General Design Criteria 19 and 60) does not comply with the criteria of General Design Criteria 19 and 60.

In view of the foregoing, we cannot find the applicant's submittal with regard to the control building ventilation system acceptable. The control building ventilation system does not comply with the acceptance criteria of SRP Section 9.4.1.

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9.4.2 Fuel Building Ventilation System (Spent Fuel Pool Area Ventilation System)

The fuel building ventilation system was reviewed in accordance with Section 9.4.2 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed, according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria except as noted below formed the basis for our evaluation of the fuel building ventilation system with respect to the applicable regulations of 10 CFR 50.

The fuel building ventilation system, which serves the entire fuel building, is designed to maintain a suitable environment for equipment operation and to limit potential radioactive release to the atmosphere during normal operation and postulated fuel handling accident conditions. The system is not required for safe shutdown of the plant in the event of a LOCA but is only required to mitigate the consequences of a fuel handling accident. The essential parts of the system are classified as seismic Category I, Quality

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Group C. All essential parts of the system are powered from independent standby emergency (Class 1E) power sources.

The system consists of four subsystems: the supply air system, the unit coolers system, the exhaust air system, and the charcoal filtration system. The supply air system consists of tornado protected outside air intake louver, with redundant isolation dampers, inlet air filter, electric heating coil, chilled water cooling coil, two 100 percent capacity supply fans, supply air ductwork, dampers, and discharge air openings. The system is seismic Category I, Quality Group C from the air intake louver through the second isolation damper. The balance of the supply system is nonseismic Quality Group D. However, seismic Category I, Quality Group C ductwork with redundant isolation dampers serves as an emergency outside air intake subsystem to distribute air throughout the building whenever the normal distribution system is shut down or unavailable. This system is also connected to the intake louver via its connection to the supply air system downstream of the intake louver.

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There are eight separate unit coolers comprising the unit coolers system; the coolers are located in various areas of the building to provide additional cooling for areas of high heat load. Each unit cooler consists of a housing, filter, chilled water cooling coil, fan and ductwork. Each unit cooler is designed to standards consistent with those for nonseismic Category I, Quality Group D components. During emergency or accident conditions, outside air is provided by the emergency air intake and is drawn into the charcoal filter rooms, spent fuel pool area and fuel pool cooling pump room to cool these areas; the air is exhausted by means of the charcoal filter exhaust fans.

The exhaust air system consists of air collection ductwork, two 100 percent capacity exhaust fans, redundant isolation dampers, and a tornado missile protected roof exhaust plenum. The system is seismic Category I, Quality Group C with the exception of the exhaust fans, dampers and ductwork between the four isolation dampers (two upstream, two downstream of the exhaust fans) that are nonseismic Category I, Quality Group D.

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The charcoal filtration system is an emergency safety feature consisting of two redundant filtration trains each containing an exhaust fan, ductwork, dampers and a filter unit. Each filter unit includes a moisture separator, electrical heating coil, pre-filter, HEPA filter, charcoal filter, and another HEPA filter. This entire system is seismic Category I, Quality Group C. There is a small fan (100 cfm capacity) designed to remove the decay heat originating in the radioactive materials captured and retained within the train. Each charcoal filtration unit contains a water spray to extinguish fires should they occur.

Ventilation of the spent fuel pool is maintained by a push-pull system in which outside air is supplied to one end by the normal air intake and exhausted from the other and by means of the normal exhaust fans. In the event of accident or emergency conditions, air for the spent fuel pool, spent fuel pool cooling pump room and charcoal filter rooms is drawn in from the outside through the emergency intake and exhausted through the charcoal filtration system by means of the charcoal filtration unit exhaust fans.

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The fuel building is maintained at a slight negative pressure (1/4 inch W.G.) to prevent escape of any contaminated air to the atmosphere. During normal operation the charcoal filter unit is bypassed. Upon a radiation level higher than a preset level in the exhaust duct, the exhaust stream is automatically diverted so that it passes through the charcoal filtration system which is automatically started. Both charcoal filtration units in the charcoal filtration system are automatically actuated and the intake air supply diverted to the emergency outside air intake in the event of any of the following isolation signals:

1. High drywell pressure
2. Reactor vessel low-low water level, and
3. A manual containment isolation signal.

These signals also isolate the normal exhaust fans. The operator can stop one of the two charcoal filtration units from the main control room, after startup, in the event only one unit is required.

During fuel handling operations exhaust air from the spent fuel pool and fuel cask areas are routed through the charcoal filter units. When only one

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unit is operating control logic causes the redundant charcoal filter unit to start automatically when the air flow drops to a preset low level.

The airflow pattern is from potentially low radioactivity areas to potentially higher radioactivity areas. Slightly more air is exhausted than is supplied, thereby preventing short-circuiting of air flow and assuring that no radioactivity escapes from the fuel building.

All essential parts of the fuel building ventilation system are seismic ~~Category I~~ Category, Quality Group C, thereby satisfying the guidelines of Position C.1 of Regulatory Guide 1.29, "Seismic Design Classification." The system is located in the fuel building which is seismic Category I, flood and tornado protected. This satisfies the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena." There are no high or moderate energy systems located near the fuel building ventilation system and adequate protection against internally generated missiles and the effects of pipe whip and fluid jets is provided by separated

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equipment locations (refer to Section 3.5.1.1 and 3.6.1 of this SER), thus complying with the requirements of General Design Criterion 4, "Environmental and Missile Design Bases."

Each unit of the plant has its own fuel building and fuel building ventilation system. There is no sharing of ventilation system functions. Therefore, the requirements of General Design Criterion 4, "Sharing of Structures, Systems and Components," are not applicable.

The applicant stated that the charcoal filtration system complies with the guidelines of Sections C.2.a through C.2.f and C.2.k of Regulatory Guide 1.52, "Design, Testing, And Maintenance Criteria For Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration And Adsorption Units of Light-Water-Cooled Nuclear Power Plants." [The applicant must provide a detailed response to show compliance with the guidelines of Positions C.2.a through C.2.f and C.2.k of Regulatory Guide 1.52.]

The applicant stated that an abnormal pressure drop across all critical components of the charcoal filtration system is alarmed in the control room and flow through units are indicated in the control room but have provided no facilities to record these readings as specified in Position C.2.g of Regulatory Guide 1.52. [The applicant must verify that records of volumetric flow and pressure drop will be kept to assure charcoal train operability in the event they need to be used, in order to comply with Position C.2.g.]

[The applicant should verify that use of Associated Air Balance Council methods in conducting duct leak tests is equivalent to compliance with Position C.2.1 of Regulatory Guide 1.52.]

A discussion of compliance with Section C.2.h of Regulatory Guide 1.52 is found in Sections 7 and 8 of this SER; compliance with Section C.2.i is discussed in Section 7 of this SER; compliance with Section C.2.j is discussed in Section 12 of this SER.

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The charcoal filtration subsystem of the fuel building ventilation system does not operate during normal operation; therefore, the guidelines of Regulatory Guide 1.140, "Design, Testing, And Maintenance Criteria For Normal Ventilation Exhaust System Air Filtration And Adsorption Units of Light-Water-Cooled Nuclear Power Plants" are not applicable.

In view of incomplete information with regard to compliance with Regulatory Guide 1.52, we cannot conclude that the fuel building ventilation system complies with the requirements of General Design Criterion 60, "Control of Releases of Radioactive Materials to the Environment."

The fuel building internal pressure is maintained at -1/4 inch W.G. so that any air containing radioactivity which is inside the building cannot be exhausted to the atmosphere without passing through the charcoal filtration units to remove the radioactive material. This negative pressure is maintained during all operating and accident conditions including refueling, thus meeting the guidelines of the portion of Position C.4 of Regulatory Guide 1.13, "Spent Fuel Storage

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Facility Design Basis." The remaining portion of Position C.4 relating to a fuel handling accident is addressed in Section 15.7.4 of this SER. This complies with the requirements of General Design Criterion 61, "Fuel Storage and Handling and Radioactivity Control."

We conclude that the fuel building ventilation system is in compliance with General Design Criteria 2, 4, 5, and 61; however, [redacted] the applicant must provide the information indicated to show compliance with General Design Criterion 60. Therefore, we cannot find that the applicant's submittal is acceptable. The applicant's submittal with regard to the fuel building ventilation system does not comply with SRP Section 9.4.2.

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9.4.3 Auxiliary and Radwaste Area Ventilation System

The auxiliary and radwaste area ventilation system was reviewed in accordance with Section 9.4.3 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed, according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria except as noted below formed the basis for our evaluation of the auxiliary and radwaste area ventilation system with respect to the applicable regulations of 10 CFR 50.

The auxiliary and radwaste area ventilation system serves the radwaste building and the auxiliary building with a separate system for each area. The radwaste building ventilation system provides ventilation for the radwaste building (which is shared by Units 1 and 2) while identical auxiliary building ventilation systems in each unit provide ventilation for the auxiliary building in that unit.

The radwaste building ventilation system (RBVS) has no safety function and does not operate on loss of offsite

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power. Accordingly, the system has been designed in accordance with Quality Group D and nonseismic Category I standards. The RBVS is separated into two major sections - the air supply system and air exhaust system. The former contains an air intake filter unit, unit coolers and distribution ductwork. Each unit cooler contains an electric heating coil, cooling coil and vaneaxial fan; water is supplied to the cooling coil from the radwaste building chilled water system (see Section 9.2.9 of this SER for further discussion of the radwaste building chilled water system). The air from the outside is filtered, conditioned by the coolers and then transported throughout the ductwork to the various parts of the radwaste building by means of the vaneaxial fans in each of the six unit coolers.

The RBVS exhaust system processes air in two separate paths; the air from tanks, centrifuges and other contaminated equipment is processed through a charcoal filter train containing (in order) a demister, electric heating coil, prefilter, HEPA filter bank, charcoal filter and HEPA filter bank all mounted within a welded steel housing. Each charcoal filter is provided with an integrally mounted water spray to extinguish any fires arising in the filtration unit.

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A centrifugal fan (4000 cfm capacity), downstream of the charcoal filter unit in the RBVS, draws the air through the ductwork and charcoal filter unit and ejects it through the roof exhaust duct. A small (100 cfm) centrifugal fan is mounted in parallel with the larger fan to remove decay heat generated in the charcoal filter unit which arises as a result of decay of radioactive material captured in the components of the charcoal filtration unit. A thermistor detection unit is installed in each charcoal filter unit to annunciate an alarm in the event of a fire. There are two parallel charcoal filter/fan units with one unit in operation and one in standby. Each charcoal filter unit contains a water spray system to extinguish fires in the unit should they occur.

The air from the general areas and equipment areas in the radwaste building utilizes a portion of the exhaust ductwork separate from that utilizing the charcoal filter units and is drawn through the ductwork by two vaneaxial 50% fans (a third fan is mounted in standby condition) and then ejected through the roof exhaust duct (the ductwork from the charcoal filtration units

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and from the vaneaxial fans is joined before the roof exhaust duct). The exhaust air system is maintained at a slight negative pressure in order to prevent unmonitored leakage of contaiminated material to the environment. Radiation monitors in the combined exhaust duct are used to detect high concentrations of radioactive particulate and/or gaseous matter in the exhaust. The applicant stated that "in the event of detection of high airbone radioactivity in the building, administrative control is required to shut down the system." [The applicant should explain whether this means that the building is to be shutdown or whether such detection requires determination by an administrative official before the building could be shutdown.]

The radwaste building ventilation system P&ID, (Figure 9.4-3a) shows that the single outlet damper leading to the roof exhaust duct is closed upon a high radiation signal. [The applicant should explain what would be the effect of an open damper (resulting from damper failure).]

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Since the radwaste building ventilation system is not safety-related nor does its failure affect any safety-related system, the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," (except for Position C.2 of Regulatory Guide 1.29, which it meets,) 4, "Environmental and Missile Design Bases," and 5, "Sharing of Structures, Systems and Components" do not apply.

The applicant notes that the charcoal filtration system in the RBVS complies with the guidance of Positions C.1.a through C.1.d and C.2.b, C.2.c and C.2.e of Regulatory Guide 1.140, "Design, Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants." A discussion of compliance with Position C.2.d is contained in Section 12 of this SER. Further, the applicant notes that housing leak tests were performed in accordance with the guidelines of Position C.2.f but the duct leak tests were not. The duct leak tests were performed in accordance with the methods of the Associated Air Balance Council. [The applicant must verify that use of the methods of ~~the~~

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the Associated Air Balance Council is equivalent to compliance with Position C.2.f of Regulatory Guide 1.140.] [The applicant must also show, how the system complies with the guidelines of Positions C.1.a through C.1.d, C.2.a, C.2.b, C.2.c, and C.2.e.]

The auxiliary building ventilation system (ABVS) is used to maintain an environment with controlled temperature and flow of air to assure personnel comfort and integrity of equipment during normal operation. Some parts of the system are required to operate during accident conditions and in the event of high radiation in the auxiliary building. The ABVS is comprised of three relatively independent parts which are:

1. The supply air system,
2. The exhaust air system, and
3. The unit coolers system

The ABVS supply air system consists of two 100% capacity (10000 cfm) vaneaxial fans (in parallel) to a prefilter, high efficiency filters, dampers and ductwork. The inlet is protected against the effects of missiles and tornadoes; the ductwork from the inlet dampers up

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to and including the two isolation dampers in series is manufactured to meet the requirements of Quality Group C and seismic Category I standards. The rest of the supply air system including the vaneaxial fans, filters and the rest of the ductwork serving to distribute the incoming air throughout the auxiliary building is designed to nonseismic ~~Category I~~ Category 1, and Quality Group D standards. Each inlet duct ends in the particular area served. There is also a part of the intake system which serves the auxiliary building electrical tunnel and is separate from the rest of the supply system. The ductwork serving this system connects to the intake ductwork downstream of the isolation dampers. The ductwork ends in the area served, the electrical tunnel. This portion of the supply system contains a vaneaxial fan (capacity 3600 cfm) to draw air from the intake and to blow it into the electrical tunnel.

Each duct in the ABVS exhaust system starts in the area of the auxiliary building which it serves. These ducts are combined into a single duct which is connected to the suction of the two vaneaxial fans in parallel

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(capacity 10000 cfm) which take the air in and blow it into the exhaust duct. The duct from the auxiliary building electrical tunnel has its own vaneaxial fan (capacity 3600 cfm), however. As a consequence, the duct from the electrical tunnel bypasses the parallel vaneaxial fans, combining directly with the downstream duct leading to the exhaust duct. An alternate avenue for exhaust which may be used in the event of high radiation in the auxiliary building or a DBA is through the standby gas treatment system (SGTS); here the single exhaust duct to the suction of the parallel vaneaxial fans branches into two ducts with a single isolation damper on each duct leading to the SGTS and additional isolation dampers on the SGTS. [The applicant should show that no single failure of the isolation dampers between the auxiliary building ventilation system and SGTS either during normal operation or accident conditions will damage or defeat the SGTS so that it could not function as required.]

The third system in the ABVS, the unit coolers systems, serves to cool the following areas in the auxiliary building:

1. Switchgear areas,
2. Main ~~steam~~ steam pipe tunnel, north end,

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3. East and west general areas, annulus mixing system, and charcoal filter room,
4. Charcoal filter rooms,
5. HPCS pump cubicle,
6. RHR pump room and heat exchanger cubicle,
7. RPCCW pumps and heat exchanger cubicles,
8. RWCU pump cubicle,
9. RCIC pumps and turbine cubicle,
10. General areas
11. LPCS pump cubicle

Each ABVS unit cooler is comprised of throwaway filters, cooling coils, a vaneaxial fan and ductwork. Cooling water is normally supplied by the service water system, described in Section 9.2-1. In the event of a loss of offsite power or a DBA, water is supplied by the safety-related standby service water system (described in Section 9.2.7 of this SER) and power to operate each cooler is supplied by an onsite safety-related power supply. The applicant states that three unit coolers serve safety-related pump rooms, as follows: one room cooler serves the room containing the HPCS pump, one unit cooler serves the room containing RHR heat

exchangers A and C, pump room A, the LPCS pump room and RCIC pump room, and one serves the room containing RHR heat exchangers B and D, room B and pump room C. It is not clear what safety-related equipment is contained in each room. [The applicant should provide the necessary clarification.]

Each unit cooler protecting an ECCS component is powered from the same emergency power source as the ECCS component. Therefore, the HPCS and RCIC rooms are cooled whenever their respective pumps operate which includes operation upon loss of offsite power, thus complying with the provisions of Item II.K.3.24 of the TMI action plan. Note that these unit coolers are considered part of the ESF ventilation systems (see SER Section 9.4.5). Table 3.2-1, Item XXXI shows that ductwork and dampers for the coolers are designed to Quality Group C and seismic Category I standards but omits the standards for the unit cooler fans. [The applicant should add this information to the table.] Table 3.2-1 shows that all unit coolers are designed to Quality Group C and seismic Category I standards. [The applicant should verify the information in Table 3.2-1.] [The applicant should show the area(s) each

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cooler serves and the number of unit coolers on the P&ID for this system.] [The applicant should provide justification for areas in the auxiliary building not cooled in the event of a DBA or high radiation signal.]

The applicant notes that the ABVS supply air system is designed to bring in fresh, clean air from outside into the various portions of the auxiliary building; the exhaust system is designed to remove the air from these areas, together with any contaminants which the room may contain by suitably separating the inlet and exhaust ducts. The building is maintained at a slight negative pressure (1/4 inch W.G.) to prevent contaminants inside the building from entering the environment without proper treatment.

Thus, during operation, the air enters the auxiliary building as fresh air containing little or no contaminants such as radioactive particles or gases. The concentration of contaminants gradually increases as it passes through the building, into the exhaust system. However, on a LOCA signal (including a manual containment isolation signal) or high radiation level within the auxiliary building, the intake is isolated from

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the rest of the supply air systems by means of the two isolation dampers in series.

As noted above, the air in the building, is normally transported to the exhaust duct. However, in the event of radioactivity in the exhaust attaining a preset level, the normal exhaust is automatically isolated and exhaust dampers to the standby gas treatment (SGTS) are opened automatically to permit treatment of the exhaust stream by the SGTS (see Section 6.5 of this SER for a more complete discussion regarding the standby gas treatment system). All of the components of the ABVS are protected against floods and tornadoes by virtue of being located within the auxiliary building.

The auxiliary building ventilation system meets the guidance of Position C.1 and C.3 of Regulatory Guide 1.29, "Seismic Design Classification," since the safety-related portions are designed in accordance with seismic Category I criteria, and the interfaces with nonseismic Category I features are designed in accordance with seismic Category I criteria, with the exception of a lack of information regarding the design of the unit cooler fans. Therefore, we cannot determine whether

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the ABVS is in compliance with the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

The auxiliary building ventilation system is protected against functional failure as a result of internally generated missiles by separation of redundant equipment. The applicant notes (in FSAR Section 3.4-1) that this system is protected against flooding as a result of a break of a fluid line within the building. Therefore, the system complies with the guidelines of General Design Criterion 4, "Environmental and Missile Design Bases."

Each unit has its own auxiliary building ventilation system and, therefore, the requirements of General Design Criterion 5, "Sharing of Structures, Systems and Components," are not applicable. It is noted that part of the exhaust system of the auxiliary building ventilation serves to exhaust air from the auxiliary building which is potentially radioactive during a DBA or which actually contains radioactive material (in the event of a high radiation signal in the auxiliary building), thus serving as part of the standby gas

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treatment system. (As such, the applicant should show how this portion of the ABVS meets the guidelines of applicable portions of Position C.2 of Regulatory Guide 1.52, "Design, Testing and Maintenance Criteria for Post Accident Engineered Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," in order to comply with the requirements of General Design Criterion 60, "Control of Releases of Radioactive Materials to the Environment.") No part of the ABVS is used as part of a cleanup system during normal operation, and so, the provisions of Regulatory Guide 1.140, "Design, Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants" do not apply.

[The applicant should also show how the auxiliary building temperature will be maintained at the minimum level of 40°F in the winter, as specified in Table 9.4-1, and also how this minimum temperature would permit normal operation and maintenance of the auxiliary building and its contents by operating personnel.]

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We find, also, that the radwaste building ventilation system does not comply with Position C.2 of Regulatory Guide 1.29, thus, not in compliance with General General Design Criterion 2 and is not in complete accordance with Position C.2 of Regulatory Guide 1.140, thus failing to comply fully with the criteria of General Design Criterion 60. [The applicant should also discuss the effect of an open outlet damper when high radioactivity is encountered in the discharge duct.]

In view of the concerns noted above, we cannot find the applicant's submittal with regard to the auxiliary ventilation and radwaste ventilation system acceptable. The auxiliary and radwaste area ventilation system does not comply with the acceptance criteria of SRP Section 9.4.3.

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9.4.4 Turbine Building Ventilation System

The turbine building ventilation system was reviewed in accordance with Section 9.4.4 of NUREG-0800 (SRP. An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed, according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria except as noted below formed the basis for our evaluation of the turbine building ventilation system with respect to the applicable regulations of 10 CFR 50.

The turbine building ventilation system (TBVS) is comprised of the following subsystems:

1. Main supply system,
2. Unit coolers system,
3. Offgas area/condensate demineralizer area ventilation system,
4. Exhaust air system,
5. Charcoal filtration system,
6. Sample room air conditioning system.

The main supply system provides filtered, conditioned air for ventilation of the various areas of the

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turbine building. The outside air enters through a single duct passing through the air conditioning unit which consists of a filter, electric heating coil, and water cooling coil. The duct is then divided in two to accommodate two vaneaxial 50% capacity (1700 cfm) fans which serve to distribute the conditioned air through the duct system leading to the various parts of the turbine building.

The unit coolers system provides air conditioning for various areas of the turbine building. Each cooler is independent of the others; each consists of a housing, throwaway type filter, cooling coil, vaneaxial fan and ductwork. Chilled water for the cooling coils is supplied by the ventilation chilled water system (see Section 9.2.9 of this SER for a further discussion of the ventilation chilled water system).

The offgas/condensate demineralizer area ventilation system is supplied with air transferred from the turbine building. The applicant states that the vault holding the offgas charcoal absorber tanks is cooled by low temperature refrigeration trains, each consisting of a compressor, refrigerant condenser, cooler, air handling

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equipment, liquid receiver, regeneration equipment together with the necessary equipment and controls. One train operates to keep the vault cool while one train remains in standby. Unit coolers are provided for some areas while some areas contain exhaust ducts leading to the separate exhaust air system which contains two 100% capacity (8300 cfm) vaneaxial fans in parallel (one operating, one on standby) blowing air into the plant exhaust duct.

The turbine building exhaust system contains exhaust ductwork from various portions of the plant combined into one duct leading to the three 50% capacity fans, each with a capacity of 22000 cfm, operating in parallel (two operating, one on standby) which blow the exhaust air into the plant exhaust duct.

The charcoal filtration system exhausts air from the two mechanical vacuum pumps used to remove air and radioactive gases from the main condenser on startup. The air is drawn through a charcoal filter unit consisting of a demister, electric heating coil, prefilter, HEPA filter, charcoal filter and HEPA filter with a 5000 cfm centrifugal fan downstream which draws the air

through the charcoal filtration unit and blows it into the plant exhaust duct. A small centrifugal fan (100 cfm) is provided to remove the decay heat generated in the charcoal filtration unit by radioactive material captured in the unit. The charcoal filtration unit contains a water spray system to extinguish fires should they occur.

The applicant stated that the charcoal filtration system complies with the guidelines of Position C.1.a through C.1.d, C.2.a through C.2.c and C.2.e of Regulatory Guide 1.140, "Design Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants." (The applicant should provide detailed information to show compliance with Positions C.1.a through C.1.d, C.2.a through C.2.c and C.2.e.)

Compliance with the guidelines of Position C.2.d is discussed in Section 12 of this SER. The applicant notes that duct leak tests are performed in accordance with the methods of the Associated Air Balance Council in lieu of Section 6 of ANSI N510-1975 as specified in Position C.2.f. (The applicant should verify that use

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of the methods of the Associated Air Balance Council in conducting duct leak tests is equivalent to compliance with Position C.2.f of Regulatory Guide 1.140.)

The applicant states that the sample room is provided with a self-contained air conditioning system consisting of throwaway filter, direct expansion cooling coil, heat coil and distribution ductwork. This unit supplies 1300 cfm for ventilation. However, this unit does not appear on the P&ID (Figure 9.4-4) for the TBVS. (The applicant should include this unit in the figure.)

The applicant states that the turbine building ventilation system is used during normal operation and is not required to operate during a design basis accident. (The applicant must show that the system meets the guidelines of Position C.2 and C.3 of Regulatory Guide 1.29, thus complying with tne requirements of General Design Crierion 2, "Design Bases for Protection Against Natural Phenomena.") The system has no safety-related function, therefore the requirements of General Design Criterion 4, "Environmental and Missile Design Bases," and General Design Criterion 5, "Sharing of Structures,

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Systems and Components," are not applicable. The applicant must meet the applicable guidelines of Regulatory Guide 1.140 in order to show compliance with the requirements of General Design Criterion 60.

In view of the foregoing, we cannot find that the applicant's submittal regarding the TBVS is acceptable. The turbine building ventilation system does not comply with the acceptance criteria of SRP Section 9.4.4.

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9.4.5 Engineered Safety Feature Ventilation Systems

The engineered safety features (ESF) ventilation systems were reviewed in accordance with Section 9.4.5 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed, according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria ~~except as noted below~~ formed the basis for our evaluation of the engineered safety features ventilation systems with respect to the applicable regulations of 10 CFR 50.

The ESF ventilation systems provide cooling for various ESF systems or functions which include:

1. Standby switchgear and battery rooms,
2. Safety-related pump rooms and the exhaust of the auxiliary building by standby gas treatment system (when high radioactivity found in exhaust stream),
3. Fuel building ventilation system (charcoal filtration units),
4. Annulus mixing system,
5. Containment ventilation system,

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6. Diesel generator building ventilation system, and

7. Standby service water pumphouse ventilation system.

Items 1, 2, 3, 4 and 5 above are reviewed in other sections of this SER; for discussion of those items see the following SER sections: For Items 1 and 2, SER Section 9.4.1; for Item 3, SER Section 9.4.2; and for Items 4 and 5, see SER Section 9.4.6. The remaining items (6 and 7) will be discussed in this Section.

9.4.5.1 Diesel Generator Building Ventilation System

Each unit has its own diesel generator building. Each of the three diesel generators in each unit is located in a separate room within the building. The diesel generator building ventilation system consists of six independent subsystems, two for each diesel generator, to assure adequate air flow in the event of a single failure. Each subsystem has an exhaust fan for the diesel generator room automatically controlled by a temperature switch to start on high ambient temperature. The diesel generator control room (one for each diesel generator) is ventilated by a dedicated supply fan, filters, and ductwork and exhausts to the diesel generator room through

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fire dampers. Each diesel generator room is also provided with a nonsafety-related and nonseismic exhaust fan to prevent stagnation of room air; each system is powered by the emergency bus served by its diesel generator. However, we cannot determine how the design accommodates minimum and maximum outside temperatures; thus, we cannot determine whether the requirements of General Design Criterion 4, "Environmental and Missile Design Bases," are met with regard to accommodating environmental conditions during normal operation and during accidents. (The applicant must provide information to show that the desired temperature range can be obtained during all operating and accident conditions and how damage from freezing during winter time is prevented when the diesel generations are not operating.)

The system is designed to seismic Category I, Quality Group C requirements and is housed in the seismic Category I, flood and tornado protected diesel generator building, thereby satisfying the requirements of General Design Criterion 2 and the guidelines of Regulatory Guide 1.29. The inlet and outlet louvers are tornado missile protected as is the diesel generator building. The system is separated

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from high energy piping systems and internally generated missiles, thereby satisfying the requirements of General Design Criterion 4 relating to missiles, pipe whip and discharging fluids. There is no sharing of systems; thus, the requirements of General Design Criterion 5 are not applicable. The inlet louvers are approximately 30 feet above grade, thereby meeting the guidance of Item 2, Subsection A, of NUREG/CR-0660, "Enhancement of Onsite Emergency Diesel Generator Reliability," and therefore, the pertinent requirements of General Design Criterion 17, "Electric Power Systems," relating to the protection of essential electrical components from failure due to the accumulation of dust and particulate material are satisfied.

9.4.5.2 Standby Service Water (SSW) Pumphouse Ventilation System

The SSW pumphouse ventilation system provides ventilation for rooms containing the safety-related SSW pumps (the SSW cooling tower pumphouse) and for rooms containing SSW switchgear. The standby service water pumps for each unit are located in separate rooms in each unit's SSW cooling tower pumphouse. A separate ventilation system is provided for each

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pump room and consists of two 100% capacity supply fans, inlet plenum, dampers, fire dampers, and associated controls. Each train is powered from the same emergency buses as the pump it serves and is automatically started when its corresponding pump is started. In the event of failure of an operating train, the redundant train is automatically initiated, thus preventing system inoperability resulting from a single failure.

Associated with each pump is a transformer and switchgear located in a separate room. A separate ventilation system is provided for each switchgear room and consists of two 100% capacity supply fans, dampers, fire dampers and associated controls. Each train is powered from the same emergency bus as the pump it serves and is automatically started whenever its corresponding pump is started. In the event of failure of an operating train the redundant train is automatically started, thus preventing system inoperability in the event of a single failure.

The systems are housed in the standby service water pumphouse which is seismic Category I, flood and

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tornado protected ~~protected~~, and the systems themselves designed to seismic Category I, Quality Group C requirements, thereby satisfying the guidelines of Regulatory Guide 1.29, "Seismic Design Classification," thus complying with the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena." The pumphouse is designed against the effects of tornado missiles and is separated from high energy piping systems and internally generated missiles, thereby satisfying a portion of the requirements of General Design Criterion 4, "Environmental and Missile Design Bases." There is no sharing of systems; thus, the requirements of Design Criterion 5, "Sharing of Structures, Systems and Components," are not applicable. (However, the applicant must provide sufficient information to enable us to determine how the desired temperature ranges are maintained in both the switchgear and pump rooms during the winter, the humidity range to be maintained in the switchgear rooms and how this humidity range is maintained in order to comply, completely, with the requirements of General Design Criterion 4, "Environmental and Missile Design Bases.")

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We find the diesel generator ventilation system and standby service water pumphouse ventilation systems in compliance with the requirements of General Design Criterion 2; since there is no sharing between units, General Design Criterion 5 is not applicable. In addition, the diesel generator ventilation system complies with the pertinent portions of General Design Criterion 17. However, the applicant must provide further information to show that these systems comply with the requirements of General Design Criterion 4. Therefore, we cannot conclude that the applicant's submittal regarding these systems is acceptable. The diesel generator ventilation system and standby service water pumphouse ventilation system do not comply with the criteria of SRP Section 9.4.5.

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9.4.6 Miscellaneous Building Heating Ventilating and Air Conditioning Systems

The miscellaneous building heating, ventilating, and air conditioning systems were reviewed in accordance with Section 9.4.3 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section were performed, according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria ~~except as noted below~~ formed the basis for our evaluation of the miscellaneous building heating, ventilating and air conditioning systems with respect to the applicable regulations of 10 CFR 50.

The miscellaneous systems considered in this section are:

1. Fire Pumphouse (Heating and Ventilation System)
2. Normal Switchgear Building (Heating Ventilation and Air Conditioning System)
3. Auxiliary Boiler Building (Heating Ventilation and Air Conditioning System)

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4. Water Treatment (Heating and Ventilation System)
5. Makeup Water Intake Structure and Switchgear House (Heating and Ventilation System)
6. Electrical and Piping Tunnels (Ventilation System)

These systems are designed to provide a suitable environment for personnel and equipment operation. They also provide filtered, tempered air for all air-conditioned areas and electrical equipment areas and also provide exhaust hoods to remove noxious fumes from the battery room and water treatment area.

