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La Crosse Boiling-Water Reactor

SAFEGUARDS REPORT

For Operating Authorization

AEC CONTRACT NO. AT (11-1)-850

RELEASED FOR ANNOUNCEMENT
IN NUCLEAR SCIENCE ABSTRACTS

VOLUME I

ALLIS-CHALMERS
ATOMIC ENERGY DIVISION

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prepared for
THE UNITED STATES ATOMIC ENERGY COMMISSION
under
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VOLUME I

ALLIS-CHALMERS
ATOMIC ENERGY DIVISION
BETHESDA, MARYLAND

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1. INTRODUCTION

This report constitutes a part of Allis-Chalmers' application for Operating Authorization of the La Crosse Boiling-Water Reactor (LACBWR). Information on the facility as actually constructed is presented, together with detailed plans and procedures for operation insofar as they have a bearing on the safety of the facility. In addition, there is a final evaluation of the likelihood and consequences of potential accidents.

Allis-Chalmers has the responsibility for the design, fabrication, construction, and start up of the reactor plant which utilizes a direct-cycle, forced-circulation boiling-water reactor cooled and moderated by light water. The reactor is to produce sufficient dry saturated steam at a turbine inlet pressure of 1265 psia for the production of 50,000 kwe (net) in the turbine-generator. Allis-Chalmers retained Sargent & Lundy Engineers as architect-engineers for the project and the Maxon Construction Company as constructors. Dairyland Power Cooperative of La Crosse, Wisconsin is furnishing the plant site and all equipment, facilities, and services necessary for a complete and operable nuclear plant, except for the nuclear steam-supply system. Dairyland will operate the facility as a base-load plant on its system for a period of ten years and will purchase steam produced by the reactor.

The plant site is near the village of Genoa, Wisconsin, on the east bank of the Mississippi River. Dairyland has an existing 14,000-kwe steam plant at this site. The reactor plant is south of the existing plant on property owned by Dairyland. The site was reviewed by the staff of the AEC and the Advisory Committee on Reactor Safeguards (ACRS) and was found acceptable for a reactor of the type and power level proposed.

The application for construction authorization was also reviewed, and it was concluded that the reactor could be constructed as proposed with reasonable assurance that it can be operated without undue hazard to the health and safety of the public. The construction approval, which was based on assumed satisfactory completion of the LACBWR research and development program, included a provision for Allis-Chalmers to inform the Commission and ACRS of the progress of this R & D program. Allis-Chalmers has complied by submitting quarterly progress reports.

Although it is intended that this report include all information required for final safety evaluation of the plant, it may be desirable to refer to supporting documents for more detail of the developmental plant components, such as the control-rod-drive mechanism, and for detailed results of accident analyses. A listing of supporting documents that comprise a part of the application follows.

- (1) La Crosse Boiling-Water Reactor Hazards Summary Report for Construction Authorization (ACNP-62574, October 1962).
- (2) Amendment 1 to the Application of the Allis-Chalmers Manufacturing Company for Construction Authorization of the La Crosse Boiling-Water Reactor (ACNP-62614, December 20, 1962).

The major portion of this document consists of answers to the questions accompanying the letter of December 4, 1962, from E. G. Case, Assistant Director of the Division of Licensing and Regulation of the AEC, to H. C. Nickel, Manager of the then Nuclear Power Department-Washington of the Allis-Chalmers Manufacturing Company. The answers contain the additional technical information which the staff of the Division of Licensing and Regulation required to complete the review of Allis-Chalmers' application for construction authorization of the LACBWR.

- (3) Amendment 2 (ACNP-63582, July 1963) - Quarterly Technical Report No. 1 Concerning Progress of the LACBWR R & D Program.

This document presents the objectives and background information for each of the major research and development programs and gives performance data on the steam dryer system selected.

- (4) Amendment 3 to the Application of the Allis-Chalmers Manufacturing Company for Construction Authorization of the La Crosse Boiling-Water Reactor (ACNP-63584, August 1963).

This document describes design changes in the shadow shielding of the LACBWR containment building and the effects these design changes have on the safety of the plant.

- (5) Amendment 4 (ACNP-63624, November 1963) - Quarterly Technical Report No. 2 Concerning Progress of the LACBWR R & D Program.
- (6) Amendment 5 (ACNP-64529, December 1963) - Quarterly Technical Report No. 3 Concerning Progress of the LACBWR R & D Program.

This document includes a preliminary description of the control-rod-drive mechanism.

- (7) Amendment 6 (ACNP-64572, June 1964) - Quarterly Technical Report No. 4 Concerning Progress of the LACBWR R & D Program.

This document includes a preliminary reliability report for the control-rod-drive mechanism, along with the results of a test of the hydraulic scram system, the final report of the steam separation R & D program, and a description of the analog simulator.

- (8) Amendment 7 (ACNP-64604, August 1964) - Quarterly Technical Report No. 5 Concerning Progress of the LACBWR R & D Program.

This document includes a discussion of the development of the sample probe of the failed-fuel-element location system and the initial results of the dynamic analysis.

- (9) Amendment 8 (ACNP-64628, November 1964) - Quarterly Technical Report No. 6 Concerning Progress of the LACBWR R & D Program.

This document includes a discussion of the fission gas separation study for the failed-fuel-element location system.

- (10) Amendment 9 (ACNP-65517, February 1965) - Quarterly Technical Report No. 7 Concerning Progress of the LACBWR R & D Program.

This document includes the final results of the dynamic analysis.

- (11) Amendment 10 (ACNP-65542, June 1965) - Quarterly Technical Report No. 8 Concerning Progress of the LACBWR R & D Program.

This document, the last of the series of quarterly technical reports, includes the final discussion of the failed-fuel-element location system.

- (12) Amendment 11 (ACNP-65543, June 1965) - Control-Rod-Drive Mechanism Design and Test Report.

This document contains the final design description of the control-rod-drive mechanism, the results of the testing program, the final reliability analysis, and the analysis of a control-rod-drop accident. This submittal complies with the terms of the construction authorization.

The individual sections of this report make reference to previous material in ACNP-62574 that is superseded. These sections point out any deviations from previous plans or systems concepts that might be significant from a safety standpoint. Appendix A to ACNP-62574 (Geology and Hydrology of a Reactor Site near Genoa, Vernon County, Wisconsin) remains as previously published.

2. SUMMARY

2.1 PLANT DESCRIPTION

The LACBWR is a nuclear power plant of nominal 50 Mw electrical output, which utilizes a forced-circulation, direct-cycle boiling-water reactor as its heat source. The plant is on the east bank of the Mississippi River in Vernon County, Wisconsin, approximately 1 mile south of the village of Genoa, Wisconsin, and approximately 19 miles south of the city of La Crosse, Wisconsin.

The reactor operates at a pressure of 1300 psia and produces 165 Mwt (see Fig. 2.1). Feedwater returning from the turbine-condenser at a temperature of 285 F is mixed with the recirculating coolant. The coolant enters the bottom of the core and boils within the core to produce ~610,000 lb/hr of steam. A portion of this steam is separated from the coolant at the steam-water interface within the reactor vessel, and the remainder of the steam is removed by 16 centrifugal steam separators inside the reactor vessel. Steam dryers in the dome of the reactor vessel remove entrained moisture and the dry saturated steam leaves the reactor vessel at 1300 psia and 577.5 F.

Inlet conditions at the turbine are 1265 psia and 574 F. Steam passes through the high-pressure element of the turbine and, after passing through a moisture separator and a reheater, enters the intermediate-pressure element and then passes to the low-pressure element. It is exhausted to the condenser at an absolute pressure of 1 in. of mercury. Gross electrical output of the turbine-generator is 55.9 Mwe. With an auxiliary load of 3.5 Mwe, the net electrical output is 52.4 Mwe and the net cycle efficiency is 31.8 percent. Condensate passes through the condensate pumps and demineralizers, feedwater heaters 1 and 2, the feedwater pumps, and feedwater heater 3 before returning to the reactor system.

The reactor and its auxiliary systems are within a steel containment building. The turbine-generator and associated equipment, the control room for both turbine and reactor controls, and plant shops and offices are in a conventional building adjacent to the containment building. Waste-handling facilities, including facilities for processing liquid wastes and for packaging, decontaminating, and temporarily storing solid wastes, are in a separate building.

The containment building has been designed to withstand the pressure resulting from the instantaneous release of all energy of the primary system. Its design pressure is 52 psig, compared to a calculated maximum pressure buildup of 48.5 psig following the maximum credible accident. The containment-building shell is designed and constructed according to the ASME Boiler and Pressure Vessel Code, Sections 11, V111, and 1X, and Nuclear Code Cases 1270 N, 1271N, and 1272N. The interior surface of the containment building above grade is lined with 9 in. of concrete, except for the hemispherical upper dome, which is unlined.

The reactor pressure vessel has an internal diameter of 8.25 ft and an overall internal height of 37 ft. The vessel is constructed of ferritic steel and is lined with stainless steel. Its design pressure is 1400 psig. It is designed, fabricated, inspected, and stamped in accordance with the ASME Boiler and Pressure Vessel Code, Section VIII (Unfired Pressure Vessels) and Nuclear Code Case 1270N.

The reactor core has an equivalent diameter of 5.09 ft and an active height of 6.92 ft. It contains a total of 72 fuel assemblies, with a nominal uranium content of 8600 kg. Each assembly is enclosed within a removable section of shroud. The channels between adjacent shrouds guide the control rods. The startup core uses both zirconium shrouds and stainless-steel shrouds. However, the stainless-steel shrouds in the initial core are to be replaced partway through first-core life by zirconium shrouds. The use of stainless-steel shrouds initially reduces the amount of reactivity to be controlled and improves the power distribution within the core.

Each fuel assembly consists of a 10 x 10 array of fuel rods 0.396 in. OD by 83 in. long (active length), on a 0.565-in. pitch. The rods consist of 0.020-in. -thick stainless-steel tubing, filled with 0.350-in. -OD uranium dioxide fuel pellets. The uranium is enriched to 3.63 percent in uranium-235.

The core includes 29 cruciform control rods with B_4C pellets as absorber. The pellets are contained in Inconel tubes, which are packed in cruciform-shaped sheaths of stainless steel; each sheath is 5/16-in. thick and has 9-3/4 in. cruciform span and an active length of 83 in. The effective multiplication factor of the cold, clean core loaded for a burnup of 12,700 Mwd/ST of contained uranium is 1.188. Total rod insertion in this condition would reduce this multiplication factor by 0.269 (which would make the reactor subcritical by 8.1 percent Δk). Even if the rod of greatest worth were not inserted, the remaining control rods could make the reactor subcritical by 1.0 percent Δk , which exceeds the shutdown margin design criterion of 0.5 percent Δk with one stuck rod.

The control-rod extensions penetrate the bottom head of the reactor vessel. Normally the rods are positioned by electro-mechanical drive mechanisms that withdraw them downward out of the core to increase reactivity and insert them upward into the core to decrease reactivity. The rods are scrambled upward by hydraulic motors.

Reactor power is normally controlled and matched to load demand by controlled forced-circulation flow of reactor water through two piping loops external to the reactor vessel. Flow control is accomplished by two 15,000 gpm, single-stage, vertical centrifugal pumps, one in each piping loop. The loops have common inlet and discharge headers so that uniform distribution of flow to the reactor-vessel inlet nozzles is maintained in the event of reduced power operation with only one pump. A flow of high-pressure demineralized water cools the pump mechanical seals and prevents leakage of primary-system water. The speed of the

recirculation pumps is variable and is controlled by the plant control system, which uses turbine inlet pressure as a primary signal. The nuclear plant follows the turbine-generator demand from 60 to 100 percent of full power without control rod movement.

All portions of the recirculation-loop piping permanently imbedded within the primary biological shield are constructed of stainless steel. A portion of the piping behind removable portions of the shield is of low alloy steel. Shutoff valves between stainless-steel and low-alloy steel sections of piping permit isolation of the low-alloy sections.

In normal operation, the reactor heat is removed by the turbine and turbine condenser. Loss of this heat sink through reactor isolation or low condenser vacuum causes reactor scram and directly or indirectly, results in automatic activation of a shutdown condenser system within the containment building, which can condense up to 10 percent of the full-power steam flow. During shutdown periods, and after initial heat removal by the main or shutdown condenser, the decay-heat cooling system removes heat from the core. An emergency cooling system automatically sprays water directly on the fuel assemblies if low reactor water level indicates possible loss of core coolant. At least 50 gpm can be delivered to the core spray nozzles by either of two pumps, and continued spray is assured by gravity flow from an overhead storage tank if the reactor pressure drops to that of the containment building.

Provisions have been made for the handling of gaseous, liquid, and solid wastes. All gaseous wastes are released from a 350 ft high stack. If activity levels are too high, off-gases from the primary system may be stored in storage tanks of sufficient capacity to store all of the gases produced by 72 hr of full-power operation. Liquid wastes are handled by the batch method, with liquids within the containment building collected in two 6000-gal retention tanks and liquids within the turbine building collected in two storage tanks of 3000- and 4500-gal capacity. Demineralizers may be used to process the wastes for release to the condenser circulating water discharge line. Release is permitted, however, only if analysis indicates that activity levels in the discharge line do not exceed the limits in 10 CFR 20. Liquid wastes too active for release may be transferred to the waste treatment building where they are concentrated by evaporation or ion exchange for eventual off-site shipment in concrete-lined drums. Solid wastes are packaged in drums for off-site shipment. The waste-handling facility includes equipment for packaging and temporary storage of solid wastes.

2.2 SUMMARY DATA

This section is a listing of general plant data.

Location

Vernon County, Wisconsin

Prime Contractor

Allis-Chalmers Manufacturing Co.
Atomic Energy Division-Bethesda

Major Subcontractors

Architect-Engine
General contractor

Sargent & Lundy Engineers
Maxon Construction Company

Utility

Dairyland Power Cooperative

Heat Balance

Thermal power	164.7 Mwt
Gross electrical output	55.9 Mwe
Net electrical output	52.4 Mwe
Net efficiency	31.8%

Steam leaving reactor vessel:

condition	saturated
maximum moisture	0.1%
pressure	1300 psia
temperature	577.5 F
flowrate	610,233 lb/hr

Steam entering turbine:

condition	saturated
pressure	1265 psia
temperature	573.9 F
flowrate	529,600 lb/hr

Feedwater to reactor:

condition	subcooled
pressure	1350 psia
temperature	285.6 F
flowrate*	610,233 lb/hr

Core

Active diameter	61 in.
Active height	83 in.
Active volume	3980 liters
Total uranium content	8600 kg
Initial U-235 enrichment	3.63%
Water/uranium dioxide volume ratio (unrodded)	2.63
Water reflector thickness	17.0 in.

*includes seal and purification water

Fuel Assemblies

No. of assemblies in core	72
No. of fuel rods per assembly	100
Arrangement of rods in assembly	10 x 10
Outside rod diameter	0.396 in.
Rod pitch	0.565 in.
Cladding material	Type-348H stainless steel
Cladding thickness	0.020 in.
UO ₂ density	10.41 g/cm ³
UO ₂ diameter	0.350 in.
Active fuel length	83 in.
End plug material	Type-348H stainless steel

Nuclear Data

Average thermal flux, initial	1.5×10^{13} n/cm ² -sec
Max/average radial flux ratio	1.47
Max/average axial flux ratio	1.87
Max/average local flux ratio	1.45
Overall peak/average flux ratio (100% power)	3.98
Core burnup:	
initial core	12,700 Mwd/ST
equilibrium core	15,000 Mwd/ST
Multiplication factor of cold, clean core (k_{eff})	1.188
Change in multiplication factor (Δk_{eff}) of startup core caused by:	
temperature rise from 68 F to 577.5 F	0.038
steam void formation and Doppler broadening	0.025
xenon and samarium poisoning	0.041
Total amount by which all control rods can change the multiplication factor of the cold, clean core (Δk_{eff})	0.269
Shutdown multiplication factor of cold, clean core with all rods in (k_{eff})	0.919
Shutdown multiplication factor of cold, clean core with all but the most valuable rod in (k_{eff})	0.990
Maximum reactivity insertion rate for rod withdrawal from a cold, clean core during normal startup (cents/sec)	6

Heat Transfer and Fluid Flow

Average power density in core	41.4 kw/liter
Max. power density in core	164.8 kw/liter
Recirculation flowrate	10,750,000 lb/hr (30,000 gpm)

Flow through fuel assemblies	28,880 gpm
Flow through control-rod channels	745 gpm
Flow area per fuel assembly	0.136 ft ²
Core-inlet velocity:	
average	6.4 ft/sec
minimum	6.0 ft/sec
maximum	6.6 ft/sec
Steam void fractions (based on water volume in fuel assembly shrouds):	
average in core	0.21
average exit	0.37
maximum exit	0.57
Inlet subcooling	18.5 F
Heat-transfer area per fuel assembly	71.71 ft ²
Heat flux:	
average	109,100 Btu/hr-ft ²
maximum at 100% power	434,700 Btu/hr-ft ²
calculated burnout	810,000 Btu/hr-ft ²
burnout safety factor at 100% power	1.86
burnout safety factor at 120% power	1.55
UO ₂ temperature:	
average at 100% power	930 F
maximum at 100% power	3850 F
maximum at 120% power	4750 F

Control Rods

Total number	29
Pitch (triangular)	9.009 in.
Shape	cruciform
Total blade span	9.750 in.
Thickness	0.313 in.
Length of poison section	83 in.
Absorbing material (80 tubes per cruciform)	B4C

Control-Rod Drive Mechanism

Number of drives	29
Type of drives	electric motor shim, hydraulic motor scram, roller-screw
Location of drives	beneath reactor vessel
Direction of rod withdrawal from core	downward
Direction of scram	upward
Velocity-normal insertion and withdrawal	20 in/min
Scram time to 90% of rod travel	2.0 sec
Scram time to full insertion	2.5 sec

Reactor Vessel

Materials of construction:

vessel	A302B carbon steel
cladding	Type-304L stainless steel
Diameter, inside	8.25 ft
Thickness of cylindrical portion (including cladding)	4 in.
Thickness of cladding	0.188 in.
Height, inside	37 ft
Design pressure	1400 psig
Design temperature	650 F
Vessel closure type	bolted, double closure seal with between-seal leakoff
time required to open	6 hr
time required to close	6 hr
Inner thermal shield:	
material	Type-304 stainless steel
thickness	1.25 in.
Weight of contained water and steam during operation	20.92 tons

2.3 RESEARCH AND DEVELOPMENT

The following research and development programs have been conducted in support of the LACBWR design:

(1) Control-Rod-Drive Mechanism. The LACBWR control rods are withdrawn and inserted from beneath the core, which necessitates careful consideration of all design requirements to insure safe and reliable operation. As part of the LACBWR development program a prototype drive mechanism was designed, built, and tested to determine its performance characteristics, mechanical integrity, and reliability. The development program indicates that the rod drive mechanism meets all safety and reliability requirements.

(2) Special Instrumentation Systems. In direct support of the LACBWR project, Allis-Chalmers undertook development programs for two special instrumentation systems: a traversing in-core flux-monitoring system to obtain periodic measurements of the core flux distribution, and a failed fuel-element location system to locate failed fuel elements within the core during full-power operation.