None of the systems has any safety-related function, nor does failure of any system compromise any safety-related system or components. Failure of any system will not prevent safe shutdown of the reactor. Therefore, no system is designed to seismic Category I standards or to Quality Group A, B, or C standards with the exception of ventilation systems for the electrical and piping tunnels which are provided with seismic Category I supports. This was done in order to maintain the structural integrity of these systems in the event of a seismic occurrence. Therefore, the systems are not required to comply with the seismic

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requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," as outlined in Regulatory Guide 1.29, "Seismic Design Classification."

General Design Criterion 5, "Sharing of Structures, Systems and Components" is not applicable because none of these systems has any safety-related function.

These systems are not designed to control release of radioactive material, therefore, General Design Criterion 60, "Control of Release of Radioactive Materials to the Environment" is not applicable.

Therefore, we find the applicant's submittal with respect to miscellaneous buildings heating ventilation, and air conditioning systems to be acceptable. These systems are in compliance with the acceptance criteria of SRP Section 9.4.3.

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9.5 Other Auxiliary Systems

9.5.1 ~~Fire Protection~~ Fire Protection Review

We have reviewed the fire protection program for conformance with the Standard Review Plan (NUREG-0800) Section 9.5-1, Fire Protection, dated July 1981. This document contains, in BTP CMEB 9.5-1, the technical requirements of Appendix A to BTP ASB 9.5-1 and Appendix R to 10 CFR 50. Because the applicant has compared his program to the latter guidelines, our report also references these guidelines.

The applicant's Fire Protection Evaluation Report, transmitted by letter dated October 1981 was in response to our request to evaluate their fire protection program against the guidelines of Appendix A to Branch Technical Position ASB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants." By letter dated June 1982 the applicant provided an evaluation against the requirements of Appendix R to 10 CFR 50.

As part of our review, we will visit the plant site to examine the relationship of safety-related components, systems, and structures in specific plant areas to both combustible materials and to associated fire detection and suppression systems. The site visit has not been conducted up to this time because the construction of the plant has not progressed to the level where such a visit would be meaningful.

Our review included an evaluation of the automatic and manually operated water and gas fire suppression systems, the fire detection systems, fire barriers, fire doors and dampers, fire protection

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The various components of the fuel pool cooling systems are located in shielded cubicles or are separated from other moderate and high energy piping systems and are thus protected against the effects of internally generated missiles and the effects of pipe whip and jets (refer to Sections 3.5.1.1 and 3.6.1 of this SER). The system is located within the fuel building, a seismic Category I structure which is protected against tornadic winds and missiles, thus meeting with the guidelines of Regulatory Guide 1.13 (Position C.2). In this manner, the fuel pool cooling system comolies with the requirements of General Design Criterion 4, "Environmental and Missile Design Bases."

Each unit of the plant is provided with a separate spent fuel pool cooling and cleanup system; thus, the requirements of General Design Criterion 5, "Sharing of Structures, Systems and Components," are not applicable.

Provisions have been made for routine visual inspection of the fuel pool cooling system components. One fuel pool cooling train is in operation at all times.

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administrative controls, and the fire brigade size and training. The objective of our review is to ensure that in the event of a fire, personnel and plant equipment would be adequate to safety shut down the reactor, to maintain the plant in a safe shutdown condition, and to minimize the release of radioactive material to the environment.

Because units 1 and 2 are of the same design except as noted, the comments made in this report apply to both units.

Our consultant, Gage-Babcock & Associates, Inc. participated in the review of the fire protection program.

9.5.1.1 Fire Protection Program Requirements

A. Fire Protection Program

The fire protection program is described in the applicant's Fire Protection Evaluation Report. The program establishes policy for the protection of structures, and components important to safety. The applicant's program conforms to the technical requirements in Appendix R to 10 CFR 50, Section II.A (BTP CMEB 9.5-1 Section C.1). We find that the fire protection program meets the guidelines of BTP CMEB 9.5-1, Section C.1, and is, therefore, acceptable.

B. Fire Hazard Analysis

The applicant provided a fire hazard analysis with the Fire Protection Evaluation Report in October 1981. This analysis did not address safe shutdown capability. Our evaluation of this fire hazard analysis is contained in the balance of this report. The applicant has committed to provide an analysis of their safe shutdown capability. GDC 3, Appendix A to 10 CFR Part 50 requires that "Fire fighting systems shall be designed to assure that rupture or inadvertent operation does not significantly impair the safety capability of those structures, systems and components."

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To satisfy the requirement, the applicant has designed components required for safe shutdown so that rupture or inadvertent operation of fire suppression systems will not adversely affect the operability of these components. Where necessary, appropriate protection is provided to prevent impingement of water spray on components required for safe shutdown. Redundant trains of components that are susceptible to damage from water spray are physically separated so that manual fire suppression activities will not adversely affect the operability of components not involved in the postulated fire. However, we are concerned that the mechanism by which fire and fire fighting systems may cause the simultaneous failure of redundant or diverse trains have been considered with design. We will require that the applicant identify such mechanisms that were considered in his fire hazards analysis and the measure taken to preclude the fire or fire suppressant induced failure of redundant or diverse safety trains.

C. Alternative Shutdown

Alternative shutdown capability has been provided for the control room. The capability is evaluated in Section V.C of this report.

D. Implementation of Fire Protection Program

The fire protection program should be operational before initial fuel loading.

9.5.1.2 ~~IV.~~ Administrative Controls

The administrative controls for fire protection consists of the fire protection program and organization, the fire brigade training, the controls over combustibles and ignition sources, the prefire plans and procedures for fighting fires, and quality assurance. The applicant has not provided sufficient information to verify compliance with our guidelines. We will require that the applicant comply with the guidelines in BTP CMEB 9.5-1 Section C.2.

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IV. Fire Brigade and Fire Brigade Training

The applicant, has not provided a sufficient description of the plant fire brigade, including equipment and training to verify the guidelines contained in BTP CMEB 9.5-1 Item C.3. We will require that the applicant follow the guidelines in BTP CMEB 9.5-1 Section C.3 in the establishment and training of the fire brigade.

9.5.1.4

V. General Plant Guidelines

A. Building Design

The applicant indicated in the Fire Protection Evaluation Report the walls that separate buildings and walls and floor/ceiling assemblies used to enclose rooms containing safe shutdown systems are 3-hour-fire-rated. However, the applicant has not indicated all fire rated assemblies are tested in accordance with ASTM E-119, "Fire Tests of Building Contruction and Materials." To assure that all fire rated assemblies will perform as indicated under fire conditions, we will require the applicant to test all assemblies in accordance with ASTM E-119.

Three-hour-fire-rated penetration seals are provided for all penetrations of fire-rated walls of floors/ceilings, tested in accordance with BTP CMEB 9.5-1, Section C.5.a.(3). Based on our review, we conclude the fire-barrier penetration seals meet the guidelines of BTP CMEB 9.5-1, Section C.5.a(3), and are, therefore, acceptable.

The door openings in fire-rated barriers are provided with fire door assemblies that have ratings commensurate with the fire ratings of the walls in which they are installed, however all doors will not be Underwriter Laboratory (UL) - labeled fire door assemblies. Door installed in fire rated assemblies required for missile protection will not be labeled assemblies. The doors are constructed as a fire door, however they are modified for missile protection. Suit-

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ability of fire doors is determined by test by nationally recognized testing laboratories, and doors not tested and not labeled cannot be relied upon for effective protection. Therefore, we will require the applicant to test and label all fire door assemblies in accordance with NFPA 252 "Fire Tests of Door Assemblies."

The HVAC penetrations of fire-rated barriers are provided with Underwriters Laboratory (UL)-labeled fire damper assemblies that have ratings commensurate with the fire ratings of the walls in which they are installed. The fire dampers are installed according to the manufacturer's directions. Three-hour-fire dampers are provided in all 3-hour-fire-rated barriers. Based on our evaluation, we conclude that the fire dampers are provided in accordance with the guidelines of BTP CMEB 9.5-1, Section C.5.a and are, therefore, acceptable.

Access and escape routes are provided for each fire area. Stairwells outside primary containment serving as access and egress routes are enclosed with fire barriers having 2-hour-fire ratings with 1-½-hour-UL-labeled fire door assemblies at all openings into the stairwell. Fire exit routes will be clearly marked and established by prefire plans. Based on our review, we conclude that the applicant's fire protection program concerning access and egress routes meets the guidelines of BTP CMEB 9.5-1, Sections C.5.a(6) and (7) and is, therefore, acceptable.

Thermal insulating materials are noncombustible. Interior walls and structural components, radiation shielding materials, and sound-proofing and interior finishes are noncombustible or listed by a nationally recognized testing laboratory, such as Factory Mutual or UL, for flame spread, smoke, and fuel contribution of 25 or less in their use configuration. We find this is in accordance with the guidelines of BTP CMEB 9.5-1, Section C.5.a and is, therefore, acceptable.

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Metal deck roof construction is either noncombustible or listed as Class I by Factory Mutual. Suspended ceilings and their supports are of non-combustible materials, and concealed spaces above the suspended ceilings are devoid of combustibles. We find this to be in accordance with the guidelines of BTP CMEB 9.5-1, Sections C.5.a(10 and (11), and, therefore, acceptable.

High voltage-high amperage transformers installed inside buildings are of the dry type.

All oil-filled transformers are installed outside buildings containing safety-related equipment. All building walls of buildings containing safety-related equipment located within 50 ft. of any oil-filled transformer have a minimum fire resistance rating of 3 hours. Each oil-filled transformer is protected with a water spray extinguishing system that is automatically actuated by heat detectors.

Based on our evaluation, we conclude that the installation of the transformers meets the guidelines of BTP CMEB 9.5-1, Sections C.5.a(12) and (13) and is, therefore, acceptable.

B. Safe Shutdown Capability

The applicant has not provided an analysis of the fire protection features which assure safe shutdown capability. We will require the applicant to follow the fire protection guidance for safe shutdown contained in BTP CMEB 9.5-1, C.5.b.

C. Alternate Shutdown Capability

Alternative shutdown capability is provided for the control room by a remote shutdown panel. We have not completed the review of the alternate shutdown capability.

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D. Control of Combustibles

Safety-related systems have been isolated or separated from combustible materials as much as possible. The storage of flammable liquids complies with NFPA 30, "Flammable and Combustible Liquids Code."

The fire protection for the diesel generator fuel oil day tanks and the reactor recirculation pump is discussed in Sections VII.A and F respectively of this report.

The turbine building does not contain any circuits or equipment of safe shutdown systems and is separated from such areas by 3-hour rated fire walls.

Based on our evaluation, we conclude combustibles have been separated from safety-related systems or provided with suppression in accordance with the guidelines of BTP CMEB 9.5-1, Section C.5.d(1) and are, therefore, acceptable.

No high-pressure gas storage containers are stored inside of buildings containing safety-related equipment. Bulk gas, including hydrogen, is stored out of doors. The hydrogen piping is not routed in areas containing safety-related systems.

Based on our evaluation, we conclude that the hydrogen storage and piping meets the guidelines of BTP CMEB 9.5-1, Sections C.5.d(2) and (5) and is, therefore, acceptable.

E. Electrical Cable Construction, Cable Trays, and Cable Penetrations

All cable traysare of steel construction. Electrical cable construction passes the IEEE-383-1974 flame test. The cables are designed to allow wetting down with fire suppression water without electrical faulting.

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Cable tray penetrations will have a fire rating at least equal to the rating of fire barriers in which they are installed. Automatic sprinkler systems have been installed in heavily cabled areas, i.e. areas containing six or more cable trays. Based on our evaluation, we find tht the electrical cable construction, cable trays, and cable penetrations meet the guidelines of BTP CMEB 9.5-1, Section C.5.e and are, therefore, acceptable.

F. Ventilation

The control room has a dedicated ventilation system designed specifically to exhaust smoke or other products of combustion. Normal plant ventilation systems will be utilized in other areas of the plant for this purpose. Portable smoke ejectors will be provided to assist in removal of the products of combustion should the normal ventilation systems be unavailable because of damper closures or other failures. We find this acceptable.

Where total flooding gas extinguishing systems are used, air intake and exhaust ventilation dampers are provided with mechanisms which close them upon initiation of gas flow. We find this acceptable. Fire-barrier ventilation openings are provided with fire dampers that close if a fire causes room temperature to exceed a set value.

Fresh air supply intakes to areas containing safety-related equipment or systems are remote from exhaust air outlets of other fire areas. Stairwells are designed to minimize smoke infiltration during a fire.

Charcoal filters have been provided with fire suppression systems in accordance with Regulatory Guide 1.52. We find this acceptable.

Based on its evaluation, we find that the ventilation system meets the guidelines of BTP CMEB 9.5-1, Section C.5.f and is, therefore, acceptable.

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G. Lighting and Communication

Emergency lighting will be installed in all areas of the plant that may need to be manned for safe shutdown operations in access and egress routes to and from all fire areas. The emergency lighting consists of fixed, self-contained flourescant or sealed-beam units with individual 8-hour minimum battery power supplies. The emergency lighting system provides illumination of all points of equipment operation needed for shutdown as well as at all points on the floor including angles and intersections of corridors, passageways and stairways of not less than 1.0 ft. candle measured at the floor. We conclude that the emergency lighting meets the guidelines of BTP CMEB 9.5-1, Section C.5.g.(1), and is, therefore, acceptable.

The applicant has provided a fixed emergency communication system which is independent of the normal plant communication system at pre-selected stations. A portable radio communications system has been provided for use by the fire brigade. Fixed repeaters have been installed for portable radio communication. The applicant has not provided a description of the fire protection provided for the fixed repeaters. We will require the applicant to meet the guidelines in BTP CMEB 9.5-1, Section C.5.g(4) for the protection of the fixed repeaters.

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V. Fire Detection and Suppression

A. Fire Detection

The fire detection systems consist of the detectors, associated electrical circuitry, and electrical power supplies. The types of detectors used are product of combustion, rate-of-rise, and fixed temperature detectors. The systems provide audible and visual alarms locally and in the control room.

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The fire detection systems are designed to conform to NFPA 72D, "Standard for the Installation, Maintenance, and Use of Proprietary Protective Signaling Systems", Class A systems. Fire detectors are installed in accordance with NFPA 72E, "Standard on Automatic Fire Detectors" in all areas of the plant containing safety related system components and cables. The systems will be continuously supervised to provide alarm and trouble indication to the control room from all detectors under single break or ground fault conditions.

Primary and secondary power supplies for fire detection system and electrically operated valves for automatic suppression systems that satisfy the provisions of Section 2220 of NFPA 72D have not been provided. The applicant has not provided sufficient information on the power supplies for the fire detection systems to verify compliance with our guidelines. We will require the applicant to provide primary and secondary power supplies for all fire detection systems and electrically operated control valves for automatic suppression systems in accordance with Section 2220 of NFPA 72D.

B. Fire Protection Water Supply System

The water supply system consists of three fire pumps; one pump is electrically driven and the other are diesel engine driven. Each fire pump is separately connected to a buried 12 water main loop around the plant. Each fire pump has a rated capacity of 1500 gpm at 150 psig. The fire pumps and controllers are UL listed. The fire pump installation has been designed and installed and will be tested in accordance with NFPA 20, "Standard for the Installation of Centrifugal Fire Pumps."

The fire pumps are located in the fire pump house. Each fire pump is separated from others with 3-hour-fire-rated barriers. Automatic sprinklers have been provided in each diesel pump room.

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The fire protection water supply system is kept pressurized at 100 psi by a fire service jockey pump and hydropneumatic tank. The fire pumps are automatically started when low pressure is sensed in the pump discharge header. All pumps are stopped manually at the fire pump house. Separate audible and visual alarms on supervised circuits are provided in the control room to monitor the fire pumps status, prime mover availability, power failure, and failure of the fire pumps to start.

The water supply for fire protection is taken from two 300,000-gallon water storage tanks. The suction piping to the three fire pumps is arranged to permit suction from either or both of the two fire water storage tanks. The greatest water demand for the fixed fire suppression system is 2000 gpm; coupled with 500 gpm for hose streams, this creates a total water demand of 2500 gpm. We find that the water supply system can deliver the required water demand with one pump out of service.

Yard hydrants are provided at intervals of 250 ft. along the fire protection water supply loop, approximately 40 ft. from the buildings. The lateral to each yard hydrant is provided with an isolation valve to facilitate hydrant maintenance and repairs without shutting down any part of the fire water supply system. Standard hose houses are provided at intervals of 500 ft. in accordance with NFPA codes. Sectional control valves are provided to isolate portions of the underground main for maintenance or repair without shutting off the supply to primary and backup fire suppression systems that serve areas containing or exposing safety-related systems.

The applicant has not provided sufficient information to verify that all valves in the fire protection water supply system are supervised in accordance with our guidelines in BTP CMEB 9.5-1, Section C.6.c(2). We will require the applicant to supervise all valves in the fire protection water supply system in accordance with our guidelines.

C. Sprinkler and Standpipe Systems

The wet pipe sprinkler systems, meet the provisions of NFPA 13, "Standard for the Installation of Sprinkler System" and NFPA 15, "Standard for Water Spray Fixed Systems for Fire Protection". The areas equipped with water suppression systems are listed on Figure 9A.2a of the applicant's FSAR.

The sprinkler systems and manual hose station standpipe connections to the looped interior fire protection headers are arranged and valved so that a single break in the water supply piping will not impair both the primary and backup fire suppression systems.

Manual hose stations are located throughout the plant in accordance with NFPA 14, "Standard for the Installation of Standpipe and Hose Systems." Hose stations are installed so that all areas of the plant can be reached with an effective hose stream with a maximum of 75 ft. of hose, with the exception of the electrical tunnels.

The electrical tunnels are protected by automatic water spray sprinkler systems as the primary fire suppression system and portable fire extinguishers are provided as the backup. To meet our guidelines on BTP CMEB 9.5-1 Section C.1.c, we will require the applicant to install manual hose stations so that all areas of each electrical tunnel can be reached with an effective hose stream.

Standpipes are 4 in. in diameter for multiple and 2-½ in. in diameter for single-hose stations. Standpipe system piping supply hose stations protecting safe shutdown equipment are seismically supported and provided with additional connections to the standby service water supply through manually operated valves. The valves are locked in the closed position.

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D. Carbon Dioxide Suppression System

High pressure carbon dioxide, automatic total flooding systems are provided for primary protection for the turbine generator bearings and exciter enclosures. The systems are activated by heat detectors which alarm and annunciate in the control room. The carbon dioxide systems may also be activated manually. The systems are designed and installed in accordance with NFPA 12 "Carbon Dioxide Extinguishing Systems."

Based on our evaluation, we conclude that the carbon dioxide extinguishing systems meets the guidelines of BTP CMEB 9.5-1, Section C.6.d and are, therefore, acceptable.

E. Halon Suppression Systems

Total flooding Halon 1301 systems are provided for the PGCC floor sections in the control room. The systems are designed to provide an initial concentration of 6 to 7% by volume of Halon 1301 within 10 seconds of initiation and sustain a 20% concentration of Halon 1301 for 20 minutes. The Halon 1301 suppression systems are to be manually initiated by the operator upon receipt of fire alarms from either the ionization smoke detectors or rate of rise thermal detectors installed in the PGCC floor sections. The Halon 1301 extinguishing system will comply with the fire protection specifications contained in the NEDO-10466 Report, Revision 2, dated March 1978, which was previously approved by us.

Based on our review, we conclude that the Halon 1301 extinguishing systems meet the guidelines of BTP CMEB 9.5-1, Section C,6.d and are, therefore, acceptable.

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F. Portable Extinguishers

Portable fire extinguishers are provided to conform with the guidelines of NFPA 10, "Standard for Portable Fire Extinguishers." We find this acceptable.

Based on our review, we conclude that these extinguishers meet the guidelines of BTP CMEB 9.5-1, Section C.6.f and are, therefore, acceptable.

9.5.1.6 ~~VII.~~ Fire Protection of Specific Plant Areas

A. Containment and Reactor Building

The reactor building is separated from adjacent buildings by three-hour fire rated barriers. Containment and reactor building fire protection features include hose stations, ionization smoke detectors, and fire extinguishers.

A lube oil fire hazard was initially through to be associated with the reactor recirculation pumps located in the containment. The pumps and motors are lubricated by oil. The lube oil systems are contained within the metal motor housing with no external parts. Hence, an engineered oil leak collection system or additional fire protection for the pumps is not required.

Based on our review, we conclude that the fire protection for containment and reactor building meets the guidelines of BTP CMEB 9.5-1, Section C.7.a and is, therefore, acceptable.

B. Control Room

The control room complex is separated from all other areas of the plant by 3-hour fire rated assemblies. Smoke detectors have been installed in the control room. All smoke detector alarms are

annunciated in the control room panel. Portable fire extinguishers inside the control room and hose stations immediately outside the control room are provided in accordance with Section C.7.b of BTP CMEB 9.5-1.

The control room utilizes the standard General Electric Power Generation Control Complex (PGCC) design. The protection of this system is reviewed in Section VI.E of this report.

The applicant has provided an alternate shut down system for the control room. The alternate shutdown system is reviewed in Section V.C of this report.

The outside air intakes for the control room ventilation system are not equipped with smoke detectors that alarm in the control room. We will require the applicant to install smoke detectors in accordance with our guidelines of BTP CMEB 9.5-1, Section C.7.b in the outside air intakes for the control room ventilation system.

C. Cable Chases

The cable chases are separated from the balance of the plant by 3-hour fire rated assemblies. Separate cable chases have been provided for each division of redundant safe shutdown system circuits.

Automatic water spray sprinkler systems are provided. Manual fire suppression capability is provided by hose stations and portable extinguishers. Early warning fire detection is provided by smoke detectors that annunciate alarm and trouble conditions in the control room.

Based on our review, we conclude that the fire protection for the cable chases is in accordance with BTP CMEB 9.5-1, Section C.5.e and is, therefore, acceptable.

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D. Switchgear Rooms

The Division I and Division II switchgear rooms are separated from each other and from other plant areas by 3-hour-fire-rated walls and floor/ceiling assemblies. Automatic fire detection provided by ionization smoke detectors. Manual protection is provided by standpipe hose stations and portable extinguishers. Floor drains have been provided in the switchgear rooms.

Based on our review, we conclude that the fire protection for the switchgear rooms is in accordance with BTP CMEB 9.5-1, Section C.7.d and is, therefore, acceptable.

E. Safety-Related Battery Rooms

The battery rooms are separated from each other and from the balance of the plant by 3-hour-fire-rated barriers. Smoke detection systems are provided in each battery room. Hose stations and portable fire extinguishers are available in the areas for fire manual suppression. The ventilation system is designed to maintain the hydrogen levels below 2%. Air flow monitors that alarm in the control room to monitor the loss of ventilation have been provided in each battery room.

Based on the above evaluation, the staff concludes that the fire protection for the battery rooms meets the guidelines of BTP CMEB 9.5-1, Section C.7.g and is, therefore, acceptable.

F. Emergency Diesel Generator Rooms

Each diesel generator is located in a different fire area separated by three-hour fire rated barriers. All cable and piping penetration through the fire rated barriers are fitted with three-hour rated penetration seals.

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A 550 gallon diesel fuel oil day tank is located in each diesel generator room. The tanks are enclosed in accordance with NFPA 37, to prevent a fuel oil spill from affecting more than one diesel generator unit, the applicant has provided water tight curbs at all entrances to and between the diesel generator rooms of sufficient height to contain a fuel oil spill from the day tank and diesel generator.

Each diesel generator room is protected by an automatic pre-action sprinkler. Heat detectors actuate the systems. Manual fire suppression capability is provided by hydrants and portable fire extinguishers.

Based on our evaluation, we conclude that the fire protection for the diesel generator rooms meets the guidelines of BTP CMEB Section C.7.i and is, therefore, acceptable.

G. Remote Safety Related Panels

The remote shutdown panel is housed in an area separated from the remainder of the plant by walls and floor/ceiling assemblies, with fire ratings of 3-hours. Ionization detectors are located at various points in the room. Manual fire suppression capability is provided by portable fire extinguishers and hose stations. We find the fire protection for this area is in accordance with the guidelines of BTP CMEB 9.5-1, Section C.7.f, and is, therefore, acceptable.

I. Other Plant Areas

The applicant's Fire Hazards Analysis addressed other plant areas not specifically discussed in this report.

We find that the fire protection for these areas is in accordance with the guidelines of BTP CMEB 9.5-1 and, is therefore, acceptable.

10 STEAM AND POWER CONVERSION SYSTEM

10.1 Summary Description

10.2 Turbine Generator

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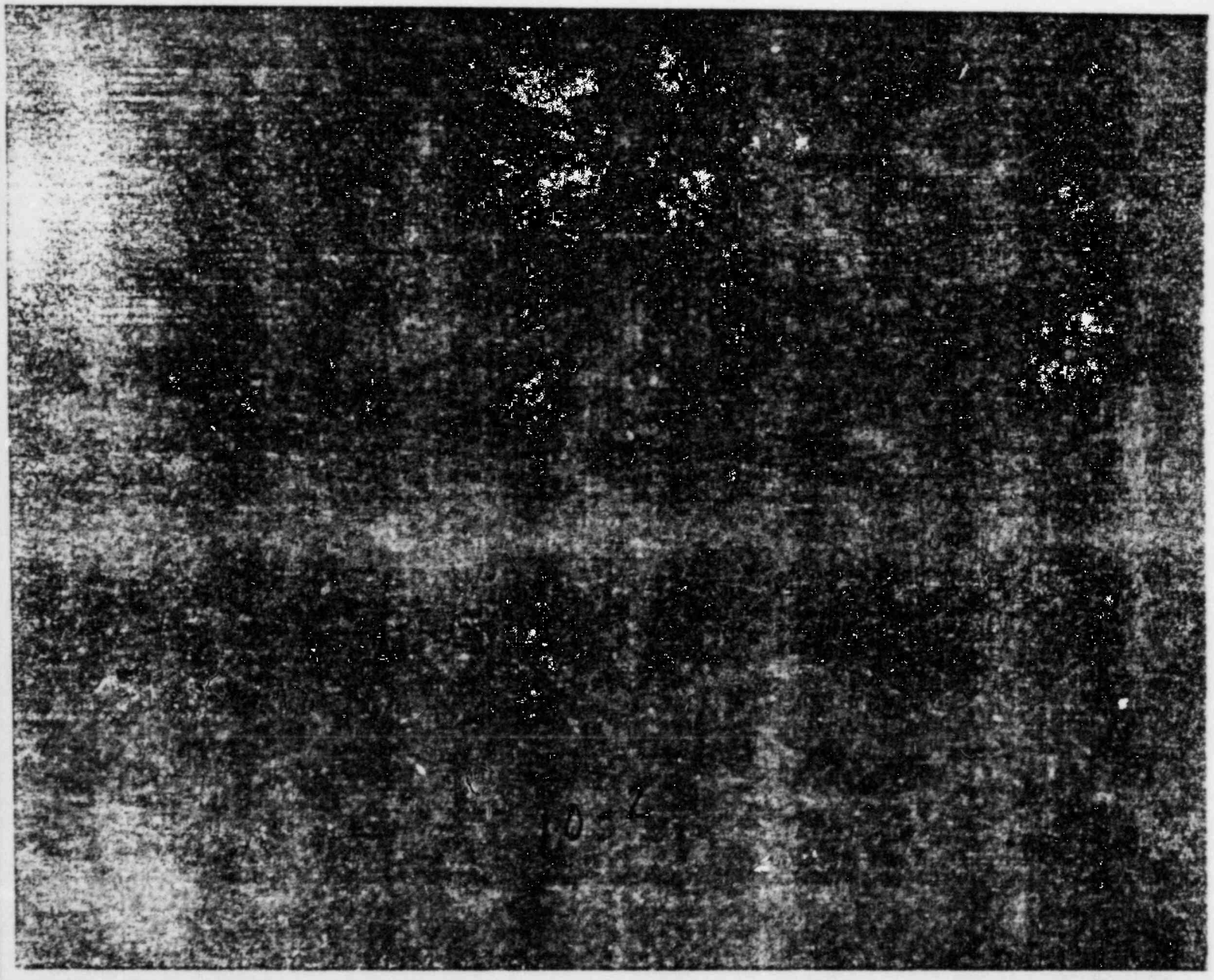
10.3 Main Steam Supply System (Up to and Including the Main Steam Block Valves)

The main steam supply system (from the containment up to and including the outermost isolation valve) was reviewed in accordance with Section 10.3 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria ~~except as noted below~~ formed the basis for our evaluation of the main steam supply system (from the containment up to the outermost isolation valve) with respect to the applicable regulations of 10 CFR 50.

10.3.1 Design Bases

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The main steam supply lines including the MSIVs and shutoff valves are designed to meet seismic Category I criteria from the reactor to the shutoff valve and are designed to Quality Group A criteria from the reactor through the outboard MSIV and beyond (up to and including the first weld outside the jet impingement wall in the auxiliary building) and Quality Group B criteria from the weld up to and including the shutoff valve in the



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auxiliary building steam tunnel. The lines pass through the drywell, containment and auxiliary buildings which are seismic Category I, flood and tornado protected structures. The guidelines of Regulatory Guide 1.29, "Seismic Design Classification," are satisfied for these portions of the main steam supply system, thus complying with the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena."

10.3.2 Description

The steam generated in the reactor vessel is routed to the high-pressure turbine by means of four main steam lines. Each main steam line contains two main steam isolation valves (MSIVs) and a shutoff (block) valve. The two MSIVs close automatically on a main steam line break, thus assuring main steam line isolation in the event of a steamline break outside containment and a concurrent siggle failure of an MSIV. One MSIV is located immediately inside of the drywell and the other immediately outside containment. The shutoff

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valve is located downstream of the outboard MSIV immediately before the steam lines leave the auxiliary building. The main steam isolation valves are designed to provide positive isolation against steam flow associated with a main steamline break. They are pneumatic and/or spring-operated (to close) fast closing (3 to 10 seconds) valves. Operating air is supplied to the valves from the instrument air system, and a seismic Category I air accumulator provides backup operating air for each valve in the event of loss of the normal instrument air supply. The MSIVs are designed to withstand the dynamic forces under the postulated steamline line break flow conditions. The main steam shutoff valves are leaktight motor-operated gate valves and are powered from separate emergency buses. They are manually actuated from the control room.

10.3.3 Evaluation

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The MSIVs which are required to function in order to assure main steam isolation are protected against the effects of high-energy pipe breaks and are qualified to function in the expected steam environment resulting from a main steamline break. Refer to Sections 3.6.1 and 3 .11 of this SER for further discussion on environmental qualification of essential equipment.

This equipment is located in tornado-missile-protected structures and is separated from the effects of internally generated missiles. Thus, the requirements of General Design Criterion 4, "Environmental and Missile Design Bases," and the guidelines of Regulatory Guide 1.117, "Tornado Design Classification," are

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satisfied. For compliance with the guidelines of Regulatory Guide 1.115, "Protection Against Low Trajectory Turbine Missiles," refer to Section 3.5.1.3 of this SER .

Based on the above, we conclude that the main steam supply system from the reactor to the main steam block valves meets the requirements of General Design Criteria 2 and 4 with respect to protection against seismic events, floods, tornadoes, missiles and environmental effects, and the guidelines of Regulatory Guides 1.117, relating to protection against tornado missiles and high and moderate-energy pipe breaks. Therefore, we find this portion of the steam supply system to be acceptable. This portion of the steam supply system is in compliance with the criteria of SRP Section 10.3.