(3) Plant Control System. The plant-control system, which regulates reactor output to follow the turbine-generator demand, is based on the use of variable-speed forced-circulation pumps. Analog-computer simulation studies demonstrated that the system will respond satisfactorily to normal load-demand changes with this method of control. The analog studies were also used to determine system response to abnormal conditions, and to determine the feasibility of various operating procedures.

In addition to these research and development programs, study programs were conducted on core physics, heat transfer, and hydraulics to provide a complete understanding of the performance characteristics of the reactor. A report on burnout heat flux design correlations for LACBWR is included as Appendix B.

2.4 PLANT SAFEGUARDS

The principal safeguards incorporated in LACBWR to insure that the reactor may be operated without undue hazards to the health and safety of the public are summarized below.

(1) Boiling-water reactors are inherently safe. The formation of additional core voids and the increases in fuel temperature upon increases in power tend to shut down the reactor before potentially dangerous power levels are reached.

(2) The reactor should operate with a high degree of stability. Its average power density of 41.4 kw/liter of core is within the range of power densities of other boiling-water reactors which have demonstrated stable operation. The high operating pressure of 1300 psia, the long thermal time constant of the uranium dioxide fuel, and the use of forced circulation are all features which promote stable operation.

(3) Three independent means of shutting down the reactor are available. The control rods are rapidly inserted by the hydraulic scram motors which are backed up by the electric shim motors. If all rods are not inserted upon an all-rod scram signal, an alarm is given and the manual boron injection system may then be used to shut down the reactor.

(4) The reactor can be cooled by four independent systems. Normally, steam produced within the reactor flows to the turbine and turbine condenser. If the reactor is isolated from this heat sink, up to 10 percent of the full-power steam flow is automatically condensed in the shutdown condenser. If the reactor core becomes uncovered, emergency cooling water will be provided automatically at either high or low pressure. During shutdown, a decay-heat cooler is also used to cool the core.

(5) In addition to the normal plant control system, overpressurization of the primary system is prevented by:

- (a) the main-steam bypass valve to the turbine condenser which, when opened, can pass 100 percent of the full steam flow
- (b) the valves to the shutdown condenser which, when opened, lead to condensation of up to 10 percent of the full steam flow
- (c) the primary-system relief valves which, when opened, release to the containment building up to 120 percent of the full steam flow

- (d) the rod control and safety system, which scrams the reactor on high pressure signal.

(6) The reactor safety system senses abnormalities throughout the plant. It actuates alarms when dangerous conditions are approached, and initiates a partial scram of 13 rods or a full scram of all rods when necessary.

(7) Radiation detectors monitor all potentially radioactive gaseous and liquid effluents. When activity release would constitute a hazard if allowed to continue, alarm signals are given and further release is prevented by closure of valves in the effluent line.

(8) The nuclear system and its associated equipment are enclosed in a steel containment building designed to limit leakage to 0.1 percent of the contained volume per day under pressure conditions of the maximum credible accident. Containment integrity is maintained through the use of airlock doors, quick-closing dampers in ducts that penetrate the shell, and manual, remote control, and automatic isolation valves. The building is lined with 9 in. of concrete to limit the radiation dose to areas outside the building in the event of a fission-product release to the containment building.

More detailed information concerning the safeguards features of the plant and their effectiveness in preventing or mitigating accidents is included in this report.

2.5 SUMMARY OF THE MAXIMUM CREDIBLE ACCIDENT (MCA)

The maximum credible accident for LACBWR was determined to be a sudden, gross loss of coolant resulting from a primary system rupture. The following specific accident conditions were assumed in calculating the consequences of such an accident:

(1) The rupture occurs in the worst location, i.e., below the core in one of the 20-in. forced-circulation lines.

(2) The core is rapidly uncovered after the rupture, and fuel temperature remains constant during the blowdown period.

(3) The reactor is generating only decay heat at the end of the blowdown period. (The reactor would scram when the water level reached a preset level above the top of the core, and the loss of moderator would make the reactor subcritical by a large amount.)

(4) There is a 30 sec delay before the emergency cooling water reaches the core through core spray nozzles.

(5) The MCA occurs when the fission-product inventory of the core is at a maximum.

With these assumptions, it is calculated that 34.9 percent of the fuel rods would eventually suffer partial melting of cladding. Pressure within the containment building would rapidly increase to 48.5 psig because of the release of all primary-system energy. Operation of the building spray system with water supplied from the overhead storage tank for 27 min after the accident would reduce this pressure to 16 psig.

Even though only 34.9 percent of the fuel rods melt, radiation dosage resulting from the MCA is calculated for areas outside the containment building on the basis of complete core meltdown. It has been assumed that all the noble gases, 50 percent of the halogens, and 1 percent of the solid fission products contained within the fuel rods are released to the containment building when the cladding is melted. It was further assumed that 50 percent of the halogens released to the containment building are either absorbed on internal surfaces or washed out of the building atmosphere.

Persons outside the containment building would receive a whole body dose from direct and scattered gamma radiation from the containment building and from the leakage plume. A 9-in.-thick concrete shield lines the interior surface of the building and reduces the direct radiation doses from the building. The leakage of fission products from the containment building would cause both external and internal doses to persons downwind of the building. The principal internal dose would be due to inhalation of the radio iodines. External doses would be primarily due to radiation from the leakage plume.

The containment building is assumed to leak the contained fission products at the maximum design leak rate of 0.1 percent per day and to leak at this rate indefinitely. Thus, no credit is taken for the reduction of containment pressure by the building spray. Leakage of halogens and gaseous fission product activity was assumed to occur for an indefinite period after the accident, whereas leakage of solid fission product activity was assumed to end after 30 days. A moderately stable meteorological regime and a low wind speed were used in these calculations.

Table 2-1 summarizes the calculated data for total whole-body doses and thyroid doses caused by inhalation of radio iodines at the exclusion radius, at the low-population zone radius, and at La Crosse, for the cases of infinite exposure and 2 hr exposure immediately after the accident.

These data are consistent with the requirements set forth in 10 CFR 100, and it is therefore concluded that reactor operation at this site will not result in undue hazard to the health and safety of the public.

TABLE 2-1

DOSES DOWNWIND FROM LACBWR FOLLOWING MCA

<u>distance, miles</u>	doses, rem			
	2 hr exposure		infinite exposure*	
	<u>Total whole-body</u>	<u>thyroid</u>	<u>total whole-body</u>	<u>thyroid</u>
0.21 (1109 ft) (exclusion radius)	23.5	291	1.15	16,300
3 (radius of low-population zone)	0.07	5.34	1.16	299
19 (distance to nearest popula- tion center, La Crosse, Wis.)	---	0.34	0.09	18.8

*30 days assumed for solids.

DAIRYLAND POWER COOPERATIVE
60,000 KW TURBINE-GENERATOR UNIT - 1250 PSIG - SATURATED - 1.0" HG ABS.

NOTE: 5% ΔP from extraction nozzles to heaters.
6% ΔP from H.P. turbine exhaust flange to I.P. turbine inlet.
2% ΔP from flash tank to heater 3.

3. SITE*

This section repeats information given in Sec. 3 of ACNP-62574, Hazards Summary Report for Construction Authorization of LACBWR, and incorporates additional data collected from an anemometer installed at the site. The reference: "Taylor, G.I., Dispersion of Matter in Turbulent Flow through a Pipe (1954)" of Appendix A of ACNP-62574 is to be replaced by the following reference: "O. Levenspiel and W. K. Smith, 'Notes on the Diffusion-type Model for the Longitudinal Mixing of Fluids in Flow,' Chemical Engineering Science, Vol. 6, pp. 227-233." There are no other changes to be made in Appendix A of ACNP-62574, which remains valid and is not repeated in this report.

3.1 LOCATION

The site is on the east bank of the Mississippi River, approximately 19 miles south of the city of La Crosse, Wisconsin, and 1 mile south of the village of Genoa, Wisconsin. The site is owned by the Dairyland Power Cooperative (DPC) and includes, besides LACBWR, the DPC Genoa Plant (a 14,000-kwe steam-electric power plant), a 161-kv and 69-kv transmission switching center, and a distribution center for 34.5 kv.⁴ The villages, towns, and cities within a 25-mile radius of the site are shown in Fig. 3.1.⁴

3.2 PRINCIPAL FEATURES IN THE PLANT VICINITY

3.2.1 Reactor Location Within Site

Figure 3.2 is an aerial view of the site before fill; Figure 3.2A is an elevated view of the site after fill. The details of plant location on the site are shown in Fig. 3.3. An area just south of the existing Genoa steam plant is filled in with sediment dredged from the river channel. This fill is used to raise the grade at the site to 19 ft above the normal river water level, and extends the east bank of the river approximately 300 ft to the west. The reactor is approximately 950 south of the existing steam plant, 300 ft from the bank of the river, and 475 ft west of the tracks of the Chicago, Burlington and Quincy Railroad.

3.2.2 Activities in the Immediate Vicinity

In addition to the LACBWR facilities, there is a 14,000-kwe steam-electric power plant kept on ready standby for emergency use. The site also contains DPC's 161-kv

*Other than the data from the anemometer, most of the information in Sec. 3 was provided by DPC.

and 69-kv. transmission switching center and a distribution center for 34.5 kv. Between 10 and 15 DPC employees work at these facilities. Approximately 46 DPC employees will work at the site upon reactor plant operation.

A portion of the filled-in area at the site is being reserved for future plant development. In the next five to ten years, DPC may develop conventional steam-power plants at this location.

No other activities are or will be carried on within the immediate plant site. Just outside the boundaries of exclusion area (see Sec. 3.3) are four family residences with an estimated four people in each house. The nearest is 1192 ft from the reactor.

A public parking lot of 100-car capacity and a boat launching ramp are approximately 3000 ft south of the reactor plant. An estimated 30 cars normally use the lot, although the lot may be full during peak holiday or weekend periods. The principal users of the lot are recreational fishermen and a few tourists and sightseers. The lot is entered by the river access road, which runs parallel to and just west of the railroad tracks. The road runs south from the main plant entrance for approximately 1500 ft, then southwesterly to the parking lot. (See Fig. 3.4.)

3.2.3 Traffic

The two main tracks of the Chicago, Burlington and Quincy Railroad pass the plant a minimum distance of 475 ft east of the reactor. There are normally five passenger and three freight trains per day in each direction, plus an average of four extra freights and two extra passenger trains per month. No significant increases in the amount of rail traffic are expected.

Wisconsin State Highway No. 35 runs north and south past the site, parallel to and east of the railroad tracks. The minimum distance between the traveled portion of the highway and the reactor center is approximately 550 ft. Average daily vehicular traffic past the site for Highway 35 as determined by 1962-1963 traffic count data was 1210 vehicles. Highway traffic past the plant is not expected to increase significantly in the future. The State Highway Commission has no immediate plans for new road construction. Moreover, the lack of plans for development of large resort installations or significant industry in the area makes any significant traffic increase unlikely.

U. S. Lock and Dam No. 8 is about 3/4 mile upriver (north) of the plant. River traffic past the site is best estimated from the data in Table 3-1.

TABLE 3-1

RIVER TRAFFIC PAST THE LACBWR SITE*

<u>year</u>	<u>total lockages</u>	<u>pleasure boat lockages</u>	<u>percent pleasure boats of total</u>
1952	1825	145	7.9
1953	1889	155	8.2
1954	2172	126	5.8
1955	2194	227	10.3
1956	2464	580	23.5
1957	2540	907	35.7
1958	2959	1341	45.3
1959	3620	1876	51.8
1960	4050	2200	54.3
1961	5151	3719	72.2
1962	5087	3683	72.5
1963	5034	3434	68.3
1964	5126	3036	59.3

The river is open for navigation approximately 9 months of the year. The navigation seasons from 1954 to 1962 are listed in Table 3-2.

TABLE 3-2

NAVIGATION SEASON

<u>year</u>	<u>opened</u>	<u>closed</u>
1954	Feb. 23	Dec. 2
1955	Mar. 9	Dec. 4
1956	Mar. 9	Nov. 28
1957	Feb. 28	Dec. 2
1958	Mar. 3	Dec. 4
1959	Mar. 19	Nov. 29
1960	Mar. 28	Dec. 3
1961	Mar. 6	Dec. 6
1962	Mar. 28	Dec. 9
1963	Mar. 31	Dec. 14
1964	Feb. 29	Dec. 8

*From counts taken at U. S. Lock and Dam No. 8.

3.24 Fencing

The reactor plant is enclosed by a fence running along the fill lines and along a line parallel to the railroad tracks. The main entrance to the plant area is west of the railroad crossing and immediately west of the point where the access road intersects the entrance road. The access road runs between the fence and the railroad tracks.

The main plant entrance has a remotely operated gate and a public address system. During normal plant operations, traffic will be permitted to enter or leave the site freely. No restrictions will be placed on traffic using the river access road. Plant security will be exercised at the main entrance to the office building. Under abnormal operating conditions, plant personnel will monitor and control traffic onto or from the site.

3.3 EXCLUSION AREA

The exclusion area is shown in Fig. 3.5. The minimum radius of the exclusion area is 1109 ft (0.21 miles), which is the area of access controlled by DPC. It consists of DPC property, highway and railroad right of way, and a portion of the river near the reactor. It does not include the four houses near the plant, mentioned in Sec. 3.2.2.

Property owned by DPC is shown in gray in Fig. 3.5 and includes lots 2, 5, 6, and 7. The small portion of lot 1 (which is not owned by DPC) that is within the exclusion radius is a steep bluff east of Highway 35 and not suitable for future development.

3.4 CHARACTERISTICS OF THE SURROUNDING REGION

3.4.1 Population

3.4.1.1 Distribution. The site is approximately 1 mile south of the village of Genoa, Wisconsin, and 19 miles south of the city of La Crosse, Wisconsin.

Genoa is separated from the plant site by a high bluff. Its population was 340 in 1940, 325 in 1960, and is expected to fall to 320 by 1965. This static or slightly declining population projection is due to the low economic base.

The community next nearest the site is Reno, Minnesota, 3 miles to the northwest, with a population of 25. No other population centers are within 5 miles of the site. The total estimated population within this 5-mile zone is 1076. Approximately 55 percent of this total represents a widely scattered farm population. The distribution of total population within the 5-mile zone is shown in Fig. 3.6.⁴

The population of cities and towns within 16 miles of the site are listed in Table 3-3.⁴ La Crosse, Wisconsin, the nearest sizable city, is beyond the 16-mile radius. The population of the La Crosse-Onalaska metropolitan area is 56,000.

TABLE 3-3

POPULATION OF CITIES AND TOWNS WITHIN 16 MILES OF LACBWR

<u>city or town and state</u>	<u>distance from site, miles</u>	<u>direction from site</u>	<u>population (1960 census)</u>
Genoa, Wisconsin	1	N	325
Reno, Minnesota	3	NW	25
New Albin, Iowa	5	S	568
Victory, Wisconsin	5.5	S	40
Stoddard, Wisconsin	6.5	N	450
Newton, Wisconsin	8.5	ENE	135
Chaseburg, Wisconsin	9	NE	219
Brownsville, Minnesota	9	NNW	298
DeSota, Wisconsin	10	S	367
Purdy, Wisconsin	10	ESE	35
Retreat, Wisconsin	11.5	SE	15
Eitzen, Minnesota	12.5	WSW	164
Caledonia, Minnesota	13.5	WNW	2287
Lansing, Iowa	14.5	S	1536
Fargo, Wisconsin	16	SE	10

TOTAL POPULATION WITHIN 16 MILES 6474

The total population within 10 and 25 miles of the site as estimated from 1960 census figures is summarized in Tables 3-4 and 3-5. Rural populations were estimated by counting the number of dwellings shown on U. S. Coast and Geodetic Quadrangle maps and assuming that five people dwell in each. Figures are summarized by quadrangle for the 10-mile radius and by state for the 25-mile radius. A population summary for radii of 5, 10, and 25 miles is given in Table 3-6.

TABLE 3-4

TOTAL POPULATION WITHIN 10 MILES OF LACBWR

<u>quadrangle</u>	<u>population</u>
Stoddard	3650
Brownsville	1610
Ferryville	1290
Waukon	<u>870</u>
TOTAL	7420

TABLE 3-5

TOTAL POPULATION WITHIN 25 MILES OF LACBWR

<u>state</u>	<u>population</u>
Iowa	9,800
Minnesota	19,200
Wisconsin*	<u>85,900</u>
TOTAL	114,900

TABLE 3-6

TOTAL POPULATION IN SITE VICINITY

<u>radius, miles</u>	<u>area, miles</u>	<u>population</u>	<u>population density, people/sq mile</u>
5	78.5	1,076	13.7
10	314.2	7,420	23.6
25**	1950.0	58,900	30.2
25***	1963.5	114,900	58.5

*Includes 56,000 for the La Crosse-Onalaska metropolitan area.

**Excluding La Crosse-Onalaska metropolitan area.

***Including La Crosse-Onalaska metropolitan area.

3.4.1.2 Low Population Zone Radius. A low population zone is defined in 10 CFR 100 as "that area immediately surrounding the exclusion area which contains residents, the total number and density of which are such that there is a reasonable probability that appropriate protective measures could be taken in their behalf in the event of a serious accident." For LACBWR, the area of this definition is that within 3 miles. The only population center within this area is the village of Genoa; the rest of the population is in widely scattered farms.

3.4.2 Agriculture

Wisconsin is predominantly an agricultural state. The reactor site is near the middle of the western edge of Vernon County, which is approximately 21 miles wide and a maximum of 47 miles long.

For the past ten years, Vernon County has had a yearly gross agricultural income of about \$22,000,000.⁴ Income from the various products is given below.

<u>source</u>	<u>amount*</u>
milk	\$11,991,000
cattle	2,110,000
hogs	1,175,000
chicken and eggs	1,161,000
other livestock products	479,000
crops	<u>5,436,000</u>
	\$22,352,000

The 521,688 acres in the country are used as follows:

<u>type of land</u>	<u>acreage</u>
crop land harvested	180,000
crop land pastured	42,000
other crop land	5,823
woodland pastured	134,843
woodland not pastured	27,955
other pasture land	70,913
other farm land	21,843
other land	<u>38,311</u>

*based on figures for 1956

Although most of Vernon County is agricultural, the western end along the Mississippi River is quite hilly and not used extensively for agriculture. The reactor site is in this area. Contrary to usual farming procedures in hilly areas, the uplands are the cultivated areas. The farmstead nearest the site is approximately 1 mile away and is shielded by a bluff approximately 500 ft high that runs along the river just east of the site.

There is no agriculture west of the site for a distance of at least 2-5/8 miles (the width of the river valley at the site). The western side of the river valley is a swampy wildlife and fish refuge. Data on agricultural activities further to the west, in Minnesota, have not been obtained, but lands immediately west of the reactor site appear not to be used extensively for agriculture.

3.4.3 Industry and Other Activities

There is very little industry in the vicinity of the reactor site. Two small commercial fishing operations were formerly near the site, one 1400 ft downstream and one 3200 ft upstream, but both have been purchased by DPC. U. S. Fish and Wildlife Service hatcheries are 3 miles south-southeast of the reactor.

Except for small commercial establishments, there is no industry in the village of Genoa. The nearest center of significant industry is the city of La Crosse, Wisconsin.