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10.4 OTHER FEATURES

10.4.1 Main Condenser

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10.4.2 Main Condenser Evacuation System

10.4.2.1 Introduction

Section 10.4.2 of the Final Safety Analysis Report (FSAR) contains information pertaining to the main condenser evacuation (air removal) system, the system design bases, and the applicable acceptance criteria.

10.4.2.2 Acceptance Criteria

The staff has reviewed the applicant's design, design criteria, and design bases for the main condenser evacuation system (MCES) for River Bend , Unit 1. The acceptance criteria used in our evaluation are those specified in the Standard Review Plan (SRP), NUREG-0800, in Section II of SRP 10.4.2. The SRP acceptance criteria include the applicable GDC (Appendix A to 10 CFR Part 50) and Heat Exchanger Institute Standard "Standards for Steam Surface Condensers". Guidelines for implementation of the requirements of the acceptance criteria are provided in the Regulatory Guides referenced in Section II of the SRP. Conformance to the acceptance criteria of the SRP provides the bases for concluding that the MCES meets the requirements of 10 CFR Part 50.

10.4.2.3 Method of Review

The MCES was reviewed in accordance with the SRP. The results of the review are disccussed below.

10.4.2.4 Design [redacted]

The main condenser evacuation system is designed to (1) establish a vacuum on the condenser, and (2) remove noncondensible gases from the main condenser and discharge them to the gaseous radwaste system.

The major components are the mechanical vacuum pumps and the steam jet air ejectors. The main condenser evacuation system is designed to minimize the potential for explosion in the piping upstream of the catalytic recombiners in the offgas system. This is achieved by mixing sufficient dilution steam via the steam jet air ejector discharge with the noncondensible gases to limit the hydrogen concentration to less than four percent by volume. The steam jet air ejectors, intercooler, and the offgas system (Section 11 of this report) are designed to withstand a detonation in the offgas system. The hydrogen concentration at the outlet of the second stage air ejector will be maintained below four volume percent hydrogen by virtue of the dilution steam. On indication of low steam pressure or low steam flow, the operating steam jet air ejector will be removed from service and the standby air ejector activated. A hydrogen analyzer will be provided on the outlet of each recombiner so that action may be taken to preclude the buildup of explosive mixtures.

Prior to startup, the mechanical vacuum pump is used to initially evacuate the main condenser to a vacuum level sufficient for the steam

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10.4.2.4 driven air ejectors to take over. Because the mechanical vacuum pumps are employed during startup operations, generally enough time has elapsed for radioactive decay to have significantly reduced the level of radioactive gases trapped within the main condenser. The mechanical vacuum pump discharges the air, along with noble gases and iodine, to the inlet of a charcoal and HEPA filtration system where the radioactive iodine and particulates are removed from the air stream prior to release at the turbine building exhaust. The exhaust is continuously sampled for acceptable low concentation of activity by isokenetic radiation monitors located in the turbine building ventilation exhaust duct.

10.4.2.5 Evaluation [redacted]

The main condenser evacuation system includes equipment and instruments to establish and maintain condenser vacuum and to prevent an uncontrolled release of gaseous radioactive material to the environment. The scope of our review included the system capability to transfer radioactive gases to the gaseous waste processing system or ventilation exhaust systems, and the design provisions incorporated to monitor and control releases of radioactive materials in effluents. The staff has reviewed the applicant's system descriptions, piping and instrumentation diagrams, and design criteria for the components of the main condenser evacuation system in accordance with the SRP.

The staff concludes that the MCES design is acceptable in that the applicant has met the requirements of GDC's 60 and 64 with respect to

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10.4.2.5 the control and monitoring of releases of radioactive materials to the environment. The applicant has met the criteria for applying appropriate industrial standard on the design of heat exchangers and air ejectors.

10.4.3 Turbine Gland Sealing System

10.4.3.1 Introduction

Section 10.4.3 of the Final Safety Analysis Report contains information pertaining to the turbine gland sealing system, the design bases, and applicable acceptance criteria.

10.4.3.2 Acceptance Criteria

The staff has reviewed the applicant's design, design criteria, and design bases for the turbine gland sealing system for River Bend, Unit No. 1. The acceptance criteria used as the basis for our evaluation are set forth in section 10.4.3 of the Standard Review Plan (SRP) NUREG-0800. The acceptance criteria are the applicable GDC (Appendix A to 10 CFR Part 50) as referenced in the SRP. Guidelines for implementation of the requirements of the acceptance criteria are provided in the Regulatory Guides identified in Section II. of the SRP. Conformance to the acceptance criteria provides the bases for concluding that the turbine gland sealing system meets the requirements of 10 CFR Part 50.

10.4.3.3 Method of Review

The turbine gland sealing system for River Bend, Unit 1, was reviewed in accordance with SRP 10.4.3. The results of the review are discussed below.

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10.4.3.4 ~~Review of Turbine Gland Sealing System~~ Design

The turbine gland sealing system (TGSS) is designed to provide a continuous supply of "clean" steam to main turbine shaft seals, the stem packings of stop valves, control valves, combined intermediate valves and bypass valves. This sealing steam is used to prevent air leaking into the steam cycle and radioactive steam leaking out of the steam cycle into the Turbine Building.

The TGSS includes the equipment and instruments to provide a source of sealing steam to the annulus space where the turbine and large steam valve shafts penetrate their casings. The scope of our review included the source of sealing steam and the provisions incorporated to monitor and control releases of radioactive material in gaseous effluents in accordance with General Design Criteria 60 and 64. We have reviewed the applicant's system descriptions and design criteria for the components of the TGSS and found them consistent with Regulatory Guide 1.26(2).

The basis for acceptance in our review has been conformance of the applicant's designs, design criteria, and design bases for the turbine gland sealing system to the acceptance criteria of SRP 10.4.3.

10.4.3.5 Evaluation [redacted]

The staff concludes that the turbine gland sealing system design is acceptable in that the applicant has met the requirements of GDC's 60 and 64 with respect to the control and monitoring of releases of radioactive materials to the environment.

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10.4.4 Turbine Bypass System

10.4.5 Circulating Water System

The circulating water system was reviewed in accordance with Section 10.4.5 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the circulating water system with respect to the applicable regulations of 10 CFR 50.

The nonsafety-related (nonseismic Category I, Quality Group D) circulating water system (CWS) is designed to remove the heat rejected from the main condenser to the atmosphere via four multicell forced draft cooling towers per unit. The CWS is not required to maintain the reactor in a safe shutdown condition or to mitigate the consequences of accidents.

The system pumps circulate water from the CWS pump structure through the condenser shells to the cooling towers. The circulating water then flows into the flume and back to the circulating water pump structure.

Four vertical circulating water pumps are located outdoors; these pumps provide a design flow of 511,560 gpm. Four multicell cooling towers are provided for each unit, designed to cool 565,000 gpm; the difference between this flow rate and the circulating water pump flow rate of 511,560 gpm is to accommodate the service water system coolant flow rate.

The applicant provided the results of an analysis of the effects of possible flooding as a result of a postulated failure of a circulating water system expansion joint or butterfly valve. Flooding in the affected unit turbine building will result from either of the above postulated failures. The analysis assumed that the entire volume of the system including the cooling tower basin would be emptied into the building. The resulting water level, 97 ft. - 6 inches, would be below any nonwatertight doors or unsealed penetrations from the adjacent auxiliary building and below the level of any safety-related equipment, thereby verifying that a total failure in the circulating water system will not result in flooding which would compromise plant safety.

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Since no safety-related equipment is affected by a postulated failure in the CWS, the requirements of General Design Criterion 4, "Environmental and Missile Design Bases," with respect to protection of safety-related systems from failure of nonsafety-related systems are satisfied. In addition, General Design Criterion 5, "Sharing of Structures, Systems and Components" is not applicable. The applicant notes that failure of any component of the CWS will not affect safety-related components, thus satisfying Position C.2 of Regulatory Guide 1.29, "Seismic Design Classification"; in this way the criteria of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena" are complied with.

Indication of leakage or potential failure in CWS components is provided to operators in the control room. The first indication would be an alarm in the control room given by the condenser retention pit redundant sump high-level alarms. Loss of condenser vacuum would also provide an early alarm if the break occurred upstream of the condenser water boxes. Operator action is necessary to trip the pumps and close the condenser

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butterfly valves in order to minimize flooding of the turbine building basement. CWS performance is monitored by pressure and temperature indicators in the control room. Other parameters are monitored by the plant computer.

Based on our review, we conclude that the circulating water system meets the requirements of General Design Criteria 2 and 4 with respect to protection of safety-related systems from failures in nonsafety-related systems. We therefore conclude that the circulating water system is acceptable. The system complies with the acceptance criteria of SRP Section 10.4.5.

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10.4.6 Condensate Filter Demineralizer System

The condensate filter demineralizer cleanup system description and piping and instrumentation diagrams have been reviewed in accordance with SRP 10.4.6.

The condensate filter demineralizer system removes corrosion products, condenser inleakage impurities, and impurities present in the condensed steam. Eight filters, in series with six parallel flow deep bed demineralizers, are provided to polish the condensate flow; six of the filters and five demineralizers are normally in operation to process 100% of the condensate flow, with the remaining units in standby.

The applicant has been asked to describe the sampling frequency, chemical analyses, and established limits for purified condensate that will be performed and the basis for these limits; the resin replacement frequency to maintain adequate capacity margin the the condensate treatment system; and the water chemistry control program to ensure maintenance of condensate demineralizer influent and effluent conductivity within the limits of Table 2 of RG 1.56, Revision 1. After receiving responses from the applicant, the staff will evaluate them and report its findings in a supplement to the SER.

NUREG-0737 II.B.3 - Post-Accident Sampling Capability

The applicant has committed (in Appendix 1A of the FSAR) to a post-accident sampling system that meets the requirements of NUREG-0737 Item II.B.3.

However, he has not yet provided the technical information required by NUREG-0737 for staff evaluation. Implementation of the requirement is not necessary before low power operation because only small quantities of radionuclide inventory will exist in the reactor coolant system and therefore will not affect the health and safety of the public. Before 5% power operation is exceeded, the applicant must demonstrate the capability to promptly obtain reactor coolant samples in the event of an accident in which there is core damage, consistent with the following conditions:

(1) Demonstrate compliance with all requirements of NUREG-0737 II.B.3 for sampling, and chemical and radionuclide analysis capability under accident conditions.

(2) Provide sufficient shielding to meet the requirement of GDC 19, assuming RG 1.3 source terms.

(3) Commit to meet the sampling and analysis requirements of RG 1.97, Revision 2.

(4) Verify that all electrically powered components associated with post-accident sampling are capable of being supplied with power and operated within 30 minutes of an accident in which there is core degradation, assuming loss of offsite power.

(5) Verify that valves that are not accessible for repair after an accident are environmentally qualified for the conditions in which they must operate.

(6) Provide a procedure for relating radionuclide gaseous and ionic species to estimated core damage.

(7) Describe the design or operational provisions to prevent high pressure carrier gas from entering the reactor coolant system from online gas analysis equipment, if it is used.

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(8) Provide a method for verifying that reactor coolant dissolved oxygen is at > 0.1 ppm if reactor coolant chlorides are determined to be > 0.15 ppm.

(9) Provide information on (a) testing frequency and type of testing to ensure long-term operability of the post-accident sampling system and (b) operator training requirements for post-accident sampling.

(10) Demonstrate that the reactor coolant system and suppression chamber sample locations are representative of core conditions.

In addition to the above licensing conditions, the staff is conducting a generic review of accuracy and sensitivity for analytical procedures and online instrumentation to be used for post-accident analysis. The staff will require that the applicant submit data supporting the applicability of each selected analytical chemistry procedure or online instrument, along with documentation demonstrating compliance with the licensing conditions, 4 months before 5% power operation is exceeded; however, review and approval of these procedures will not be a condition for full-power operation. If the staff's generic review determines a specific procedure is unacceptable, the staff will require the applicant to make modifications as determined by the generic review.

If the applicant does not satisfactorily address the above items on post-accident sampling capability, the staff will condition the plant operating license.

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10.4.7 Condensate and Feedwater System

The condensate and feedwater system was reviewed in accordance with Section 10.4.7 of NUREG-0800(SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed, according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria ~~except as noted below~~ formed the basis for our evaluation of the condensate and feedwater system with respect to the applicable regulations of 10 CFR 50.

The condensate and feedwater system includes all components and equipment from the condenser outlet to the connection to the reactor vessel and to the heater drain system. The system serves no safety function and is therefore classified as nonsafety-related (Quality Group D, nonseismic Category I). However, the portion of the system between the reactor vessel and the auxiliary building wall (in the steam tunnel) is safety-related and design to seismic Category I, Quality Group A criteria from the reactor to the outboard containment isolation valve, and is designed to seismic Category I, Quality Group B criteria from the outboard containment isolation valve through the feedwater shutoff valves to the auxiliary building wall in order to assure feedwater system isolation under accident conditions.

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Each of the two main feedwater lines contains a spring load piston actuated check valve outside of containment, held open by air pressure during normal operation that serves as the outboard containment isolation valve, ~~inside~~ inside containment, there is an inboard isolation check valve; there are also two motor operated shutoff valves with each shutoff valve powered from a separate emergency bus on each feedwater line, outside of containment. Thus, feedwater isolation is assured in the event of a single failure in any isolation valve. The safety-related portion of the system is located in the seismic Category I, flood and tornado-protected auxiliary building and reactor building. Thus, the guidelines of Position C.1 of Regulatory Guide 1.29, "Seismic Design Classification" are met, thus complying with the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena." The essential equipment is separated from the effects of internally generated missiles and is qualified to function in a steamline break environment. Thus, the requirements of General Design Criterion 4, "Environmental and Missile Design Bases," are satisfied. Refer to Section 3.6.1 and 3.11 for further discussion of environmental qualification of essential equipment and protection against postulated piping failures. The feedwater system is not shared between units; therefore, the requirements of General Design Criterion 5, "Sharing of Structures, Systems and Components," are not applicable.

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The feedwater system is not required to transfer heat under accident conditions and, therefore, General Design Criterion 44, "Cooling Water," 45, "Inspection of Cooling Water System," and 46, "Testing of Cooling Water System," are not applicable.

Based on the above, we conclude that the condensate and feedwater system meets the requirements of General Design Criteria 2 and 4 with respect to its protection against natural phenomena, missiles and environmental effects, and is, therefore, acceptable. The condensate and feedwater system complies with the acceptance criteria of SRP Section 10.4.7.

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[redacted] and 64 with respect to the control and monitoring of releases of radioactive materials to the environment.

11 RADIOACTIVE WASTE MANAGEMENT

11.1 [redacted] Source Terms

The radioactive waste management systems for River Bend, Units 1 and 2, are designed to provide for the controlled handling and treatment of liquid, gaseous and solid wastes. The liquid radioactive waste system has been sized so that it is shared between both Units 1 and 2. (Note: Construction of Unit 2 has not been started). The system processes wastes from equipment and floor drains, sample wastes, decontamination and laboratory wastes, and chemical wastes generated from both units. The gaseous radioactive waste system provides holdup capacity to allow decay of short lived noble gases from the main condenser air ejector and treatment of ventilation exhausts through HEPA filters and carbon adsorbers as necessary to reduce releases of radioactive materials to "as low as is reasonably achievable" levels in accordance with 10 CFR Part 20 and 10 CFR Part 50.34a. The solid radioactive waste system has not been designed yet and, accordingly will be the subject of separate correspondence. At this time, the radioactive waste management review area includes the process and effluent radiological monitoring and sampling systems provided for the detection and measurement of radioactive materials in plant process and effluent streams.

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11.1.1 Acceptance Criteria

The staff has reviewed the applicant's design, design criteria, and design bases for the radioactive waste management systems for the River Bend station. The acceptance criteria used as the basis for our evaluation are set forth in the SRP, NUREG-0800, in Section II of SRPs 11.1, 11.2, 11.3, 11.4, and 11.5. These acceptance criteria include the applicable GDC (Appendix A to 10 CFR Part 50), Section 20.106 of 10 CFR Part 20, Appendix I to 10 CFR Part 50, and ANSI Standard N13.1, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities". Guidelines for implementation of the requirements of the acceptance criteria are provided in the ANSI Standards, Regulatory Guides, and other documents identified in Section II of the SRP. Conformance to the acceptance criteria provides the bases for concluding that the radioactive waste management systems meet the requirements of 10 CFR Part 20 and 10 CFR Part 50.

11.1.2 Liquid and Gaseous Effluent Source Terms

The estimated releases of radioactive materials in liquid and gaseous effluents were calculated by the applicant using the BWR GALE Code described in NUREG-0016. The staff has reviewed these source terms and found them consistent with the guidelines of SRP 11.1. The applicant's source terms were used in our evaluation. These source terms are given in Table 11.1-1 of the FSAR. The principal parameters used in our calculations are given in Table 11.1 of this SER.

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11.1.3 Method of Review

The River Bend Nuclear Generating Station has three radioactive waste management systems; a shared Liquid Waste Management System that serves both Units 1 and 2; separate Gaseous Waste Management Systems for each unit; and a Solid Waste Management System, the design details of which have not yet been determined. Additionally, the River Bend Station has several radiation monitoring subsystems used for radiological monitoring of processes and plant effluents. These systems are reviewed in accordance with the applicable portions of the Standard Review Plan Sections 11.1, 11.2, 11.3, 11.4, and 11.5. The results of these reviews are discussed below.

11.2 Liquid Waste Management Systems

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11.2.1 Design Bases

The liquid radwaste systems include the equipment necessary to control the releases of radioactive materials in liquid effluents in accordance with GDC's 60 and 61 of Appendix A of 10 CFR Part 50 and 10 CFR Part 50.34a. Capacities of principal components considered in the liquid waste processing system evaluation are listed in Table 11.2-1 of the FSAR. The staff concludes that the design of the liquid waste management system is acceptable and meets the requirements of 10 CFR Part 20, Section 20.106, 10 CFR Part 50, Section 50.34a, GDC's 60 and 61 and 10 CFR Part 50, Appendix I, as referenced in the SRP. This conclusion is based on the following:

1. The applicant has met the requirements of Section II.A of Appendix I of 10 CFR Part 50 with respect to dose limiting objectives by proposing a liquid radwaste treatment system that is capable of maintaining releases of radioactive materials in liquid effluents such that the calculated individual doses for two unit operation in an unrestricted area from all liquid pathways of exposure are less than 5 millirems/yr to the total body and less than 15 millirems/yr to any organ.
2. The staff has concluded that the applicant has met the requirements of the Commission's September 4, 1975 Annex to Appendix I of 10 CFR Part 50 with respect to meeting the "as low as reasonably achievable" criterion, and is, therefore exempt from the cost-benefit analysis required by Section II.D of Appendix I to 10 CFR Part 50.

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3. The applicant has met the requirements of 10 CFR Part 20, Section 20.106 since we have considered the potential consequences resulting from reactor operation and have determined that the concentrations of radioactive materials in liquid effluents in unrestricted areas will be a small fraction of the limits in 10 CFR Part 20, Appendix B, Table II, Column 2.

4. The applicant has met the requirements of General Design Criterion 60 and 61 with respect to controlling releases of radioactive material to the environment since we have considered the capabilities of the proposed liquid radwaste treatment system to meet the demands of the plant due to anticipated operational occurrences and have concluded that the system capacity and design flexibility are adequate to meet the anticipated needs of the plant. We have reviewed the applicant's quality assurance provisions for the liquid radwaste systems, the quality group classifications used for system components, and the seismic design applied to structures housing these systems. The design of the systems and structures housing these systems meet the criteria as set forth in Regulatory Guide 1.143. We have reviewed the provisions incorporated in the applicant's design to control the release of radioactive materials in liquids due to inadvertent tank overflows and conclude that the measures proposed by the applicant are consistent with the criteria as set forth in Regulatory Guide 1.143.

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11.2.2 System Description

The liquid radioactive waste management system is a shared system that accommodates liquid waste from both Units 1 and 2. The system consists of process equipment and instrumentation necessary to collect, process, and monitor and recycle or dispose of radioactive liquid wastes. The liquid wastes from operation of River Bend, Unit Nos. 1 and 2, will be collected and processed in three major subsystems and a miscellaneous supporting subsystems. The high purity liquid radwaste system is designed to collect and process wastes from equipment drains and other non-contaminated sources of reactor coolant. The low purity liquid radwaste system is designed to collect and process floor drain waste

and other potentially contaminated waste. The regenerant chemical radwaste sub-system receives chemical waste from the condensate and radwaste demineralizer chemical regeneration subsystem. It processes the residue from spent resin regeneration, concentrates the radioactive materials and delivers them for solid radwaste disposal. The phase separator/backwash system processes spent resins and filter sludges. The liquid is separated from the sludge and is sent to the low purity radwaste system while the sludge is transferred to the solid waste systems for disposal.

The liquid radwaste processing systems are designed to store processed water for eventual reuse in the plant. On occasion, for water inventory management, procesed water will be released under controlled conditions to the environment. A radiation monitor in the plant discharge line will automatically terminate liquid waste discharges if radiation measurements exceed a predetermined level.

The equipment drain subsystem processes the low conductivity, high purity wastes. Wastes will be processed through a back wash filter, a cation demineralizer, an anion demineralizer, and a mixed bed demineralizer before stored in the recovery sample tanks. From these tanks the processed water is stored in the condensate storage tanks or a portion can be dicharged to the environment. The licensee estimates the equipment drain subsystem will receive waste input flow from both units at approximately 58,000 gallons per day. These inputs are con-

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sistent with Standard Review Plan guidelines. Adequate initial tank storage (100,000 gallons) is provided to accommodate flexible operation and the process flow rate (100 gpm) is suffucient to treat the daily inputs within an adequate time frame (10 hrs).

The floor drain subsystem processes higher conductivity, low purity wastes. Waste is initially collected in tanks (75,000 gal, total) which later will be processed through a back wash filter, an evaporator (if required), a cation, anion and mixed bed demineralizer in series prior to being collected in the recovery tanks. The applicant estimates the floor drain subsystem waste input flow from both units to be approximately 12,930 gallons per day. The design capacity of the floor drain subsystem, when the evaporator is not being used, is approximately 144,000 gallons per day (100 gpm). When the evaporator is used, the process flow is limited to 35 gpm; this flow rate is sufficient to process the daily inputs within an acceptable time duration (10 hours).

The chemical waste subsystem will process resin regenerant from the condensate and radwaste demineralizers and also waste from laboratory drains. Wastes are first collected in regenerant waste tanks followed by filtratrion by a strainer and flat bed filter and then concentrated by an evaporator where water is separated out and sent to the high purity waste system leaving a high concentrated mixture of resin regenerant and solids on the evaporator bottoms for disposal to the solid radwaste system.

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The applicant has estimated an average 2 units input of 5310 gallons per day to the chemical waste system. These inputs are consistent with the standard review plan guidelines. The maximum process flow rate through the evaporator (35 gpm) is sufficient to process the daily chemical waste in less than three hours. The regenerate waste storage tanks (50,000 gallons) are sufficient to handle surge flows while processing.

In addition to these three main systems, a backwash and phase separator system is provided to remove the solids from filter and demineralizer backwashes. Backwashes from radwaste system filters and strainers are first decanted. The separated water is transferred to the low purity radwaste system. The tank bottoms are then either processed through a flatbed filter and/or sent to the centrifuge. The water separated is sent to the low purity radwaste system and the flatbed filter and centrifuge sludges are sent to the solid radwaste system for disposal. The phase separator system receives filter back flush from the reactor water cleanup system filters and also demineralizer spent resins. Water is decanted from the phase separator and sent to the high purity radwaste systems. The tank bottoms are either processed through a flatbed filter and/or sent to the centrifuge. Separated water is sent to the high purity radwaste system and sludge is sent to the solid radwaste system for disposal.

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In our evaluation of the liquid radioactive waste management system, we considered: (1) the capability of the system for keeping the levels of radioactivity in effluents "as low as is reasonably achievable" based on expected radwaste inputs over the life of the plant, (2) the capability of the system to maintain releases below the limits in 10 CFR Part 20 and also to limit radiological doses to less than allowed by Appendix I to 10 CFR 50, during periods of fission product leakage at design levels from the fuel, (3) the capability of the system to meet the processing demands of the station during anticipated operational occurrences, (4) the quality group and seismic design classification applied to the equipment and components and structures housing the system, and (5) the design features that are incorporated to control the releases of radioactive materials in accordance with General Design Criterion 60.

The estimated releases of radioactive materials in liquid effluents were calculated by the applicant using the BWR-GALE Code described in NUREG-0016[redacted].

The BWR-GALE Code is a computerized mathematical model for calculating the routine releases of radioactive material in effluents from boiling water reactors (BWR). The code has been in use since 1976 for all BWR licensing reviews. The calculations in the code are based on (1) data generated from operating reactors, (2) field and laboratory tests, (3) standardized coolant activities derived from American Nuclear Society (ANS) 18.1 Working Group recommendations[redacted], (4) release and transport mechanisms that result in the

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appearance of radioactive material in liquid streams, and (5) the River Bend Station radwaste system design features used to reduce the quantities of radioactive materials ultimately released to the environs. The principal parameters used in these calculations are given in Table 11.1 of this SER. The liquid source term is given in Table 11.1-1 of the applicant's Final Safety Analysis Report.

11.2.3 Radioactive Releases

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11.3 Gaseous Waste Management Systems

11.3.1 ~~System Description and Review Discussion~~ Design BASIS

The gaseous radioactive waste processing and plant ventilation systems are designed to collect, store, process, monitor, and discharge potentially radioactive gaseous wastes which are generated during normal operation of the plant. The systems consist of equipment and instrumentation necessary to reduce releases of radioactive gases and particulates to the environment. The principal sources of gaseous waste are the effluents from the gaseous waste processing system, condenser mechanical vacuum pumps, and ventilation exhausts from the radwaste building, reactor building, auxiliary building, drywell, and turbine building.

11.3.2 System Description

The gaseous waste processing system (GWPS) for River Bend, Unit 1, is a main condenser air ejector low temperature refrigerated charcoal system. This system collects, processes and stores fission product gases removed from the main condenser via air ejectors and then processes these gases through a recombiner, cooler, and dryer, followed by noble gas adsorption and decay through a series of eight refrigerated charcoal delay tanks. The gas discharge from the delay tanks passes through a HEPA filter and is discharged to the environment through the offgas building vent.

Ventilation exhaust air from the reactor containment building and drywell purge is treated by charcoal filtration prior to discharge to the

environment. The principal exhaust path is a treated 7,000 cfm flow through one HEPA-charcoal treatment system. An alternate or backup exhaust flow rate of 25,000 cfm is also provided but this exhaust flow is discharged through the Standby Gas Treatment System HEPA-charcoal filters.

During normal operation, the continuous 7,000 cfm treated exhaust is operational whereas during refueling the containment building and drywell purge exhaust is increased to 25,000 cfm. The HEPA-charcoal filtration unit consist of moisture separator gas heating elements for humidity control; the charcoal adsorber is a 4 inch deep bed unit with minimum residence time of 0.50 seconds.

The Annulus Pressure Control system exhausts 2,000 cfm from the annulus between the containment wall and the shield wall. This exhaust gas is discharged directly to the environment. Should activity be detected in the reactor plant discharge stack the annulus pressure control system is isolated and the Standby Gas Treatment System (SGTS) takes over processing 2,500 cfm from the annulus and exhaust through a HEPA-charcoal filtration system prior to release to the environment. The charcoal adsorbers are 4 inch deep bed units with a minimum residence time of 0.50 seconds.

Exhaust flow from general spaces of the radwaste building, control room, and fuel handling building is passed through particulate filters.

Gases released from radwaste system equipment vents (such as tanks, evaporator reliefs, phase separators, and solid waste fill casks) are passed through redundant HEPA-charcoal filtration units. The filtration unit consist of particulate removal, moisture removal and heater element for humidity control and 4 inch deep bed charcoal with a minimum gas residence time of 0.50 seconds.

Exhaust flow from general spaces of the auxiliary building is unfiltered. However, this exhaust flow will be diverted through the standby gas treatment charcoal filtration system if activity is detected in the exhaust stream.

Exhaust flow from the fuel building is normally unfiltered unless activity is detected in the fuel building exhaust duct. If the latter is the case, then the fuel building exhaust flow is diverted throgh redundant HEPA-charcoal filtration units prior to release to the atmosphere. The filtration unit consist of particulate, moisture separation and heater elements for humidity control, and also contain 4 inches of charcoal with a minimum residence time of 0.50 seconds.

The turbine building ventilation exhaust is unfiltered. However, the main condenser mechanical vacuum pump discharge is processed through a single HEPA-charcoal fitlration unit consisting of particulate filtration, moisture separator and heater elements for humidity control, and a 4 inch deep charcoal bed with a minimum residence time of 0.50 seconds.

The major source of gaseous radwaste during normal plant operation will be offgas from the main condenser. River Bend, Unit No. 1, will have an offgas system consisting of catalytic hydrogen-oxygen recombiners, a ten-minute holdup pipe to permit decay of short-lived radioisotopes, a refrigerated charcoal (RECHAR) delay train, and a HEPA filter. All essential components will be redundant except for the eight charcoal beds in two parallel four bed series. The recombiners are sized to maintain hydrogen concentration below four percent; dual hydrogen analyzers are provided to monitor hydrogen concentration and, additionally, the offgas system is designed to withstand the effects of a hydrogen detonation. The hydrogen analyzers are a nonsparking type and designed to withstand a hydrogen explosion. The eight charcoal beds are designed to be operated at zero degrees Fahrenheit. The gaseous waste processing system and the plant ventilation systems are described in Sections 11.3 and 9.4 of the FSAR[100], respectively, with design details provided separately in proprietary documents.

In our evaluation of the gaseous radwaste management system, we considered the following SRP criteria: (1) the capability of the system for keeping the levels of radioactivity in effluents "as low as is reasonably achievable" based on expected radwaste inputs over the life of the plant, (2) the capability of the system to maintain releases below the limits in 10 CFR Part 20 during periods of fission product leakage at design levels from the fuel, (3) the capability of the

system to meet the processing demands of the station during anticipated operational occurrences, (4) the seismic design classification applied to the equipment and components and structures housing the system, (5) the design features that are incorporated to control the releases of radioactive materials in accordance with General Design Criterion 60, and (6) the potential for gaseous releases due to hydrogen explosion in the gaseous radwaste system.

11.3.3 Radioactive Releases

The estimated releases of radioactive materials in gaseous effluents were calculated by the applicant using the BWR-GALE Code described in NUREG-0016, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors (BWRs)".

The BWR-GALE Code is a computerized mathematical model for calculating the routine releases of radioactive material in effluents from boiling water reactors (BWR). The code has been in use since 1976 for all BWR licensing reviews. The calculations in the code are based on (1) data generated from operating reactors, (2) field and laboratory tests, (3) standardized coolant activities derived from American Nuclear Society (ANS) 18.1 Working Group recommendations[6], (4) release and transport mechanisms that result in the appearance of radioactive material in gaseous streams, and (5) the River Bend Station radwaste system design features used to reduce the quantities of radioactive materials ultimately released to the environs. The

principal parameters used in these calculations are given in Table 11.1 of this SER. The gaseous source term is provided in Table 11.1-1 of the applicant's FSAR.

We have reviewed the applicant's quality assurance provisions for the gaseous radwaste systems, the quality group classifications used for system components, the seismic design criteria applied to the design of the system, and of structures housing the radwaste systems. The design of the system and structures housing these systems meet the criteria as set forth in Regulatory Guide 1.143 and referenced in the SRP.

We have reviewed the provisions incorporated in the applicant's design to control releases due to hydrogen explosions in the gaseous radwaste system and conclude that the measures proposed by the applicant are adequate to prevent the occurrence of an explosion or to withstand the effects of a hydrogen detonation.

We have reviewed the provisions incorporated in the applicant's design to collect airborne radioactive materials in the normal ventilation exhaust systems during normal plant operation, including anticipated operational occurrences. We find the design of air filtration and adsorption units consistent with Regulatory Guide 1.140, as referenced in the SRP.

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~~11.2.3 Evaluation Findings~~

The staff concludes that the design of the gaseous waste management system is acceptable and meets the requirements of 10 CFR Part 20, Section 20.106, 10 CFR Part 50, Section 50.34a, General Design Criteria 3, 60, and 61, and 10 CFR Part 50 Appendix I, as referenced in the SRP. This conclusion is based on the following findings:

1. The applicant has met the requirements of GDC 60 and 64 with respect to controlling releases of radioactive material to the environment by assuring that the design of the gaseous waste management system includes the equipment and instruments necessary to detect and to control the release of radioactive materials in gaseous effluents. Capacities of principal components considered in the gaseous waste processing system evaluation are listed in Table 11.2.

2. The applicant has met the requirements of Appendix I of 10 CFR Part 50 by meeting the "as low as is reasonably achievable" criterion as follows:

   a. Regarding Sections II.B and II.C of Appendix I, we have considered releases of radioactive material (noble gases, radioiodine and particulates) in gaseous effluents for normal operation including anticipated operational occurrences based on expected radwaste inputs over the life of the plant. We have determined that the proposed gaseous waste management system is capable of limiting releases of radioactive

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materials in gaseous effluents such that the calculated individual doses in an unrestricted area from all pathways of exposure are less than 5 mrems to the total body or 15 mrems to the skin and less than 15 mrems to any organ from releases of radioiodine and radioactive material in particulate form.

b. The applicant has met the requirements of the Commission's September 4, 1975, Annex to Appendix I to 10 CFR Part 50 with respect to meeting the "as low as reasonably achievable" criterion, and is therefore exempt from the cost-benefit analysis required by Section II.D of Appendix I to 10 CFR Part 50.

3. The applicant has met the requirements of 10 CFR Part 20 since we have considered the potential consequences resulting from reactor operation with "a fission product release rate consistent with a noble gas release rate to the reactor coolant of 100 uCi/MWt-sec at 30 minutes decay" for a BWR and determined that under these conditions, the concentrations of radioactive materials in gaseous effluents in unrestricted areas will be a small fraction of the limits specified in 10 CFR Part 20, Appendix B, Table II, column 1.

4. We have considered the capabilities of the proposed gaseous waste management system to meet the demands of the plant due to anticipated operational occurrences and have concluded that the system capacity and design flexibility are adequate to meet these demands.

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5. We have reviewed the applicant's quality assurance provisions for the gaseous waste management system, the quality group classifications used for system components, the seismic design applied to the design of the system, and of structures housing the radwaste system. The design of the system and of structures housing the system meet the criteria as set forth in Regulatory Guide 1.143.

6. We have reviewed the provisions incorporated in the applicant's design to control releases due to hydrogen explosions in the gaseous waste management system and concluded that the measures proposed by the applicant are adequate to prevent the occurrence of an explosion and to withstand the effects of a hydrogen detonation in accordance with General Design Criterion 3 of Appendix A to 10 CFR Part 50.

11.4 Solid Waste Management System

Presently, Gulf States Utilities have not provided a solid waste management system design for review. They have stated their conceptual solid waste methodology which is to contract for solid waste processing and disposal from one of the various NRC approved vendors offering this service. This approach to solid waste processing is acceptable to the staff. However, the vendor, the solid waste process, plant-vendor interface design requirement, environmental protection features, and solid waste volumes to be disposed are required to be approved by the NRC. The overall solid waste system will be reviewed in accordance

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11.4 with the Standard Review Plan (Rev. 2, dated July 1981) Section 11.4. An appendix to the FSAR should be submitted by the licensee addressing the solid waste system. This submittal should be made by June 30, 1983. A separate safety evaluation report on this system will be issued after receipt of this appendix.

11.5 Process and Effluent Radiological Monitoring And Sampling Systems

11.5.1 ~~System Description and Review Discussion~~ Des. Bases

The process and effluent radiological monitoring systems are designed to provide information concerning radioactivity levels in systems throughout the plant, indicate radioactive leakage between systems, monitor equipment performance, and monitor and control radioactivity levels in plant discharges to the environs.

11.5.2 System Description

Table 11.3 provides the proposed locations of continuous monitors. Monitors on certain effluent release lines will automatically terminate discharges in the event that radiation levels exceed a predetermined value. Systems which are not amenable to continuous monitoring, or for which detailed isotopic analyses are required, will be periodically sampled and the samples will be analyzed in the plant laboratory.

All airborne radioactive releases to the environs from River Bend are from three principal points: (1) the main plant vent (which releases at a 290 ft elevetion the containment/drywell purges, Turbine building exhaust and offgas discharge, Standby Gas Treatment exhaust, pipe tunnels

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and auxliary building exhaust), (2) the Radwaste Building Exhaust Vent at 211 ft. elevation, and (3) the Fuel building exhaust vent at 177 ft. elevation.

Redundant radiation monitors are installed in these vents. They sample effluent air downstream of the last point of mixing and provide continuous readout of noble gas and particulate activity. In addition, charcoal adsorbers are provided to sample the air stream for radioiodine. Sampling flow is through a regulated isokinetic probe.

In addition to the main plant vent discharge point monitors, radiation monitors are also provided in the exhaust stream of the Auxiliary building, containment purge, Standby Gas Treatment, Annulus pressure control exhaust, Turbine building, offgas building, and mechanical vacuum pump exhaust. These monitors are used to identify sources of airborne activity prior to mixing and dilution in the main plant exhaust vent.

11.5.3 Effluent Monitoring and Sampling

The principal radioactive liquid effluent release point at River Bend is the cooling tower circulating water blowdown. Potentially radioactive inputs to the circulating water discharge will be continuously monitored for radioactivity prior to discharge. Since this is a high flow rate system, the radioactive inputs to the system will be highly diluted. Also, the principal radioactive input streams to the blowdown leg of the circulating water system are continuously monitored (process

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monitors), and periodically sampled and analysed. These input streams so monitored are the emergency service water (ESW), the liquid radwaste effluent discharge, and the liquid waste water treatment effluent. The liquid radwaste discharge process monitors provide alarm signals for termination of discharge of the liquid radwaste system in the event a predetermined Technical Specification limit is exceeded.

11.5.4 Process Monitoring and Sampling

Our review included the locations and types of effluent and process monitoring provided for River Bend, Unit No. 1. Based on the plant design and on continuous monitoring locations and intermittent sampling locations, we concluded that all normal and potential release pathways will be monitored. The applicant's description indicates that the process and effluent monitoring system design meets the guidelines in Regulatory Guide 4.15 for quality assurance. We also determined that the sampling and monitoring provisions are adequate for detecting radioactive material leakage to normally uncontaminated systems and for monitoring plant processes which could affect radioactivity releases. On these bases, we consider that the monitoring and sampling provisions meet the requirements of General Design Criteria 60, 63, and 64 and the guidelines of Regulatory Guide 1.21, and, therefore, meet the acceptance criteria of the SRP.

11.5.5 Evaluation Findings

The staff concludes that the process and effluent radiological monitoring instrumentation and sampling systems are acceptable and meet the

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relevant requirements of 10 CFR Part 20, Section 20.106, and General Design Criteria 60, 63, and 64, as referenced in the SRP. This conclusion is based on the following: The process and effluent radiological monitoring and sampling systems include the instrumentation for monitoring and sampling radioactivity, contaminated liquid and gaseous process and effluent streams. With the exception of solid waste, our review included the provisions proposed to sample and monitor all station effluents in accordance with General Design Criterion 64, the provisions proposed to provide automatic termination of effluent releases and assure control over discharges in accordance with General Design Criterion 60, the provisions proposed for sampling and monitoring plant waste liquid and gaseous process streams for process control in accordance with General Design Criterion 63, and the provisions for conducting sampling and analytical programs in accordance with the guidelines in Regulatory Guides 1.21 and 4.15. The applicant has provided a description of equipment for sampling and monitoring process and effluent streams during postulated accidents in accordance with the guidelines in NUREG-0737.

The review included piping and instrument diagrams and process flow diagrams for the liquid and gaseous processing systems and also for ventilation systems, and the location of monitoring points relative to effluent release points as shown on the site plot diagrams.

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[redacted] Except for the solid waste system, we have determined that the applicant's designs, design criteria, and design bases for the process and effluent radiological monitoring instrumentation and sampling systems provided for normal operation meet the guidelines and industry standards referenced in SRP 11.5. The monitoring instrumentation for the solid waste system will be addressed in a separate SER. See section 11.4.

TABLE 11.1

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PRINCIPAL PARAMETERS FOR
THE CALCULATION OF RADIOACTIVE EFFLUENTS
FROM RIVER BEND, UNIT NO. 1

Thermal power	3039 MWt
Capacity factor	0.80
Total steam flow (10^6 lb/hr)	13.1
Mass water, reactor vessel (10^6 lb)	0.44
Cleanup demineralizer flow (10^6 lb/hr)	0.124
Fraction of feedwater through condenser demineralizer	0.72
Air ejector offgas holdup (hours)	0.17
Reactor building iodine release fraction	0.1
Reactor building particulate release fraction	0.01
Auxiliary building iodine release fraction	0.1
Auxiliary building particulate release fraction	0.01
Radwaste building iodine release fraction	0.1
Radwaste building particulate release fraction	0.01
Charcoal delay system	
Krypton adsorption coefficient (cm^3/g)	105
Xenon adsorption coefficient (cm^3/g)	2410
Krypton holdup time, days	2.9
Xenon holdup time, days	68
Mass of charcoal (tons)	48
Liquid waste inputs	
High purity system	
29,000 gallons per day	
0.158 Primary Coolant Activity (PCA)	
0.01 Discharge fraction	
0.66 Days, collection	
0.14 Days, decay	
10^3 Decontamination Factor (DF) Iodine	
10^2 Decontamination Factor (DF) Cesium	
10^3 Decontamination Factor (DF) Other Nuclides	
Low purity system	
6470 gallons per day	
0.001 Primary Coolant Activity (PCA)	
0.10 Discharge fraction	
3.1 Days, collection	
0.4 Days, decay	
10^4 Decontamination Factor (DF) Iodine	
10^5 Decontamination Factor (DF) Cesium	
10^5 Decontamination Factor (DF) Other Nuclides	
Condensate Regenerant	1700 gals/day
Chemical waste system	
1310 gallons per day (include radwaste demineralizer regenerant)	
3.6 Primary Coolant Activity (PCA)	
0.1 Discharge fraction	
5.0 Days, collection	
0.34 Days, decay	
10^4 Decontamination Factor (DF) Iodine	
10^5 Decontamination Factor (DF) Cesium	
10^5 Decontamination Factor (DF) Other Nuclides	

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TABLE 11.2

DESIGN PARAMETERS OF PRINCIPAL COMPONENTS CONSIDERED IN THE EVALUATION OF LIQUID AND GASEOUS RADIOACTIVE WASTE TREATMENT SYSTEMS

COMPONENT	NUMBER	CAPACITY, EACH
LIQUID SYSTEMS[a]		
EQUIPMENT (HIGH PURITY) DRAIN PROCESSING SYSTEM		
Waste Collector Tank	4	25,000 gallons
Travelling Bed Pre-Coat Filter	1	100 gal/min
Demineralizer (anion, cation, mixed bed)	3	100 gal/min
Waste Sample Tanks	3	19,800 gallons
Evaporator (optional, shared with low purity)	1	35 gal/min
FLOOR (LOW PURITY) DRAIN PROCESSING SYSTEM		
Floor Drain Collector Tank	4	25,000 gallons
Travelling Bed Pre-Coat Filter	1	100 gal/min
Evaporator	1	35 gal/min
Demineralizer (anion, cation, mixed bed)	3	100 gal/min
Sample Tank (same tanks as high purity)[a]	3	19,800 gallons
CHEMICAL WASTE PROCESSING SYSTEM		
Chemical Waste Tank 2	2	25,000 gallons
Concentrator (Evaporator) 1	1	35 gal/min
Demineralizer (same as high purity systems)[a]	3	100 gal/min
GASEOUS SYSTEMS[a]		
RECHAR SYSTEM		
Catalytic Recombiner	2	350 psig, 900°F Design
Condenser	1	350 psig, Shell Design Pressure 900 F°Design Temp.
Charcoal Adsorber Beds	8	Carbon Steel, 350 psig Design Pressure, Design Temp. 0°F Operating Temp.
Charcoal	-	-50°F/250°F 48 tons Activated Carbon

[a]Quality Group and Seismic design in accordance with Regulatory Guide 1.143.

Table 11.3

PROCESS AND EFFLUENT RADIATION MONITORING SYSTEMS

Monitor Location (Number)	Monitor Type	Detector Type	Sensitivity and Range	Isotope	Alarm Set Point	Trip/High Set Point	Function
A. Safety-Related Systems							
Fuel building ventilation exhaust (two)	Gaseous Particulate	(Later) (Later)	10^{-7} to 10^{5} μCi/cc 10^{-11} to 10^{-2} μCi/cc	Xe133 I131	(Later) (Later)	Tech. Spec. Tech. Spec.	Monitors effluent release for R.G. 1.21 report generation. Activates Cat. I filters. PAMs[1]
Main plant exhaust duct (two)	Gaseous Particulate	(Later) -	10^{-7} to 10^{5} μCi/cc 10^{-11} to 10^{-2} μCi/cc	Xe133, Kr85 I131	(Later) (Later)	Tech. Spec. Tech. Spec.	Monitors effluent releases for R.G. 1.21 report generation. PAMs.
Containment atmosphere (two)	Gaseous Particulate	(Later) (Later)	10^{-6} to 10^{-1} μCi/cc 10^{-10} to 10^{-5} μCi/cc	Xe133, Kr85 I131	(Later) (Later)	Tech. Spec. Tech. Spec.	Monitors containment for airborne radiation RCPB leak detection[3].
Standby gas treatment system effluent (two)	Gaseous Particulate	(Later) (Later)	10^{-7} to 10^{5} μCi/cc 10^{-11} to 10^{-2} μCi/cc	Xe133, Kr85 I131	(Later) (Later)	Tech. Spec. Tech. Spec.	Monitors SGTS function. Containment effluent PAMs.
Reactor bldg. annulus ventilation (two)	Gaseous Particulate	(Later) (Later)	10^{-6} to 10^{-1} μCi/cc 10^{-10} to 10^{-5} μCi/cc	Xe133 I131	(Later) (Later)	Tech. Spec. Tech. Spec.	Monitors airborne levels in the annulus. Activates SGTS.
Drywell atmosphere (two)	Gaseous Particulate	(Later) (Later)	10^{-6} to 10^{-1} μCi/cc 10^{-10} to 10^{-5} μCi/cc	Xe133, Kr85 I131	(Later) (Later)	Tech. Spec. Tech. Spec.	Monitors drywell for airborne radiation. RCPB leak detection[3].
Main control room intakes (two per intake)	Gaseous -	(Later) -	10^{-6} to 10^{-1} μCi/cc -	Xe133, Kr85 -	(Later) -	10CFR20, App B	Monitors incoming control room air. Activates Cat. I filters.
RHR heat exchanger service water (two)	Liquid -	(Later) -	10^{-6} to 10^{-1} μCi/cc -	Cs137 -	(Later) -	Tech. Spec. -	Monitors effluent from heat exchangers for contamination.

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Table 11.3
(Continued)

Monitor Location (Number)	Monitor Type	Detector Type	Sensitivity and Range	Isotope	Alarm Se Point	Trip/High Set Point	Function
Main steam line (four)	Gamma[a]	Ionization Chamber	1-10^4 mr/hr	N16	(Later)	Tech. Spec.	Monitors main steam lines for fuel damage and carry over to turbine building. Scrams reactor and isolates steam lines.
B. Systems Required for Plant Operation							
Radwaste bldg. ventilation exhaust (two)	Gaseous Particulate	(Later) (Later)	10^{-7} to 10^{2} μCi/cc 10^{-11} to 10^{2} μCi/cc	Xe133, Kr85 I131	(Later) (Later)	Tech. Spec. Tech. Spec.	Monitors effluent releases for RG 1.21 report generation. PAMs.
Fuel pool cooling pumps discharge (two)	Liquid -	(Later) -	10^{-6} to 10^{-1} μCi/cc -	Cs137 -	(Later) -	Tech. Spec. -	Monitors FPC cooling water for fuel damage.
Turbine plant component cooling water (one)	Liquid -	(Later) -	10^{-6} to 10^{-1} μCi/cc -	Cs137 -	(Later) -	Tech. Spec. -	Monitors TPCCW discharge for detection of radioactive inleakage.
Liquid radwaste effluent (one)	Liquid -	(Later) -	10^{-6} to 10^{-1} μCi/cc -	Cs137 -	(Later) -	Tech. Spec. -	Monitors radwaste effluent discharge to environment. Isolates system at trip level.
Cooling tower blowdown line (one)	Liquid -	(Later) -	10^{-6} to 10^{-1} μCi/cc -	Cs137 -	(Later) -	Tech. Spec. -	Monitors service water discharge to environment.
Auxiliary building ventilation (one)	Gaseous Particulate	(Later) (Later)	10^{-6} to 10^{-1} μCi/cc 10^{-10} to 10^{-5} μCi/cc	Xe133 I131	(Later) (Later)	Tech. Spec. Tech. Spec.	Monitors airborne radiation levels in auxiliary building exhaust.
Containment purge exhaust (one)	Gaseous Particulate	(Later) (Later)	10^{-6} to 10^{-1} μCi/cc 10^{-10} to 10^{-5} μCi/cc	Xe133, Kr85 I131	(Later) (Later)	Tech. Spec. Tech. Spec.	Monitors function of containment purge filter. Isolates purge.

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Table 11.3
(Continued)

Monitor location (Number)	Monitor Type	Detector Type	Sensitivity and Range	Isotope	Alarm Set Point	Trip/High Set Point	Function
Mechanical vacuum pump discharge (one)	Gaseous -	(Later) -	10^{-6} to 10^{-1} µCi/cc -	Xe133, Kr85 -	(Later) -	Tech. Spec. -	Monitors mechanical vacuum pump discharge for radiation level.
Turbine building ventilation (one)	Gaseous Particulate	(Later) (Later)	10^{-6} to 10^{-1} µCi/cc 10^{-10} to 10^{-8} µCi/cc	Xe133, Kr85 I131	(Later) -	Tech. Spec. -	Monitors airborne radiation levels in turbine building exhaust.
Fuel pool cooling demineralizer outlet (one)	Liquid -	(Later) -	10^{-6} to 10^{-1} µCi/cc -	Cs137 -	(Later) -	Tech. Spec. -	Monitors function of demineralizer discharge fuel pool cleanup system.
Reactor plant component cooling water (one)	Liquid -	(Later) -	10^{-6} to 10^{-1} µCi/cc -	Cs137 -	(Later) -	Tech. Spec. -	Monitors RPCCW discharge for detection of radioactive inleakage.
Radwaste reboiler clean steam outlet (one)	Online Steam -	(Later) -	10^{-6} to 10^{-1} µCi/cc -	N16 -	(Later) -	(Later) -	Detects contamination of clean steam from reboiler tube leaks.
Seal steam evaporator clean steam outlet (one)	Online Steam - -	(Later) - -	10^{-6} to 10^{-1} µCi/cc - -	N16 - -	(Later) - -	(Later) - -	Detect contamination of clean steam from evaporator tube leaks.
Condensate demineralizer and off gas bldg. ventilation (one)	Gaseous Particulate	(Later) (Later)	10^{-6} to 10^{-1} µCi/cc 10^{-10} to 10^{-8} µCi/cc	Xe133, Kr85 I131	(Later) (Later)	Tech. Spec. Tech. Spec.	Monitors airborne radiation levels in cond./demin. and off gas building exhaust.
Turbine gland seal discharge (one)	Gaseous -	(Later) -	10^{-6} to 10^{-1} µCi/cc -	Xe133, Kr85 -	(Later) -	Tech. Spec. -	Monitors turbine gland seal discharge for contamination.

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AMENDMENT 7

Table 11.3
(Continued)

Monitor Location (Number)	Monitor Type	Detector Type	Sensitivity and Range	Isotope	Alarm Set Point	Trip/High Set Point	Function
Off gas pretreatment (one)	Gaseous(2)	GM Tube	1 to 10^4 mr/hr	Xe133, Kr85	(Later)	Tech. Spec.	Monitors process before treatment.
		-	-	-	-	-	
Off gas post-treatment (one)	Gaseous(2)	GM Tubes	10 to 10^6 cpm	Xe133, Kr85	(Later)	Tech. Spec.	Monitors process after treatment. Isolates discharge.
		-	-	-	-	-	

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(1)PAM - Post-accident monitor.
(2)Monitors are not part of digital radiation monitoring system, supplied separately as part of process systems.
(3)Reactor coolant pressure boundary leak detection, in accordance with Regulatory Guide 1.45.

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12 RADIATION PROTECTION

The staff has evaluated the proposed radiation protection program presented in FSAR Chapter 12 against the criteria set forth in SRP 12 (NUREG-0800). The radiation protection measures at River Bend are intended to ensure that internal and external exposures to station personnel and contractors as a result of plant conditions, including anticipated operational occurrences, will be within applicable limits of 10 CFR 20, and will be as low as is reasonably achievable (ALARA).

The basis of the staff's acceptance of the River Bend radiation protection program is that doses to personnel will be maintained within the limits of 10 CFR 20, Standard for Protection Against Radiation. The applicant's radiation protection designs and program features are consistent with the guidelines of RG 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will be as Low as is Reasonably Achievable" (Revision 3). On the basis of its review of the FSAR, the staff has concluded that the radiation protection measures incorporated in the design will provide reasonable assurance that occupational doses will be maintained below the limits of 10 CFR 20 and ALARA both during plant operation and during decommissioning, provided the open issues described herein are satisfactorily resolved.

12.1 Ensuring that Occupational Exposures Are as Low as Is Reasonably Achievable

The staff has audited the policy considerations, design considerations, and operational considerations in the FSAR against the criteria in SRP 12.1. The staff review consisted of ensuring that the applicant had either committed to following the criteria of the RGs and staff positions referenced in SRP 12.1 or provided acceptable alternatives. In addition, the staff selectively reviewed the FSAR against the SRP acceptance criteria using review procedures identified in the SRP. This selective review found the plant acceptable in these areas. Details of the review follow.

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12.1.1 Policy Considerations

The applicant provides a management commitment to ensure that the River Bend station will be designed, constructed, and operated in a manner consistent with RGs 8.8; 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable"; and 1.8, "Personnel Selection and Training." The applicant has identified the specific corporate plan to implement that policy and specified in detail the plant design and integrated the layout, shielding, ventilation, monitoring, traffic control, security access control, and health physics aspects to ensure that the overall design is conducive to maintaining exposure ALARA. The Plant Manager is responsible for the overall effectiveness of the ALARA efforts. Each supervisor at the plant is responsible for ensuring that doses are ALARA. The Radiation Protection Supervisor provides technical assistance regarding ALARA implementation and reports to management on the results of ALARA efforts. These policy considerations are consistent with those outlined in RGs 8.8 and 8.10 and, therefore, are acceptable.

12.1.2 Design Considerations

The objective of the plant's radiation protection design is to maintain individual and collective doses to plant workers--including construction workers--and to members of the general public ALARA and to maintain individual doses within the limits of 10 CFR 20. The general arrangements and shielding provisions of River Bend are in accordance with RG 8.8 and are designed to provide levels of dose to operating personnel that are ALARA.

To reduce radiation exposures, the applicant has incorporated general design considerations to reduce (1) the need to enter radiation fields, (2) the time of exposure when entry is necessary, and (3) the dose rate during exposure. Example of these design considerations include

(1) redundancy of equipment or components to reduce the need for immediate repair when radiation levels may be high

(2) design of equipment, piping, and valves to minimize the buildup of radioactive material and to facilitate flushing of crud traps

(3) locating equipment, instruments, and sampling sites in zones with the lowest radiation as practicable

(4) equipping systems and other pieces of equipment that are subject to crud buildup with connectors that can be used for flushing the system to eliminate potential hot-spot buildup

(5) utilizing welded connectors instead of flanged or threaded connectors, high-quality valves, valve packings, and gaskets to minimize leakage and spillage of radioactive materials

(6) providing means to control contamination and to facilitate decontamination of potentially contaminated areas where practicable

(7) providing for draining, flushing, and remote cleaning of equipment and piping containing radioactive material

(8) providing adequate space in labyrinth entrances for easy access to radioactive pump, equipment, and valve rooms

These design considerations conform with the guidelines of SRP 12.1.2 and are acceptable. The applicant will provide at a later date information regarding the continuing radiation protection reviews to ensure that ALARA objectives are met in the design change and construction process.

12.1.3 Operational Consideration

The applicant has committed to include in the operational procedures measures for reducing exposure. The criteria for implementation of those measures are consistent with the guidance of RGs 8.8 and 8.10.

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The applicant has committed that operating and maintenance personnel will follow specific plans and procedures to ensure that ALARA goals are achieved in the operation of the plant. High radiation exposure operations will be preplanned and carried out by personnel trained in radiation protection, using proper equipment. During such activities, personnel will be monitored for exposure to radiation and contamination. When major maintenance, repair, surveillance, and refueling tasks are completed, results will be factored into the radiation protection procedures to make changes in future job procedures and techniques.

The staff concludes that the policy considerations, design considerations, and operational considerations are adequate to ensure that occupational radiation doses will be ALARA in accordance with RGs 8.8 and 8.10 and are acceptable.

12.2 Radiation Sources

The staff has audited the contained sources and airborne radioactive material source terms provided in FSAR Section 12.2 and Chapter 11 against the criteria in SRP 12.2 (NUREG-0800). These source terms are used as inputs for dose assessment and for the design of the shielding and ventilation systems. The staff review consisted of ensuring that the applicant had either committed to following the criteria of the RGs and staff positions referenced in SRP 12.2 or provided acceptable alternatives. In addition, the staff selectively compared source terms for specific systems used by the applicant against those used for plants of similar design. This selective review found the River Bend source terms equivalent to those used at other plants. Details of the review follow.

12.2.1 Contained Sources ~~Airborne Radioactive Material Sources~~

Inside the containment during power operation, the greatest potential for personnel dose comes from nitrogen-16, noble gases, and neutrons. Outside the containment and inside the containment after shutdown, the primary sources of personnel exposures are fission products from fuel clad defects and activation and corrosion products. The shielding source terms for normal operation are based on an offgas rate of 0.1 Ci/sec after 30 minutes of decay. The coolant and corrosion activation product source terms are based on measurements at

operating BWRs. Neutron and prompt gamma source terms are based on reactor core physics calculations and applicable reactor operating experience. The contained radiation source terms are comparable to those estimated by other applicants with BWR designs and are acceptable. The accident source terms based on NRC short-term lessons-learned recommendations in NUREG-0737 have not been provided. This is an open item.

The applicant has not provided a tabulation of normal expected airborne radioactivity concentrations in equipment cubicles, corridors, and operating areas as a result of equipment leakage. This is an open item. The ventilation systems will be designed to maintain exposures within the limits of 10 CFR 20 limits and are acceptable.

12.3 Radiation Protection Design Features

The staff has audited the facility design features, shielding, ventilation, and radiation and airborne monitoring instrumentation contained in the FSAR against the criteria in SRP 12.3. The staff review consisted of ensuring that the applicant has either committed to following the criteria of the RGs and staff positions referenced in SRP 12.3 or provided acceptable alternatives. In addition, the staff selectively reviewed the FSAR against the acceptance criteria of the SRP using the SRP review procedures. This review found the plant acceptable in these areas. Details of the review follow.

12.3.1 Facility Design Features

The radiation protection design features at River Bend are intended to help maintain occupational radiation exposures ALARA consistent with 10 CFR 10.1(c) and the dose-limiting provisions of 10 CFR 20.101, 20.103, 20.203, and 20.207, as well as RGs 8.8 and 8.10. Many of these design features have been incorporated as a result of the applicant's radiation protection design review and from radiation exposure experience gained during the operation of other nuclear power plants. Information on the reviews to ensure that ALARA objectives are being met in the design change process and during construction have not been provided. This is an open item.

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Some examples of design features that have been incorporated into the design to reduce the need for maintenance and other operations in radiation fields include the following:

(1) Valve galleries are provided with labyrinths for personnel protection.

(2) Tanks are provided with sloped or conical bottoms and bottom outlet connections.

(3) Labyrinth entrance way shields or shielding doors are provided to reduce radiation streaming.

(4) Piping is designed to minimize low points and dead legs.

(5) Filters are provided with means to remotely backflush the filter cake.

The applicant has provided radiation zone maps showing the expected maximum total (neutron plus gamma) radiation levels during normal operations and refueling for areas inside the plant. These zone maps include the routinely visited areas in the containment, including the areas near major drywell shield penetrations. The reactor water cleanup system, standby liquid control system, and traversing incore probe (TIP) are not in the line of sight of the drywell purge penetrations or drywell personnel lock or equipment. The penetrations through the drywell walls were tabulated on all shielding design details as shown on diagrams. The control rod drive (CRD) control area and containment personnel lock areas maximum dose rates were also given. The applicant has also provided an estimation of average daily personnel time in these areas. The radiation dose rates anticipated in these areas indicate doses can be maintained below the limits of 10 CFR 20.101 and, therefore, are acceptable.

The description of the features incorporated in the plant design to maintain occupational radiation exposure ALARA by minimizing the buildup of activated corrosion products will be provided at a later date.

12.3.2 Shielding

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The shielding was designed to meet the criteria of the radiation dose rate zone system discussed above. The applicant's shielding design methods--including the use of source terms, cross-section data, shield and source geometries, and radiation transport calculational schemes--are consistent with generally accepted practice. The applicant has committed that the shield design and construction will be consistent with RG 1.69, "Concrete Radiation Shields for Nuclear Power Plants." The basic radiation transport analysis used for the applicant's shield design is consistent with RP-8A, "Radiation Shielding Design and Analysis Approach for LWR," May 1975. The applicant also used shielding information from operating nuclear plants as input data for the shield design calculation. The staff finds the shielding design and methodology acceptable.

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The radiation shielding will be designed to provide protection against radiation for operating personnel, both inside and outside the plant, and for the general public. The following are several of the shielding design features incorporated into River Bend:

(1) All systems containing radioactivity are identified and shielded based on the access requirements of adjacent areas.

(2) Penetrations are placed so that they do not pass through the shield wall in a direct line with the radiation source in order to prevent shielding.

(3) Radioactive piping is routed through high radiation areas where practicable, or in shielded pipe chases in low radiation areas.

(4) Labyrinth entrance way shields or shielding doses are provided for each compartment from which radiation could stream to access areas and exceed the radiation zone dose limits for those areas.

These shielding techniques are designed to maintain personnel relation exposures ALARA, in accordance with the provisions of RGs 8.8 and 8.10 and are acceptable.