Since small local schools in the area have recently been abandoned in favor of centralized county schools, there is no attended public school within three miles of the site. There is a small parochial school of approximately 180 children in Genoa.

The tourist industry in Vernon County is not important. There are no established resorts in the Genoa area, and the development of large-scale permanent resort installations in the area is not expected. The nearest significant recreational boat dock is 14.5 miles downstream at Lansing, Iowa. Tourist and recreational facilities are more fully developed in the La Crosse area, where some future expansion is expected.

In general, the population and activities within Vernon County have remained stable for the past 50 years. Farming predominates in the area and is expected to remain the principal activity. Even the farm population, however, has been declining. In 1956 it was 29 percent below the 1934 level.

3.4.4 Meteorology

3.4.4.1 General Features. A recording anemometer was installed on the roof of the DPC Genoa steam plant. The wind direction and wind speed data collected are

incorporated in this report. Data from the Weather Bureau Airport Station at the La Crosse Municipal Airport were used as preliminary indications of site weather features and are included for comparison.

The similarity between topographical features at the site and at the La Crosse Airport indicated that the weather features at each place should be similar. Both areas are on the eastern side of the Mississippi River valley, west of bluffs approximately 500 ft high. The major differences are in the width of the river valley and the proximity of the eastern bluffs. The valley is approximately 5 miles wide at the airport and approximately 2-5/8 miles wide at the site. The valley wall is approximately 2 miles east of the airport station and approximately 1000 ft east of the reactor.

The 500-ft bluffs of the Mississippi River valley channel the winds. Airport data showed prevailing southerly winds for eight months of the year and prevailing north-west winds during January, February, March, and April. Site data showed that the wind blows predominately from the north-northwest or from the south-southeast throughout the year. In both cases channeling effects are apparent. However, wind speeds at the site were lower than those at the airport. In summer, a tendency toward air stagnation in the river valley causes periods of hot humid weather, and in winter, cold air drainage into the valley brings extremes of cold. Some deepening and intensification of inversion conditions may be influenced by the formation of a stagnant layer of cold air on the valley floor.

The site area exhibits a typically continental type of climate. Thus, inversion frequencies are closely related to the diurnal cycle. Nighttime inversions occur under conditions of light winds and low sky cover. These inversions tend to be broken up during the day. Inversions occur about 25 to 30 percent of the time in the spring and summer months and between 35 and 40 percent of the time in the fall and winter.² The higher inversion frequency in the fall and winter is due in part to the longer duration of the stable nocturnal periods in winter and to minimum storminess in fall.² Flow of arctic air masses over the area, with associated subsidence inversions, also tends to increase the depth and intensity of winter inversions.

3.4.4.2 Meteorological Data. A U. S. Weather Bureau station has been maintained at the La Crosse Municipal Airport since 1950. Before this, weather stations were maintained at various locations in La Crosse. The general climatological data in Table 3-10 are based mostly on data collected at the municipal airport and somewhat on data collected at other La Crosse stations. Upper air wind data were obtained from information collected at St. Paul, about 132 miles northwest of the site. Surface wind data were obtained from the La Crosse Municipal Airport. Discussions of inversion frequency are based on information in Ref. 2.

TABLE 3-7

WIND DIRECTIONAL FREQUENCY AND SPEED AT LACBWR SITE*

(Example: During October the wind is from the North 10.64 percent of the time. Its speed is < 12 mph 90.21 percent of the time, 12-24 mph 9.79 percent of the time, and its average speed is 6.20 mph.)

wind	directional frequency, % of time				0 - 12 mph, % of time				12 - 24 mph, % of time				> 24 mph, % of time				average speed, mph			
	Jan.	April	July	Oct.	Jan.	April	July	Oct.	Jan.	April	July	Oct.	Jan.	April	July	Oct.	Jan.	April	July	Oct.
N	4.31	6.83	7.71	10.64	95.08	92.63	95.50	90.21	4.92	5.26	4.50	9.79	0.00	2.11	0.00	0.00	3.41	5.21	5.07	6.20
NNE	0.42	0.86	0.76	1.64	100.00	100.00	100.00	100.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	1.83	1.08	1.73	1.95
NE	0.14	0.58	0.35	0.22	100.00	100.00	100.00	100.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	1.00	2.00	0.60	0.00
ENE	0.14	0.36	0.76	0.15	100.00	100.00	100.00	100.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	1.00	0.09	0.00
E	0.14	0.79	0.69	0.37	100.00	100.00	100.00	100.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.50	3.73	2.30	0.60
ESE	0.42	0.72	1.67	0.60	100.00	100.00	100.00	100.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.67	2.80	2.54	0.63
SE	5.93	5.04	7.71	5.58	94.05	75.71	96.39	93.34	5.95	24.29	2.70	6.67	0.00	0.00	0.90	0.00	4.96	7.69	5.45	6.44
SSE	28.04	19.71	29.38	28.20	80.61	68.98	91.25	82.59	19.14	29.93	8.75	17.41	0.25	1.09	0.00	0.00	7.36	9.44	5.60	8.16
S	11.65	15.54	13.26	14.36	93.33	74.08	96.33	88.60	6.67	25.46	3.66	11.40	0.00	0.46	0.00	0.00	4.98	8.50	4.81	6.45
SSW	3.39	7.34	4.51	4.91	100.00	89.21	98.47	95.45	0.00	10.78	1.54	4.55	0.00	0.00	0.00	0.00	3.06	5.33	2.75	3.86
SW	2.90	3.38	2.85	2.90	97.56	68.09	100.00	94.87	2.44	31.91	0.00	5.13	0.00	0.00	0.00	0.00	3.41	8.57	2.49	2.51
WSW	2.97	3.38	3.19	1.79	100.00	95.75	100.00	100.00	0.00	4.26	0.00	0.00	0.00	0.00	0.00	0.00	3.07	3.55	1.57	3.17
WNW	5.23	3.45	3.26	2.68	97.30	95.83	100.00	94.44	2.70	4.17	0.00	5.56	0.00	0.00	0.00	0.00	5.01	3.83	1.74	5.14
NW	9.11	8.85	5.00	6.03	83.72	90.24	100.00	87.65	16.28	9.76	0.00	12.35	0.00	0.00	0.00	0.00	7.67	6.36	2.96	5.37
W	4.31	3.17	2.15	1.49	100.00	97.73	100.00	95.00	0.00	2.27	0.00	5.00	0.00	0.00	0.00	0.00	3.89	2.82	1.13	3.05
NNW	20.90	20.00	16.74	18.45	69.93	73.74	83.40	78.23	30.07	21.22	15.77	21.77	0.00	5.04	0.83	0.00	8.87	9.38	7.12	7.50

*Data was taken January, April, and July 1963 - 1964, and October 1962 - 1963.

TABLE 3-8

24-HR SITE WIND DATA FOR JANUARY AND JULY OF 1963-64

<u>time interval</u>	<u>average wind speed, m/sec</u>	<u>wind from SE*, % of time</u>	<u>wind from E*, % of time</u>	<u>wind from NE*, % of time</u>
12 m. -3 a.m.	2.01	37	2.8	20
3 a.m. -6 a.m.	1.87	36	1.5	20
6 a.m. -9 a.m.	2.16	35	2.2	22
9 a.m. -12 n.	2.92	28	0.3	23
12 n. -3 p.m.	3.26	20	0.0	22
3 p.m. -6 p.m.	3.08	27	1.2	19
6 p.m. -9 p.m.	2.50	32	2.2	23
9 p.m. -12 m.	2.26	32	1.5	21

*For half-radian on either side of indicated direction

3.4.4.3 General Climatology. Temperatures in the La Crosse region are typical of the extremes of a continental climate, the extremes being more marked because of the river-valley location. Average temperatures vary from 19.2 F in the three months of winter to 71.4 F in the summer months. A maximum temperature of 108 F was recorded in July 1936, with a maximum low of -43 F recorded in January 1873.

Monthly precipitation in the area averages between 2.0 and 4.2 in. between March and October, and 1 to 2 in. for the rest of the year. Average yearly precipitation is 31.2 in. Monthly snow and sleet averages between 5 and 14 in. from November through March, the largest amount normally occurring during March. The normal annual amount of snow and sleet is 43.5 in.

3.4.4.4 Winds. Figures 3.7 and 3.8 give wind roses for La Crosse Municipal Airport and for the site. Four months of the year are used to represent the four seasons for the site. Table 3-7 gives site data and average wind speeds.

The river valley tends to channel the winds. The prevailing wind at the airport is from the south, but in winter tends to be northwesterly. Predominant winds at the site are from either the south-southeast or the north-northwest. The river valley runs almost directly south from La Crosse but bends northwest north of the city. The river valley is north-south at Genoa. The prevailing winds are therefore subject to channeling at Genoa. The channeling effects of the river valley may also be seen by comparing the surface wind roses (Figs. 3.7 and 3.8) with the upper air wind roses at St. Paul (Fig. 3.9)⁴. The upper air wind roses show a more uniform wind distribution.

Data from hourly observations at the La Crosse Airport (Fig. 3.10) during June, July, and August of 1961 indicated the wind tends to shift toward the east during the night. The anemometer data in Table 3-8 show an increased southeasterly wind frequency at night but show that the easterly shift is not as pronounced as indicated by the airport data. The shift is attributed to cold air drainage down the walls of the valley at night, which may increase the inversion frequency.

3.4.4.5 Inversions. The percentage of inversion frequencies for the entire United States is plotted as isopleths in Fig. 1 A-E of Ref. 2. These data are summarized below for the La Crosse area.

TABLE 3-9

INVERSION FREQUENCY FOR LA CROSSE AREA

<u>season</u>	<u>inversion frequency (% of total hours)</u>
winter	35 - 40
spring	30
summer	25 - 30
fall	35 - 40
ANNUAL	30 - 35

The La Crosse area is within the general Central Plains delineation of Ref. 2 and has a continental type of climate. The tendency for continental areas is low-level stability at night and instability during the day. Inversion frequencies are highest for the La Crosse area during fall and winter. This may be from causes such as: (1) the nocturnal period is longer in winter, and (2) fewer storms in the fall mean lack of highspeed winds that break up inversions.

The site is in an area influenced by arctic air masses during the winter. The effects of strong radiation cooling associated with snow cover and of subsidence inversions associated with cold air masses could produce deeper and more intense surface inversions in the winter. During late winter and early spring, especially with snow on the ground, the advection of warmer air from the south could increase low-level inversions. However, this condition would be the initial stage of a warm front that would eventually produce low-level instability by storms and precipitation.

TABLE 3-10

CLIMATOLOGICAL SUMMARY

METEOROLOGICAL DATA FOR THE CURRENT YEAR																																				
MONTH	TEMPERATURE							Degree days	PRECIPITATION						RELATIVE HUMIDITY			WIND					Percent of possible sunshine	Average sky cover sunrise to sunset	NUMBER OF DAYS											
	AVERAGES			EXTREMES					Total	Greatest in 24 hrs	Date	SNOW, SLEET			Midnight CST	6:00 A.M. CST	Noon CST	6:00 P.M. CST	Average hourly speed	Prevailing direction	FASTEST MILE				SUNRISE TO SUNSET			TEMPERATURE								
	Daily Maximum	Daily Minimum	Monthly	Highest	Date	Lowest	Date		Total	Greatest in 24 hrs	Date	Total	Greatest in 24 hrs	Date	Midnight CST	6:00 A.M. CST	Noon CST	6:00 P.M. CST	Average hourly speed	Prevailing direction	Speed	Direction			Date	Clear	Partly Cloudy	Cloudy	Precipitation 0.01 in. or more	Snow, Sleet 1.0 in. or more	Thunderstorms	Heavy fog	90° and above	32° and below	32° and below	0° and below
JAN	14.8	-2.2	6.3	43	8	-31	15	1819	0.67	0.42	11-12	13.3	8.7	11-12	70	73	63	66	9.8	S	35	NW	20	7.3	5	6	20	10	3	0	0	0	28	31	20	
FEB	24.9	4.6	14.8	40	17+	-15	21	1405	0.49	0.22	27	5.3	2.6	27	80	74	59	61	10.9	S	37	NNW	20	6.0	9	8	11	6	1	0	2	0	21	28	11	
MAR	45.1	26.0	35.6	76	30	3	9+	906	2.25	0.71	25-26	14.3	8.2	5	77	79	64	63	10.7	S	28	NW	20	6.9	6	8	17	14	3	1	1	0	3	20	0	
APR	59.8	38.2	49.0	78	16	27	4	475	2.46	0.69	18	0.7	0.7	22	73	73	45	49	11.9	S	41	SSW	3	5.9	9	7	14	10	0	3	0	0	0	8	0	
MAY	67.6	47.3	57.5	85	7	32	1	250	2.04	0.81	12	0.0	0.0		73	77	51	51	11.2	SE	52	ESE	10	6.7	5	11	15	7	0	4	0	0	0	1	0	
JUN	83.3	59.3	71.3	98	30	45	12+	34	2.35	0.69	8-9	0.0	0.0		79	82	50	47	9.2	S	63	NNW	8	5.3	8	15	7	9	0	10	1	8	0	0	0	
JUL	84.6	62.8	73.7	99	1	52	4	1	5.14	2.14	12-13	0.0	0.0		86	86	51	52	8.6	SE	35	NNW	16	4.9	11	13	7	9	0	5	2	4	0	0	0	
AUG	78.7	59.1	68.9	88	7	46	18	29	3.40	1.32	23	0.0	0.0		90	91	59	57	7.8	S	63	NW	2	5.8	8	12	11	10	0	8	9	0	0	0	0	
SEP	72.6	53.1	62.9	85	18	37	13	114	3.92	1.40	18-19	0.0	0.0		92	94	65	67	8.1	S	35	NE	11	6.5	7	11	12	8	0	4	6	0	0	0	0	
OCT	71.3	51.1	61.2	93	6	35	29	151	1.46	0.68	24-25	0.0	0.0		79	84	55	58	8.3	S	38	NW	2	4.9	14	7	10	6	0	4	3	1	0	0	0	
NOV	48.8	33.1	41.0	62	3	17	-30	714	1.90	1.61	21-22	0.1	0.1	22	79	83	64	67	11.0	S	32	NW	1	6.8	6	8	16	5	0	2	3	0	2	14	0	
DEC	19.7	5.2	12.5	46	7	-18	20	1623	0.49	0.18	11-12	7.6	2.8	11-12	78	75	63	67	9.4	S	26	NNW	9	5.9	11	6	14	8	3	0	0	0	25	31	14	
YEAR	55.9	36.5	46.2	99	JUL 1	JAN -31	15	7521	26.57	2.14	JUL 12-13	41.3	8.7	JAN 11-12	80	81	57	59	9.7	S	63	NW	2+	6.1	99	112	154	102	10	41	27	13	79	133	45	

NORMALS, MEANS, AND EXTREMES

	TEMPERATURE							Normal degree days	PRECIPITATION										RELATIVE HUMIDITY				WIND				Pct. of possible sunshine	Mean sky cover sunrise to sunset	MEAN NUMBER OF DAYS																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																														
	NORMAL			EXTREMES					Normal Total	Maximum monthly	Year	Minimum monthly	Year	Maximum in 24hrs.	Year	SNOW, SLEET					Midnight CST	6:00 A.M. CST	Noon CST	6:00 P.M. CST	Mean hourly speed	Prevailing Direction			FASTEST MILE*			SUNRISE TO SUNSET			Precipitation 0.01 in. or more	Snow, Sleet 1.0 in. or more	Thunderstorms	Heavy fog 90° and above	TEMPERATURES																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																				
	Daily maximum	Daily minimum	Monthly	Record highest	Year	Record lowest	Year									Mean total	Maximum monthly	Year	Maximum in 24 hrs.	Year									Mean total	Maximum monthly	Year	Maximum in 24 hrs.	Year	Speed					Direction	Year	Clear	Partly Cloudy	Cloudy	90° and above	32° and below	32° and below	0° and below																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																																												
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Means and extremes in the above table are from the existing location (or last comparable location). Annual extremes have been exceeded at prior locations as follows: Highest temperature 108 in July 1936; lowest temperature - 43 in January 1873; maximum monthly precipitation 12.09 in October 1900; minimum monthly precipitation 0.01 in December 1943; maximum precipitation in 24 hours 7.23 in October 1900; maximum monthly snowfall 39.6 in January 1929; fastest mile of wind 69 from Southwest in October 1949.

- (a) Length of record, years.
- (b) Climatological standard normals based on the period 1931-1960.

*Data is from "Local Climatological Data with Comparative Data", U. S. Department of Commerce, La Crosse, Wisconsin, 1958.

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Inversion conditions normally occur at night throughout the year. Data in Ref. 2 show that for any inland site the probability of inversions for more than 20 hr is less than 0.1 percent. For periods longer than the nocturnal period, inversions are significantly shorter. These statements also hold for the river valley. Daytime heating of the valley walls and floor tends to create low-level instability.

How much the river valley may increase the percentage of inversion frequencies is uncertain. From experience, major parameters favoring nocturnal radiation-inversions are light winds and clear skies. The amount of sky cloud cover is relatively independent of the river valley, although light winds during the night are favored by the valley location. Data from Ref. 2 give percent frequencies of nighttime wind speeds below 7 mph for the general La Crosse area and for the La Crosse airport station. These data are summarized below.

TABLE 3-11

FREQUENCY OF NIGHT WINDS OF 7 mph OR LESS, PERCENT

<u>season</u>	<u>general area</u>	<u>La Crosse airport station</u>	<u>site</u>
winter	~ 40	44	60
spring	~ 30	39	—
summer	50 - 60	66	83
fall	~ 40	46	—

There is an increase in the frequency of low winds within the river valley, especially at the reactor site. This indicates a higher inversion frequency at the site than the 30 - 35 percent given for the La Crosse area in Table 3-9.

The drainage of cold air into the valley at night can be expected to intensify surface-based inversions. However, inversions at ground level in the site vicinity should be more influenced by sky cover and wind-speed conditions. During the day, inversions should be broken up by solar heating and warm air currents up the valley walls. One effect of the deepened inversion is a prolonging of fumigation during early morning hours. Inversion lids could form over the valley for as long as 1 hr.

3.4.4.6 Tornadoes. Tornadoes are less likely to occur in Wisconsin than in states to the southwest, south, and the immediate southeast.³ Tornado occurrence at the site is higher than for the state as a whole. Figure 4 of Ref. 3 presents tornado zones in the U. S., based on data collected between 1920-49, in terms of the number of tornadoes reported in each 50 mile square area. The region of the most tornadoes reports more than 15 tornadoes per 50-mile square. For the same period, the La Crosse area

reported approximately 10 tornadoes per square. From these data and from the fact that the average tornado involves an area of 3 square miles, there is an annual probability of 1/2000 for a tornado to strike a given square mile in the La Crosse area. The chance that a tornado would pass directly over the reactor site is even smaller.

3.4.5 Geology and Hydrology

At the request of the U. S. Atomic Energy Commission, a report entitled "Geology and Hydrology of a Reactor Site Near Genoa, Vernon County, Wisconsin" has been prepared by Alfred Clebsch, Jr., and Eric L. Meyer of the U. S. Department of the Interior Geological Survey, Water Resources Division. This report is included as Appendix A of ACNP-62574. The data and conclusions given below pertaining to the geologic and hydrologic characteristics of the site are based on the information in that report.

3.4.5.1 Topographic Features. The reactor site is in the Mississippi River valley. In the vicinity of the site, this valley is deeply cut into highly dissected uplands. From La Crosse to Prairie du Chien, the valley varies between 2-1/2 and 4-1/2 miles in width. The valley walls rise sharply 500 to 600 ft from river level.

There is little or no agricultural use of the river valley floor, which consists primarily of marshy land, islands between river channels, and extensions of a low-lying flood plain cut by ponds, sloughs, and meandering stream channels. Numerous short, steep-sided valleys that have been cut into the uplands by tributary streams intercept the main river valley. The valley walls of both the main and tributary channels are wooded. The flat upland areas and some of the tributary valleys are cultivated and grazed.

The main channel of the river varies greatly in width above and below the site. A series of dams are operated by the U. S. Army Corps of Engineers for navigational purposes. Above dam No. 8 (about 3/4 mile north of the site), the river is nearly four miles wide. Between dam No. 8 and the site, the river is 1500 to 2000 ft wide. Below the site, the river is relatively narrow for a distance of 20 miles, then gradually widens as the river approaches dam No. 9, 33 miles south of the site.

3.4.5.2 Drainage. The site is on a filled-in area south of the Genoa steam plant. Therefore drainage at the site must be provided. There is allowance for runoff from the high valley walls to the east. The site is favorably located with respect to this runoff, however, because of two short valleys east of the bluffs bordering the site. One valley drains to the north and one to the south, so that only precipitation that falls on the bluff adjacent to the site and on a small portion of the upland area

contributes to runoff directly across the site. This runoff is presently channeled along the highway and railroad to prevent interference with traffic. No problems of flash floods are foreseen at the site.

3.4.5.3 Downstream Water Use. For a distance of 40 miles downstream of the site, virtually all municipal water supplies for cities and towns along the river are obtained from ground water. On the basis of readily available published records, the nearest major city using the river water for direct human consumption is Davenport, Iowa, about 195 miles downstream. The nearest user of river water for industrial purposes is the steam-power plant in Lansing, Iowa, about 15 miles downstream. River water is used at this plant for condenser cooling. There is no other known user of river water for industrial purposes between the reactor site and Prairie du Chien, 40 miles downstream.

3.4.5.4 Streamflow.

3.4.5.4.1 Flood Frequency. The magnitude and probability of Mississippi River floods at the site have been estimated. From data recorded since 1873, the annual probability of a flood having a total river discharge of 220,000 cfs is 1/100. There is a lower probability for floods of greater magnitude.

River water level rise at the site from a flood of 220,000 cfs was computed from water level records of the U. S. Army Corps of Engineers at Genoa. For this condition, the water level at the site is estimated to be 635.2 ft. The highest flood since 1873 made the maximum water level at Genoa 634.5 ft, for an estimated water level of 634.4 ft at the site. The grade elevation at the filled site is 639 ft. The annual probability of a flood of sufficient magnitude to raise the river-water to the fill level is less than 1/200.

This information is based on data contained in Appendix A of ACNP-62574, which covers the period from 1873 to 1962.

3.4.5.4.2 Magnitude and Duration of Low Flow. The degree of river contamination from the release of radioactivity from the site is more dependent on river flow than on any other hydrologic condition and is highest for a low-flow condition. Low flow at the site occurs in fall and winter; the most frequently recorded lowest monthly average flow occurs in February. Minimum flows have also been recorded in August and September, during drought. Records of minimum and average flows maintained over the period of 1930-1955 at the U. S. Geological Survey station at La Crosse are summarized in Table 3-12. Low flows at the site should vary only slightly from those at La Crosse.

TABLE 3-12

SUMMARY FLOW DATA FOR MISSISSIPPI AT LA CROSSE STATION
(1930 - 1955)

<u>condition</u>	<u>discharge, cfs</u>
all-time low flowrate (Dec. 30 and 31, 1933)	3,200
median of annual minimum flowrates (averaged over 1 day)	8,100
average monthly flowrate (exceeded in 95% of months)	9,000
overall average flowrate, 1930-1955	27,970

3.4.5.5. Dilution of Possible Contaminants. When contamination is continuously added to a stream, the concentration at a point downstream approaches the ratio of the rate of contaminant addition and the rate of stream flow. This ratio would be the concentration of contaminants uniformly mixed with stream flow south of the site. Uniform mixing is estimated to be essentially complete at between 10 and 30 miles downstream. This is well north of the nearest known user of the river water for human consumption. The maximum continuous release of radioactivity from the reactor site would occur from the maximum credible accident (see Sec. 14.3.18). The contamination of the Mississippi River from this release is discussed in Sec. 14.3.15.1.

If contamination is added to the river as a slug, either instantaneously or over a few hours, the concentration downstream will be a function of distance and time after introduction. Conservative estimates have been made of the maximum concentration per curie released for McGregor, Iowa, approximately 40 miles downstream (see Appendix A of ACNP-62574). For a river flow of 8000 cfs, this value equals 8×10^{-7} $\mu\text{c/ml/curie}$ released. The effects of the worst slug release are also evaluated in Sec. 14.3.15.1.

3.4.5.6. Geology and Ground Water. A description of the subsurface geology near the site has been obtained from rock samples collected during well drilling for the hatchery of the U. S. Fish and Wildlife Service. Valley sand fill extends to a distance of approximately 170 ft. Eau Claire sandstone of Cambrian age underlies the valley fill to a depth of approximately 300 ft and is underlain by Mt. Simon sandstone extending to a measured depth of 635 ft. Areas close to the site but not within the river valley are characterized by a sublayer of Mt. Simon sandstone. Other formations overlie the Mt. Simon in the following order: The Eau Claire, Galesville, and Franconia Sandstones, and the Trempealeau formation.

Deep wells penetrating the Mt. Simon layer flow at the surface, indicating an artesian head above the level of the river valley floor. Use of water from these artesian

aquifers has been limited because the chemical quality of this deep water is poorer than that from shallow aquifers. As a result, there has been no extensive withdrawal of water and no serious decrease in artesian head. Therefore, accidentally released contaminants cannot enter the artesian aquifer.

Information obtained from five domestic wells east of the site indicate that the water table is near the surface and that the gradient is toward the river. It is not considered possible that pumping from wells in the plant vicinity could have depressed the water table at the site and caused a reversal in the water table gradient. Thus, spillages occurring at the site would percolate towards the river rather than towards shallow wells in the site vicinity.

3.4.5.7 Seismicity. The site is in a tectonically stable area where no significant earthquakes have occurred during the history of earthquake observation. The Uniform Building Code, 1958 edition, includes the site in Zone O (no damage).

REFERENCES

1. "Local Climatological Data with Comparative Data," U. S. Department of Commerce, La Crosse, Wisconsin, 1963.
2. C. R. Hosler, "Low Level Inversion Frequency in the Continental U. S.," Monthly Weather Review, Vol. 89, Sept. 1961.
3. S. D. Flora, Tornadoes of the United States (rev. ed.), University of Oklahoma, Norman, Oklahoma, March 1954.
4. Small Pressurized Water Reactor, Dairyland Power Cooperative Site Report, Gibbs and Hill, Inc., July 18, 1960.

General Site Location Map

FIG. 3.1

AERIAL VIEW OF SITE BEFORE FILL

FIG. 3.2

ELEVATED VIEW OF SITE AFTER FILL

FIG. 3.2A

FIG. 3.3

Access Map

FIG. 3.4

Exclusion Map

FIG. 3.5

Population Distribution

FIG. 3.6

LEGEND:
 ○ per cent calms
 □ 0-12 mph
 ▤ 12-24 mph
 ■ 24 & above

NOTE:
 La Crosse Municipal Airport located approximately 19 miles north of site.

Yearly Variation in Surface Winds, La Crosse Municipal Airport

FIG. 3.7

SEASONAL VARIATION IN SURFACE WINDS, LACBWR SITE

FIG. 3.8

500 m
above sea level

Dec.-Jan.-Feb.

1000 m
above sea level

500 m
above sea level

Mar.-Apr.-May

1000 m
above sea level

500 m
above sea level

June-July-Aug.

1000 m
above sea level

500 m
above sea level

Sept.-Oct.-Nov.

1000 m
above sea level

LEGEND:
 ○ per cent calms
 — 2-7 meters/second
 — 8-14 meters/second
 — 14 & above meters/second

Yearly Variation in Upper Air Winds, St. Paul, Minnesota

FIG. 3.9

24-hr Variation in Surface Winds During June, July, and August 1961, La Crosse Municipal Airport

FIG. 3.10

4. REACTOR

Sections 4 and 5 of this report supersede Section 4 of ACNP-62574, "Hazards Summary Report for Construction Authorization of the LACBWR." Changes from the preliminary design concept are within the overall LACBWR design objectives and criteria and have not compromised plant reliability or safety. Other than final design concepts for the control-rod-drive mechanisms (Sec. 8.5), steam separators, steam dryers, and the failed-fuel-element location and emergency core spray assemblies (all of which were derived from the research and development programs), the only significant design change is the substitution of boron carbide (B_4C) in place of the silver-indium-cadmium alloy for the control-rod absorber material. This change was made primarily because of problems of procurement for silver-indium-cadmium control rods; the boron carbide absorber material has performed satisfactorily in operating reactors and does not cause design complications for LACBWR.

The final reactor design and the final design calculations have incorporated a number of revisions to the original design (refer to ACNP-62574) and performance characteristics. The final design and performance characteristics are described in the following subsections.

4.1 GENERAL DESCRIPTION

The major components of the reactor assembly (see Fig. 4.1) are as follows: reactor vessel, core support skirt, plenum separator plate, bottom grid assembly, thermal shield, fuel element shrouds, vertical posts, control rods, fuel elements, steam separators, thermal shock shield, flow deflector, emergency core spray ring, and steam dryer. The pressure vessel is a right cylinder with a removable top head for core access. The four inlet nozzles in the bottom head receive incoming coolant, and the four outlet nozzles in the cylindrical section of the vessel discharge the recirculated water back to the coolant loops. Two steam nozzles near the top of the cylindrical portion of the vessel discharge the separated steam.

The thermal shield is supported by a ring that is an integral part of the vessel wall. The thermal shield reduces the level of gamma radiation heating at the vessel wall. The core contains the fuel and consists of fuel elements, control rods, fuel element shrouds, and vertical posts. The fuel elements are supported by the fuel element shrouds, each fuel element nozzle fitting into the bottom flange of the fuel element shroud. This arrangement permits individual fuel assemblies to be installed and removed without disturbing adjacent fuel assemblies, and provides for rearrangement of fuel elements to equalize fuel burnup and prolong fuel lifetime. Core reactivity is controlled by 29 control rods. The control rods are inserted from the bottom of the core by control-rod-drive mechanisms operating through the control-rod nozzles. Core reactivity decreases with rod insertion and increases with withdrawal.

4.2 REACTOR VESSEL

4.2.1 Design Criteria

The vessel is designed, constructed, and tested in accordance with the requirements of Section VIII of the ASME Boiler and Pressure Vessel Code (1962 Edition), and applicable nuclear code cases, and is stamped in accordance with Code Case 1270 N.

In addition, the vessel has been stress analyzed in accordance with Tentative Structural Design Basis for Reactor Pressure Vessels and Directly Associated Components (Dept. of Commerce, No. PB-151987). Primary stresses are limited by the ASME Code. The sections of the vessel have been designed so that the primary stresses do not exceed the "S" values of Table UCS-23 in Section VIII of the ASME Code.

In the case of structural discontinuities, the secondary stresses are such that the sum of the primary and secondary stresses does not exceed 90 percent of the yield strength of the material or 60 percent of the specified minimum ultimate strength at the temperature condition considered.

There is no absolute limit on stress concentration. Stress concentration will be considered as a fatigue problem, and the effects will be considered in a fatigue analysis. The stress concentration is limited on a fatigue basis to an accumulative use factor of 0.8. A design objective has been to keep the stress concentrations at a minimum.

For multiaxial stress conditions, the Maximum Shear Stress Theory has been used to calculate the maximum stress intensity at a given point. The following terms are as defined in the Bureau of Ships Code, A Tentative Design Basis for Nuclear Reactor Vessels and Associated Components: stress intensity, structural discontinuity, stress concentration, and primary and secondary stress.

All construction materials conform to the requirements of applicable ASTM specifications and to Allis-Chalmers Specification 41-473 (see Appendix A). A summary list of stress analyses and a tabulation of results from code calculations and stress calculations are included in Appendix A. This appendix also contains material procurement specifications for pressure vessel and internals fabrication requirements and acceptance standards, along with fabrication drawings giving the location and details of major vessel components and welding joints.

The vessel design requirements are given in Table 4-1.

TABLE 4-1
VESSEL DESIGN REQUIREMENTS

Pressure Vessel - Design and Operating Conditions*

Internal pressure (psia)	1415 (1300)
Temperature (° F)	650 (577.5)
Incident neutron flux on pressure vessel at core centerline (neutrons/cm ² -sec, > 1 Mev)	1.9×10^{10} (1.53×10^{10})
Design lifetime (years)	20

Weight Loads (tons)

Core	22
Core support structure	6
Water during refueling	143
Control rods and drives	9
Thermal shields and attached components	35

Dynamic Loads

Vertical load at each control-rod nozzle during reactor scram (10 g deceleration), lb	5000
Time for application of scram load, sec	0.150
Seismic loading conditions for U. S. Coast and Geodetic Survey Major Damage Classification, Zone 0	In accordance with Uniform Bldg. Code (1958 Edition)

External Pipe Reactions

	<u>8-in. nozzle</u>	<u>16-in. nozzle</u>
Axial load	8,000 lb	11,000 lb
Bending moment	90,000 in.-lb	160,000 in.-lb

* Numbers in parentheses denote operating conditions.

Primary System Transients

(a) Normal Startup-Shutdown Cycle*

Total number of cycles:	150
Temperature	
range:	120 F to 577.5 F
maximum rate:	100 F/hr startup 150 F/hr shutdown
Pressure	
range:	14.7 psia to 1300 psia
maximum rate:	Corresponding to 150 F/hr change in saturation temperature

(b) Steady State Operation - Normal Load Change

Number of cycles for load increase:	7300
Number of cycles for load decrease:	7300
Pressure	
range:	
load decrease	1300 - 1325 psia
load increase	1300 - 1275 psia
rate:	2 psi/min
Temperature:	Ranges and rates correspond to changes in saturation tempera- ture for the above pressure changes

(c) Reactor Scram

Number of cycles:	250
Pressure	
range:	1300 - 1100 psia
rate:	20 psi/sec
Temperature:	Ranges and rates correspond to changes in saturation tempera- ture for the above pressure changes

* Coolant in flange region will be steam.

(d) Maximum Credible Accident

Number of cycles:

1

Pressure

range:

1300 - 65 psia

rate:

200 psi/sec

Temperature

range:

577.5 F - 270 F

rate:

20 F/sec

(e) Turbine Trip

Number of cycles:

60

Pressure

range:

1300 - 1340 psia

rate:

21 psi/sec

Temperature:

Range and rate correspond to changes in saturation temperature for the above pressure changes

(f) Bypass Valve Opening (Fails fully open and turbine stop does not close)

Number of cycles:

20

Pressure

range:

1300 - 1100 psia

rate:

20 psi/sec

Temperature:

Range and rate correspond to changes in saturation temperature for the above pressure changes

4.2.2. Vessel Description

The reactor pressure vessel (Fig. 4.2) provides containment for the primary system pressure and receives and positions the reactor core and internal structures.

The reactor pressure vessel consists of a cylindrical, plate-welded shell section with a formed integral hemispherical bottom head and a removable hemispherical top head which is bolted to a mating flange on the vessel shell to provide for vessel closure. The vessel has an overall inside height of 37 ft, an inside diameter of 99 in., and a nominal wall thickness of 4 in. (including 3/16 in. of integrally-bonded stainless-steel cladding).

The vessel is supported by a structure at the lower end of the vessel shell near the forced-circulation nozzles. The vessel cylindrical shell and bottom head contain penetrations for the following: forced-circulation coolant inlet and outlet, steam outlet, experimental access, emergency cooling, liquid level, control-rod drive mechanisms, in-core flux monitoring, and blowdown. The vessel closure head contains penetrations for experimental access, instrumentation access, vent line nozzles and liquid level probes. There are also penetrations for the failed-fuel-element location system at the flange connecting the cylindrical portion and the top head. Figure 4.2 lists all penetrations. Thermal sleeves are provided in the forced-circulation inlet nozzles and the emergency cooling water inlet nozzle as protection from excessive thermal stress. A ring located on the inner circumference of the vessel between the inlet and outlet forced-circulation nozzles forms an integral part of the vessel wall and supports the vessel internals structure. Support pads are provided in the bottom head to support the core structure.

The reactor vessel is of ferritic steel (ASTM A-302-GR-B plates), with integrally bonded Type-304L stainless-steel cladding. The flanges and large nozzles are ferritic steel (ASTM A336) forgings, according to special Code Case 1236-1. These forgings and vessel seams are weld-overlayed with stainless steel of a surface chemistry equivalent to Type-304 stainless steel, except that the overlay is stabilized with a small amount of columbium.

The blowdown and emergency cooling nozzles are fabricated of the same forging material as the large nozzles and are lined with an Inconel sleeve welded to the forging. The other small nozzles are made of Inconel pipe, attached to the vessel with a partial-penetration-type weld.

Each nozzle is provided with a transition piece of the same material as the pipe to which it is attached in the field.

The shell flange forging is machined for seal closure and head bolting. The closure head consists of a formed center section and a forged flange.

The reactor vessel shell exterior surface is covered with a 2-in. layer of fiberglass thermal insulation. The fiberglass insulation for the vessel closure head is jacketed in carbon steel for removal as a unit, to permit access to the studs.

The vessel closure is of the bolted, double-gasket type. The two concentric soft iron core gaskets are jacketed with austenitic stainless steel for corrosion protection. The space between the gaskets is connected to a fluid leakoff line for determining the eventual leakage past the inner gasket. The closure head is secured with studs that are prestressed to a predetermined load by hydraulic bolt tensioners.

Thermocouples are mounted on the exterior surface of the vessel shell and closure head for monitoring material temperatures during operation.

4.3 REACTOR INTERNALS

The reactor internals consist of the following: a thermal shield, a core support skirt, a plenum separator plate, a bottom grid assembly, steam separators, a thermal shock shield, a steam dryer with support structure, an emergency core spray and failed-fuel-element location assembly, and the reactor core. A summary of stress analyses for main components that are subjected to significant loads is included in Appendix A. A description of the internals follows.

4.3.1 Thermal Shield (Figs. 4.3 and 4.4)

The main function of the thermal shield is to reduce the gamma radiation on the reactor vessel wall. The thermal shield is a cylinder, 95-in. in ID and 1.25 in. thick. The shield extends from the support lug on the vessel wall (61.5 in. above the bottom apex of the reactor vessel) to an elevation 240 in. above the apex.