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Shielding for the spent fuel transfer canal was reviewed. The applicant has committed that all plant radiation sources capable of producing radiation levels in excess of 100 rads per hour will be shielded and clearly marked, indicating that potentially lethal radiation fields are possible. If other than permanent shielding is used, administrative controls will be initiated and local audible, and visible alarming monitors must be installed to alert personnel if temporary shielding is removed.

The ventilation system at River Bend is designed to ensure that plant personnel are not inadvertently exposed to airborne radioactive material exceeding the limits and action points in 10 CFR 20.103. The applicant intends to maintain personnel exposures ALARA by: (1) maintaining air flow from areas of potentially low airborne contamination to areas of higher potential concentrations; (2) ensuring negative or positive pressures to prevent exfiltration or infiltration of potential contaminants, respectively; and (3) locating ventilation system intakes so that intake of potentially contaminated air from other building exhaust points is minimized.

These design criteria are in accordance with the guidelines of RGs 8.8 and 1.52, "Design, Testing, and Maintenance Criteria for Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants." Some examples of exposure reduction features in the ventilation system are

(1) Ventilation fans and filters are provided with adequate access space to permit servicing with minimum personnel radiation exposure. The ventilation system is designed to allow rapid replacement of components.

(2) Ventilation ducts are designed to minimize the buildup of radioactive contamination within the ducts to the maximum extent practicable. Welded seams are used to join ductwork segments, and internal obstructions are avoided wherever practicable.

(3) Potentially contaminated equipment is designed to minimize the potential for airborne contamination during maintenance operations.

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12.3.3 Ventilation

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

12.3.4.1 Area Radiation Monitoring Instrumentation

The applicant's area radiation monitoring system is designed to: (1) monitor the radiation in areas where radiation levels could become significant and where personnel may be present; (2) alarm when the radiation levels exceed preset levels to warn of excess radiation levels; and (3) provide a continuous record of radiation levels at key locations and throughout the plant. To meet these objectives, the applicant plans to use 40 area monitors per unit, with an additional 8 monitors shared, located in areas where personnel may be present and where radiation levels could become significant. The area radiation monitoring system is equipped with local and remote audio and visual alarms and a facility for central recording.

The applicant has provided area radiation monitors around the fuel storage areas to meet the requirements of 10 CFR 70.24 and to be consistent with the guidance of RG 8.12, "Criticality Accident Alarm Systems."

The objective of the applicant's area radiation monitoring system is to provide the capability to assess the radiation hazard in areas that may be occupied during the course of an accident. The installed instruments have emergency power supplies, and the portable instruments are placed to be readily accessible to personnel responding to an emergency. The system is designed (1) for use in the event of an accident in terms of usable instrument range and the environment the instrument can withstand, and (2) to meet the provisions of Section II.F.1 of NUREG-0737 and RG 1.97.

12.3.4.2 Airborne Radioactivity Monitoring Instrumentation

The design objectives of the applicant's airborne radioactivity monitoring system are: (1) to assist in maintaining occupational radiation exposure to airborne contaminants ALARA; (2) to check on the integrity of systems containing radioactivity; and (3) to warn of inadvertent release of airborne radioactivity to prevent undue exposure of personnel. The applicant will install airborne

radioactivity monitors to work in areas where there is a potential for airborne radioactivity. The applicant has not committed that these airborne radioactivity monitors have the capability to detect 10 MPC-hours of radioactivity in any compartment that has a possibility of containing airborne radioactivity and that may be occupied by personnel. This is an open item. The applicant will provide portable continuous air monitors when needed to monitor air in areas not provided with fixed airborne radioactivity monitors. All airborne and area radioactivity monitors will be calibrated at regular intervals in accordance with station procedures. The applicant has not provided information on the high-range monitors that will be installed in accordance with TMI Action Plan Item II.F.1-3. This is an open item.

The objectives and location criteria of River Bend area and airborne radiation monitoring systems are in conformance with those portions of 10 CFR 20.201, and 70.24 and RGs 1.97, 8.2, and 8.8 related to airborne radioactivity monitoring.

12.4 Dose Assessment

The staff has audited the applicant's dose assessment for the River Bend against the criteria in SRP 12.3. The staff review consisted of ensuring that the applicant had either committed to following the criteria of the RGs and staff positions referenced in SRP 12.3 (NUREG-0800) or provided acceptable alternatives. The applicant has not provided an assessment of the doses that will be received by plant and contractor personnel. This information will be provided at a later date.

12.5 Operational Radiation Protection Program

The staff has audited the organization, equipment, instrumentation and facilities, and procedures for radiation protection contained in the FSAR against the criteria of SRP 12.5. The plant's health physics program objectives are to provide reasonable assurance that the limits of 10 CFR 20 are not exceeded, to further reduce unavoidable exposures, and to ensure that every reasonable effort is made to maintain occupational radiation doses ALARA. The staff review consisted of ensuring that the applicant had either committed to following the

criteria of the RGs and staff positions referenced in SRP 12.5 or provided acceptable alternatives. The review selectively compared the applicant's FSAR against the specific areas of review using the review procedures identified in the SRP. This selective review found the plant acceptable in these areas. Details of the review follow.

It is the responsibility of the health physics group to prepare and recommend procedures for controlling radiation doses and intakes within the limits of 10 CFR 20.101 and 20.103 for all routine and nonroutine activities and to ensure that such exposures are kept ALARA; to provide radiation protection controls for personnel and operations on the site; to provide radiation surveys of station areas and maintain records of results; to assist in the station training program; to provide, maintain, and calibrate radiation detection instrumentation; to provide, maintain, and issue protective clothing; to assist in shipping and receiving all radioactive materials; to assist in decontamination of personnel and equipment; to be responsible for the respiratory protection program; and to develop and maintain the personnel dosimetry system.

12.5.1 Organization

The applicant's organization will include health physics professionals and technicians. The Radiation Protection Supervisor will have the responsibility for implementing the health physics programs and procedures and the program for maintaining exposure ALARA. The Plant Manager is responsible for the overall effectiveness of the radiation protection program. It is the responsibility of the plant supervisory personnel to ensure that plant personnel are made aware of the management commitment to keep all occupational radiation exposure ALARA. An outline of the qualifications of the Radiation Protection Supervisor and of the individual who will act as the backup for the RPM have not been provided. Therefore, this is an open item. Additionally, the River Bend plant organization chart shows that the Radiation Protection Supervisor does not have direct access to the Plant Manager in all radiation protection matters. This apparent lack of direct access to the Plant Manager is also an open item.

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12.5.2 Equipment, Instrumentation, and Facilities

The radiation protection features at River Bend include a radiochemistry lab, personnel decontamination areas, an access control point, counting room, calibration room, respirator testing facility, office, and laundry. These facilities are sufficient to maintain occupational radiation exposures ALARA and are consistent with the provisions of RG 8.8.

Equipment to be used for radiation protection purposes includes portable radiation survey instruments, personnel monitoring equipment, fixed and portable area and airborne area radioactivity monitors, laboratory equipment, air samplers, respiratory protective equipment, and protective clothing. The number and types of equipment to be used are adequate, meet the criteria of RG 1.97 for portable survey meters, and provide reasonable assurance that the applicant will be able to maintain occupational exposures ALARA.

All station personnel entering controlled radiation areas will be assigned thermoluminescent dosimeter (TLD) badges and pocket dosimeters. Special neutron surveys or badges will be provided when plant personnel enter neutron areas, when these surveys or badges are required by 10 CFR 20. Neutron dosimetry will be provided in accordance with RG 8.14.

All personnel assigned TLDs or film badges are also required to wear direct reading dosimeters when entering the controlled area. The readings from these dosimeters will be used to keep a running total of an individual's dose before TLD or film badge processing. Plant visitors wear a TLD and a self-reading dosimeter when they are in controlled areas or are escorted by an individual wearing such personnel dosimetry devices. Appropriate caution signs, labels, and signals will be provided, in accordance with 10 CFR 20.203 and 20.204 or in accordance with Standard Technical Specifications. Whole-body counts of all plant personnel will be conducted on a scheduled basis and other bioassays will be provided when deemed necessary, in accordance with 10 CFR 20.103 and RG 8.26. Records of surveys, personnel monitoring, and bioassay will be maintained in accordance with 10 CFR 20.102, 20.201, 20.202, 20.401, 20.402, 20.405, 20.407,

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and 20.408, as well as RGs 8.7, 8.27, and 8.29. All radiation exposure information will be processed and recorded in accordance with 10 CFR 20.

The applicant to date has not submitted the information required by NUREG-0737 Item III.D.3.3, "Improved In-Plant Iodine Instrumentation Under Accident Conditions." The staff will review this information when it is provided. This is an open item.

12.5.3 Procedures

The applicant has described the station procedures that will be used to implement the radiation protection program. The procedures described are for access control, radiation work permits, radiation surveys, personnel monitoring, bioassay, radiation protection training, contamination control, methods of maintaining exposures ALARA, and reviews of the effectiveness of the health physics program. Procedures are also developed to administer and control conditions of radiation work permits; to post radiation areas; to establish radiation access control zones; to control all radioactive material entering or leaving the plant site; and to train all visitors in radiation protection policies and procedures; and to meet the guidance of RG 1.33. For long-term repair tasks, consideration will be given to setting up a communications network such as sound-powered telephones or closed-circuit television to assist supervising personnel in checking work progress from an area of lower radiation. For complex tasks with exceptionally high radiation levels, dry runs, mockups, and specialized training will be utilized to familiarize the workers with the operations they must perform.

Based on the information presented in the FSAR and the applicant's responses to staff questions, the staff concludes that the applicant intends to implement a radiation protection program that will maintain inplant radiation exposures within the applicable limits of 10 CFR 20 and will maintain exposures ALARA in accordance with RG 8.8.

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13 CONDUCT OF OPERATIONS

13.1 Organizational Structure ~~and Qualifications~~

13.1.1 Management and Technical Support Organization

13.1.2 Operating Organization

13.1.3 Qualifications of Nuclear Plant Personnel

13.2 Training

13.2.1 Plant staff ~~Licensed Operator~~ Training Program

13.2.2 Replacement and Retraining ~~Training for Nonlicensed Plant Staff~~

13.2.3 Applicable NRC Documents

13.3 Emergency Planning

13.4 ~~Secretariat~~ Review and Audit

13.4.1 Onsite Review ~~Station Operations Review Committee~~

13.4.2 Independent Review ~~Nuclear Safety Audit and Review Committee~~

13.4.3 Audit Program ~~Independent Safety Engineering Group~~

13.4.4 Evaluation

13.5 Plant ~~Station Administrative~~ Procedures

13.5.1 Administrative Procedures

13.5.2 Operating and Maintenance Procedures

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13.5.2.1 General

The staff has reviewed the applicant's plan for development and implementation of operating and maintenance procedures to determine the adequacy of the applicant's program for ensuring that routine operating, offnormal, and emergency activities are conducted in a safe manner. The staff reviewed information in the FSAR, the applicant's response to NRC TMI Action Plan Items (NUREG-0660 and NUREG-0737), and supplemental information obtained during the review according to SRP 13.5.2 (NUREG-0800). The review consisted of an evaluation of (1) the applicant's procedure classification system for procedures that are performed by licensed operators in the control room and the classification system for other operating and maintenance procedures; (2) the applicant's plan for completion of operating and maintenance procedures during the initial plant-testing phase to allow for correction before fuel load; (3) the applicant's program for compliance with RG 1.33, Revision 2 regarding the minimum procedural requirements for safety-related operations; (4) compliance with ANSI N18.7-1976/ANS 3.2; and (5) the applicant's program for compliance with NUREG-0660 Item I.C.1, "Guidance for the Evaluation and Development of Procedures for Transients and Accidents," for the development of emergency operating procedure guidelines. The applicant's program for compliance with NUREG-0737 Item I.C.1 for the development of emergency operating procedures will be reviewed and addressed in a supplement to this report.

13.5.2.2 Operating and Maintenance Procedure Program

The applicant has committed in the FSAR to a program in which all activities are to be conducted in accordance with detailed written and approved procedures meeting the requirements of RG 1.33, Revision 2, and ANSI N18.7-1976/ ANS 3.2.

The applicant uses the following categories of procedures for those operations performed by licensed operators in the control room:

- system operations (includes offnormal operations)
- abnormal operations

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- general operations
- emergency operations
- alarm response
- temporary

Other procedures cover the following areas:

- radiation protection
- radioactive waste management
- chemistry/radiochemistry
- plant security
- emergency plan implementation
- instrument control
- general maintenance
- material control
- fire protection
- reactor engineering
- plant engineering

The staff review disclosed that the applicant's program for use of operating and maintenance procedures meets the relevant requirements of 10 CFR 50.34, and is consistent with RG 1.33 and ANSI N18.7-1976/ANS 3.2. Therefore, the staff concludes that the applicant's program is acceptable.

13.5.2.3 Reanalysis of Transients and Accidents; Development of Emergency Operating Procedures

Through letters of September 13 and 27, October 10 and 30, and November 9, 1979, the staff required licensees of operating plants, applicants for operating licenses, and licensees of plants under construction to perform analyses of transients and accidents, prepare emergency procedure guidelines, upgrade emergency procedures, and conduct operator retraining. Emergency procedures must be consistent with the actions necessary to cope with the transients and accidents analyzed. Analyses of transients and accidents were to be completed

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in early 1980, and implementation of procedures and retraining were to be completed 3 months after emergency procedure guidelines were established; however, some difficulty in completing these requirements has been experienced. Clarification of the scope of the task and appropriate schedule revisions were included in NUREG-0737 Item I.C.1.

In a submittal dated June 30, 1980, the BWR Owners Group provided a draft of generic emergency operating procedure guidelines for BWRs. The guidelines were developed to comply with NUREG-0660 Item I.C.1(3), as clarified by NUREG-0737, and incorporated the requirements for short-term reanalysis of small-break LOCAs and inadequate core cooling Items I.C.1(1) and I.C.1(2). Additional information requested by the staff was submitted by the Owners Group on January 31, 1981. The staff will complete its review of this additional information before making a final conclusion on the acceptability of these guidelines for implementation for all BWRs.

In FSAR Table 1A-1, the applicant committed to implement a program of emergency operating procedures based on the GE Owners Group guidelines, when they are approved by the staff, to meet the requirements of NUREG-0737 Item I.C.1. In addition to the GE Owners Group efforts, the staff has prepared draft guidelines for long-term upgrading of Emergency Operating Procedures (NUREG-0799) in accordance with NUREG-0660 Item I.C.9. These guidelines, as revised during the resolution of public comments, should be used in the preparation of the River Bend emergency operating procedures. Therefore, the staff does not plan to conduct a pilot monitoring review of selected emergency operating procedures, as described in NUREG-0660 Item I.C.8, for River Bend. However, the staff will review the applicant's submittal for compliance with the final staff guidelines resulting from resolution of public comments of NUREG-0799. The staff review of the applicant's submittal for emergency operating procedures will be completed before an operating license is issued and will be addressed in a supplement to this report.

In accordance with NUREG-0737 Item I.C.7, NSSS vendor review of low power testing, power ascension, and emergency operating procedures is necessary to further verify adequacy of the procedures. FSAR Table 1A-1 and Section 14.2.9

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of the FSAR state that the NSSS vendor, GE, will review the startup test procedures and emergency operating procedures before these procedures are implemented. The startup test procedures encompass the low power testing and the power ascension testing phases. The applicant has committed to ensuring these reviews are complete before fuel load. The staff will confirm that this review is completed before a full power licensa is issued.

13.6 Industrial Security

The applicant has submitted security plans entitled "River Bend Station Physical Security Plan," "River Bend Station Safeguards Contingency Plan," and "River Bend Station Security Training and Qualification Plan," for protection against radiological sabotage. The Security and Contingency plans have been reviewed in accordance with SRP 13.6 (NUREG-0300). ~~The Training and Qualification Plan is currently being reviewed by the staff.~~

As a result of the evaluation ~~the Contingency Plan has been approved~~ certain portions of ~~the Security Plan and Training and Qualification Plan have~~ were identified as requiring additional information to satisfy the requirements of 73.55 and Appendix B of 10 CFR 73. ~~The applicant is expected to make commitments acceptable to the staff that, when formally incorporated into these planning documents, will bring them into compliance with the Commission's regulations in 10 CFR 50 and 73.~~

~~The staff will review the implementation of these plans while it is in progress to ensure conformance with the performance requirements of 10 CFR 73.~~ An ongoing review of the progress of the implementation of these plans will be performed by the Staff to assure conformance with the performance requirements of 10 CFR Part 73. The identification of vital areas and measures to control access to these areas, as described in the plan, may be subject to amendments.

The applicant's security plans are being protected from unauthorized disclosure in accordance with section 73.21 of 10 CFR Part 73

The applicant has filed revisions to the Physical Security, Safeguards Contingency, and Security Training and Qualification Plans which satisfied these requirements. The revised plans, as amended by the condition set forth in the Safeguards portion of the proposed Operating License, are considered to comply with the Commission's regulations contained in 10 CFR Parts 50, 70, and 73.

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14 INITIAL TEST PROGRAM

The testing activities to be performed on safety-related systems and other selected systems at the River Bend station consist of three major phases: preliminary, preoperational, and initial startup.

The preliminary test phase begins as installation and/or construction of the individual structures, systems, and components nears completion. The prime objective of this phase is to verify that construction activities associated with the respective structure, system, or component have been completed and documented. Another function is to verify that the components within a system can be put into operation safely. The testing requirements associated with this phase verify installation integrity and component and system functional characteristics, and ensure that the systems, structures, and components are ready for preoperational testing. These tests, in general, include instrument calibration; electrical continuity and megger checks; pump and motor rotation and vibration checks; hydrostatic testing; cleaning; and flushing.

The preoperational test phase normally begins after preliminary testing of individual components and systems or subsystems is completed. This phase includes the tests required to demonstrate that structures, systems, and components perform satisfactorily in all modes of operation and that they are ready to support fuel loading and initial startup testing. The preoperational test phase ends at the start of fuel loading; however, the possibility exists that some nonsafety-related preoperational tests may be conducted after fuel loading. For structures, systems, and components not tested before fuel load, proper notification and justification will be provided before the start of fuel load. Preoperational tests are performed on any system that is safety related; acceptance tests are performed on all other systems. Plant operating personnel are to obtain hands-on experience during both the preoperational and acceptance testing.

The initial startup test phase of the test program begins with the preparation for fuel load and extends through 100% rated power and warranty demonstrations. The initial startup of testing is divided into five areas: fuel load, open vessel, initial heatup, power ascension, and rated power warranty run. Testing performed during this phase of the program ensures that fuel loading is accomplished in a safe manner, confirms the plant design basis, demonstrates that the plant can withstand anticipated transients and postulated accidents, and ensures that the plant can be safely brought to rated power and sustain power operation.

The staff review of FSAR Chapter 14 concentrated on the administration of the test program and the completeness of the preoperational and startup tests, and the SER-CP was reexamined to determine the principal design criteria for the plant and to identify any specific concerns or unique design features that would warrant special test consideration. Also reviewed were Licensee Event Report Summaries for operating reactors of similar design; Standard Technical Specifications for GE BWRs; NUREG-0660, NUREG-0694, and NUREG-0737; and the startup test reports for other BWRs.

SRP 14 (NUREG-0800) was used to determine the acceptability of the applicant's test program. The staff review covered several aspects of the initial test program including the following major considerations:

(1) Test procedures have been developed using input from the NSSS vendor, the architect-engineer, the applicant's engineering staff, and other equipment suppliers and contractors as needed. Operating experiences at similar plants were factored into the development of the test procedures.

(2) Tests are being conducted using approved test procedures. Administrative controls cover the completion of test prerequisites, the completion of necessary data sheets and other documentation, and the review and approval of modifications to test procedures. The applicant stated that administrative procedures also cover implementation of modifications or repair requirements identified by the tests as necessary and any necessary retesting.

(3) The results of each test are reviewed for technical adequacy and completeness by personnel including the NSSS vendor and architect-engineer, as appropriate. Preoperational test results are reviewed before fuel load, and the startup test results from each activity or power level will be reviewed before proceeding to the next activity or power level.

(4) Normal plant operating and emergency procedures are used in performing the initial test program, thereby verifying the correctness of the procedures to the extent practicable.

(5) The applicant's schedule for conducting the initial test program allows adequate time to conduct all preoperational and startup tests. Preoperational test procedures will be available for NRC Regional Administrator review at least 60 days before scheduled implementation. Startup test procedures will be available for review not less than 60 days before the scheduled fuel load date.

(6) The abstract of each test procedure in FSAR Chapter 14 was reviewed. The staff verified that there were test abstracts for those structures, systems, components, and design features that: (a) will be used for shutdown and cooldown of the reactor under normal plant conditions and for maintaining the reactor in a safe condition for an extended shutdown period; (b) will be used for shutdown and cooldown of the reactor under transient (infrequent or moderately frequent events) conditions and postulated accident conditions and for maintaining the reactor in a safe condition for an extended shutdown period following such conditions; (c) will be used for establishing conformance with safety limits or limiting conditions for operation that will be included in the facility Technical Specifications; (d) are classified as engineered safety features or will be relied on to support or ensure the operations of engineered safety features within design limits; (e) are assumed to function or for which credit is taken in the accident analysis of the facility, as described in the FSAR; or (f) will be used to process, store, control, or limit the release of radioactive materials.

(7) The test objectives, prerequisites, test methods, and acceptance criteria for each test abstract were reviewed in sufficient detail to establish that the functional adequacy of the structures, systems, components, and design features will be demonstrated.

(8) The test program's conformance with applicable RGs was reviewed. In addition to RG 1.68, Revision 2, the review included RGs 1.20, Revision 2; 1.30, (Safety Guide 30), Revision 0; 1.37, Revision 0; 1.41, Revision 0; 1.52, Revision 2; 1.68.1, Revision 0; 1.68.2, Revision 1; 1.80, Revision 0; 1.95, Revision 1; 1.108, Revision 1; 1.116, Revision 0-R; 1.128, Revision 1; 1.139, Draft 1, Revision 0; and 1.140, Revision 0.

The applicant made a number of changes to the initial test program because of staff comments. Examples of these changes include

(1) Testing was added to ensure that components move to the correct fail-safe position on loss of instrument air.

(2) Acceptance criteria for the major preoperational and startup tests were expanded to ensure that quantifiable and referenceable data were available to establish that the actual test objectives have been attained.

(3) Testing was added to demonstrate operability of the fuel rod cooling and cleanup system.

(4) Testing was added to verify the operability of the battery exhaust fans.

(5) Testing was added to ensure that containment recirculation fans will be operational during post-LOCA conditions.

(6) Test descriptions were modified to more fully demonstrate the ability of the emergency diesel generators and their support equipment to provide reliable emergency power.

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(7) Testing was added to demonstrate that, for hot containment penetrations where coolers are not used, concrete temperatures do not exceed design limits.

(8) Testing was added to demonstrate the capability of the emergency lighting system to operate properly.

(9) Dynamic and static load testing was added to verify that the fuel handling and containment building polar crane are capable of performing their intended functions safely.

(10) Testing was added to demonstrate that required air flows were achieveable for the following ventilation systems: auxiliary building, radwaste building, fuel building, and turbine building.

The following items are still unresolved:

(1) The applicant has not furnished technical responses to staff questions related to conformance with RGs 1.68 (Question 640.1); 1.68.2 (Question 640.2); and 1.95 (Question 640.3).

(2) The applicant has stated that vibration level or piping movement limit criteria for the residual heat removal and feedwater systems have not yet been established. Those criteria will be required before the responses to Questions 640.6 and 640.39 can be considered acceptable.

(3) The acceptance criteria for the low pressure core spray and high pressure core spray preoperational tests must be modified to reference or define in greater detail what constitutes an acceptable core spray pattern (Question 640.6).

(4) A test description is needed for the post-accident monitoring instrumentation. The instrumentation to be used is currently being developed; therefore the description is to be provided later (Question 640.8, 1.j (22)).

(5) The fuel handling and vessel servicing equipment preoperational test must be modified to ensure that the associated lifting and rigging equipment is static tested at 125% of rated load and operationally tested at 100% of rated load (Question 640.13).

(6) The control building HVAC preoperational test must be expanded to include testing of the chlorine monitors (Question 640.18).

(7) The Class 1E 125-volt system preoperational test must be modified to demonstrate that the loads will function as necessary to ensure plant safety at a battery terminal voltage equal to the acceptance criterion that has been established for minimum battery terminal voltage for the discharge load test (Question 640.25).

(8) Additional testing described in Appendix E of the BWR Owners Group letter from D. B. Waters to D. G. Eisenhut, dated February 4, 1981 (regarding NUREG-0737 I.G.1) should be added to FSAR Chapter 14.

Based on its review, the staff has concluded that if acceptable responses to the above-listed unresolved items are made, the initial plant test program is acceptable and meets the requirements of: 10 CFR 30.53 relating to initial testing of radiation detection and monitoring instruments; 10 CFR 50.34(b)(6)(iii) requiring inclusion of plans for preoperational testing and initial operations in the FSAR; 10 CFR 50, Appendix B, Section XI requiring a test program to ensure that all testing required to demonstrate that structures, systems, and components will perform satisfactorily in service is identified and performed in accordance with written test procedures that incorporate the requirements and acceptance limits contained in applicable design documents; and 10 CFR 50, Appendix J, Section III.A.4 requiring a preoperational measurement of the overall integrated leak-tightness of the primary reactor containment building under specified pressure conditions.

The staff has further concluded that if acceptable responses to the outstanding items are made, the initial test program described in the River Bend application will meet the acceptance criteria of SRP 14.2, and successful completion of the

program will demonstrate the functional adequacy of plant structures, systems, and components.

This review and evaluation was performed with the assistance of Battelle Pacific Northwest Laboratories personnel.

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15 TRANSIENT AND ACCIDENT ANALYSIS

Introduction

Two groups of design basis events are evaluated in this section: anticipated operational occurrences and accidents. In order for the analysis of events in either group to be acceptable, it is required that a conservative model of the reactor be used, and that all appropriate systems whose operation (or postulated misoperation) would affect the event be included. Anticipated operational occurrences are expected to occur during the life of the power plant and are analyzed to assure that they will not cause damage to either the fuel or to the reactor coolant pressure boundary and to assure that the radiological dose is maintained within Title 10, Code of Federal Regulations, Part 20 (10 CFR 20) guidelines. Design basis accidents are not expected to occur, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. They are analyzed to determine the extent of fuel damage expected and to assure that reactor coolant pressure boundary damage, beyond that assumed initially by the design basis accident, will not occur, and that the radiological dose is maintained within 10 CFR 100 guidelines.

For loss-of-coolant accidents, the acceptance criteria for the emergency core cooling system specified in Title 10, Code of Federal Regulations, Part 50.46 are: (1) the peak cladding temperature must remain below 2200 degrees Fahrenheit; (2) maximum cladding oxidation must nowhere exceed 17 percent of the total cladding thickness before oxidation; (3) total hydrogen generation must not exceed one percent of the hypothetical amount that would be generated if all the metal in the cladding cylinders, excluding the cladding surrounding the plenum volume, were to react; (4) the core must be maintained in a coolable geometry; (5) calculated core temperatures after successful initial operation of the emergency core cooling system shall be maintained acceptably low and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

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15.1 Anticipated Operational Occurrences

Anticipated operational occurrences are those transients resulting from single equipment failures or single operator errors that might be expected to occur during normal or planned modes of plant operation. The acceptance criteria for these transients are based on General Design Criteria 10, 15, and 20. Criterion 10 specifies that the reactor core and associated control and instrumentation systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. Criterion 15 specifies that sufficient margin shall be included to assure that design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences. Criterion 20 specifies that a protection system be provided that automatically initiates appropriate systems to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation including anticipated operational occurrences. Specific acceptance criteria (Standard Review Plan) for the moderate frequency transients are:

(1) Pressures in the reactor coolant and main steam systems should be maintained below 110% of the design values according to ASME Boiler and Pressure Vessel Code, Section III, Article NB-7000, "Overpressure Protection." For RBS, which has a design pressure of 1250 pounds per square inch gauge, the pressure should not exceed 1375 pounds per square inch gauge during any anticipated operational occurrence.

(2) Fuel cladding integrity should be maintained by ensuring that the reactor core is designed to operate with appropriate margin to specified limits during any conditions of normal operation, including the effects of anticipated operational occurrences. For boiling water reactors, the minimum value of the critical power ratio reached during the transient should be such that 99.9% of the fuel rods in the core would not be expected to experience boiling transition during core wide transients. This limiting value of the minimum critical power ratio, called the safety limit, will vary for different plants and/or product lines. For RBS, the value is 1.06 (see Section 4.4).

(3) An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

(4) An incident of moderate frequency in combination with any single active component failure, or single operator error, should not result in loss of function of any fission product barrier other than the fuel cladding. A limited number of fuel rod cladding perforations is acceptable.

The applicant used the following conservative assumptions with respect to initial power, scram reactivity, reactivity coefficients, and power profiles in the analyses:

(1) Initial power 104.4 percent of rated power (corresponding steam flow rate 105 percent of rated steam flow rate).

(2) Scram reactivity characteristics accounting for end-of-cycle conditions that result in the most conservative effects. The slowest allowable control rod scram motion is assumed with a scram worth shape for all the control rods fully withdrawn at the end-of-cycle.

(3) Scram reactivity calculations incorporating a 20% safety conservatism factor (i.e., rod worth of 0.80 x end-of-cycle rod worth).

This includes the effect of a stuck control rod as required by General Design Criterion 26.

(4) Core burnup selected to yield the most limiting expected combination of moderator temperature coefficient, void coefficient, Doppler coefficient, axial power profile, and radial power distribution.

The transients analyzed involved the following reactor scrams:

(1) Reactor vessel high pressure,

(2) Reactor vessel low water level,

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(3) Turbine stop valve closure,

(4) Turbine control valve fast closure,

(5) Main steam line isolation valve closure,

(6) Neutron monitoring system scram.

Time delays to trip for each scram signal were included in the analyses.

The transient events were categorized in terms of the following system parameter variations:

1. Decrease in Core Coolant Temperature

   Transients analyzed in this group included loss of feedwater heating, feedwater control failure to maximum demand, pressure regulator failure in the open direction, inadvertent opening of a safety/relief valve and inadvertent residual heat removal shutdown cooling operation at reduced pressure and temperature conditions of the reactor.

2. Increase in Reactor Pressure

   Transients in this group included generator load rejection and turbine trip with and without turbine bypass, inadvertent MSIV closure, loss of condenser vacuum, loss of auxiliary power transformer, loss of all grid connections and loss of all feedwater flow. Failure of residual heat removal shutdown cooling was considered for conditions of reduced pressure and temperature of the reactor.

3. Decrease in Reactor Coolant Flow Rate

   Transients in this group included trip of one or both recirculation pump motors and recirculation flow control failure to decrease flow.

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4. Reactivity and Power Distribution Anomalies

Transients in this group included rod withdrawal error at power, abnormal startup of an idle recirculation loop and recirculation flow controller failure with increasing flow.

5. Increase in Reactor Coolant Inventory

The transient analyzed was inadvertent startup of the high pressure core spray pump (feedwater flow control failure to maximum demand was covered in Category 1).

6. Decrease in Reactor Coolant Inventory

Inadvertant opening of a safety relief valve (covered in Category 1).

The analysis of continuous control rod withdrawal during power operation was made with the three-dimensional BWR simulator code described in NEDO-20953, "Three-Dimensional BWR Core Simulator," which has been approved by NRC.

The analyses of the other abnormal operational transients were performed using the REDY computer code described in General Electric Topical Report NEDO-10802, "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor."