When placed in the vessel, the thermal shield provides a minimum annulus of 0.5 in. between the outside of the shield and the inside of the vessel wall. At an elevation of 38.8 in. above the bottom of the thermal shield, an annular ring plate is provided to support and position the steam separators and to provide a seal that prevents recirculating flow from bypassing the steam separators. At an elevation of 201.5 in. above the bottom apex, lugs are provided to support the eight tie rods that laterally support the core lateral support structure. At the top of the shield, dowels are provided to position the steam separators and to align the thermal shock shield. Tapped holes are provided to receive the bolting of the thermal shock shield and the radial guide supports for the flow deflector. Two rows of 16 equally-spaced 4-in. diameter holes, 38.8 and 177.6 in. above the bottom of the shield, provide flow to cool the vessel wall. Four 15.87-in.-diameter holes, 13 in. above the bottom of the shield, pass recirculating flow to the outlet nozzles. All components of the thermal shield are of AISI Type-304 stainless steel. The thermal shield is aligned by four 19-9 DL alloy dowels, located off the support lug attached to the vessel wall.

4.3.2 Core Support Skirt (Fig. 4.5)

The primary function of the core support skirt is to transmit the weight of the core and core structure to the vessel bottom head. It also provides flow separation between the inlet and outlet plenum, and a proper flow distribution to the fuel assemblies. In addition, the core support skirt supports the plenum separator plate.

The skirt is a cylinder, 68 in. in ID and 1 in. thick. The core support skirt is bolted to the 12 Inconel supports that are welded to the inside of the vessel bottom head 21.5 in. above the vessel bottom apex. The skirt extends 76.25 in. above this elevation. A 2.5-in.-thick AISI Type-304 stainless steel flange at the top of the skirt receives the bolting of the bottom grid assembly. At an elevation of 40 in. above the bottom of the skirt, there is a 1.5 in.-thick stainless steel support ring to support the inner diameter of the plenum separator plate. An Inconel flange at the bottom of the skirt is used in bolting the skirt to the support pads. Twelve cutouts, equally spaced about the lower periphery of the skirt, receive incoming coolant flow from the inlet nozzles. A 1/2-in.-thick cylindrical baffle extends around the inside bottom of the skirt to channel the flow through the cutout spaces. The baffle is perforated with 353 1-in.-diameter holes that provide proper flow distribution to the fuel assemblies.

All components of the core support skirt are fabricated of AISI Type-304 stainless steel, except for the bottom flange, which is Inconel. The core support skirt supports the bottom grid assembly, fuel assemblies, vertical posts, fuel element shrouds, and the core lateral support structure. The finished weight of the core support skirt is approximately 6200 lb.

4.3.3 Plenum Separator Plate (Fig. 4.6)

The plenum separator plate provides a flow seal between the inlet and outlet plenums of the reactor vessel. The plate is bolted about the outside periphery to the vessel support ring 61.5 in. above the vessel bottom apex, and about the inside circumference to a rim on the core support skirt at an elevation 0.025 in. higher than the outside support. This arrangement minimizes any leakage between the inlet and outlet plenums. The plenum separator plate is a 1/2-in.-thick annular plate with a 70.5 in. ID and a 98.2 in. OD. The plate is fabricated entirely of Inconel.

4.3.4 Bottom Grid Assembly (Fig. 4.7)

The bottom grid assembly receives and supports the vertical posts, fuel element shrouds, fuel assemblies, and the core lateral support structure and transfers the weight of these components to the core support skirt. In addition, the bottom grid assembly provides a guide for the control rods and allows the proper amount of coolant flow through the control-rod channels.

The bottom grid assembly consists of top and bottom plates, 77.2 in. apart, connected by 29 cruciform-shaped guides and eight cylindrical tie rods. The cruciform guides transmit the weight of the supported components of the bottom plate and protect the control rods from lateral coolant flow that might cause excessive control-rod vibration.

The eight tie rods transmit the load from the bottom plate to the outer periphery of the top plate near the support. The load is thus transmitted to the core support skirt. The tie rods consist of Schedule-160, 4-in. stainless-steel pipes equally spaced about the outer periphery of the bottom grid assembly. Threaded plugs are at both ends of the tie rods. Each upper tie rod plug is threaded into a nut which is welded on the underside of the upper plate. Each lower plug is inserted through holes provided in the lower plate. It is fastened to the underside of this plate by a nut.

The top plate of the bottom grid assembly (see Fig. 4.8) is 70.75 in. in diameter and 1-3/4 in. thick. It has 29 cruciform-shaped openings (see Fig. 4.9) for control-rod passage and 80 circular openings to receive and position the fuel element shrouds (and fuel assemblies). Four dowels of 19-9 DL alloy are provided at each fuel element shroud location for holddown purposes. In addition, AISI Type-304 stainless steel dowels at the 29 control-rod locations align the control-rod guides. Tapped holes are provided for bolting of the guides and the core lateral support structure to the top plate. Additional holes receive the 52 vertical posts.

The bottom plate of the assembly (see Fig. 4.10) is a 2-1/4-in.-thick plate, 63 in. in diameter, reinforced by a grillage of 1-in.-by-6-1/4-in. intersecting beams. Forty-four 3-1/2-in.-diameter holes in the bottom plate admit coolant flow, and nine holes of 0.687 in. diameter receive piping for the in-core monitoring system.

Control-rod scabbards (see Fig. 4.11) in the bottom plate admit the proper amount of flow to the control-rod channels and also provide a bearing surface for the control rods. Tapped holes and dowels are provided in the bottom plate to align the control-rod guides. Nine guide tubes extend between the two plates to permit installation and to support the in-core monitoring tubing. All components of the bottom grid assembly are fabricated of AISI Type-304 stainless steel. Dowels and bolting are of 19-9 DL alloy.

The bottom grid assembly is fastened to the core support skirt about the outside circumference of the top plate with ten 1.25-in. stainless-steel socket head cap screws and two 1.0 in. dowels of 19-9 DL alloy. In position, the bottom grid assembly is enclosed within the core support skirt. The finished weight of the assembly is approximately 10,000 lb.

4.3.5 Steam Separators (Fig. 4.12)

An estimated 50 percent of the generated steam is separated at the two-phase interface. Sixteen centrifugal-downflow-type steam separators mounted around the inside of the thermal shield separate the rest of the steam and limit the amount of steam carry-under into the recirculation pumps. Each steam separator is essentially a 10-in. tube, 196.5 in. long, of 11-gauge (0.12 in.) AISI Type-304 stainless steel.

The two scroll-shaped inlets, through which the steam-water mixture enters, open to tangential slits in the tube, 1.1 in. wide and 72 in. long. The top of the steam separator (above the inlets) has a 10/5 reducer and a section 5 in. in diameter and 31.5 in. long through which the separated steam is discharged. The bottom water outlet nozzle is cone-shaped, tapering from a 10.0-in. to a 7.25-in. diameter in its 12-in. length. The flow is discharged partly through the cylindrical bottom opening and partly through six triangular side openings along the length of the cone. The overall length of each of the steam separators is 242 in. Except for a small percentage of flow bypassing the outer periphery of the core, the entire recirculation flow passes through the steam separators. A discussion of the steam separator research and development program is found in Amendment No. 6 to Application for Construction Authorization (ACNP-64572).

4.3.6 Thermal Shock Shield (Fig. 4.13)

The thermal shock shield provides a water seal between the saturated steam above the two-phase interface and the dry steam discharge annulus and also supports the steam dryer and its support structure. It is a right cylinder that extends from 240 in. (the top of the thermal shield) to 340.25 in. above the bottom apex. The thermal shock shield has a 97-in. ID and a 1/4-in.-thick wall.

Eight 1-in.-thick lugs at the bottom of the shock shield are used to bolt the shock shield to the thermal shield. A flange 1.5 in. thick at the top of the shock shield receives the bolting to the steam dryer support structure. Eight vertical reinforcing ribs about the inner periphery of the thermal shock shield increase the rigidity of the structure. Slots at the bottom of the shield permit the four flow deflector support guides and the sixteen steam separator guides to extend from the thermal shield. The finished weight of the thermal shock shield is approximately 4000 lb. All components of the thermal shock shield are AISI Type-304 stainless steel. Bolts and dowels are of 19-9 DL alloy.

4.3.7 Steam Dryer and Support Structure (Fig. 4.14)

The steam dryer support structure, bolted to the thermal shock shield 340.25 in. above the vessel bottom apex, supports the steam dryer. The structure is a right cylinder with an 80-in. inside diameter and a 1/2-in.-thick wall. A 1-in.-thick bottom flange is provided to fasten the structure to the thermal shock shield. The top of the dryer support structure is welded to the steam dryer assembly.

The steam dryer reduces the moisture content of the steam to a minimum quality of 99.9 percent at the discharge side of the dryer. The steam dryer is formed by 8 Chevron-type vane banks welded to each other and contained within a welded frame structure of face plates and top and bottom annular plates. The total steam dryer discharge area is 21.5 ft². Except for Inconel vanes, the steam dryer assembly and support structure are

entirely of AISI Type-304 stainless steel. The steam dryer and support assembly is fastened to the shock shield with forty-eight 1.0-in. 19-9 DL studs, welded to the shock shield upper flange, and two 1.0 in. 19-9 DL dowels press-fitted into the shock shield upper flange.

The steam dryer closure plate primarily provides a closure between the steam dryer discharge area and the vessel dome, to prevent wet steam from bypassing the steam dryer. The closure plate is an annular plate 0.109 in. thick, which acts as a membrane between the steam dryer and the reactor vessel flange and is designed to accommodate the differential thermal expansion between the internal components and the reactor vessel.

A ring is welded about the outer periphery of the closure plate. At operating conditions the ring is wedged between the vessel flange inside diameter and the seal surface of the closure head flange, thus preventing excessive moisture from penetrating the dry steam annulus. About the inner periphery, the closure plate is clamped between a 1.0-in.-thick annular plate and the top of the steam dryer. A stainless-steel asbestos-wound gasket is inserted between the closure plate and the steam dryer upper plate to prevent wet steam and condensate water from entering the dry steam annulus at the discharge of the dryer. The clamping force is provided by forty-eight 3/4-in. 19-9 DL alloy studs, welded to the steam dryer upper plate.

The annular clamping plate is provided with radial slots to discharge condensate water from the top of the closure plate into four 1/2-in. drain pipes. The water entering the drain pipes is then discharged below the steam dryer assembly.

4.3.8 Emergency-Core-Spray and Failed-Fuel-Element-Location Assembly (Figs. 4.15 through 4.20)

This assembly consists of a 73-tube bundle, an upper support, intermediate tube sheets, tie rods, a flow deflector, and an emergency-cooling ring. A functional description of the system is given in Secs. 5.2.6 and 5.2.7.

The upper support is an annular ring plate with ten parallel bars extending across the plate opening. The 73 tubes are attached on the bars in proper orientation to the fuel assembly locations. The tube bundle is laterally supported between the top of the steam dryer and the top of the core by four radial supports that are bolted to the thermal shield and fitted about the periphery of one of the tube sheets formed by the lateral flow deflector baffle.

The tube bundle is vertically supported by the upper support ring plate on the top of the steam dryer and is fastened to the steam dryer closure plate clamp ring by four bolts and two locating dowels. Eight additional tubes can be inserted into the bundle if the eight spare fuel element shrouds are loaded with fuel assemblies.

4.3.8.1 Emergency-Core-Spray System. The emergency-cooling core spray tubes are inserted into the failed-fuel location system tubes through tees and are kept concentric along the entire length of the tubes by means of radial spacers. The emergency-core spray tubes protrude beyond the failed-fuel system tubes to discharge through nozzles directly above each fuel assembly. The spray nozzles have a 3/32-in. orifice diameter and a 120 degree spray angle. They are made by Spraying Systems Company, Bellwood, Illinois (No. 1/8 GG8W). At the upper part of the assembly, each emergency-core spray tube passes through one leg of a tee arrangement, where it is secured to the tee with a Swagelok fitting; thus the core spray system is isolated from the failed-fuel detection system. The tubes are then connected to flow regulating valves mounted directly on the emergency-core-spray distribution ring. The emergency-core-spray distribution ring is a doughnut-shaped ring, 52-in. in diameter, of AISI Type-304 stainless steel. It is supported from the upper tube sheet of the combined assembly.

The water supply to the emergency-core spray ring is provided by a 1-1/4-in. pipe between the reactor vessel penetration and the spray ring. A bolted flange connection directly below the emergency-core spray ring facilitates installation and removal of the combined systems during refueling.

The section of the 1-1/4-in. line between the reactor vessel penetration and the 1-1/4-in. pipe at the steam dryer support is of flexible tubing to accommodate the differential expansion between the stainless-steel internals and the carbon-steel vessel. The flexible tubing is attached to the emergency-cooling nozzle inside the vessel by a 1-1/4-in. union, threaded on a mating part welded to the emergency-cooling nozzle thermal sleeve. An access hole in the steam dryer support facilitates assembly and disassembly of the connection.

4.3.8.2 Failed-Fuel-Element Location System. Each failed-fuel-element system tube extends downward from a tee arrangement to the top of the fuel assembly and terminates with a probe arrangement to collect samples of fission gas released from any fuel assembly that might fail. The channel is formed by the annulus of the two concentric tubes of the failed-fuel detection and emergency-cooling spray system. The channel runs from the probe above the fuel element to the tee arrangement of the upper part of the bundle.

The failed-fuel element system line is attached to one leg of the tee with a Swagelok fitting, and extends radially to a quick disconnect mounted on the steam dryer closure plate.

Flexible tubes below the steam dryer closure plate are attached to the quick disconnect by a Swagelok fitting, and to the reactor vessel flange penetration by a pipe fitting threaded into the penetration. Swagelok fittings are provided about the outer periphery of the vessel flange to connect the probe lines to the failed-fuel detection system.

4.3.8.3 Flow Deflector (Figs. 4.15 and 4.17). The lateral flow deflector, an integral part of the combined assembly, consists of two parallel plates, 0.125 and 0.250 in. thick and 0.9 in. apart, separated by pipe spacers. The bottom plate is 20 ft-0 in. above the vessel apex. Each plate is perforated with 1/2-in.-diameter holes on a 1-in. square pitch. The deflector primarily produces the quiescent water level necessary for efficient natural steam water separation, but it also provides radial support for the failed-fuel element detection and emergency-cooling core spray systems assembly.

4.3.9 Core and Core Structure

The reactor core consists of fuel assemblies, fuel element shrouds, control rods, vertical posts, and the lateral support structure as described in the following subsections. The core plan is shown in Fig. 4.21.

4.3.9.1 Fuel Assemblies (Figs. 4.22 and 4.23). The core is loaded with 72 fuel assemblies, each of 100 fuel rods in a 10 x 10 array, spaced on a 0.565-in. square pitch. There are provisions for eight additional fuel assemblies. Each fuel rod is a stainless-steel tube, loaded along 83 in. of its length with uranium dioxide fuel pellets, with a space between the last fuel pellet and the top end plug. The fuel pellets are enriched in U-235 to 3.63 percent and are compacted to a density of 10.4 g/cc. The enrichment is based on an average first core burnup of 12,000 Mwd/2000 lb of contained uranium. Each fuel rod is 0.396 in. in OD. The AISI Type-348H stainless-steel cladding is 0.020 in. thick and the uranium dioxide pellets are 0.350 in. in diameter, leaving a nominal radial gap of 0.003 in. between the fuel and the cladding. The tube-wall thickness is adequate to withstand the pressure of the released fission gases accommodated at the upper end of each rod. A stainless-steel spring in each of these end spaces prevents gaps between pellets. The loaded tubes are filled with 70 percent helium and 30 percent air at STP to ensure a satisfactory heat-transfer medium between the pellets and the cladding. The pellets have dished end faces to prevent differential expansion relative to the cladding.

The fuel pellets are uranium dioxide of low enrichment and consist of a mixture of UO₂ powder, densifier, and lubricant, mechanically pressed into a cylindrical shape. The pellets are baked to remove volatile lubricant and are fired at between 1550 and 1700 C to increase the minimum density to 95 percent of theoretical. The finished pellets are loaded into the fuel tube, the tube is filled with helium at atmospheric pressure, and the end plugs are welded to the tube by automatic shielded-arc equipment.

The nominal outside diameter of a fuel pellet is 0.350 in., and the length varies from a minimum of 0.350 in. to a maximum of 1.050 in. The initial core of 72 fuel assemblies nominally contains 8600 kg (but not less than 8514 kg) of elemental uranium.

A fuel rod assembly consists of a fuel rod and two end plugs. Each fuel assembly is composed of 88 regular fuel rod assemblies, eight holddown fuel rod assemblies, and four segmented fuel rod assemblies. A regular fuel rod assembly consists of fuel pellets, a 3.93-in. space with spring, and end plugs with bullet-shaped tips that rest in the upper and lower fuel assembly grids. A holddown fuel rod assembly consists of fuel pellets, a 3.67-in. space with spring, and end plugs with a threaded tip. These assemblies are retained in the fuel-assembly grids by nuts that thread onto the end-plug tips, tying the 100-tube bundle to the end grid assemblies. A segmented fuel rod assembly consists of four separate segments (three with pellets and a 1-in. gap, and one with pellets and a 3.93 in. gap with spring). The segments are joined together by stainless-steel fittings that lock the intermediate grids in place at three positions along the length of the assembly.

The spacer grids, placed at intervals of approximately one-fourth the length of the fuel rod, maintain proper fuel rod spacing.

There is mechanical contact between the intermediate grids and the fuel rods; a 1/4-in. slot at the corners of the intermediate grid provides water contact with corner fuel rods to minimize potential hot spots in the area. A button on the outer surface of the grids maintains a minimum clearance of 0.060 in. between the outer rods and the fuel assembly shrouds. Both the upper and lower fuel-assembly end fittings are AISI Type-304 stainless steel. The upper end fitting has an opening for the fuel-assembly handling tool. The grids of the end fittings have a 0.060-in. flange that maintains the proper spacing between the tops of adjacent fuel assemblies.

Fission-gas release, thermal and mechanical stresses, and vibration analyses of the fuel rods are summarized in Appendix A.

4.3.9.2 Fuel Assembly Shrouds (Fig. 4.24). The fuel assembly shrouds enclose the fuel assemblies laterally, channel the coolant flow (thus preventing cross flow within the reactor core), and provide guide channels for the control rods. The shrouds also provide lateral support for the fuel assemblies and a seating surface for the fuel assembly nozzles. Each fuel assembly shroud consists of a hollow square can, 105 in. long. The nominal inside wall-to-wall distance is 5.467 in., and the wall thickness is 0.078 in. A 0.820-in. flange is located at the bottom of each of the shrouds. The inside diameter of the bottom flange is machined so that the fuel assembly provides shroud holddown during normal operating conditions. The flange is fastened to the can by 12 machine screws, which are tack-welded to the shroud. The bottom flange acts as a flow seal to prevent significant leakage into the control rod channels.

The rotating lock, a 1/2-in.-thick plate on top of each flange, provides positive hold-down. The lock is rotated under four lugs that form part of the bottom grid assembly and extend through cutouts in the bottom flange of the shrouds. This lock provides positive hold-down of the shroud under all conditions. It is held in the locked position by a tab on the fuel assembly inlet nozzle that fits into machined slots in the rotating lock. Four 1-in.-diameter holes are provided at the top of each of the shrouds for shroud removal.