The General Electric Company developed a new computer code called ODYN to more adequately model overpressurization transients. The ODYN code has been reviewed by the staff and found acceptable. (Safety Evaluation for the General Electric Topical Report-Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors, NEDO-24154 and NEDE-24154-P, Volumes I, II, III, June 1980).

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The applicant analyzed the following transients using ODYN:

(1) For Thermal Limit Evaluation

(a) Feedwater controller failure - maximum demand,
(b) Generator load rejection without bypass, and
(c) Turbine trip without bypass.

(2) For American Society of Mechanical Engineers Overpressure Protection

(a) Main steam isolation valve closure with position switch scram failure (main steam isolation valve closure with flux scram).
(b) Loss of condenser vacuum.
(c) Loss of all grid connections.

The transient resulting in the highest system pressure was a generator load rejection without turbine bypass which resulted in a peak system pressure about 144 pounds per square inch below the allowable maximum pressure of 1375 pounds per square inch gauge. Nota, however, that a further degraded transient is analysed for the American Society of Mechanical Engineers overpressure protection report. This is an inadvertant main steam isolation valve closure with postulated failure of the position switch scram (see Section 5.2.2).

For transients categorized under Decrease in Core Coolant Temperature, the most severe transient is the loss of feedwater heating event with manual flow control. The resultant minimum critical power ratio reached is 1.06 and the peak vessel presssure is 276 pounds per square inch below the ASME code limit. For this transient, the applicants assumed a 100 degree Fahrenheit drop in feedwater temperature. The applicant has assumed operation of a high simulated thermal power trip (STPT) to initiate scram for this event. STPT is the primary protection system trip in mitigating the consequences of this event. STPT is a safety grade system and is designed to be single failure proof. We requested additional information concerning the STPT system. We will report on this item in a supplement to the SER.

Inadvertent safety/relief valve opening causes a decrease in reactor coolant inventory and results in a mild depressurization event which has only a slight effect on fuel thermal margins. Changes in surface heat flux are calculated to be negligible indicating an insignificant change in minimum critical power ratio. Thus, the transient is found to be acceptable. The effect of inadvertent safety/relief valve opening on suppression pool temperature is treated in Section 6.2.

The applicants were asked to justify that operation with partial feedwater heating to extend the cycle beyond the normal end of cycle condition would not result in a more limiting change in minimum critical power ratio than that obtained using the assumption of normal feedwater heating. We require that analyses be provided prior to operation in this mode if a decision is made to operate in this mode. Until such analyses are provided, we will condition the license from operation in this mode.

For transients categorized under Decrease in Reactor Coolant System Flow Rate, the most severe transient is that resulting from simultaneous trip of both recirculation pump motors. As the pumps coast down, the core void fraction increases, causing level swell in the reactor vessel and a decrease in neutron flux. Turbine trip occurs because of high water level. The minimum critical power ratio remains well above the safety limit and there is a small increase in reactor pressure.

Increased recirculation flow because of flow control failure or startup of an idle recirculation pump would result in reactivity anomalies. These events are not limiting transients and neither primary pressure boundary nor fuel damage criteria are exceeded. The transient which could cause unplanned addition to coolant inventory is the inadvertent actuation of the high pressure core spray system. The high pressure core spray system actuation has a small effect, because its flow is small compared to the recirculation flow. The transient has little effect on fuel thermal margins and on reactor system pressure. (The discussion of the rod withdrawal at power should be inserted here.)*

*LPM to add

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Of the transients listed under Increase in Reactor Pressure, the limiting transients are generator load rejection without bypass and turbine trip without bypass which result in minimum critical power ratios of 1.07 and 1.08, respectively, both above the MCPR safety limit and therefore acceptable.

The applicants have proposed, however, that the generator load rejection without bypass and the turbine trip without bypass transients be classified as infrequent events. The reclassification of these events has been under review by the staff and has not been approved. We require that these events be categorized as moderate frequency events and that the operating limit minimum critical power ratio be adjusted as necessary so that the safety limit 1.06 is not violated by these transients.

In analyzing anticipated operational transients, the applicants have taken credit for plant operating equipment which is not normally reviewed by us because it is not considered essential for safety. We have discussed the application of this equipment generically with General Electric. Based on these discussions, it is our understanding that the most limiting transient, that takes credit for this equipment is the excess feedwater event. Further, it is our understanding that the only plant operating equipment that plays a significant role in mitigating this event (excess feedwater) is the turbine bypass system and the level 8 high water level trip (closes turbine stop valves). In order to assure an acceptable level of performance for RBS, our position is that this equipment be identified in the plant Technical Specifications with regard to availability, setpoints, and surveillance testing. The results of our review of the Technical Specifications with respect to the level 8 trip and turbine bypass system will be addressed in Chapter 16.0.

15.2 Accidents

The applicants analyzed a pump shaft seizure and pump shaft break accident. Reactor scram is sufficient to preclude violating the safety limit minimum critical power ratio (1.06) and, therefore, no fuel damage occurs. The reactor vessel presssure is maintained below the specified limit throughout the event. However, the analyses included the use of non-safety grade equipment, in particular, the turbine bypass system. We require that the applicant provide confirmation that precludes the use of non-safety grade equipment during these accidents will not result in violation of the safety limit minimum critical power ratio.

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15.3 Radiological Consequences of Accidents

In order to demonstrate the adequacy of the plant's engineered safety features, the applicant calculated the offsite consequences which could result from the occurrence of each of several design-basis accidents and presented the results of these computations in the FSAR. The staff has independently performed similar calculations to confirm the effectiveness of these engineered safety features.

The computed doses of these postulated design-basis accidents are listed in Table 15.1. Where these doses are significantly dependent on plant conditions at the time of the accident, or arise through multiple leakage paths, separate doses for extreme conditions or separate dose contributions from individual leakage paths have been listed. The computed doses in Table 15.1 are in a form which allows direct comparison with the dose guidelines of 10 CFR Part 100.11; they are expressed as rem of thyroid and whole body exposure for a 2-hour period at the exclusion area boundary (EAB, 914 m from the reactor centerline) and for a 30-day period at the boundary of the low population zone (LPZ, 4023 m from the reactor centerline). As described in Section 2 of this report, the EAB and LPZ have been found to conform to the definitions in 10 CFR Part 100.3(a) and (b).

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As discussed in section 2.3.4 of this report, Regulatory Guide 1.145 specifies modifications to be made to the treatment of atmospheric transport and dispersal parameters contained in Regulatory Guides 1.3 & 1.25. In all other respects, staff review methods and computations are in full accord with the Regulatory Guides and Standard Review Plan Sections (NUREG-0800) referenced in each of the following accident descriptions.

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15.3.1 Loss-of-Coolant Accident (LOCA)

In the River Bend Station FSAR, the applicants presented an evaluation of the radiological consequences of a design basis LOCA which showed that the distances to the exclusion area and low population zone boundaries are sufficiently large to meet the guidelines set forth in 10 CFR Part 100, paragraphs 11(a)(1) and (2). The staff's review of the River Bend Station design has confirmed the applicant's findings while considering a more diverse range of possible radioactive leakages than did the applicants.

The staff's postulated LOCA for the River Bend Station followed methods outlined in the Appendices to Standard Review Plan 15.6.5. In applying these methods, the staff viewed the primary containment as being effectively surrounded by a secondary containment system capable of intercepting any primary containment leakage and filtering it prior to release to the environment. This view was adopted following the conclusion, as discussed in Section 6.5, that the standby gas treatment, fuel building charcoal filtration, and main steam line positive leakage control systems were adequately designed to prevent any leakage from bypassing the secondary containment.

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The paths for radioisotope transport to the environment evaluated by the staff are the leakage of contaminated air from the primary containment atmosphere, and the leakage of water containing dissolved and suspended contamination from pipes and ESF equipment within the secondary containment. Both leakages result in contamination of the secondary containment atmosphere, and the consequences of their filtered exhaust to the environment have been listed separately in Table 15.1 as "containment" and "ESF" leakage, respectively.

The numerical values of other assumptions used to compute the LOCA doses are listed in Table 15.2, and have been derived using Regulatory Positions 1.a through 1.f of Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors." Primary containment liquid capable of dissolving core iodine has been taken as the sum of the water inventories of the reactor vessel and the suppression pool, and 10 minutes of ECCS injection from the condensate storage tank. The leak rate chosen for ESF equipment is that used by the staff in past evaluations, and represents the largest flow which could plausibly exist over the duration of the accident without discovery and isolation of this source. The applicant did not provide any estimate for this source. Whole body doses resulting from ESF leakage have been computed by assuming

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immediate release to the environment of xenon produced by the decay of the cumulative iodine leaked into the secondar containment.

The LOCA doses in Table 15.1 conform to Regulatory Guide 1.3 and Standard Review Plan Section 15.6.5 used in their computation by conservatively estimating potential consequences. Most of the 0-2 hour doses result from the assumption of instantaneous mixing within the primary containment, while most of the thyroid doses result from the assumption that iodine is present in a volatile, though chemically unstable, form throughout the course of the accident. The 30-day whole body doses are also conservative, in that they assume a large leak to persist without remedial action being taken to control the large volume of accumulated leakage.

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15.3.2 Control Rod Drop Accident

For the postulated control rod drop accident, it is assumed that a bottom entry control rod has been fully inserted and becomes stuck in this position, unknown to the reactor operator. The control rod drive is then assumed to be uncoupled from the control rod and fully withdrawn. Subsequently, the control rod becomes unstuck and falls from the core, inserting an amount of reactivity corresponding to its reactivity worth at a rate consistent with the maximum rate at which a control rod can drop.

In evaluating the radiological consequences of this accident, the applicant assumed that the most reactive control rod assembly is dropped out of the core, causing 770 rods (2.1 percent of the core) to fail.

River Bend Station is equiped with a rod sequence control system which limits the amount of positive reactivity increase due to sudden withdrawal, such that no fuel would be expected to fail as a result of the postulated rod drop accident. The fuel damage assumed is intended to conservatively encompass unexpected misloaded fuel assemblies.

The staff used Regulatory Guide 1.77 for the amount of activity

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accumulated in the fuel-clad gap and used the assumptions and methods specified in the Appendix to SRP Section 15.4.9 to compute the offsite radiological consequences of this accident. These consequences are listed in Table 15.1, and the staff assumptions are listed in Table 15.3. The resulting doses shown in Table 15.1 are well within the 10 CFR Part 100 guidelines and, therefore, meet the acceptance criteria of the Appendix to SRP Section 15.4.9. Based upon our findings, we conclude that the River Bend Station is effectively designed to control any release of radioactive fission products following a control rod drop accident.

15.3.3 Fuel Handling Accident

A postulated fuel handling accident was evaluated using assumptions contained in Regulatory Positions C.1.a through k of Regulatory Guide 1.25 and the procedures specified in SRP Section 15.7.4 (NUREG-0800). Assumptions having numerical values are listed in Table 15.4. In addition, the specified assumptions postulate a single dropped fuel assembly, the kinetic energy of which is expended with perfect mechanical efficiency in breaking open the maximum possible number of fuel rods. Instantaneous release of noble gases and radioiodine vapor from the gaps of the broken rods occurs as gas bubbles passing up through the water covering the fuel, with all

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radioactivity reaching the building atmosphere (either containment or fuel handling building) being exhausted to the environment within two hours by the engineered safety feature filtered exhaust systems described in Chapter 6.

The offsite doses computed using these assumptions are listed in Table 15.1, and are well within the guideline dose limits of 10 CFR Part 100. The staff, therefore, concludes that the plant features designed to mitigate the consequences of fuel handling accidents are adequate, and meet the criteria in SRP Section 15.7.4.

In the staff's review of the spent fuel cask drop accident, it was noted that the potential vertical drop distances onto hard surfaces were less than 30 feet and the cask is not permitted to travel over spent fuel or safety-related equipment. Because the most severe drop is within the designed strengths of the barriers to fission product release, the radiological consequences of a spent fuel cask drop accident have not been evaluated (as specified in SRP Section 15.7.5).

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15.3.4 Main Steam Line Break Accident

For this postulated accident, it is assumed that one of the four main steam lines breaks instantaneously along its circumference at a location downstream of the outermost isolation valve. The main steam isolation valves (MSIV's) start to close at 0.5 seconds on a high flow signal and are fully closed at 5.5 seconds. The flow restrictor on each steam line limits the mass loss from both ends of the break, and the subsequent full closure of the MSIV's terminates the mass loss for this event. The fuel damage is predicted to occur as a result of this accident. The applicant has estimated a total of 80,562 pounds of integrated mass loss from the break. In calculating the radiological consequences the staff, however, followed the guidance of SRP Section 15.6.4 in which 140,000 pounds of primary coolant is assumed to be released prior to MSIV closure. The staff also followed Regulatory Guide 1.5, assuming that all the iodine and noble gases in this leaked coolant are released to the environment without credit for plateout or holdup, and that the release occurs at a height of 30 meters with fumigation meteorological conditions.

Two cases of this accident have been analyzed. The first assumes that the break occurs when the primary coolant iodine concentration is at the Technical Specification equilibrium

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limit of 0.2 microcuries/gm dose equivalent I-131 (DEI). The second assumes that the break occurs when the primary coolant iodine concentration is at the maximum permissible limit or "spike limit" of 4.0 microcuries/gm DEI. These limits are specified in NUREG-0123, "Standard Technical Specifications for General Electric Boiling Water Reactors."

Our asumptions are listed in Table 15.5. The calculated doses are listed in Table 15.1. For the first case, the doses are only a small fraction of the 10 CFR Part 100 guideline values. For the second case, doses are well within the limits of 10 CFR Part 100. These results meet the acceptance criteria of SRP Section 15.6.4 and, therefore, the staff concludes that the design provides appropriate protection against such an accident.

15.3.5 Small Line Break Accident

This section will be supplied when the outstanding question (450.5) is responded to by the applicant.

TABLE 15.1

RADIOLOGICAL CONSEQUENCES OF SELECTED DESIGN BASIS ACCIDENTS

Postulated Accident	0-2 Hour Doses EAB*, Rem Thyroid	0-2 Hour Doses EAB*, Rem Whole Body	0-30 Day Doses LPZ**, Rem Thyroid	0-30 Day Doses LPZ**, Rem Whole Body
Loss of coolant				
containment leakage	50.	16	38.	4.
ESF leakage	4.	1	4.	2.
Total	54.	17	41.	6.
Control rod drop	2.	0.2	1.	0.05
Steam Line Break				
Case I	13.	<0.1	0.3	<0.1
Case II	26.	<0.1	7.	<0.1
Loss-of-coolant accident doses during intervals within first 30 days at LPZ				
0-2 hours			4.1	1.3
2-8 hours			7.0	2.1
8-24 hours			4.4	2.0
1-4 days			12.3	.8
4-30 days			14.1	.3

* exclusion area boundary (EAB)
** low population zone (LPZ)

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TABLE 15.2

ASSUMPTIONS USED TO CALCULATE LOCA DOSES

Power level (megawatts thermal)	3039
Equilibrium core life (years)	3
Core fractions released (percent)	
to primary containment atmosphere:	
Noble gases	100
Iodines	25
to primary containment liquids:	
Iodines	50
Total primary containment liquids (cubic meters)	4488
Containment atmosphere leak rate (%/day)	0.26
Containment liquid leak rate (gpm)	1
Evolved into secondary containment atmosphere from liquid releases (percent)	
Xenon	100
Iodine	10
Iodine ESF filter efficiency (percent)	99

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TABLE 15.3

ASSUMPTIONS USED FOR THE ROD DROP ACCIDENT

Power level (megawatts thermal)	3039
Fraction of failed fuel (percent)	2.1
Peaking factor of failed fuel	1.5
Percent of failed fuel iodine and noble gases released to coolant	10
Percent of coolant activity reaching condenser	
Iodines	10
Noble gases	100
Activity released to the environment from the condenser (percent/day)	1

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TABLE 15.4

ASSUMPTIONS USED IN THE FUEL HANDLING ACCIDENT

Power level (megawatts thermal)	3039
Fuel rods ruptured	123
Delay between shutdown and accident (hours)	24
Fraction of core fission products in ruptured rods	0.0034
Peaking factor	1.5
Release from rods (percent):	
Krypton-85	30
Other noble gases	10
Iodines	10
Percent iodine removed by passage through spent fuel pool	99
ESF exhaust filter iodine efficiency (percent)	99

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TABLE 15.5

ASSUMPTIONS USED IN THE STEAM LINE BREAK ACCIDENT

Power level (megawatts thermal)	3039
Fuel damage	none
Isolation valve closure time (seconds)	5.5
Total primary coolant released (pounds)	140,000
Equilibrium primary coolant concentration (microcuries/gm-dose equivalent I-131)	0.2
Maximum primary coolant concentration (microcuries/gm-dose equivalent I-131)	4.0
X/Q's at 30 meters height (sec/cubic meters)	
0-2 hour @ EAB (Fumigation)	2.0 E-4*
0-8 hour @ LPZ (Fumigation)	5.1 E-5*

* E-4 equals 10

15.4 Rod Wd. Rdrwl Events

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15.4.1 Inadvertent Rod Withdrawal at Low Power

The applicant has examined the design of the rod control system to ascertain if a single failure can lead to the uncontrolled withdrawal of a control rod during refueling and during startup and low power operation. During refueling, interlocks ensure that all control rods are inserted while fuel is being handled over the core. When no fuel is being handled, a maximum of one rod may be withdrawn. However, the control system is designed (see Section 4.3.2) so that the core is subcritical with the highest worth rod withdrawn. The removal (from the top) of a control rod is not physically possible without removing the four fuel assemblies that surround the rod.

The uncontrolled withdrawal of a rod during reactor startup is prevented by the rod pattern control system function of the rod control and information system (RCIS). This system is single-failure proof and enforces the banked position withdrawal sequence. Thus rod withdrawals other than those permitted in normal operation will be precluded.

The possibilities for single failures of the reactor control system that could result in the uncontrolled withdrawal of control rods under low power startup conditions have been reviewed. The RCIS prevents the withdrawal of control rods in any but the design sequence. Thus, the consequences of any rod withdrawal are within the bounds of those for normal operation.

The staff, therefore, finds that GDC 20, which requires that protection be automatically initiated, and GDC 25, which requires that a single failure of not result in violation of specified fuel design the protection system ~~acts to violation of specified fuel design limits~~ have been satisfied.

~~15~~.4.2 Rod Withdrawal Error at Power

The "at power" rod withdrawal error transient occurs when the operator makes a procedural error and withdraws a rod continuously. Although procedures call for moving rods in small increments (a few notches) in the power range and prescribe the sequence of such withdrawals, no hardware enforces this sequence.

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Instead, the rod withdrawal limiter mode of the RCIS acts to limit the amount by which a control rod or control rod gang may be moved in a single action.

A statistical generic analysis of this event has been performed for BWR/6 reactors and is reported in Appendix 15B of GESSAR II. It is also reported in Appendix 15B of the FSAR for the Perry nuclear plant (Cleveland Electric Illuminating Co.). (June, 1981)

For the statistical analysis, anticipated operating states (including rod patterns) are developed for several cycles of a BWR/6. These states are examined to establish the anticipated MCPR values as a function of core power.

A Technical Specification curve of allowable initial MCPR (IMCPR) as a function of power is then defined so that there is margin between this limit and the expected plant performance. The change in MCPR (ΔMCPR) is then calculated for ganged rod withdrawal events that are initiated from reactor states (rod patterns) used to establish the IMCPR curve (analyses have shown that ganged withdrawals are always more limiting). The ΔMCPR values are then expressed as a function of withdrawal distance and core power, and withdrawal distances are determined as a function of initial power, so that there is a 95% probability with 95% confidence that the safety limit MCPR for the core will not be violated. The curve of withdrawal limit as a function of initial power is then divided into two segments--above 70% of full power and between 20% and 70% of full power--and constant values of the withdrawal distance defined for each segment. The rod withdrawal limiter is then designed to enforce these limits, which are 12 in. above 70% of full power and 24 in. between 20% and 70% of full power. Below 20% of full power, rod motions are constrained by the rod pattern control system mode of the RCIS.

The staff has reviewed the analysis of the rod withdrawal event at power and concludes that it is acceptable. This conclusion is based on the following:

(1) The BWR simulator code, which has been reviewed and approved by the staff, was used to perform the analyses.

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(2) The data base used for the analyses covered the range of BWR/6 sizes and exposure cycles.

(3) The allowed withdrawal distances are conservative with respect to the calculated results.

(4) The Technical Specification values of the permitted IMCPR as a function of core power are conservative with respect to the calculated values.

It should be noted that this analysis is not applicable to a control cell loading strategy or high-energy/high-discharge exposure cycles. Use of such cycles in the River Bend reactor will require that compliance checks be performed and documented to demonstrate the applicability of the generic analysis.

The possibilities for single failures of the reactor control system that could result in uncontrolled withdrawal of control rods beyond normal limits under power operation conditions have been reviewed. The scope of the review has included investigations of possible initial conditions and the range of reactivity insertions, the course of the resulting transients, and the instrumentation response to the transient. The methods used to determine the peak fuel rod response and the input into that analysis--such as power distributions, rod reactivities, and reactivity feedback effects of moderator and fuel temperature changes--have been examined.

The staff concludes that the applicant has met the requirements of GDC 10 that the specified acceptable fuel design limits are not exceeded, GDC 20 that the reactivity control systems are automatically initiated so that specified acceptable fuel design limits are not exceeded, and GDC 25 that single malfunctions in the reactivity control systems will not cause the specified acceptable fuel design limits to be exceeded. These requirements have been met by comparing the resulting extreme operating conditions and response for the fuel (fuel duty) with the acceptance criteria for fuel damage (for example, critical power ratio, fuel temperatures, and clad strain limits should not be exceeded, to ensure that fuel rod failure will be precluded for this event. The basis for acceptance in the staff review is that the applicant's analysis

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of maximum transients for single-error control rod malfunctions have been confirmed, that the analytical methods and input data are reasonably conservative, and that specified acceptable fuel design limits will not be exceeded.

15.4.3 ~~Inadvertent Loading~~ Operation of a Fuel Assembly ~~in~~ in an Improper Position - Fuel Misloading Event

Two sorts of fuel misloading events may be considered: misorientation of a fuel assembly in its proper location and loading a fuel assembly into an improper location. The first of these events has trivial consequences for a C-lattice in the first cycle because the assembly fuel design is symmetric with respect to rotation. The slight tilt in the assembly caused by the misorientation has a negligible effect on the thermal-hydraulic performance of the fuel in the first cycle and tends to improve that performance in succeeding cycles.

The initial core consists of three bundle types with average enrichments that are high, medium, or low with correspondingly different gadolinia concentrations. The fuel bundle loading error involves interchanging a bundle of one enrichment with another bundle of a different enrichment. The following fuel loading errors can be conceived for an initial core:

(1) A high-enriched bundle is misloaded into a low-enriched bundle location.

(2) A medium-enriched bundle is misloaded into a low-enriched bundle location.

(3) A low-enriched bundle is misloaded into a high-enriched bundle location.

(4) A low-enriched bundle is misloaded into a medium-enriched bundle location.

(5) A medium-enriched bundle is misloaded into a high-enriched bundle location.

(6) A high-enriched bundle is misloaded into a medium-enriched bundle location.

Because all low-enriched bundles are located on the core periphery, the two possible fuel loading errors consisting of the misloading of high- or medium-enriched bundles into a low-enriched bundle location (Types 1 and 2 above) are not significant. In these cases, the higher reactivity bundles are moved to a region of lower importance, resulting in an overall improvement in performance.

Of the remaining types of misloadings, the one with the greatest consequences is the last. The other types consist of misloading a bundle into a location intended for a higher enrichment. In this case, if the lower enrichment bundle is loaded next to a low power range monitor (LPRM) string, the detector response will be different from that in a properly loaded four-bundle array. Two effects are present--a reduction in detector response because of the low-enrichment bundle and an increase because of the softened neutron spectrum resulting from the decreased thermal absorption present (because of the lower enrichment). The latter effect dominates and the detector reads too high--a conservative direction.

For the last type of event--loading a high-enrichment bundle into a medium-enrichment location--the initial effect is small because there are different amounts of gadolinium in two bundles so that their reactivities are similar. However, as the gadolinium burns out, the high-enrichment bundle becomes increasingly more reactive than the medium-enrichment bundle. If, at the end of cycle, one of the mirror four-bundle arrays (assumed to be instrumented) is placed on operating limits, the array containing the misloaded bundle will exceed limits.

The consequences of this event have been evaluated using the BWR simulator code, which has been reviewed and approved by the staff. The results of the analysis show that a change in critical power ratio of 0.10 and a change in linear heat generation rate of 1.3 kW/ft occur for this event. These changes are well below the operating margin to fuel thermal limits. The staff concludes that the analysis of the fuel misloading event is acceptable.

The staff has evaluated the consequences of a spectrum of postulated fuel loading errors and concludes that the analyses provided by the applicant have

shown for each case considered that either the error is detectable by the available instrumentation (and hence remediable) or the error is undetectable but the offsite consequences of any fuel rod failures are a small fraction of 10 CFR 100 guidelines.

The staff concludes that the applicant has met the requirements of GDC 13 with respect to providing adequate provisions to minimize the potential of a misloaded fuel assembly going undetected and has met 10 CFR 100 with respect to mitigating the consequences of reactor operations with a misloaded fuel assembly. These requirements have been met by providing acceptable procedures and design features that will minimize the likelihood of loading fuel in a location other than its designated place.

15.4.4 Rod Drop Accident

The postulated rod drop accident occurs when a rod which has been stuck in the upper portion of the core becomes disconnected from the rod drive, the drive is subsequently withdrawn, and the rod becomes unstuck and falls rapidly onto the drive. This results in a power excursion that could, under certain circumstances, result in local fuel damage.

The consequences of the rod drop accident depend chiefly on the amount of reactivity inserted into the core by the dropping rod and on the initial thermal hydraulic conditions of the core. Dependence on rod drop speed, Doppler feedback coefficient, the shape of the scram curve, and the scram speed are less pronounced. The analysis of the rod drop accident has been performed on a generic basis by GE and is reported in NEDO-10527, "Rod Drop Accident Analysis for Large Boiling Water Reactors," and Supplements 1 and 2 to that report. The calculation is performed under the following conservative assumptions:

(1) No thermal-hydraulic feedback is assumed.

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(2) The least negative Doppler coefficient that is anticipated is used.

(3) The rod drop speed is assumed to be that measured for the rod design plus three standard deviations.

(4) The scram speed is the Technical Specification value.

(5) The shape of the scram curve is assumed to be that which starts with all rods out of the core. This configuration results in the longest delay before significant reactivity is inserted into the core.

In addition, the calculational model contains conservatism. For example, the axial flux shape is assumed to remain constant throughout the excursion. This means that the energy deposition in the hot pellet is maximized.

The enthalpy rise in the hot pellet is plotted as a function of the worth of the dropped rod in NEDO-10527, Supplement 1. For the design calculation described above, a rod worth of approximately 1.4% reactivity change is required to produce an enthalpy rise of 280 cal/gm, which is the staff acceptance criterion.

To assess the extent of the conservatism in the assumption of no thermal-hydraulic feedback in the design calculations, the staff consultant, Brookhaven National Laboratories, performed a series of calculations that included this effect. The results are reported in BNL-NUREG-28109, "Thermal-Hydraulic Effects on Center Rod Drop Accidents in a Boiling Water Reactor." These results show that if thermal-hydraulic feedback is included in the calculations, the resulting enthalpy rise is less than 140 cal/gm for a rod worth of 1.4% reactivity change. Thus, it may be concluded that a large conservatism factor exists in the design calculations.

The River Bend reactor has a rod pattern control system that enforces the banked position withdrawal sequence. This sequence has been described in NEDO-21231, "Banked Position Withdrawal Sequence." This document shows that the maximum worth of a potential dropped rod during withdrawal of the first

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50% of the rods is 0.62% reactivity change during the first cycle and less during succeeding cycles.

The maximum worth of a potential dropped rod during withdrawal of the second 50% of the rods is 0.75% reactivity change during the first cycle and 0.83% reactivity change during the equilibrium cycle. Using the largest of these values, the design calculation would predict an enthalpy rise of 135 cal/gm. The calculation with thermal-hydraulic feedback would predict less than 75 cal/gm.

In view of the low value of maximum fuel enthalpy, cladding failure would not be expected. However, for purposes of evaluating environmental consequences, it is conservatively assumed that 770 rods (1.7%) suffer cladding failure. Similarly, the low value of fuel enthalpy would result in an insignificant pressure pulse in the core.

The staff concludes that the analysis of the rod drop accident is acceptable. The applicant met the requirements of GDC 28 with respect to preventing postulated reactivity accidents that could result in damage to the RCPB greater than limited local yielding or cause sufficient damage that would significantly impair the capability to cool the core. The staff has evaluated the applicant's analysis of the assumed control rod drop accident and finds the assumptions, calculational techniques, and consequences acceptable. Because the calculations predict peak fuel enthalpies less than 280 cal/gm, prompt fuel rupture with consequent rapid heat transfer to the coolant from finely dispersed molten UO_2 was assumed not to occur. The pressure surge was, therefore, calculated on the basis of conventional heat transfer from the fuel and resulted in a pressure increase below "Service Limit C" (as defined in Section III of the ASME Code) for the maximum control rod worths assumed. The staff believes that the calculations contain sufficient conservatism, both in the initial assumptions and in the analytical models, to ensure that primary system integrity will be maintained.

In evaluating the radiological consequences, the staff used RG 1.77 for the amount of activity accumulated in the fuel-clad gap and used the assumptions

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and methods specified in the Appendix to SRP 15.4.9 to compute the offsite radiological consequences of this accident. These consequences are listed in Table 15.1, and the staff assumptions are listed in Table 15.2. The resulting doses shown in Table 15.1 are well within the 10 CFR 100 guidelines and, therefore, meet the acceptance criteria of the Appendix to SRP 15.4.9. Based upon its findings, the staff concludes that the River Bend station is effectively designed to control any release of radioactive fission products following a control rod drop accident.

Table 15.2 Assumptions used for the rod drop accident analysis

Parameter	Value
Power level, MWt	3039
Fraction of failed fuel, %	2.1
Peaking factor of failed fuel	1.5
Failed fuel iodine and noble gases released to coolant, %	10
Coolant activity reaching condenser, %	10
Iodines	
Noble gases	100
Activity released to the environment from the condenser (%/day)	1

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15.5 Postulated Radioactive Releases Resulting From Liquid Tank Failures

~~Postulated Radioactive Releases Due to Liquid Tank Failures~~

~~Introduction~~

The applicant's analysis of the radioactive liquid waste tank failure accident appears in Section 15.7.3 of the FSAR. An evaluation of this accident was performed and the applicant's analysis was reviewed, in accordance with review specified in NUREG-0800, SRP Section 15.7.3.

~~Acceptance Criteria~~

The staff has conducted an independent evaluation of the consequences of component failures for radioactive-liquid-waste-components located outside the reactor containment which could result in releases of liquid containing radioactive materials to the environs.

The principal criteria governing acceptance in our review were (1) General Design Criterion 60, as it relates to the radioactive waste management systems being designed to control releases of radioactive materials to the environment; and (2) 10 CFR Part 20, as it relates to effluents to unrestricted areas. Tanks and associated components

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containing radioactive liquids outside containment are considered acceptable, by the criteria of SRP 15.7.3, if failure does not result in radionuclide concentrations in excess of the limits in 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply in an unrestricted area.

[redacted]

The River Bend Station has several radioactive liquid waste processing and storage tanks which are encompassed by the review conducted under SRP 15.7.3. The results of the reviews are discussed below.