The startup core has a total of 80 shroud cans, with the probable distribution of 45 zirconium alloy shroud cans and 35 AISI Type-304 stainless-steel shroud cans (including the spares). The internal components of each shroud are also AISI Type-304 stainless-steel. Cores subsequent to the initial core will contain only zirconium-alloy canned shrouds. The total weight of an assembled fuel assembly shroud is approximately 55 lb for the zirconium-alloy type and 60 lb for the stainless-steel type.

4.3.9.3 Core Vertical Posts (Fig. 4.25). The 52 vertical posts provide lateral support for the fuel element shrouds within the core, thereby maintaining the control-rod channel passages. Each of the vertical posts is 102.25 in. long (not including the bottom pin) and is provided with a tapered dowel at the bottom. This dowel locates and supports each post in the top plate of the bottom grid assembly, 97.5 in. above the bottom apex of the reactor vessel. A 1/2-in.-diameter hole centrally drilled in nine of the vertical posts receives the tubing from the in-core monitoring system. A tapped hole in each of the posts facilitates installation and removal. Twenty-eight of the posts receive bolting from the horizontal grids. Each post on the outer periphery of the core is provided with a tapered edge at the point of engagement with the core lateral support structure. Except for pads 1 in. long at the top, center, and bottom, the posts within the core are undercut by 0.030 in., which provides for easier assembly of core components and allows cross flow between the control-rod channels. The vertical posts are of zirconium alloy. The pins for insertion into the bottom grid assembly are of AISI Type-304 stainless steel. The total weight of each vertical post is approximately 60 lbs.

4.3.9.4 Core Lateral Support Structure (Figs. 4.26 through 4.29). The core lateral support structure, which is at the top of the bottom grid assembly and outlines the core, provides lateral support for the core and the core components. The core lateral support structure consists of a top and bottom flange, connected by structural angles. Guides are provided on the top flange to guide and support the vertical posts on the outside of the core.

4.3.9.5 Control Rods (Figs. 4.30 and 4.31). There are 29 cruciform-shaped control rods in the core. The control rods move vertically (upward to decrease reactivity and downward to increase reactivity) in each of the control-rod channels formed by four fuel element shrouds and four vertical posts. The poison section consists of 80 annealed Inconel-600 tubes, 0.206 in. in diameter, which contain boron carbide pellets that

are 0.157 in. in diameter and sintered to a minimum of 67 percent of theoretical density. The rod geometry is that of a 9.750-in. cruciform span, 0.313 in. thick and 83 in. in active length, with an extension shaft of AISI Type-304 stainless steel. The tubes are packed in a 0.051-in.-thick perforated sheath of AISI Type-304 stainless steel.

The holes in the sheath are 0.75 in. in diameter and are spaced on a 10-in. vertical pitch. The holes are arranged so that the spaces between the sheath and the poison tubes have flow communication with the space between the sheath and the fuel assembly shrouds. The sheath is welded to the central spider and end connectors. The control rod has rollers at the top that prevent the sheath from contacting the fuel assembly shroud. The minimum clearance between the sheath and shroud, if the roller is touching the shroud, is 0.059 in. An annealed AISI Type-304 stainless-steel cruciform shaft, 3 in. by 3 in. and 100.125 in. long, is welded to the lower end of the poison section. The other end of this stainless-steel section terminates in an end stud that fastens the control rod to a drive mechanism by means of a ball lock mechanism. The control rod can be removed from the top of the reactor vessel by disengaging the ball lock mechanism and attaching a lifting tool to a handle on the top of the absorber section.

4.4 CORE NUCLEAR CHARACTERISTICS

4.4.1 Design Criteria and Characteristics

The design requirements and objectives pertinent to the core physics analysis include:

- (1) a plant availability factor of 90 percent, including no more than 12 days of shutdown time per year for fuel management and routine plant maintenance;
- (2) a uniform core power distribution with minimum peaking;
- (3) low-enriched UO_2 fuel clad with stainless steel;
- (4) movable control rod blades sufficient in worth when unirradiated to make the fully loaded cold clean core subcritical by an adequate safety margin and to provide an average fuel exposure of not less than 10,000 Mwd per short ton of contained uranium, without reliance on burnable poison or the addition of liquid poison to the coolant;
- (5) fuel assemblies containing uniformly enriched fuel in a uniform geometry, and a reactor vessel and internals design that accommodates that geometry for full-power operation;
- (6) an average exposure of 12,000 Mwd for each short ton of uranium in the first-core fuel and 15,000 Mwd per short ton in the equilibrium condition;

(7) a negative temperature coefficient of reactivity for all moderator temperatures associated with reactor operation; and

(8) a core subcritical by at least 0.5 percent Δk in the cold, clean condition even if the control rod of maximum worth is withdrawn from the core.

In order to meet these criteria, the LACBWR startup core design has been chosen in accordance with a control-rod management program, and a study of fuel lifetime and the isotopic change caused by fuel exposure has been performed. The design startup core configuration contains 28 stainless-steel shrouds and 44 zirconium shrouds in the arrangement shown in Fig. 4.32. The use of the stainless-steel shrouds reduces flux peaking and, in combination with control rod and fuel management described in Sec. 4.5, promotes power flattening over the entire operating cycle.

After fuel exposure of approximately 6300 Mwd/short ton, stainless-steel shrouds will be replaced by zirconium shrouds. This operation will add approximately 5 percent Δk to the core. The excess lifetime reactivity will then be less than the excess reactivity (8 percent) of the startup core, and thus core shutdown margins will be met at this time.

Table 4-2 gives the core characteristics for the startup fuel loading (based on unrodded core unless otherwise noted). The characteristics are tabulated for the true startup configuration of mixed composition shrouds, and for a hypothetical core with all zirconium shrouds. The latter case represents the most reactive core that could be achieved by insertion of a complete, new fuel loading of an enrichment equivalent to the initial core, with all zirconium shrouds in place. It can be seen that shutdown criteria are met for the hypothetical core, although power distribution would be adversely affected.

The average fuel exposure achieved by the first-core fuel is 12,700 Mwd/ST, compared with the design requirement of 12,000 Mwd/ST. This exposure is with the 3.63 percent U-235 fuel enrichment of the final design. The final enrichment value resulted from a refined analysis of the disadvantage factor in the shroud and water gap between elements and from the calculation of experimental critical core configurations with fuel and moderator similar to LACBWR. The experimental data came from the Savannah Criticals (4-1) and Consumers Power Criticals. (4-2)

TABLE 4-2

LACBWR CORE CHARACTERISTICS

	STARTUP CORE	
	With 28 s.s. and 44 Zr shroud elements, actual case	With all Zr shroud elements, hypothetical case*
average thermal flux in core, n/cm^2 -sec	1.5×10^{13}	1.5×10^{13}
average fast flux in core, n/cm^2 -sec	4.4×10^{13}	4.4×10^{13}
peak-to-average power ratios (rodded as in Fig. 4.34)		
local	1.45	1.45
axial (hot, operating)	1.87	1.87
radial	1.47	1.62
overall (hottest core location)	3.98	4.39
average fuel exposure, Mwd/short ton U	---	12,700
multiplication constants, k_{eff}		
cold, clean	1.188	1.241
hot, clean (zero voids)	1.150	1.201
hot, operating (100% power, 15% avg voids**)	1.125	1.175
hot, operating, with equilibrium xenon and samarium	1.084	1.132
cold, clean, control rods fully inserted	0.919	0.931
cold, clean, most important control rod removed	0.990	0.990

*This case represents the highest core reactivity achievable with the design fuel load.

**Based on total coolant volume in core

STARTUP CORE

reactivity coefficients

temperature coefficients, $\Delta k/k/^{\circ}\text{C}$

	With 28 s.s. and 44 Zr shroud elements, actual case	With all Zr shroud elements, hypothetical case*
cold core (20 C), unrodded	$+3.8 \times 10^{-5}$	$+3.3 \times 10^{-5}$
cold core (20 C), rodded critical	-1.1×10^{-4}	-1.3×10^{-4}
hot core (303 C), zero void	-4.8×10^{-4}	-5.0×10^{-4}

void coefficients, $\Delta k/k/\%\text{void}$

	With 28 s.s. and 44 Zr shroud elements, actual case	With all Zr shroud elements, hypothetical case*
hot core, zero void	-1.6×10^{-3}	-1.7×10^{-3}
hot core, 15% average void	-2.3×10^{-3}	-2.4×10^{-3}

power coefficients, $\Delta k/k/^{\circ}\text{C}$

	With 28 s.s. and 44 Zr shroud elements, actual case	With all Zr shroud elements, hypothetical case*
303 C fuel temperature, 15% void	-2.3×10^{-5}	-2.4×10^{-5}
526 C fuel temperature, 15% void	-2.0×10^{-5}	-2.0×10^{-5}

nuclear constants

mean life of prompt neutrons, sec

	With 28 s.s. and 44 Zr shroud elements, actual case	With all Zr shroud elements, hypothetical case*
cold core	4×10^{-5}	4×10^{-5}
hot core, zero void	5×10^{-5}	5×10^{-5}
hot core, 15% void	5×10^{-5}	5×10^{-5}

effective delayed neutron fraction

	With 28 s.s. and 44 Zr shroud elements, actual case	With all Zr shroud elements, hypothetical case*
cold core	0.0071	0.0071
hot core, zero void	0.0073	0.0073
hot core, 15% void	0.0075	0.0075

average decay constant, sec^{-1} 1-group

	With 28 s.s. and 44 Zr shroud elements, actual case	With all Zr shroud elements, hypothetical case*
cold core	0.47	0.47
hot core, zero void	0.47	0.47
hot core, 15% void	0.47	0.47

*This case represents the highest core reactivity achievable with the design fuel load.

	STARTUP CORE	
	With 28 s.s. and 44 Zr shroud elements, actual case	With all Zr shroud elements, hypothetical case*
thermal utilization (f)		
cold core***	0.737	0.770
hot core, zero void***	0.770	0.804
hot core, 15% void***	0.781	0.816
fission neutrons generated per thermal neutron absorbed in fuel (η)		
cold core	1.840	1.840
hot core, zero void	1.826	1.826
hot core, 15% void	1.820	1.820
fast fission factor (ϵ)		
cold core	1.029	1.029
hot core, zero void	1.039	1.039
hot core, 15% void	1.044	1.044
resonance escape probability, (p)		
cold core	0.884	0.884
hot core, zero void	0.836	0.835
hot core, 15% void	0.815	0.814
fast non-leakage probability		
cold core	0.965	0.965
hot core, zero void	0.946	0.946
hot core, 15% void	0.935	0.936
thermal non-leakage probability		
cold core	0.998	0.998
hot core, zero void	0.996	0.996
hot core, 15% void	0.995	0.995

*This case represents the highest core reactivity achievable with the design fuel load.

***Effective value is averaged for the entire core.

STARTUP CORE

	With 28 s.s. and 44 Zr shroud elements, actual case	With all Zr shroud elements, hypothetical case*
neutron age (τ), cm ²		
cold core	37.8	37.8
hot core, zero void	59.4	59.4
hot core, 15% void	72.2	72.2
diffusion length (L), cm		
cold core	2.0	2.0
hot core, zero void	4.2	4.2
hot core, 15% void	5.3	5.3
geometric buckling, (B_g), cm ⁻²		
cold core	0.984×10^{-3}	0.984×10^{-3}
hot core, zero void	0.960×10^{-3}	0.960×10^{-3}
hot core, 15% void	0.948×10^{-3}	0.948×10^{-3}
initial fuel enrichment, %	3.63	3.63
volume fractions (unit cell)		
uranium dioxide	0.237	0.237
water (unrodded cell)	0.622	0.622
control rods	0.073	0.073
stainless-steel cladding	0.058	0.058
stainless-steel shrouds	0.017	
zirconium shrouds	0.027	0.044
structure	0.031	0.031
voids (between clad and fuel)	0.008	0.008

*This case represents the highest core reactivity achievable with the design fuel load.

4.4.2 Reactivity Coefficients

Reactivity coefficients for LACBWR are negative, as shown in Table 4-2. The temperature and void coefficients are delayed, while the power coefficient is prompt. The derivation of time constants for the transfer of heat from the fuel to the coolant is given in Sec. 3.2 of Appendix D to ACNP-62574 (Amendment No. 6 to Application for Construction. Authorization Docket No. 115-5). The derivation of void redistribution dynamics is described in Sec. 3.8 of the same appendix. The k_{eff} was calculated as a function of temperature and as a function of voids (with constant temperature). The data were plotted, and the coefficients were obtained by taking the derivative at the temperature or void percent of interest. The power coefficient was obtained by calculating the reactivity change due to the Doppler effect associated with a small change in the average fuel temperature. Since the negative reactivity coefficients are of greater magnitude with control rods inserted, the physics analysis was performed for the unrodded core, the most conservative method of analysis for considerations of safety. However, the temperature coefficient was also analyzed for the cold-critical rod configuration to ascertain that it would be negative for all operating conditions.

The temperature, void, and power coefficients cannot be physically separated, but they have arbitrarily been separated for this study by application in the following ranges:

(1) temperature coefficient: moderator and fuel vary between ambient temperature and operating moderator temperature, with zero voids.

(2) void coefficient: moderator and fuel are each at operating temperature, with voids varying from zero to average operating value.

(3) power coefficient: moderator is at operating temperature, voids are at the operating average value, and fuel temperature is varied between operating moderator temperature and operating fuel temperature.

Although an unrodded cold core has a positive temperature coefficient, the coefficient is negative for the actual critical condition (i.e. with control rods in a bank or partial bank) because of the increase in reactivity worth of the control rods as the coolant temperature rises. This increase in rod worth is more than enough to compensate for the reactivity gain experienced by the unrodded core as temperature increases from 20 C.

4.4.3 Control-rod Worth

The LACBWR control-rod analysis was performed using two methods: the area absorption method⁽⁴⁻³⁾ and the transport boundary method⁽⁴⁻⁴⁾. Calculation techniques were verified by comparison with rod worth measurements in The Consumers Power Reactor, which has a core and control rods similar to those of LACBWR.

The total reactivity worth ($\Delta k/k$)* of all rods in a cold, clean core with zirconium shrouded fuel assemblies was calculated to be 25.0 percent, with a k_{eff} (unrodded) of 1.241 and a k_{eff} (rodded) of 0.931. The rod worth is less (22.6) for the startup core configuration, but the shutdown margin is greater with a k_{eff} (rodded) of 0.919, because of the stainless-steel shrouds. Since the core will have an average fuel burnup greater than 12,000 Mwd per short ton, the rod control requirement for exposure of not less than 10,000 Mwd per short ton is met.

A multiplication-constant calculation was performed to ascertain that the design criterion of 0.5 percent Δk minimum subcriticality with the maximum worth rod withdrawn from the LACBWR core in the cold, clean condition is satisfied. The results for a core with all zirconium shrouds showed that the worth in $\Delta k/k$ for the 28 least important control rods was 20.2 percent, resulting in a k_{eff} of 0.99 (1 percent subcritical).

4.4.4 Power Distribution

LACBWR power distributions were obtained by multiplying a local peaking factor by an overall macroscopic peak-to-average power ratio. The local peaking factor is defined as the power ratio between the fuel rod producing maximum energy and the fuel rod producing average energy in a 10 x 10 rod fuel assembly. Since the core will contain both stainless and zirconium shrouded assemblies, local peaking factors for both cases were determined by a three-group analysis. The local peaking factor is 1.35 for the stainless-steel shrouded assembly, as compared to 1.45 for the zirconium shrouded assembly.

The local peaking analysis was performed with the two-dimensional PDQ-2-90 code. The cell was mocked up as shown in Fig. 4.33. The shroud region was arbitrarily homogenized as 10 percent water and 90 percent steel, and, as an additional conservative measure in the calculation, no boiling was assumed to occur outside the shroud. Region 5 in the mockup is the area associated with the rods in the cell corners without posts, where the maximum power in the fuel region occurs. The zirconium posts were explicitly represented, whereas the region inside the shrouds is a homogenized fuel region consisting of UO₂, stainless-steel cladding, void, and the water surrounding the rods.

Reactor operation in accordance with the fuel management and control-rod program described in Sec. 4.5 gives a power distribution that flattens as fuel is consumed. Therefore, the startup core at the beginning of life presents the most severe power distribution test in LACBWR. Since the control rods in the core at the start of life will be fully inserted, the calculation for power distribution can be separated into a one-dimensional axial calculation and a two-dimensional X-Y calculation at a height where the hottest element is producing maximum power. The calculations were performed using the two-energy-group cross sections in the Wanda code for axial distribution, and

$$*\Delta k/k = \frac{k_{eff} \text{ unrodded} - k_{eff} \text{ rodded}}{k_{eff} \text{ unrodded}}$$

three-energy-group cross sections in the PDQ code for the X-Y geometry case. Control rods were represented explicitly. In the initial axial calculation a variable axial void was used; the initial X-Y calculation was performed assuming a uniform void. The core configuration for this calculation is shown in Figs. 4.32 and 4.34, and the radial power distribution of the X-Y geometry is indicated in Fig. 4.34. The axial power distribution is presented in Fig. 4.35. These initial calculations yielded a maximum peak-to-average power of 3.98.

Further analysis of the initial core was performed in conjunction with the fuel management studies. The X-Y calculations were repeated using the void distribution which resulted from a power-void iteration, with the X-Y plane taken through the point of peak axial flux. These results are about eight percent lower than the values of Fig. 4.34. However, the maximum peak-to-average power ratio of 3.98 was retained as a conservatism for the thermal performance analysis.

4.5 FUEL MANAGEMENT AND CONTROL-ROD PROGRAM

4.5.1 LACBWR Operating Program

Two of the major objectives for LACBWR are long lifetime and power flattening. To achieve these goals the core was designed to have removable shrouds. Thus in the initial core, which has maximum positive reactivity, there are 28 elements with stainless-steel shrouds as opposed to the later (equilibrium) condition in which all shrouds are zirconium. The initial core power flattening benefits derived from the stainless shrouds include the following:

- (1) Local power peaking factors are reduced.
- (2) The macroscopic initial power distribution is flattened.
- (3) By carefully choosing the location and number of stainless-steel shrouds (Fig. 4.32), reactivity can be adjusted to allow use of the desired control-rod pattern, allowing the reduction of the axial macroscopic peak-to-average power ratio.

Thus, the stainless-steel shrouds are used to shape the power distribution in the initial core, and to permit the desired control rod configuration, which depends on the control-rod program for the operating reactor. In LACBWR the operating objective is to have a uniform core burnup. In an unperturbed core the initial power distribution would approximate a cosine shape axially and a J_0 shape radially. As burnup proceeds, a natural flattening of the macroscopic power peaks occurs reducing the peak-to-average power ratios. In the operating reactor, axial flattening is achieved by using control rods either fully inserted or removed and radially, power flattening is achieved by the slight poisoning of the central region of the core, resulting from initially placing the stainless-steel shrouds near the center of the core.

The key factor, however, for attainment of uniform fuel burnup is the control-rod operating program. The reactor is controlled with 29 cruciform rods — a central rod with others distributed symmetrically about it. At power after xenon buildup, the rods must control 8 percent excess reactivity in the startup core. Insertion of seven control rods (the central rod and six designated outside rods) will accomplish this. The six rods display half core symmetry such that a rotation of 90 degrees will insert control rods that were previously removed and will remove rods that were previously inserted, while maintaining an identical overall control-rod pattern in the core. The control-rod program illustrated in Fig. 4.36 will achieve uniform burnup in this manner. The central rod is used as the regulating rod and is removed as required to insert reactivity; this operating mode contributes to power flattening.