[redacted]

The applicant submitted two failure cases that are considered limiting. One case is the failure of the regenerant evaporator releasing the evaporator bottoms to the floor of the radwaste building. Even though the radwaste bulding floor drain sump pump would normally transfer the spilled contents to the floor drain tanks, the pump may not be available. Therefore, the applicant analysis postulated the entire contents of the evaporator to leak out of the radwaste building and migrate to ground water. The regenerant evaporator contains the largest inventory and concentration of radionuclides that could be transported to the groundwater from the liquid radwaste system.

The second case postulated by the applicant is failure of a full condensate storage tank at its greatest potential radionuclide concentration which could occur during refueling operations.

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has performed

The staff ~~is presently performing~~ an independent analysis, accounting for hydrologic considerations to verify the applicants conclusions. The results of this analysis ~~will be available by the end of April for incorporation into the Final SER~~. Based on review of the licensee's submittal, ~~and subject to the verification mentioned above~~, the two failure cases represent the worst case conditions and also yield acceptable activity concentrations at the nearest unrestricted drinking water source as defined by 10 CFR 20, Appendix B, Table II, Column 2.

~~5. Conclusions~~ Findings [redacted]

[redacted] ~~(to be included in addendum to Final SER)~~.

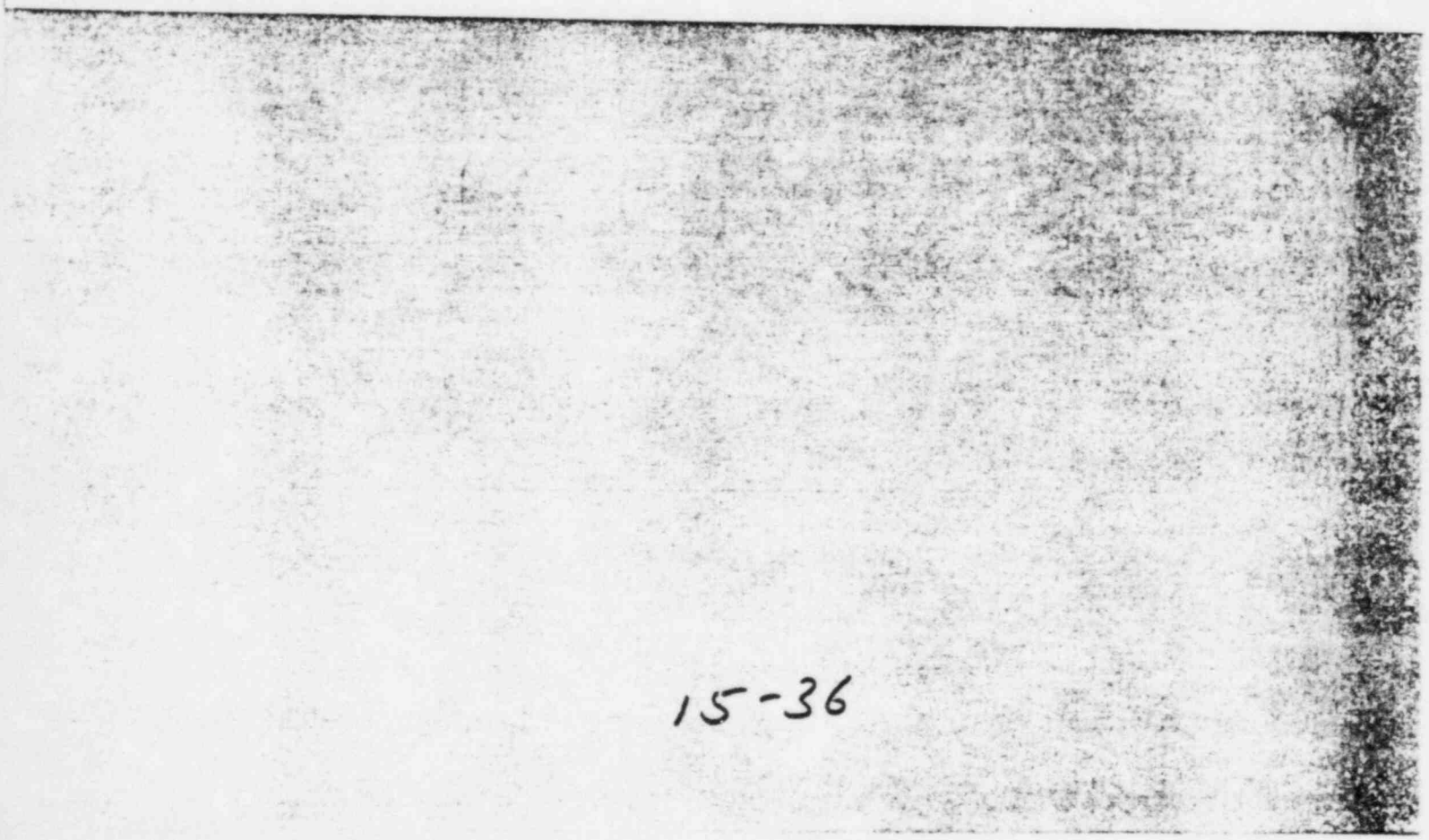
Confirm, using conservative groundwater travel time and dilution estimates, that the concentrations of all radionuclides present will be sufficiently below the 10 CFR 20 limits.

15.6 Anticipated Transients Without Scram

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Anticipated transients without scram (ATWS) are events in which the scram system (reactor trip system) is postulated to fail to operate as required. This subject has been under generic review by the Commission staff for several years.

In December 1978, Volume 3 of NUREG-0460, "Anticipated Transient Without Scram for Light Water Reactors," was issued describing the proposed type of plant modifications the staff believes necessary to reduce the risk from ATWS to an acceptable level. The staff asked the industry to supply generic analyses to confirm the ATWS mitigation capability described in Volume 3 of NUREG-0460. Subsequently, the staff recommended to the Commission that rulemaking be used to determine any future modifications necessary to resolve ATWS concerns and recommended a required schedule for implementation of such modifications. River Bend station is subject to the Commission's decision in this matter. The staff expects that the necessary plant modifications will be implemented 1 to 4 years after a Commission decision on ATWS. As a prudent course, to further reduce the risk from ATWS events before completion of any plant modifications deemed necessary by the Commission, the staff requires that the following steps be taken:



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(1) An emergency operating procedure should be developed for an ATWS event, including consideration of scram indicators, rod position indicators, average power range flux monitors, reactor vessel level and pressure indicators, relief valve and isolation valve indicators, and containment temperature, pressure, and radiation indicators. The emergency operating procedures should be sufficiently simple and unambiguous to permit prompt operator recognition of an ATWS event.

(2) The emergency operating procedure should describe actions to be taken in the event of an ATWS including consideration of manually scramming the reactor by using the manual scram buttons, changing the operation mode switch to the shutdown position, tripping the feeder breakers on the reactor protection system power distribution buses, scramming individual control rods from the back of the control room panel, tripping breakers from plant auxiliary power source feeding the reactor protection system, and valving out and bleeding off instrument air to scram solenoid valves. These actions must be taken immediately after detection of an ATWS event. Actions should also include prompt initiation of the residual heat removal system in the suppression pool cooling mode to reduce the severity of the containment conditions and actuation of the standby liquid control system if a scram cannot be made to occur.

Early operator action as described above, in conjunction with the recirculation pump trip, would provide significant protection for some ATWS events, namely those which occur (1) as a result of common mode failure in the electrical portion of the scram system and some portions of the drive system, and (2) at lower power levels where the existing standby liquid control system capability is sufficient to limit the pool temperature rise to an acceptable level.

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In a letter from J. E. Booker to D. F. Eisenhut dated April 15, 1983, which was in response to Generic Letter 82-33, "Supplement 1 to NUREG-0737, Requirements for Emergency Response Capability," the applicant committed to implement EOPs based on the GE Owners' Groups guidelines. The GE Owners Groups Guidelines include a "Reactivity Control" guideline. The staff approved the Guidelines for implementation in a letter from D. F. Eisenhut to T. Dente dated February 4, 1983. The staff therefore finds the applicants' commitment to develop EOPs based on the GE Owners' Group guidelines sufficient to assure adequate ATWS procedural guidance. ~~[Will not be available to the plant's operators]~~ A review of the River Bend ATWS procedure will not be required, and this item is considered complete for River Bend. However future recommendations from the Task Force investigating the Salem ATWS occurrence will be appropriately applied to River Bend.

Insert on Dec. 15 as the last paragraph
Delete the last paragraph that is there now.

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15.6.1 Deviations From Standard Review Plan Chapter 15

Our review of Section 15 has been performed in compliance with the guidelines of the revised Standard Review Plan (SRP), NUREG-0800, with the following exception:

The SRP requires that the transient and accident scenarios should include transients combined with single failures. The applicant has not analyzed anticipated transients with single failure/operator error as required by the SRP. We require the applicant to analyze the most limiting transients combined with a single failure.

The SRP states that the proposed plant technical specifications are reviewed. We have not yet reviewed all of the plant technical specifications since they are not available. A review will be performed when the completed specifications are submitted by the applicant prior to fuel loading. A detailed review of all of the specifications will probably not be performed since it is expected that approved standard technical specifications will be used at RBS.

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15.7 TMI Action Plan Requirements

15.7.1 II.F.1 Attachment 1, Noble Gas Effluent Monitor

The applicant's initial submittal did not contain sufficient information for the staff to conduct a review, and thus the submittal was unacceptable.

15.7.2 II.F.1 Attachment 2, Sampling and Analysis of Plant Effluents

The applicant did not supply sufficient information for the staff to conduct a review, so the applicant's initial submittal was unacceptable.

15.7.3 III.D.1.1 Integrity of Systems Outside Containment Likely To Contain Radioactive Material For Pressurized-Water Reactors and Boiling-Water Reactors

The applicant has committed to developing procedures and a scheduled maintenance program to monitor leakage and to reduce detected leakage to as-low-as-practical levels for systems outside the containment that could contain radioactivity.

The River Bend station program to reduce leakage to as-low-as-practical levels for all required post-accident systems outside the containment that could contain highly radioactive fluid will consist of:

(1) Monitoring drain sumps to ascertain gross leakage occurring from the systems included in this program.

(2) Determining leak rates for containment isolation valves by type "C" leak rate tests as defined in 10 CFR 50, Appendix J, paragraph II.H. Type "C" leak rate tests will be performed during the preoperational test program and periodically after initial fuel load in accordance with 10 CFR 50, Appendix J, paragraph III.D.3.

(3) Reducing leakage from miscellaneous components (vents, drains, valve packing, pump gland seal leak offs, etc.) to as-low-as-practical levels

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during initial system operations as part of the systems preoperational test. After fuel load, these components will be monitored as part of the River Bend station surveillance test program.

(4) Where it is not possible, practical, or permissible* to make direct inspections (high radiation areas, no provisions for testing the component, etc.) providing for indirect inspections or a suitable substitute. Indirect inspections may consist of, but are not limited to, inspecting floor areas and equipment drain cups for wetting that would occur if leakage were present, and monitoring the associated equipment or floor drain sump for excessive flow or fill rates.

The above program is considered acceptable. It will satisfy the action plan requirement when the initial leak rates are provided.

*Based on ALARA requirements.

16 TECHNICAL SPECIFICATIONS

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The Technical Specifications in a license define certain features, characteristics, and conditions governing operation of a facility that cannot be changed without prior approval of the staff. The finally approved Technical Specifications will be made a part of the Operating License. Included will be sections covering definitions, safety limits, limiting safety system settings, limiting conditions for operations, surveillance requirements, design features, and administrative controls.

The Technical Specifications for this facility will be based on "Standard Technical Specifications for General Electric Boiling Water Reactors" (NUREG-0123, Rev. 4). This document has been updated from earlier revisions as a result of continued discussion with General Electric and other licensees with General Electric BWRs.

The staff is working with the applicant to prepare a draft of the Technical Specifications for the River Bend facility. On the basis of its review to date, the staff concludes that normal plant operation within the limits of the Technical Specifications will not result in offsite exposure in excess of the 10 CFR 20 limits. Furthermore, the limiting conditions for operation and surveillance requirements will ensure that necessary engineered safety features will be available in the event of malfunctions within the facility.

During its review of the River Bend application, the staff identified certain issues that must be included in the Technical Specifications as a condition of staff acceptance. These issues are identified in various sections of this SER. Most of the issues that the staff has identified as being required to be included in the River Bend Technical Specifications are already addressed in NUREG-0123. Those issues that are not included in NUREG-0123 will be added to the Technical Specifications being prepared for the River Bend facility.

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17 QUALITY ASSURANCE

17.1 ~~General~~ Quality Assurance During Design and Construction

The quality assurance (QA) program for the operations phase of the River Bend station is described in FSAR Section 17. The staff has reviewed the program against acceptance criteria in SRP 17.2, and applicable QA-related RGs referenced in SRP 17.2 to determine if the program complies with the requirements of 10 CFR 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." The specific quality-related RGs committed to are identified in Table 17.1.

17.1.1 Organization

The Vice President-River Bend Nuclear Group (RBNG) is responsible for the safe and reliable operation of River Bend station, including the overall responsibility for the policies, goals, and objectives of the QA program.

The Director-Quality Assurance reports both administratively and functionally to the Vice President-RBNG (Figure 17.1). This relationship ensures that GSU QA personnel who are responsible for checking, auditing, or otherwise verifying that quality-related activities have been performed are independent of personnel directly responsible for performing the activities associated with schedules or costs.

The Director-Quality Assurance is authorized to evaluate how all activities at the station are conducted, with respect to quality, by means of checks, reviews, audits, surveillance, and/or inspections. He performs this evaluation on a planned and periodic basis to verify that the QA program is being effectively implemented. He is responsible for periodically evaluating and reporting the status and adequacy of the QA program to the appropriate management. He

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Table 17.1 RGs Applicable to QA Program

RG	Revision/Date	Title
1.8	1R	Personnel Selection and Training
1.30	8/72	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electrical Equipment
1.33	2	Quality Assurance Program Requirements (Operation)
1.37	3/73	Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants
1.38	2	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants
1.39	2	Housekeeping Requirements for Water-Cooled Nuclear Power Plants
1.58	1	Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel
1.64	2	Quality Assurance Requirements for the Design of Nuclear Power Plants
1.74	2/74	Quality Assurance Terms and Definitions
1.88	2	Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records
1.94	1	Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants
1.116	0-R	Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems

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Table 17.1 continued

RG	Revision/Date	Title
1.123	1	Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants
1.144	1	Auditing of Quality Assurance Programs for Nuclear Power Plants
1.146	8/80	Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants

has the authority and organizational freedom to identify quality problems; initiate, recommend, or provide solutions through designated channels; and verify implementation of solutions. He has the authority, delineated in writing, to initiate stop work actions or control further processing, delivery, or installation of nonconforming material through appropriate channels as described in the applicable quality assurance procedure.

The Director-Quality Assurance is assisted in carrying out his responsibilities by the QA department staff, consisting of Quality Systems, Operational QA, and Engineering QA divisions (Figure 17.2).

The Director-Quality Assurance has delegated the responsibility of the QA program during operations to the Operational Quality Assurance Supervisor (OQAS).

The OQAS has direct responsibility for ensuring implementation of the GSU QA program and additions and changes thereto at the station. He reports directly to the Director-Quality Assurance and maintains a working interface and communication with the Plant Manager.

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Figure 17.1 River Bend Nuclear Group organization

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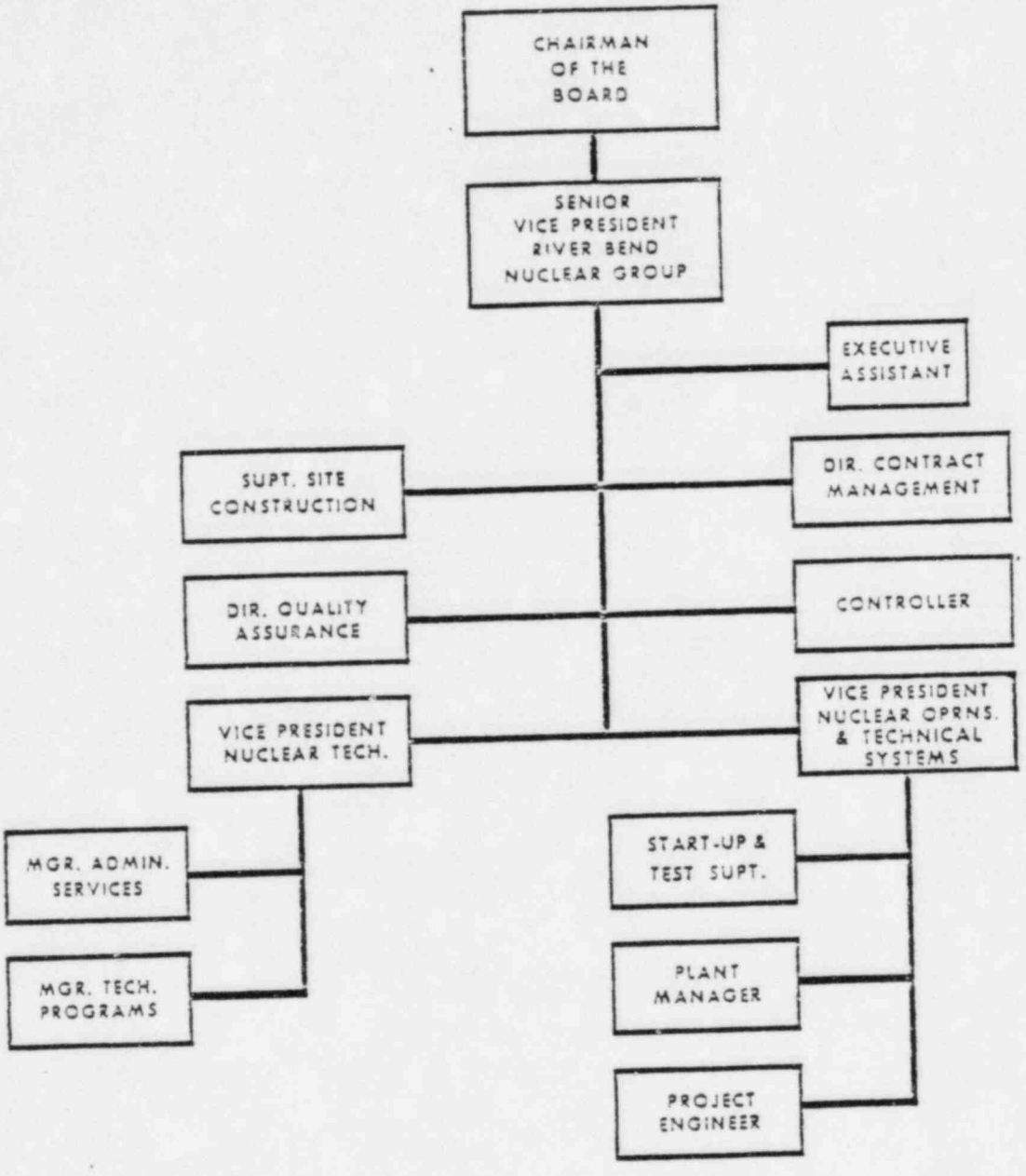


FIGURE 17.1

River Bend ~~Nuclear~~ Group ~~Organization~~

type caption

The principal responsibilities of the OQAS are (1) establishing and implementing all station QA procedures and instructions; (2) conducting inspections and audits; (3) assisting in developing and maintaining the QA training and indoctrination program to ensure adequate training is provided to those personnel performing activities affecting quality; (4) reviewing and concurring with the QA/quality control programs of suppliers, contractors, and consultants, when the purchase requisition for their services originates at the station; (5) analyzing nonconformance reports for quality trends; and (6) evaluating and reporting the status and adequacy of the quality assurance program at the station to the Director-Quality Assurance, and to the Plant Manager on a periodic basis. He and his staff have the authority delineated in writing to initiate stop work action through channels or control further processing, delivery, or installation of nonconforming material.

The QA organization is responsible for (1) reviewing and approving the Operational QA Manual and reviewing and concurring with operations administrative procedures that detail activities within the scope of the QA program; (2) reviewing procurement documents and concurring that appropriate quality assurance requirements are specified; (3) reviewing and approving suppliers' QA program requirements; (4) performing reviews and evaluations of procurement sources to determine their capability to meet quality assurance requirements; (5) performing surveillance and evaluations at suppliers' facilities to verify continued compliance with the quality assurance requirements of the procurement documents; and (6) performing surveillance and audits of all quality-related activities performed onsite, offsite, and at suppliers' facilities. Personnel performing quality-related activities have the authority to: (1) identify quality problems, which includes stop work decisions by the quality assurance organization; (2) initiate, recommend, or provide solutions to quality problems; (3) verify implementation of solutions; and (4) control further processing, delivery, installation, or utilization of nonconforming items until further dispositioning has occurred.

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Figure 17.2 River Bend QA Organization

* Located on site

---- Direct Communication

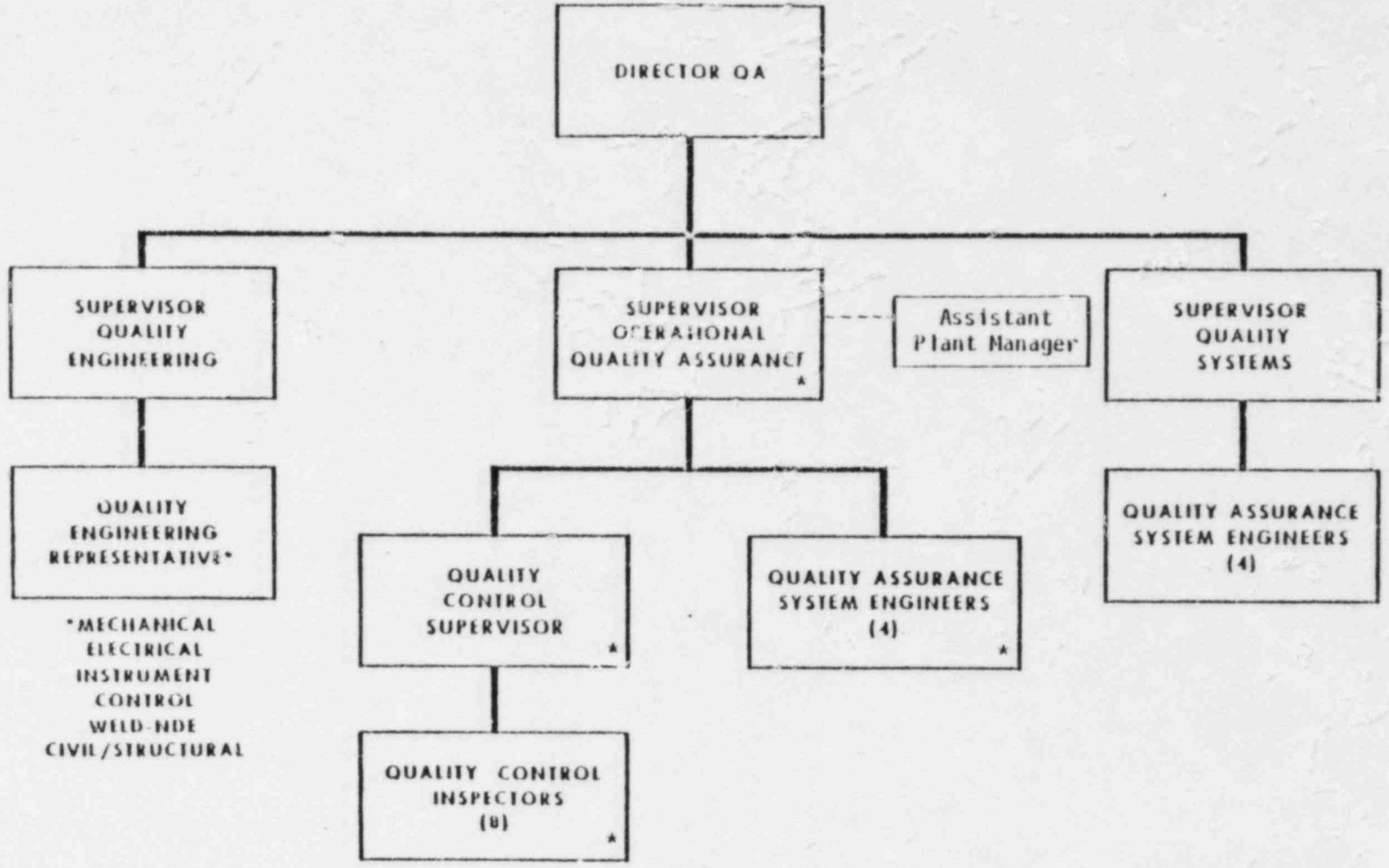


FIGURE 17.2
River Bend Quality Assurance Organization

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17.1.2
~~17.3~~ Quality Assurance Program

GSU has structured its QA program for the operations phase to satisfy Appendix B to 10 CFR 50 and the provisions of the RGs. The QA program for the operation of River Bend is implemented by written policies, procedures, and instructions. These documents control quality-related items in accordance with the requirements of Appendix B to 10 CFR 50 and with applicable regulations, codes, and standards. The QA organization is responsible for ensuring that quality-related procedures and instructions completely and adequately satisfy QA requirements. In addition, QA personnel conduct inspections and audits to verify the effective implementation of the QA program.

The GSU QA program requires that implementing procedures and instructions encompass detailed controls for (1) translating codes, standards, regulatory requirements, Technical Specifications, engineering, and process requirements into drawings and specifications, procedures, and instructions; (2) developing, reviewing, and approving procurement documents, including changes; (3) prescribing all quality-related activities by documented instructions, procedures, drawings, and specifications; (4) issuing and distributing approved documents; (5) purchasing items and services; (6) identifying materials, parts, and components; (7) performing special processes; (8) inspecting and/or testing materials, equipment, processes or services; (9) calibrating and maintaining measuring and test equipment; (10) handling, storing, and shipping of items; (11) identifying the inspection, test, and operating status of items; (12) identifying and dispositioning nonconforming items; (13) correcting conditions adverse to quality; (14) preparing and maintaining quality assurance records; and (15) auditing of activities that affect quality.

The indoctrination and training program established by GSU ensures that personnel performing activities affecting quality are knowledgeable in QA requirements, implementing procedures, and instructions and that they have competence and skill in the performance of their quality-related activities. It also provides for retraining of personnel to ensure maintenance of their competence and skills.

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Quality is verified through checking, review, surveillance, inspection, testing, and audit of quality-related activities. The QA program requires that quality verification be performed by individuals who are not directly responsible for performing the quality-related activities. Inspections are performed by qualified personnel in accordance with procedures and instructions.

The QA organization is responsible for the establishment and implementation of the audit program. Audits are performed in accordance with pre-established written procedures by qualified personnel not having direct responsibilities in the areas being audited. Periodic audits will be performed by the QA department to evaluate all aspects of the QA program including the effectiveness of the QA program implementation.

The QA program requires the review of audit results by the person having responsibility in the area audited and to determine and take corrective action where necessary. Continued deficiencies, or failure to implement corrective action, will be reported in writing by the QA organization to the appropriate executives within GSU.

Followup audits are performed to determine that nonconformances and deficiencies are effectively corrected and that the corrective action precludes repetitive occurrences. Audit reports are prepared and issued to responsible management for review and assessment.

The Vice President-RBNG, at least once a year, appoints a qualified auditing organization that is independent of the GSU QA department to conduct a management audit of the GSU QA program. The appointed organization reports directly to the Vice President-RBNG. The Vice President-RBNG reviews result summaries submitted to him by the Director-Quality Assurance. This review of the summaries of the management audit results allows the Vice President-RBNG to evaluate the effectiveness of the QA program.

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17.2 ~~Conclusion~~ Quality Assurance during the Operations Phase

Based on its detailed review and evaluation of the QA program description in FSAR Section 17, the staff concludes that the QA program for operations is acceptable, except that certain additional information and clarifications are necessary in regard to the list of structures, systems, and components that fall under the control of the QA program. ~~This additional material must be in FSAR Section 17 so the staff can complete its determination of the acceptability of the QA program. GSU has been made aware of the specific open items. This SER will be amended upon submittal of an agreed revision to FSAR Section 17.~~ These items have been identified in the PSAR and are undergoing review by [redacted] the staff. At the completion of this review additional information will be required from the applicant. This is considered an open item.

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APPENDIX C
UNRESOLVED SAFETY ISSUES

C.1 Introduction

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

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

These issues are sometimes called "generic safety issues" because they are related to a particular class or type of nuclear facility rather than to a specific plant. Certain of these issues have been designated as "unresolved safety issues" (NUREG-0410, "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants," dated January 1, 1978). However, as discussed above, such issues are considered on a generic basis only after the staff has made an initial determination that the safety significance of the issue does

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not prohibit continued operation or require licensing actions while the longer term generic review is under way.

C.2 ALAB-444 Requirements

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

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

This appendix is specifically included to respond to the Decision of the Atomic Safety and Licensing Appeal Board as enunciated in ALAB-444, and as applied to an operating license proceeding, Virginia Electric and Power Company (North Anna Nuclear Power Station, Unit Nos. 1 and 2), ALAB-491, 8 NRC 245 (1978).

C.3 Unresolved Safety Issues

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

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UNRESOLVED SAFETY ISSUES PLAN

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

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

SECTION 3 - UNRESOLVED SAFETY ISSUES

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

In response to the reporting requirements of the new Section 210, the NRC staff submitted NUREG-0410 to Congress on January 1, 1978. This NUREG describes the NRC generic issues program. The NRC program was already in place when PL 95-209 was enacted and is of considerably broader scope than the unresolved safety issues plan required by Section 210. In the letter transmitting NUREG-0410 to the Congress on December 30, 1977, the Commission indicated: "the progress reports, which are required by Section 210 to be included in future NRC annual reports, may be more useful to Congress if they focus on the specific Section 210 safety items."

It is the NRC's view that the intent of Section 210 was to ensure that plans were developed and implemented on issues with potentially significant public safety implications. In 1978, the NRC undertook a review of more than

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issues addressed in the NRC program to determine which issues fit this description and qualify as unresolved safety issues for reporting to the Congress. The NRC review included the development of proposals by the NRC staff and review and final approval by the NRC Commissioners.

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

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

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

All of the issues addressed in the NRC program were systematically evaluated against this definition as described in NUREG-0510. As a result, 17 unresolved safety issues addressed by 22 tasks in the NRC program were identified. ~~The issues are listed below. Progress on these issues was first discussed in the 1978 NRC Annual Report. The number(s) of the generic task(s) (for example, A-1) in the NRC program addressing each issue is indicated in parentheses following the title.~~

~~Unresolved Safety Issues (Applicable Task Nos.)~~

~~(1) Waterhammer (A-1)~~

~~(2) Asymmetric Blowdown Loads on the Reactor Coolant System (A-2)~~

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A-47 Safety Implications of Control Systems
A-48 Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment

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

In addition to the four issues identified above, the Commission approved another issue, A-49, Pressurized Thermal Shock, as an unresolved safety issue in December 1982.

In the view of the staff, the 27 unresolved safety issues listed above are the substantive safety issues referred to by the Appeal Board in ALAB-444 when it spoke of "...those generic problems under continuing study which have... potentially significant public safety implications." ~~Eight~~ Seven of the 27 tasks identified with the unresolved safety issues are not applicable to River Bend because they apply to PWRs only. These tasks are A-2, A-3, A-4, A-5, A-12, A-26 and A-49. Also, Tasks A-6, A-7, and A-8 apply only to Mark I or Mark II BWR containments. With regard to the remaining tasks that are applicable to this facility, the NRC staff has issued NUREG reports providing its proposed resolution of ~~seven~~ eight of these issues (Table 1). Each of these has been addressed in this Safety Evaluation Report or will be addressed in a future supplement. The table below lists those issues and the section of this Safety Evaluation Report in which they are discussed.