After the central rod is removed, a new configuration consisting of four outside control rods and a fully inserted central rod, is used to control the core. The procedure is continued until the core can no longer maintain criticality. At that time, the stainless-steel shrouds are replaced with zirconium shrouds. The added reactivity requires the insertion of five control rods for criticality. Reactor operation will then continue in the same manner until the core goes subcritical, at which time the first fuel replacement will occur. The 24 elements whose fuel has been exposed longest are removed from the core. The 24 next-longest-exposed elements will be placed in the inner zone. The remaining elements will be placed in specific locations around the outside. The 24 fresh elements will then be placed in the empty locations. Elements in the two outside zones are placed in a manner that approximates a uniform mixture (see Figs. 4.37 through 4.42). The process is repeated until all of the initial fuel is removed from the core and the reactor goes into equilibrium-cycle operation.

The results of nuclear calculations in the X-Y geometry for the proposed operating program are presented in Table 4-3, and show that the startup core with equilibrium xenon is about 1 percent supercritical at power with seven control rods inserted. At 5500 Mwd/MT, the core is 0.3 percent supercritical, with three control rods inserted. From rod worth data, it can conservatively be predicted that the core could operate with stainless-steel shrouds to 7000 Mwd/MT before going subcritical.

The next calculation, at 7000 Mwd/MT, shows the core approximately 1 percent subcritical with five control rods inserted, with zirconium in place of the stainless-steel shrouds. At 10,000 Mwd/MT, the core is just critical with the central rod inserted. Finally, a calculation at 10,000 Mwd/MT shows the core 1.7 percent supercritical with all control rods removed.

Neglecting the special cases, the "radial" peak-to-average power ratio varied from 1.33 to 1.62 with an initial value of 1.40. Because of the time-step method of approach and the detailed core geometry of actual control-rod programming, non-uniform power distributions and non-uniform fuel burnup are well represented in the calculation.

The calculated data for axial cases, as presented in Table 4-4, are based on the assumption that the core initially contained only zirconium shrouds. Because of the in-out rod program, this assumption affects only the multiplication constant, not the power distribution. However, the object of this study is to calculate the end-of-life core, with all the zirconium shrouds and all the control rods removed. The rods-removed condition is the one mocked up by the calculation. Table 4-4 shows that the axial peak-to-average power steadily decreases with exposure—from 1.83 at the start of life to 1.18 at 10,000 Mwd/Mt. At an average exposure of 10,000 Mwd/Mt, k_{eff} is 1.021, which is very close to 1.017, the value predicted by the X-Y geometry calculation. These calculations, which take into account non-uniform exposure effects, indicate a core capability of more than 10,000 Mwd/Mt average exposure before any fuel need be removed.

TABLE 4-3

SUMMARY OF LACBWR FUEL EXPOSURE CALCULATION

(X-Y GEOMETRY)

average core exposure, Mwd/MT	number of control rods inserted	core multiplication constant	elemental peak-to-average power ratio	notes
0	7	1.042	1.40	no Xe or Sm; 28 s.s. shrouds
0	7	1.010	1.38	
2,000	7	0.994	1.35	
*2,000	7	1.005	1.73	control rods rotated 90 deg
4,000	7	0.977	1.33	
4,000	5	0.998	1.46	
5,500	3	1.003	1.35	
7,000	5	0.989	1.62	28 s.s. shrouds replaced with Zr shrouds
10,000	1 (center)	1.000	1.33	
**10,000	0	1.017	1.70	

*The elemental peak-to-average power ratio in this case is high because of the long exposure before rotating the control rods. The power distribution is flatter if the rod rotation is made after a smaller burnup increment.

**The elemental peak-to-average power ratio in this case is unreasonably high because the central regional exposure was based on a central rod fully inserted at all times. In fact, the central rod (regulating rod) will be withdrawn periodically, increasing the exposure of the region and causing a reduction in the peak-to-average power ratio.

TABLE 4-4

SUMMARY LACBWR FUEL EXPOSURE CALCULATION (AXIAL)

<u>average core exposure, Mwd/Mt</u>	<u>core multiplication constant</u>	<u>axial peak/average power ratio</u>
0 (Eq. xenon)	1.143	1.83
1,000	1.127	1.73
2,000	1.112	1.59
3,000	1.104	1.44
4,000	1.094	1.39
5,000	1.083	1.32
6,000	1.071	1.29
7,000	1.059	1.27
8,000	1.046	1.23
9,000	1.033	1.20
10,000	1.021	1.18

Figure 4.43 is a plot of the core multiplication constant, with equilibrium xenon, bases on axial non-uniform burnup effects only. After shutdown, transient xenon concentration results in a maximum reactivity loss of 0.74 percent. It can be estimated from the slope of Fig. 4.43 (at $\approx 10,000$ Mwd/MT average exposure) that this reactivity loss corresponds to 600 Mwd/MT of fuel exposure. Thus, it can be expected that the LACBWR core will be capable of overriding peak xenon until the final 600 Mwd/MT of core life. Figure 4.44 is a plot of the transient xenon poisoning, which is at a maximum 7 hr after shutdown.

4.5.2 Fuel Lifetime

The fuel lifetime requirement in LACBWR stipulates a first-core average fuel exposure of 12,000 Mwd per short ton, and an equilibrium core fuel exposure of 15,000 Mwd per short ton. Since the core will not use a batch loading scheme, only a fraction of the fuel will be removed from the core at each refueling. This method of refueling is chosen so that freshly inserted fuel can be used to drive fuel previously exposed. Therefore the average first core exposure is the sum of the exposures, upon removal, received by all the fuel assemblies initially in the core, divided by the total number of assemblies in the core. Therefore, from the initial loading, the first fuel removed will have relatively lower exposure and the last of the fuel removed will have relatively higher exposure.

In actual operation the startup core will have 8 percent Δk for lifetime. At a fuel exposure of 6360 Mwd/short ton, zirconium shrouds replace the stainless-steel shrouds, causing a step increase of 5 percent Δk . At the first refueling, another reactivity step increase will occur as fresh fuel replaces spent fuel.

Figure 4.45 shows the change in multiplication constant with fuel exposure for an unrodded, uniformly exposed core. The solid line represents the value for the initial startup core followed by the all-zirconium shrouded core up to the first fuel change. The broken line continuation of the solid line represents the change in the multiplication constant, neglecting shroud management and refueling. Table 4-5 presents the isotopic fuel change, i.e., the depletion of U-235 and the buildup of plutonium (Pu-239, Pu-240 and Pu-241), according to exposure.

TABLE 4-5
VARIATION OF FUEL COMPOSITION WITH EXPOSURE

exposure (Mwd/ST)	U-235 enrichment, %	g Pu/initial kg U		
		Pu-239	Pu-240	Pu-241
0	3.63	---	---	---
2,780	3.24	1.47	0.10	0.02
5,730	2.90	2.46	0.29	0.11
8,600	2.59	3.12	0.49	0.25
11,640	2.31	3.55	0.66	0.44
14,300	2.06	3.84	0.81	0.62

The fuel element exposure according to the fuel management program is presented in Table 4-6. As a conservatism, it was assumed that the first one-third of fuel would be removed at an average core exposure of 10,000 Mwd/Mt. The fuel was divided into three groups of 24 assemblies each, according to fuel exposure. The tabulation presents the average exposure of all the fuel in the core, the average exposure of each of the three groups of fuel, and the exposure of the fuel removed from the core after each operating period.

TABLE 4-6

LACBWR FUEL EXPOSURE DATA

core average	exposure, Mwd/MT			
	1st third	2nd third	3rd third	removed third
10,000	11,760	9,574	8,687	
6,087	9,574	8,687	0	11,760
10,087	14,313	11,898	4,050	
5,316	11,898	4,050	0	14,313
9,316	16,034	8,095	3,823	
3,973	8,095	3,823	0	16,034

The average exposure of the first-core fuel is the average exposure of the fuel removed from the core, or 14,000 Mwd/MT (12,700 Mwd/ST). This value is the sum of the last column of Table 4-6, divided by 3. The average uranium enrichment at this exposure is 2.21 percent, and the plutonium content is 4.90 g per kg of contained uranium.

4.6 THERMAL AND HYDRAULIC CHARACTERISTICS

4.6.1 Full Power Performance

Reactor full power (rated power) conditions are defined as 165 Mwt of nuclear heat, a reactor recirculating coolant flow of 30,000 gpm (10,750,000 lb/hr), a reactor steam pressure of 1300 psia, and a feedwater temperature of 291.6 F. During full power operation of the reactor, the parameters are expected to differ slightly from these nominal values because of the difference between the actual and the calculated values of steam line pressure drop and steam plant thermal efficiency. Reactor components are designed so that the plant produces the required electrical output, using appropriate margins to compensate for uncertainties in design.

Figure 4.46 gives the fluid temperatures and flowrates of recirculating water, steam, feedwater, and auxiliary streams during full power operation; the reactor internal flow distribution is shown in Fig. 4.47. The pressure of the subcooled water, the slight amount of vapor in recirculating water leaving the vessel (less than 2 percent by volume), and the slight amount of water in the steam (less than 0.1 percent by weight) do not appreciably affect the energy balance. Feedwater is returned to the forced-circulation outlet header, giving maximum subcooling in the outlet header and minimizing the possibility of steam flashing. Seal water and purification water are returned to the circulation pump suction line. The water flow withdrawn from the control-rod nozzles (Station 4, Fig. 4.46) is greater than the seal water flow to the control-rod nozzles (Station 8). The excess is withdrawn from the reactor inlet plenum. Measurements of flowrate and temperature will be made on the external primary fluid lines shown in Fig. 4.46.

The nuclear heat energy input is 165 Mwt, 157 Mwt being generated in the fuel elements and 8 Mwt in the control rods, internal components, and the vessel. The forced-circulation pumps add an additional 0.4 Mwt to the recirculating water.

The net energy used in the steam cycle is 164.5 Mwt. This energy is obtained by an energy balance of primary fluid stream enthalpies and heat loss from the hot primary system. The energy balance is:

$$\begin{aligned} (\text{Nuclear Heat}) + (\text{Pump Work}) &= (\text{Energy to Steam Plant}) \\ &+ (\text{Energy Rejected to Purification System}) \\ &+ (\text{Heat Loss from Primary System}) \end{aligned}$$

Calculated full power performance data are tabulated in Table 4-7.

TABLE 4-7

CALCULATED REACTOR PERFORMANCE AT FULL POWER

Fuel Assemblies

active fuel length, in.	83
flow channel equivalent dia, ft.	0.0525
flow area per assembly, ft ²	0.136
total core flow area, ft ²	9.8
heat transfer area per assembly, ft ²	71.71
total heat transfer area, ft ²	5163

Heat Utilization

nuclear heat, Mwt	165.0
heat gain from pumps, Mwt	0.4
heat loss from vessel, Mwt	0.1 (a)
heat loss in purification system, Mwt	0.8
heat used in steam cycle, Mwt	164.5

(a) Includes heat loss within reactor cavity only.

Table 4-7 - Calculated Reactor Performance at Full Power (cont'd)

Water Flow Distribution in Reactor

	lb/hr	gpm
forced-circulation pump discharge	10,750,000 (b)	30,000 (b)
flow through fuel assemblies	10,348,680	28,880
flow bypassing fuel assemblies:		
control rods	266,960	745
seal water system	8,650	24
purification system	20,000	56
plenum separator plate leakage	44,790	125
leakage to control rods	60,920	170
flow leaving steam separators	9,900,230	27,630
flow bypassing steam separators:		
flow through thermal shield annulus	143,330	400
leakage by core grid plate	21,500	60
flow leaving vessel through forced-circulation outlet nozzles	10,109,850 (c)	28,215 (c)

Forced-Circulation System Head Loss, ft

friction pressure loss in forced-circulation loop	45.0 (d)
core inlet plenum	5.0
loss through core	6.0
core exit plenum	2.0
steam separators	2.0
water outlet plenum	1.0
total	61.0

Reactor Pressure, psia	1,300
Steam Flow, lb/hr	611,500
Feedwater Return Temperature, °F	291.6
Core Inlet Subcooling, Btu/lb	18.5
Core Average Power Density (kw/liter).	41.4

(b) Performance based on rated flow of 30,000 gpm. Pumps are capable of producing 33,000 gpm at maximum pump speed.

(c) Includes 125 gpm for leakage from inlet plenum to outlet plenum.

(d) Includes expansion from inlet nozzles and contraction to outlet nozzles, as well as pressure drop in external forced-circulation loop.

Table 4-7 - Calculated Reactor Performance at Full Power (cont'd)

Fuel Heat Flux, Btu/(hr)(ft²)

average	109,100
maximum	434,700
burnout at 100 percent power	810,000

Heat Flux Peaking Factors

radial	1.49
axial	1.87
local	1.43
overpower	1.20

Minimum burnout safety margin at 100 percent power	1.86
--	------

Minimum burnout safety margin at 120 percent power	1.55
--	------

Steam-void Fraction (e)

average in core	0.21
average at core exit	0.37
maximum at core exit	0.57

Steam Quality at Core Exit, fraction

average	0.06
maximum	0.11

Water Velocity at Core Inlet, ft/sec

minimum	6.0
average	6.4
maximum	6.6

Fuel Temperature, ° F

average in core	930 (f)
average of rod cross section at peak flux	2,315 (f)
maximum in core at 120 percent power	4,750 (f)

(e) Based on volume of coolant within shrouds.

(f) The effective thermal conductivity is 1.1 Btu/(hr)(ft)(°F).

4.6.2 Forced-Circulation System Flow Characteristics

The forced-circulation pump speed can be controlled by positioning the scoop tube in the variable-speed fluid coupling to regulate the pump speed and the flowrate, within pump performance limitations. The pumps were designed to deliver 30,000 gpm (10,750,000 lb/hr) at 1140 rpm, against a system head of 85 ft. The calculated forced-circulation system head loss is 61 ft for full power operation (30,000 gpm flow). Therefore, the pumps can produce larger flowrates at full power.

The calculated head for the various pump speeds and the calculated system head loss are plotted against circulation flow in Fig. 4.48. Pump characteristics will be determined by tests prior to installation. The design calculation of system head loss is for operation at 100 percent power (i.e., a flow of 30,000 gpm entering the reactor), and for the flow distribution shown in Fig. 4.47. The maximum and minimum system head losses are based on the estimated accuracy of the design pressure drop calculations.

The pressure drop in the external circuit, calculated with conventional pipe friction factors and head losses for valves and fittings, was 45 ft (including expansion from the reactor inlet nozzle and contraction to the outlet nozzle). Pressure drop calculations are estimated to be accurate within +5 percent and -10 percent. A greater margin is allowed for the lower limit since the formulas usually give a slightly conservative value for pressure drop.

The head loss within the reactor, regarding the water inlet plenum as zero, is indicated in Fig. 4.47. The major head losses are through the fuel assemblies (6 ft) and from the water inlet to the core inlet plenums (5 ft). Head losses through the fuel along with void and flow distribution were calculated by computer. A leeway of ± 15 percent is allowed in the calculation for core losses. Pressure loss between the water inlet and core inlet plenum is primarily a result of the pressure drop (calculated within an estimated accuracy of ± 5 percent) through the holes in the flow distribution cylinder and the bottom grid plate. Pressure drop through the steam separators was experimentally determined to be 2 ft maximum and 1.5 ft minimum. The remaining 3 ft is a result of plenum and turning losses, calculated to an estimated accuracy of within ± 10 percent.

The maximum and minimum system head losses, within the accuracies stated, are 54.6 ft and 64.7 ft at full power conditions. The calculated recirculation flow at full pump speed ranges between 32,000 and 34,000 gpm (Fig. 4.48). The most probable flowrate is 33,000 gpm. Minimum permissible flowrates for various power levels are given in Sec. 4.6.8.

4.6.3 Hydraulic Design Considerations for Reactor Internal Components

The final design of reactor internal components satisfies physics, thermal, hydraulic and mechanical design requirements. The major hydraulic considerations in the design of the reactor internal components are described below.

(1) Water flows to the fuel elements through a perforated flow distribution cylinder between the water inlet plenum and the core inlet plenum (see Detail G, Fig. 4.49). The holes in the flow distribution cylinder are sized to supply the required coolant for the outer 28 fuel elements. The holes in the bottom grid plate are sized and located to supply coolant to the other 44 fuel elements. With this arrangement, there will be little cross flow in the core inlet plenum, and the resulting pressure distribution at the core support plate will be nearly uniform. The maximum calculated variation in the inlet pressure between an inner and outer fuel element is less than 0.01 ft of water at full power, while friction pressure drop in fuel assemblies is 6 ft of water.

(2) Restriction of control rod flow to a range that supplies sufficient water to prevent overheating and excessive bypassing of coolant around the fuel elements is accomplished by a flow restriction in the bottom grid plate (Detail B, Fig. 4.49). The restriction is sized to provide water slightly in excess of that required for the hottest fully inserted control rod. The maximum flow is calculated for the maximum clearance area (based on manufacturing tolerances) and for a fully withdrawn control rod. With this control-rod flow arrangement, the largest proportion of the control-rod flow pressure loss occurs in the flow restriction. Friction losses in the control-rod shroud and the active core region do not significantly affect the flow. For the minimum clearance area with the hottest control rod fully inserted, the flow is 750 lb/hr per rod; for the maximum clearance area with a control rod withdrawn, the flow is 1000 lb/hr per rod.

(3) Water flow distribution within the fuel elements must be determined with the greatest possible accuracy to predict minimum burnout ratio safety considerations and steam void distribution for core physics calculations. The fuel elements are contained within shrouds, so there is no crossflow between elements. Fuel-coolant leakage (fuel coolant is at a higher pressure than control-rod coolant) has been minimized, consistent with other requirements in the design of internal components. Calculated maximum leakage flows are given in Sec. 4.6.4.

(4) Internal components must be designed to preclude malfunctions caused by hydraulic and pressure forces. The flow distribution cylinder in the water inlet plenum prevents large hydraulic forces that would result if the inlet nozzle flow impinged directly on the control-rod shrouds. The weight of the fuel elements is 3.5 times the lifting force exerted on them by full-power water and steam flow. The thicknesses of the control-rod and fuel-element shrouds are sufficient to withstand the pressure differential between fuel coolant and control-rod coolant without significant deformation.

The separator plate between inlet and outlet plenums is designed with an initial deflection and sufficient thickness to maintain contact with the seat on the core support skirt when the pressure difference between water inlet and outlet plenums is at a maximum.

(5) Reactor vessel temperature must be kept uniform. The cylindrical portion and upper head are in contact with saturated water or saturated steam. The lower head, in contact with subcooled water, is made uniform by assuring good mixing of reactor inlet water. About 40 percent of the recirculated water flows into the region below the bottom grid plate and the vessel head. A small quantity of core inlet water (8650 lb/hr) is withdrawn with the control-rod seal water through the seal-water return lines attached to control-rod nozzles. Control-rod nozzles in the vicinity of the vessel head are thus kept at nearly the same temperature as the vessel head.