~~It is noted that~~ A-46 deals with seismic qualification of equipment in operating plants and does not apply. River Bend ~~is located, not in this category~~. River Bend was designed on the basis of current seismic design criteria, and commitments for seismic equipment qualification are in accordance with the latest codes and standards (see Sections 3.9.2 and 3.10 of this SER).

A-11 NUREG-0744, "Resolution of the Task A-11 Reactor Vessel Materials Toughness Safety Issue," Vols. I and II, Rev. 1, Oct. 1982

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Task No.	NUREG Report and Title	Safety Evaluation Report Section*
A-9	NUREG-0460, Vol. 4, "Anticipated Transients Without Scram for Light Water Reactors"	15.3.8
A-10	NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking"	*
A-24	NUREG-0588, Revision 1, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment"	3.11
A-31	SRP 5.4.7 and BTP 5-1 "Residual Heat Removal Systems" incorporate requirements of USI A-31	5.4.3
A-36	NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants"	9.1.4
A-39	NUREG-0802,* "Safety/Relief Valve-Quencher Loads Evaluation Report BWR Mark II and III Containments"	6.2
A-42	NUREG-0313, Revision 1, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping"	*

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

~~*This report is scheduled for publication in the third quarter of 1982. The report documents NRC acceptance of the safety/relief valve tests proposed by General Electric in Report #22A7000, Rev. 1, Appendix 3B for GESSAR II 238 Nuclear Island, dated November 25, 1980.~~

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The remaining issues applicable to this facility are:

A-1 Waterhammer
~~A-11 Reactor Vessel Materials Toughness~~
A-17 Systems Interaction in Nuclear Power Plants
A-40 Seismic Design Criteria
A-43 Containment Emergency Sump Reliability
A-44 Station Blackout
A-45 Shutdown Decay Heat Removal Requirements
~~A-46 Seismic Qualification of Equipment in Operating Plants~~
A-47 Safety Implications of Control Systems
A-48 Hydrogen Control Measures and Effect of Hydrogen Burns on Safety Equipment

The task Action Plans for the generic tasks up to and including A-40 above are included in NUREG-0649, "Task Action Plans for Unresolved Safety Issues Related to Nuclear Power Plants." Task action plans for later tasks were issued individually as indicated in the table below.

Task Action Plans for Selected USIs

Task Number	Issue Task Action Plan
A-43	1/81
A-44	7/80
A-45	10/81
	6/82 (Revision 1)
A-46	5/82
A-47	6/82*
A-48	12/82*

Update 2

~~*Draft task action plans have been developed and are scheduled to be issued on these dates.~~

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Each task action plan provides a description of the problem; the staff's approach to its resolution; a general discussion of the bases upon which continued plant licensing or operation can proceed pending completion of the task; the technical organizations involved in the task and estimates of the manpower required; a description of the interactions with other NRC offices, the ACRS and outside organizations; estimates of funding required for contractor-supplied technical assistance; prospective dates for completing the task; and a description of potential problems that could alter the planned approach or schedule.

In addition to the task action plans, the staff issues the "Office of Nuclear Reactor Regulation Unresolved Safety Issues Summary, Aqua Book" (NUREG-0606), on a quarterly bases which provides current schedule information for each of the unresolved safety issues. It also includes information relative to the implementation status of each unresolved safety issue for which technical resolution is complete.

The staff has reviewed the unresolved safety issues listed above as they relate to River Bend. Discussion of each of these issues, including references to related discussions in the Safety Evaluation Report is provided in Section C.4. The staff has satisfactorily concluded its review for all but ~~Task A-46, "Seismic Qualification of Equipment in Operating Plants," and~~ Task A-48, "Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment." It will discuss resolution of these issues in a supplement to this SER. Based on its review, the staff concludes for the reasons set forth in Section C.4, that, with the exception of Task ~~A-46 and~~ A-48, there is reasonable assurance that River Bend can be operated before the ultimate resolution of these generic issues without endangering the health and safety of the public.

C.4 Discussions of USIs as they Relate to River Bend Units 1 & 2

This section provides the NRC staff's evaluation of River Bend for each of the applicable unresolved safety issues. This includes the staff's bases for licensing before ultimate resolution of these issues. NRC conclusions are based in part on information provided by the Licensing Review Group II in their letter dated January 25, 1982.

BWR's has been essentially eliminated through the use of Jockey pumps (to eliminate voiding) eliminate/A line and vacuum breakers.

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Task A-1 ~~WATERHAMMER~~

Waterhammer events introduce large hydraulic loads, or pressure pulses, into a fluid system, and are the result of rapid condensation of steam pockets, steam-driven slugs of water, pump startup into voided lines, and improper (or sudden) valve closures. Since 1969, about 150 waterhammer events have been reported in nuclear power plants. Where waterhammer has occurred in water lines, the principal damage has been to pipe hangers and snubbers. Waterhammer in ~~steam generators with top feed rings has been basically eliminated by use of J-tubes, feedwater piping changes, and limits on auxiliary feedwater flow.~~ Examination of reported events reveals that approximately one-half of the waterhammer hammer events have occurred in preoperational testing or during the first year of operation. This suggests that a learning process during initial plant operation is involved. About one-half of the underlying causes are operational or plant procedure related. The remainder are related system design and systems interfaces. ~~Examination of reported events also shows that approximately one-third of the current operating plants have reported waterhammer occurrence, and that these events have been about equally distributed between BWRs and PWRs.~~ The increasing frequency of waterhammer events noted in the early 1970's (when an increasing number of plants were coming online) is not supported by current experience. Finally, in none of the waterhammer incidents reported has there been a release of radioactive material or disabling of safety systems.

Resolution of safety concerns related to the waterhammer issue is ~~being pursued under Task A-1. This study, which~~ is nearing completion, ~~confirms the situation described above.~~ The perspective gained is that (1) total elimination of waterhammer is not possible due to the coexistance of steam, water and voids inherent in nuclear power plant operation, (2) prediction of waterhammer under all possible operational scenarios is not feasible due to the extent of studies required and analysis code limitations. Rather, resolution of Task A-1 appears possible if occurrence of waterhammer is accepted and plant operational preventive measures are taken. Concluding Task A-1 studies show the following key elements necessary for reducing frequency and severity of waterhammer: (1) increase operator awareness of possible occurrence; (2) assure that proper plant operation and maintenance

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procedures are in effect; (3) utilize void surveillance instrumentation for void detection in those systems having a history of occurrence; (4) incorporate design features which have demonstrated success in elimination of waterhammer (that is, "keep-fill" systems using jockey pumps, proper vent design and maintenance, operational procedures designed to minimize contact between steam and cold water).

In order to protect the River Bend emergency core cooling system against the effects of waterhammer, the Emergency Core Cooling Systems (ECCS) are equipped with jockey pumps that provide a continuous supply of water to the ECCS discharge piping. These jockey pumps keep the ECCS lines water-filled so that the ECCS pumps will not start pumping into voided lines and steam will not collect in the ECCS piping. To ensure that the ECCS lines remain water filled, vents have been installed and a Technical Specification requirement to periodically vent air from the lines has been imposed. Further assurance for filled discharge piping is provided by pressure instrumentation at the piping high point. An alarm sounds in the main control room if the pressure falls below a predetermined setpoint indicating difficulty maintaining a filled discharge line. Should this occur, or if an instrument becomes inoperable, the required action is identified in the Technical Specifications.

Although the probability of a large pipe break is very small and initiation of a waterhammer occurrence from such a break is even less likely, adequate core cooling is ensured by the current ECCS design and redundancy. Protection against effects of pipe breaks and waterhammer loads are incorporated in piping design codes which require consideration of impact loads and dynamic loads in the piping system design.

In the event that in concluding Task A-1, efforts identify design considerations or plant operational requirements, in addition to those currently planned for the River Bend plant, followup corrective action will be required. Efforts-to-date have not identified design features or operational measures beyond those planned for implementation.

Based on the foregoing, the staff concludes that River Bend Station can be operated prior to final resolution of this generic safety issue without undue risk to the health and safety of the public.

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Task A-17 Systems Interactions in Nuclear Power Plants

The staff's systems interaction program was initiated in May 1978 with the definition of Unresolved Safety Issue A-17 (Systems Interactions in Nuclear Power Plants) and was intensified by Task Action Plan (NUREG-0660) Item II.C.3, "Systems Interaction." The concern arises because the design, analysis, and installation of systems are frequently the responsibility of teams of engineers with functional specialties such as civil, electrical, mechanical, or nuclear. Experience at operating plants has led to questions of whether the work of these functional specialists is sufficiently integrated to enable them to minimize adverse interactions among systems. Some adverse events that occurred in the past might have been prevented if the teams had ensured the necessary independence of safety systems under all conditions of operation.

The applicant has not described a comprehensive program that separately evaluates all structures, systems, and components important to safety for the three categories of adverse systems interactions, which are (1) spatially coupled, (2) functionally coupled, (3) and humanly coupled. However, there is assurance that River Bend can be operated without endangering the health and safety of the public. The plant has been evaluated against current licensing requirements that are founded on the principle of defense-in-depth. Adherence to this principle results in requirements such as physical separation and independence of redundant safety systems as well as protection against hazards such as high-energy line ruptures, missiles, high winds, flooding, seismic events, fires, human errors and sabotage. These design provisions are subject to review against the Standard Review Plan (NUREG-0800) which requires interdisciplinary reviews of safety-grade equipment and addresses different types of potential systems interactions. Also, the quality assurance program that is followed during the design, construction, and operational phases for each plant contributes to the prevention of introducing adverse systems interactions. Thus, the current licensing requirements and procedures provide an adequate degree of plant safety.

The NRC staff's current procedures assign primary responsibility for review of various technical areas to specific organizational units and secondary responsibility to other units where there is a functional interface. Designers follow somewhat similar procedures and provide the analyses of systems and interface

reviews. Task A-17 investigators have been developing methods that could discover adverse systems interactions that were not considered by current review procedures. The first phase of this study began in May 1978 and was completed in February 1980 by Sandia National Laboratories (SNL) under contract to the NRC staff.

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

SNL used fault-tree analysis on a selected light water reactor design to identify component failure combinations (cut-sets) that could result in loss of a safety function. The cut-sets were further reduced by incorporating six linking systems failures into the analysis. The results of the SNL effort indicated a few potentially adverse systems interactions within the limited scope of the study. The staff reviewed the interactions for safety significance and generic implications. The staff concluded that no corrective measures needed to be implemented immediately, except for the potential interaction between the power-operated relief valve and its block valve. This interaction had been separately identified by the evaluations of the TMI-2 accident while SNL was studying the selected light water reactor. Because corrective measures were already being implemented, no separate measures were needed under Unresolved Safety Issue A-17.

The applicant has taken some initial steps toward the performance of a separate evaluation of River Bend from a multidisciplinary point of view. A ~~Systems Interaction Survey Team has been~~ multidisciplinary team will be formed to analyze the "as-built" plant in areas where spatially coupled systems interactions could adversely affect safety-grade inspection equipment.

The "NRC Action Plan Developed as a Result of the TMI-2 Accident," NUREG-0660, provides for a systems interaction follow-on study in Section II.C.3, "Systems Interactions." Since April 1980, the Office of Nuclear Reactor Regulation has

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intensified the effort both by broadening the study of methods to discover potential systems interactions and by preparing guidance for audit reviews of selected plants for systems interactions. The staff's recent experience provides a basis from which it is developing a more efficient review process for potential systems interactions. The process will provide for a resolution of Unresolved Safety Issue A-17, assimilate operating reactor experience, and rank identified systems interactions by their relative importance to safety.

It is expected that the development of systematic ways to identify, rank, and evaluate systems interactions will go further to reduce the likelihood of intersystem failures resulting in the loss of plant safety functions. A comprehensive program is expected to use analytical methods, visual inspections, experience feedback, and experiments for hidden dependencies. The industry's current experience with systems interaction reviews is fragmented. Experience like that gained by the Phase I study is an essential ingredient to the staff's considerations of a comprehensive systems interaction program. After the resolution of Unresolved Safety Issue A-17, the staff will determine whether the applicant must perform further evaluations for adverse systems interaction.

Task A-40 Seismic Design Criteria - Short-Term Program

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

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The seismic design basis and seismic design of River Bend have conformed to existing licensing criteria and requirements. The staff's review of River Bend to these criteria is discussed in Sections 2.5.2 and 3.9.2.2 of this SER. Should the resolution of Task A-40 indicate that a change is needed in these licensing requirements, all operating reactors, including River Bend, will be reevaluated on a case-by-case basis.

Accordingly, the staff has concluded that River Bend can be operated before ultimate resolution of this generic issue without endangering the health and safety of the public.

Task A-43 Containment Emergency Sump Reliability

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

The concern addressed by Task A-43 for boiling water reactors is limited to the potential for degraded emergency core cooling system performance as a result of thermal insulation debris that may be blown into the suppression pool during a loss-of-coolant accident and cause blockage of the RHR pump suction lines. A second concern, potential ~~vortex formation~~ air ingestion, is not considered a serious concern for Mark III containment because of the depth of the emergency core cooling system suction lines and the low approach velocities. The minimum emergency core cooling system suction submergence is 10 ft. Tests have shown that submergences in excess of 10 feet do not result in significant air ingestion. With regard to potential blockage of the intake lines, the likelihood of any insulation being drawn into an emergency core cooling system pump suction line is very small. The potential debris in the drywell could only be swept into the suppression pool through the horizontal vents. Any pieces reaching the

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pool would tend to settle on the bottom and would not be drawn into the pump suction because the suction centerline is 10 ft above the pool bottom. In addition, boiling water reactors employ strainers on the suction with flow areas 200 % larger than the suction piping.

Accordingly, the staff concludes that River Bend can be operated before ultimate resolution of this generic issue without endangering the health and safety of the public.

Task A-44 Station Blackout

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

Task A-44 involves a study of whether or not nuclear power plants should be designed to accommodate a complete loss of all alternating current power, that is, a loss of both the offsite and the emergency diesel generator alternating current power supplies. This issue arose because of operating experience regarding the reliability of alternating current power supplies. A number of operating plants have experienced a total loss of offsite electrical power, and more occurrences are expected in the future. During each of these loss-of-offsite power events, the onsite emergency alternating current power supplies were available to supply the power needed by vital safety equipment. However, in some instances, one of the redundant emergency power supplies has been unavailable. In addition, there have been numerous reports of emergency diesel generators failing to start and run in operating plants during periodic surveillance tests.

A loss of all alternating current power was not a design-basis event for the River Bend facility. Nonetheless, a combination of design, operating, and testing requirements has been imposed to ensure that this unit will have substantial

and, in a few cases there has been a complete loss of AC power. ~~Complete loss of~~ AC power was restored in a few minutes without serious consequences.

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resistance to a loss of all alternating current and that, even if a loss of all alternating current power should occur, there is reasonable assurance the core will be cooled. These design, operating, and testing requirements are discussed below.

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

If offsite alternating current power is lost, three diesel generators ~~per unit~~ and their associated distribution systems will deliver emergency power to safety-related equipment. The staff's review of the design, testing, surveillance, and maintenance provisions for the onsite emergency diesels is described in Sections 8.3 and 9.6 of this SER. Staff requirements include preoperational testing to ensure the reliability of the installed diesel generators.

If both offsite and onsite alternating current power are lost, boiling water reactors may use a combination of safety/relief valves and the reactor core isolation cooling system to remove core decay heat without reliance on alternating current power. These systems ensure that adequate cooling can be maintained for at least 2 hrs, which allows time for restoration of alternating current power from either offsite or onsite sources.

In addition to the above, the Commission has determined that some interim measures should be taken at all plants to accommodate a station blackout pending resolution of the issue. Consequently, the NRC requested (Generic Letter 81-04, dated February 25, 1981) a review of plant operation to determine the applicant's capability to mitigate a station blackout event and properly implement, as necessary, emergency procedures and training programs for station blackout events. Appropriate review of the procedures and training programs for station blackout events will be completed before fuel load.

~~conducted~~ in accordance with the provisions of Regulatory Guide 1.108. In addition, the applicant has been required to implement a program for enhancing diesel generator reliability to ensure the long-term reliability of the diesel generators. This program resulted from recommendations of NUREG/CR-0660, "Enhancement of Onsite Emergency Generator Reliability."

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Based on the above considerations, the staff concludes that there is reasonable assurance that River Bend can be operated before the ultimate resolution of this generic issue without endangering the health and safety of the public.

Task A-45 Shutdown Decay Heat Removal Requirements

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

The River Bend reactor has various methods for the removal of decay heat. As discussed above, the decay heat is normally rejected to the turbine condenser and condensate is returned to the vessel by the feedwater system. The reactor core isolation cooling system provides an alternate means of supplying makeup water to the vessel. This turbine-driven pump takes suction from the reactor core isolation cooling storage tank and pumps to the vessel. If the condenser is not available (for example, loss of offsite power), heat can be removed by means of the safety/relief valves to the suppression pool. Also, high-pressure core spray (HPCS) is provided if the reactor core isolation cooling system is not available. The staff reviews of the RCIC and HPCS systems are described in Sections 5.4.7 and 6.3, respectively.

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If the reactor core isolation cooling and high-pressure core spray are unavailable, the reactor system pressure can be reduced by the automatic depressurization system (ADS) so that cooling by the residual heat removal system can be initiated. When the condenser is not used, the heat rejected to the suppression pool is subsequently removed by the residual heat removal system. The staff review of the ADS is described in Section 6.3.

The reactor core isolation cooling and high pressure core spray systems at River Bend have improvements over comparable systems at older boiling water reactors. The reactor core isolation cooling system has been upgraded to safety-grade quality (now required for all boiling water reactors), and the high pressure core spray is powered by its own dedicated diesel, so it can operate with an assumed loss of all other sources of alternating current power. Also, the residual heat removal system contains three pumps; the flow capacity of any single pump is sufficient to easily remove the decay heat. The staff review of the RHR system is described in Sections 5.4.7 and 6.3.

Following the TMI accident, the industry performed and documented extensive analyses of feedwater transients and small-break loss-of-coolant accidents to support acceptability of current designs. A report of these analyses was provided to the NRC in NEDO-24708A, Revision 1, dated December 1980. The staff's assessment of current designs related to loss-of-feedwater transients and small loss-of-coolant accidents is contained in NUREG-0626, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in GE-Designed Operating Plants and Near-Term Operating License Applications."

On the basis of these considerations, the staff concludes that River Bend can be operated before the ultimate resolution of this generic issue without endangering the health and safety of the public.

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Task A-47 Safety Implications of Control Systems

This issue concerns the potential for transients or accidents being made more severe as a result of control system failures or malfunctions. These failures or malfunctions may occur independently or as a result of the accident or transient under consideration. One concern is the potential for a single failure, such as a loss of a power supply, short circuit, open circuit, or sensor failure, to cause simultaneous malfunction of several control features. Such an occurrence could conceivably result in a transient more severe than those transients analyzed as anticipated operational occurrences. A second concern is for a postulated accident to cause control system failures that would make the accident more severe than analyzed. Accidents could conceivably cause control system failures by creating a harsh environment in the area of the control equipment or by physically damaging the control equipment. Although it is generally believed that such control system failures would not lead to serious events or result in conditions that safety systems could not safely handle, indepth studies have not been rigorously performed to verify this belief. The potential for an accident that would affect a particular control system, and effects of the control system failures, may differ from plant to plant. Therefore, it is not possible to develop generic answers to all these concerns; it is possible to develop generic criteria that can be used for future plant-specific reviews. The purpose of this unresolved safety issue task is to verify the adequacy of existing criteria for control systems or propose additional generic criteria (if necessary) that will be used for plant-specific review.

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

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

Also, the applicant has been requested (NRC Information Notice 79-22, "Qualification of Control Systems," September 17, 1979) to review the possibility of consequential control system failures that exacerbate the effects of high-energy line breaks (HELBs) and adopt new operator procedures, where needed, to ensure that the postulated events would be adequately mitigated. As part of the review, the staff is also evaluating the qualification program to ensure that equipment that may potentially be exposed to HELB environments has been adequately qualified or an adequate basis has been provided for not qualifying the equipment to the limiting hostile environment. The staff's evaluation of the applicant's response to Information Notice 79-22 and the adequacy of the qualification program are reported in Sections 7.7.2.1 and 3.10 of this SER, respectively.

With the recent emphasis on the availability of postaccident instrumentation (Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident"), the staff reviews evaluate the designs to ensure that control system failures will not deprive the operator of information required to maintain the plant in a safe shutdown condition after any anticipated operational occurrence or accident. The applicant was requested to evaluate its control systems and identify any control systems whose malfunction could impact plant safety. The applicant is requested to document the degree of interdependence of these identified control systems and identify the use (if any) of common power supplies and the use of common sensors or common sensor impulse lines whose failure could have potential safety significance. The status of these reviews and the staff's evaluation are documented in Section 7.5.2.2 of the SER.

In addition, IE Bulletin 79-27 ("Loss of Non-Class 1E Instrumentation and Control Power System Bus During Operation," November 30, 1979) was issued to the applicant requesting that evaluations be performed to ensure the adequacy

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of plant procedures for accomplishing shutdown on loss of power to any electrical bus supplying power for instruments and controls. The results of this review are documented in this SER, Section 7.5.2.4.

The subtask of this issue concerning the reactor overfill transient in boiling water reactors is currently under review by the BWR Owners Group of which the applicant is a member. Pending ultimate resolution of this item, the applicant has incorporated in the River Bend design a commercial-grade high-level trip (Level 8) of the reactor core isolation cooling, high-pressure core spray, and feedwater systems to prevent the occurrence of overfill transients.

On the basis of these above considerations, the staff concludes that there is reasonable assurance that River Bend can be operated before the ultimate resolution of this generic issue without endangering the health and safety of the public.

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

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

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

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

The accident at TMI-2 on March 28, 1979 resulted in hydrogen generation well in excess of the amounts specified in 10 CFR 50.44. It became apparent to NRC that specific design measures are needed for handling larger hydrogen releases, particularly for small, low-pressure containments. As a result, hydrogen control requirements have been developed. These requirements were published in a proposed rule in the Federal Register on December 23, 1981, 46 FR 62281.

For plants with Mark III containments, such as River Bend, the proposed rule specified that either it must be demonstrated that the containment can withstand hydrogen burns or explosions or a detailed evaluation of possible hydrogen control measures must be performed and the selected measures installed.

River Bend will comply with this rule through use of a hydrogen ignition system. This system consists of glow plug igniters distributed throughout the containment and drywell, which will ignite hydrogen at low concentrations and thereby maintain the concentration of hydrogen below its detonable limit and prevent potential containment overpressure.

To collectively evaluate the concerns associated with the hydrogen issue for Mark III containments, an owners' group has been formed. This group is sponsoring analytical work with General Electric, Offshore Power Systems, and others. Current evaluations of this group indicate that containment pressures will remain well below the failure point as the result of the postulated hydrogen release and burn.

The staff is currently reviewing (1) the hydrogen ignition system and the (2) applicant's analysis of the ability of essential equipment to survive the hydrogen burn environment. The results of this review will be provided in a supplement to this SER

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APPENDIX D

ACRONYMS AND INITIALISMS

ABVS	Auxiliary Building Ventilation System
ACI	American Concrete Institute
ACRS	Advisory Committee on Reactor Safeguards
ADS	Automatic depressurization System
ADV	atmospheric dump valve
AEC	Atomic Energy Commission
AISC	American Institute of Steel Construction
ALARA	as low as reasonably achievable
ANL	Argonne National Laboratory
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASCE	American Society of Civil Engineers
ASHRAE	American Society of Heating, Refrigeration and Air Conditioning Engineers
ASLB	Atomic Safety and Licensing Board
ASME	American Society of Mechanical Engineers
ATWS	anticipated transients without scram
BOL	beginning of life
BOP	balance of plant
BTP	Branch Technical Position
BWR	boiling water reactor
CAMS	containment atmospheric monitoring system
CAOC	constant axial control mode
CBCW	control building chilled water
CBS	containment building spray

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CCW	component cooling water
CCWS	component cooling water system
CFR	Code of Federal Regulations
CHF	critical heat flux
CIS	containment isolation signal
CIV	containment isolation valve
CO	condensation oscillation
CP	construction permit
CPI	Central Pressure Index
CRD	control rod drive
CRDS	control rod drive system
CRT	cathode-ray tube
CS	containment spray
CSAS	containment spray actuation signal
CSS	containment spray system
CST	condensate storage tank
CTMW	cooling tower makeup water
CVCS	chemical and volume control system
CVN	Charpy V-notch
CWS	circulating water system
DBA	design-basis accident
DCRDR	detailed control room design review
DECLG	double-ended cold leg guillotine
DEI	dose equivalent iodine
DES	Draft Environmental Statement
DGB	diesel generator building
DHR	decay heat removal
DHRS	decay heat removal system
DLF	dynamic load factor
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
DOE	Department of Energy
D/t	diameter-to-thickness

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EAB	exclusion area boundary
EAS	essential auxiliary support
ECC	emergency core cooling
ECCS	emergency core cooling system
EFPY	effective full-power year
EFW	emergency feedwater
EFWS	emergency feedwater system
EHC	electrohydraulic control
EMT	electrical metallic tubing
EOC	emergency operations center
EOF	emergency operations facility
EOL	end of life
~~EPA~~	~~electrical penetration area~~
EPG	emergency procedures guides
EPRI	Electric Power Research Institute
EPZ	emergency planning zone
EQ	equipment qualification
ESF	engineered safety feature
ESFAS	engineered safety feature actuation signal
FAA	Federal Aviation Administration
FBCFS	fuel building charcoal filtration system
FEMA	Federal Emergency Management Agency
FES	Final Environmental Statement
FIVP	feedwater isolation valve pit
FMEA	failure modes and effects analysis
FSAR	Final Safety Analysis Report
GAO	General Accounting Office
GDC	General Design Criterion(a)
GE	General Electric Company
GESSAR	General Electric Standard Safety Analysis Report
GSU	Gulf States Utilities Co.

HAZ	heat-affected zone
HCOG	Hydrogen Control Owners Group
HCU	hydraulic control unit
HED	human engineering discrepancy
HELB	high-energy line break
HEPA	high efficiency particulate air
HPCS	high pressure core spray
HPI	high-pressure injection
HPSI	high-pressure safety injection
HV	heating and ventilation
HVAC	heating, ventilation, and air conditioning
HVC	heating, ventilation, and cooling
I&C	instrumentation and control
IBA	intermediate break accident
IE	Office of Inspection and Enforcement
IEEE	Institute of Electrical and Electronics Engineers
INEL	Idaho National Engineering Laboratories
INPO	Institute of Nuclear Power Operations
ISEG	Independent Safety Engineering Group
ISI	inservice inspection
LANL	Los Alamos National Laboratory
LED	light-emitting diode
LER	licensee event report
LHGR	linear heat generation rate
LLNL	Lawrence Livermore National Laboratory
LO	lubricating oil
LOCA	loss-of-coolant accident
LPCI	low pressure coolant injection
LPCS	low pressure core spray
LPG	liquefied petroleum gas
LPMS	loose parts monitoring system
LPRM	low power range monitor

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LPZ	low population zone
LWR	light-water reactor
MCB	main control board
MCC	motor control center
MCES	main condenser evacuation system
MCPR	minimum critical power ratio
MCRHS	main control room habitability system
MM	Modified Mercalli
MMI	Modified Mercalli Intensity
MPC	maximum permissible concentration
MSIV	main steam isolation valve
msl	mean sea level
MSLB	main steamline break
MWt	megawatts thermal
mybp	million years before present
NEMA	National Electrical Manufacturers Association
NEPA	National Environmental Policy Act
NFPA	National Fire Protection Association
NNS	nonnuclear safety
NOAA	National Oceanic and Atmospheric Administration
NPSH	net positive suction head
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NSARC	Nuclear Safety Audit and Review Committee
NSSS	nuclear steam supply system
NTOL	near-term operating licensees
NWS	National Weather Service
OBE	operating basis earthquake
OD	outside diameter
OHLH	overhead heavy load handling
OL	operating license

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ORNL	Oak Ridge National Laboratory
OQAS	operational quality assurance supervisor
PA	public address
PAB	primary auxiliary building
PCCW	primary component cooling water
PCI	pellet cladding interaction
PCT	pellet cladding temperature
PDA	preliminary design assessment
PDT	primary drain tank
PID	proportional integral derivative
P&ID	piping and instrumentation diagram
PIS	process instrumentation system
PMF	probable maximum flood
PMH	probable maximum hurricane
PMP	probable maximum precipitation
PORC	Plant Operations Review Committee
PORV	power-operated relief valve
PRT	pressurizer relief tank
PSAR	Preliminary Safety Analysis Report
PSI	preservice inspection
PSRV	pseudo-relative velocity
PSTF	pressure suppression test facility
PVLCS	penetration valve leakage control system
PWR	pressurized-water reactor
QA	quality assurance
QC	quality control
RAI	request for additional information
RBNG	River Bend Nuclear Group
RRVS	radwaste building ventilation system
RC	reactor coolant
RCCA	rod cluster control assembly

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RCDT	reactor coolant drain tank
RCIC	reactor core isolation cooling
RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RDMS	radiation data management system
RFCS	recirculation flow control system
RG	Regulatory Guide
RGS	radwaste gas system
RGWS	radioactive gaseous waste system
RHR	residual heat removal
RHRS	residual heat removal system
RMS	radiation monitoring system
RO	reactor operator
RPCCW	reactor plant component cooling water
RPM	radiation protection manager
RPS	reactor protection system
RPT-?	(5.2.2)
RPV	reactor pressure vessel
RQD	rock quality designation
RTS	reactor trip system
RWCS	reactor water cooling system
RWCU	reactor water cleanup
RWST	refueling water storage tank
SAFDL	specified acceptable fuel design limits
SAR	Safety Analysis Report
SBA	small-break accident
SEP	systematic evaluation program
SER	Safety Evaluation Report
SER-CP	Safety Evaluation Report issued at Construction Permit stage
SFA	standard fuel assembly
SG	steam generator
SGBS	steam generator blowdown system

SGTR	steam generator tube rupture
SGTS	standby gas treatment system
SI	safety injection
SIAS	safety injection actuation signal
SIS	safety injection system
SLCS	standby liquid control system
SORC	Station Operations Review Committee
SPCM	suppression pool cooling mode
SPDS	safety parameter display system
SPF	standard project flood
SQRT	Seismic Qualification Review Team
SRC	Safety Review Committee
SRO	senior reactor operator
SRP	Standard Review Plan
SRSS	square root of the sum of the squares
SRV	safety relief valve
SSE	safe shutdown earthquake
SSLPS	solid state logic protection system
SSPS	solid state protection system
SSSP	site-specific spectra project
SSWS	standby service water system
STA	shift technical advisor
STPT	simulated thermal power trip
SV	safety valve
SWS	service water system
SWST	sodium hydroxide storage tank
TA	tower actuation
TDC	thermal diffusion coefficient
TGSS	turbine gland sealing system
TIP	traversing incore probe
TLD	thermoluminescent dosimeter
TMI-2	Three Mile Island Unit 2

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TPCCW	turbine plant component cooling water
TSC	Technical Support Center
UHS	ultimate heat sink
UL	Underwriters Laboratory
UPS	uninterrupted electric power system
USGS	U.S. Geological Survey
USI	unresolved safety issue
UT	ultrasonic inspection
WATS	wide-area telephone system
WMS	waste management system
WNP-2	Washington Public Power Supply System Nuclear Project No. 2
WPB	waste processing building