4.6.4 Water and Steam Flow Distribution

Figure 4.49 shows the main reactor water and steam flow paths and the reactor fluid plenums. Flow through controlled clearances (controlled flow) and leakage flows are shown in the enlarged sections of Fig. 4.49. The magnitudes of the various reactor flows at full power are shown in the flow diagram of Fig. 4.47.

Water from the forced-circulation pumps (10,750,000 lb/hr) enters the water inlet plenum through the four 16-in.-diameter nozzles. The boundaries of the water inlet plenum are formed by the flow distribution cylinder, the vessel, the separator plate between inlet and outlet plenums, and the bottom grid plate. Water from the inlet plenum flows into the active core region in two main parallel paths. The first flow (10,409,600 lb/hr) goes through the holes in the flow distribution cylinder and the bottom grid plates (see Fig. 4.50), into the core inlet plenum, and from there enters the 72 fuel assemblies. The core inlet plenum is bounded by the core grid plate, bottom grid plate, flow distribution cylinder, core support skirt, and control-rod shrouds (see Fig. 4.51). Table 4-8 summarizes inlet flow distribution and pressure drops, and the corresponding flow areas, coolant flow rates, and water velocities.

The second flow (266,960 lb/hr) is through the clearance between the scabbard and control-rod shaft at the bottom grid plate (see Fig. 4.52), into the 29 control-rod shrouds, and into the control-rod channels in the active core region. Control-rod channels are formed by the fuel-element shrouds and vertical support posts. Steam is generated in the active core region within the fuel elements and control-rod channels. Water and steam flow from the active core into the core outlet plenum (above the active core), where at least 50 percent of the generated steam is separated by gravity and is released to the steam dome (the space above the two-phase water level). The remainder of the steam and water flows into the 16 steam separators.

TABLE 4-8

SUMMARY OF FULL-POWER FLOW DISTRIBUTION AND PRESSURE DROP IN REACTOR INLET PLENUMS

connecting plenum, Fig. 4.51	upstream plenum area,* in. ²	minimum restricted area,* in. ²	downstream plenum area,* in. ²	coolant flowrate,* gpm	velocity in restriction, ft/sec	plenum-to- plenum pressure drop, ft of water
1 to 3	630	125	275	2200	6	0.84
1 to 2	1900	507	1200	5175	3.5	0.43
2 to 3	104	104	275	2445	8	0.41
3 to 4	275	207	300	4645	7.4	0.93
4 to 5	350	106	350	4470	13.2	3.06
2 to 5	1200	86	1000	2730	10.4	4.40

* Based on one quadrant.

The peripheral shroud and the perforated plate baffle above the core aid steam separation in the core outlet plenum. The peripheral shroud (Fig. 4.28) has rectangular openings with vanes on the outside that serve as a slight restriction to flow from the core outlet plenum to the separator inlet. The top of the shroud is at the same elevation as the top of the separator inlet. The perforated baffle (of 19 percent open area) above the core (Fig. 4.17) is below the two-phase level during full power operation and will decrease the two-phase turbulence. The benefits of the peripheral shroud and baffle were developed from experimental steam separation work done at Greendale Laboratory and described in Appendix C of Amendment No. 6 (ACNP-64572).

The remaining steam is separated in the separators. Water is discharged from the lower end of the separators into the water outlet plenum, which is bounded by the core support skirt, thermal shield, plenum separator plate, and separator support plate. Steam is discharged from the upper end of the separators into the steam dome. The water leaves the outlet plenum through holes in the thermal shield that are adjacent to the four 16-in.-diameter outlet nozzles. The thermal shield outlet holes are rounded and enlarged with respect to the outlet nozzle diameter to minimize outlet pressure losses. Calculations indicate that there will be no cavitation in the reactor outlet line. The steam flows from the steam dome into the steam outlet plenum through the steam dryers, where water is de-entrained to a steam quality of at least 99.9 percent. The steam leaves the reactor through two 8-in.-diameter nozzles.

Cooling water for the cylindrical portion of the reactor vessel enters the annulus between the thermal shield and the vessel through sixteen 4-in.-diameter holes in the thermal shield. The inlet holes are located slightly above the separator support plate. Water flows upward in the annulus by natural circulation and is returned to the core outlet plenum through sixteen 4-in.-diameter holes near the top of the thermal shield. Part of the coolant (143,300 lb/hr) flows downward in the annulus, through the controlled clearance between the thermal shield and the forced-circulation outlet nozzles (see Detail F, Fig. 4.49).

A small amount of coolant (8650 lb/hr) flows from the water inlet plenum into the 29 control-rod nozzles to keep nozzle and vessel temperature the same. This flow is controlled by valves in the seal water return lines.

Leakage flows are water flows which bypass the fuel elements, control rods, or steam separators. Leakage occurs at the mating surfaces of the reactor internal components, as shown in Fig. 4.49. The nominal leakage flow rates expected during full power operation are included in Fig. 4.47. Maximum leakage flows for full power operation are given in Table 4-9.

TABLE 4-9

LEAKAGE FLOW RATES

<u>leakage path</u>	desig. detail in Fig. 4.49	pressure drop, ft	<u>clearance, in.</u>		<u>flowrate, lb/hr</u>	
			<u>nominal</u>	<u>maximum</u>	<u>nominal</u>	<u>maximum</u>
1. water inlet plenum to water outlet plenum	D	16.0	0.005	0.010	44,790	98,550
2. core inlet plenum to control-rod shrouds	A and H	5.0	0.004	0.008	35,835	98,550
3. fuel elements to con- trol-rod channels	C	5.0	0.003	0.005	25,085	44,790
4. side reflector to water outlet plenum	E	2.0	0.015	0.030	21,500	43,000

4.6.5. Steam Void Distribution

The steam void distribution in various regions above the core grid plate is shown in Figs. 4.53 and 4.54. The abscissa in Fig. 4.54 refers to the reactor internal component elevations shown in Fig. 4.53. The steam voids plotted in Fig. 4.54 are the average in the region for a given elevation. The calculation of steam voids in the fuel assemblies is discussed in Sec. 4.6.7.1.1.

The steam voids in the fuel assemblies are dependent on recirculation flow and power level. The steam voids in the thermal-shield-to-vessel annulus and in the control-rod channels are dependent on the steam voids in the side reflector plenum (Fig. 4.49). The average voids in the thermal-shield-to-vessel annulus and in the control-rod channels are slightly higher than steam voids in the reflector (Fig. 4.54), because of good flow communication with the side reflector and additional voids resulting from heat generation in the control rods and vessel. Flow in the control-rod channels is maintained by the buoyant force resulting from the difference in density between coolant in the side reflector and coolant in the channels.

The steam voids in the side reflector are calculated conservatively to make the calculated steam voids in the control-rod channels and the thermal-shield-to-vessel annulus conservatively high. The slip ratio (steam velocity to water velocity) is calculated from Petrick's

correlation (4-5) for downflow, assuming 50 percent of the steam generated in the core is carried into the separators. Tests on the prototype separator indicate that only about 40 percent of the steam will be carried into the separators. The steam voids calculated for the top of the separator inlet are assumed to be uniform down to the bottom of the separator inlet. Actually, the steam voids will decrease from a maximum at the top of the inlet to nearly zero at the bottom of the inlet.

The steam voids in the side reflector below the bottom of the separator inlets will be negligibly small since the velocity in this region is less than 0.2 ft/sec. For water velocities less than 1 ft/sec, significant amounts of steam cannot be carried downward.

The average steam voids in the core outlet plenum (see Fig. 4.54) are assumed to be 40 percent. Actually, the steam voids will increase from 36 percent at the outlet of the fuel assemblies to a value higher than 40 percent at the two-phase interface. The main effect of higher voids in this region is a higher two-phase operating water level. However, a conservatively large estimate of steam voids in the side reflector tends to offset an underestimate of voids in the core outlet plenum. The two-phase level will be measured during reactor operation.

4.6.6 Pressure Distribution

The pressure distribution in the reactor is used in determining the required thicknesses of some of the reactor internal components (such as fuel-element shrouds and the separator plate between the water inlet and outlet plenums). The pressure distribution is also used in determining leakage and controlled flowrates.

Figure 4.55 shows pressure distribution in the various regions as a function of vessel height. The pressure decrease from the bottom of the vessel, in regions having upward flow (fuel-element and control-rod regions), is the sum of the head loss resulting from flow and from static head of water. The pressure decrease in regions having downward flow (side reflector) is the difference between the static head and the head loss resulting from flow. The head loss in the various regions is described in Sec. 4.6.2, and the static head in various regions is obtained using the steam void distribution (described in Sec. 4.6.5).

Pressure within a fuel element is higher than that within control-rod channels (Fig. 4.55), so leakage is from fuel coolant to control-rod coolant. The maximum pressure difference between the inside and outside of a fuel element shroud is 5 ft (at the bottom of the shroud). The pressure difference between control-rod channels and the side reflector is small since there is good flow communication between them. The pressure difference across the plenum separator plate is 16 ft.

4.6.7 Design Criteria and Calculated Performance

4.6.7.1 Fuel. Thermal and hydraulic design criteria pertaining to the fuel assemblies are as follows:

(1) The exit steam void fraction from the fuel generating the highest power shall be less than 70 percent at any steady-state power level, and the average power density in the core shall be less than 45 kw/liter for 100 percent power operation. The limitation on exit void fraction and power density ensures operation in a hydrodynamically stable region. (4-6)

(2) The maximum UO_2 temperature shall be less than the melting point (4950 F) for steady-state operation at 115 percent of rated power. This criterion limits the UO_2 temperature to that for other operating UO_2 -fueled reactors.

(3) The ratio of minimum burnout heat flux to maximum fuel element heat flux at every point in the core shall be greater than 1.5, based on limit curves (Sec. 4.6.7.1.3), for steady-state or transient operations. This is the most important thermal criterion for the fuel assemblies, since fuel cladding may fail if fuel element heat flux exceeds burnout heat flux.

Calculations of full power performance determined steam void distribution, velocity distribution, and fuel temperature distribution in the core. This information is used in other analyses in addition to ensuring that thermal and hydraulic criteria are met. Steam void and fuel temperature distributions are used in core physics calculations. Fuel temperature distribution is used in stress analysis of fuel elements.

4.6.7.1.1 Steam Void and Velocity Distribution. Steam void fraction and velocity distribution are calculated with a digital computer code. This code evaluates void and velocity in five radial positions and in ten axial segments of five fuel elements. The power level in each of the five radial positions is selected to approximate the actual transverse (radial) power distribution in the core. With a slight modification of input orders, the code is applicable to natural-circulation or forced-circulation reactors.

The heat flux distributions used in performance calculations are shown in Figs. 4.34 and 4.35. These distributions are based on the initial core and on the rod insertion described in Sec. 4.5. The calculation of flux distribution is dependent on void distributions, and the void distribution calculation is dependent on flux distribution. Several iterations of flux and void distributions were made to achieve satisfactory convergence. Figure 4.35 shows the flux distributions and the void distributions used to compute them.

The steam void distributions (Figs. 4.65 and 4.57) are calculated for five regions in the core. Fuel elements of comparable power are grouped in the same region for the calculation. The total power of all elements must be the same for the approximated distribution and for the distribution calculated by physics. Thus, the powers used for some elements in the calculation are slightly less than the values calculated by physics, and powers used for other elements are slightly greater than the calculated values. The maximum exit void fraction calculated for the element of highest normalized power is 0.575, whereas the limiting criterion is 0.70. For operation at 120 percent power and 100 percent flow, the maximum exit void fraction is 0.66 (Sec. 4.6.8).

The inlet water velocity varies from 6.00 ft/sec for the element of highest power to 6.61 ft/sec for the element of lowest power. The lower velocity is a result of the increased two-phase friction in the high-power element. The minimum burnout ratio will occur in the highest-power element, since burnout heat flux decreases as velocity decreases in the low steam quality range. The maximum exit steam quality is 0.112.

The pump head required to force the water through the core at 100 percent power is equal to the pressure drop caused by flow in the core. There is no available net thermal head, since the average void fraction in the core is about the same as the average void fraction in the downcomer (Fig. 4.54). For the results shown in Fig. 4.56, the required pump head for the core is 6.0 ft of water, with an average steam void fraction of approximately 20 percent in both the side reflector and core.

4.6.7.1.2 Fuel Temperature Distribution. The fuel temperature distribution is calculated with a digital computer code for five regions of the core and 10 axial segments. Robertson's ⁽⁴⁻⁷⁾ thermal conductivity data and method of analysis are used. This method includes neutron flux variation within a fuel rod. An interface thermal conductance of 1000 Btu/(hr)(ft²)(°F) is used for the cladding-to-UO₂ interface. This value is conservative since it is based on the initial gap between UO₂ and the cladding. Post-irradiation examination of UO₂ fuel shows that the oxide cracks and then expands against the cladding.

For the hottest fuel rod section in the reactor, the Robertson method gives an effective thermal conductivity of 1.1 Btu/(hr)(ft)(°F) for heat conductance from the fuel rod center to the inside surface of the cladding. This effective conductivity value is for cracked oxide since it is based on irradiated fuel. More recent UO₂ fuel irradiations ⁽⁴⁻⁸⁾ indicate that the effective UO₂ thermal conductivity may be 10 to 15 percent higher than the values reported by Robertson.

The UO₂ temperature distribution in the core is shown in Fig. 4.58. The temperatures shown are the average in a fuel assembly, as a function of length, for the five different power regions shown in Fig. 4.56. The average UO₂ temperature in the core is 930 F.

The UO_2 temperature distribution in the hottest rod for 100 percent power and 120 percent power is shown in Figs. 4.59 and 4.60. The maximum temperature at 120 percent power, even including the conservative factors discussed above, is 4750 F, which is less than the design criteria of 4950 F at 115 percent power.

4.6.7.1.3 Minimum Burnout Ratio. The minimum burnout ratio at a given power is defined as the ratio of the burnout heat flux for the fluid conditions at that power level to the maximum heat flux in a fuel rod at any location in the core. This ratio is an index of the safety margin that exists for those operating conditions. In determining burnout heat flux, experimental data from electrically heated test sections are used. These tests simulate the fuel-element geometry and duplicate the pressure, mass velocity, and water inlet subcooling. Local flux peak factors and factors that account for overpower during control maneuvers are included in the determination of maximum heat flux.

The large body of data accumulated in the last decade is utilized in the selection of an appropriate burnout heat flux correlation to determine the minimum burnout ratio. For this reason, a report on burnout heat flux design correlations for LACBWR is included as Appendix B. This report includes the basis for selection of correlations for the power range of 60 to 120 percent, as well as for the reduced flows and pressures encountered during startup. The design correlation for the range of power operation is that given in APED-3892⁽⁴⁻⁹⁾ and reproduced below, with the limits of its application. This correlation gives the lower limit of experimental burnout data. Most of the experimental data used to derive the correlations were obtained for annular test sections, but some data for multirod test sections are included. The pressures tested ranged from 1000 to 1400 psia, and the range of the mass velocities and steam qualities tested includes the range of these quantities encountered in LACBWR during power operation between 60 and 120 percent.

The burnout heat flux correlations used in LACBWR design are:

$$\frac{q''_{bo}}{10^6} = 0.705 + 0.237 \left(\frac{G}{10^6} \right) \quad x < x_1 \quad (1)$$

$$\frac{q''_{bo}}{10^6} = 1.634 - 0.270 \left(\frac{G}{10^6} \right) - 4.710 \cdot x \quad x_1 < x < x_2 \quad (2)$$

$$\frac{q''_{bo}}{10^6} = 0.605 - 0.164 \left(\frac{G}{10^6} \right) - 0.653 \cdot x \quad x_2 < x \quad (3)$$

where q_{bo}'' is the burnout heat flux, G is the coolant mass velocity (lb/hr-ft²) and x is the steam quality, expressed in terms of coolant mass flowrate, as follows:

$$x_1 = 0.197 - 0.108 \left(\frac{G}{10^6} \right) \quad (4)$$

$$x_2 = 0.254 - 0.026 \left(\frac{G}{10^6} \right) \quad (5)$$

At a pressure (P) other than 1000 psia, the burnout heat flux is corrected by adding a pressure-dependent term:

$$(q_{bo}'')_P = (q_{bo}'')_{1000} + 440 (1000 - P) \quad (6)$$

The range of application for Equations 1 through 6 is limited to recirculation flowrates of 40 percent to 120 percent (G from 0.2×10^6 to 1.2×10^6 lb/hr-ft²) and pressures from 600 to 1400 psia. The lower mass velocity is less than that recommended in APED-3892. However, a check of more recent data for low mass velocities showed that the correlation can be extended to this mass velocity, as discussed in Appendix B.

The closest approach to burnout during steady-state operation will be at 120 percent power. The design correlation and experimental data applicable to LACBWR are compared for conditions close to those that will exist for 120 percent power operation (Fig. 4.61). The experimental data at various pressures lie above the design correlation line for the corresponding pressures. Data from Matzner⁽⁴⁻¹⁰⁾ for a 12-rod bundle gave burnout heat fluxes below the correlating lines, but they are not considered applicable to LACBWR since the rod spacing was much smaller and the rods were on a triangular pitch. Experimental data from Janssen et al⁽⁴⁻¹¹⁾ for an eccentric rod in a tube showed that, as the rod was moved from the concentric position to various eccentric positions, the burnout heat flux began to decrease for minimum spacings of less than 0.060 in. between the rod and tube. For spacings of 0.060 in. or greater, there was no appreciable decrease in burnout heat flux for the eccentric positions from that for the concentric position. The minimum clearance between an outer rod and the fuel-element shroud in LACBWR is 0.060 in., to minimize the possibility of burnout caused by bundle eccentricity in the shroud. These data are discussed further in Appendix B.

The burnout ratios for the hottest fuel rod at 100 percent and 120 percent power are given in Fig. 4.62. The peak heat flux is calculated for a corner rod in the fuel element generating the greatest power. The conservative assumptions used in calculating heat flux and steam quality are:

(1) Calculation of the local peaking factor assumes water in the control-rod channel, though, actually, some steam may be present (see Sec. 4.6.7.2).

(2) The flow through the fuel assemblies is assumed to be 93.3 percent of the rated recirculation flow (30,000 gpm), whereas the calculated minimum flow through the fuel assemblies is 94.7 percent based on the maximum clearances permitted by manufacturing tolerances at all mating surfaces that restrict fuel bypass flows.

(3) The steam quality for 120 percent power is calculated using a flowrate of 100 percent. When the reactor is on automatic control, flowrate will increase as power is increased (see Sec. 4.6.8). Since the pumps can deliver a flow of at least 110 percent, the flow would be at least 110 percent at 120 percent power, except for rapid transients. For rapid transients, the heat flux on the rod surface lags behind the increase in neutron flux.

The minimum calculated burnout ratio is 1.55 at 120 percent power and 1.86 at 100 percent power (Fig. 4.62), meeting the criteria of a minimum burnout ratio greater than 1.5 at 120 percent power. During operation, recirculation flow and the flux distribution within the core will be measured so that actual burnout ratios can be accurately computed.

4.6.7.2 Control Rods. Thermal and hydraulic design criteria pertaining to the control rods are as follows.

(1) The maximum steam volume fraction leaving control-rod channels will be less than 0.60 at 100 percent power. This criterion is imposed to preclude the possibility of flow instabilities.

(2) The minimum ratio of burnout heat flux in the external coolant channel to maximum heat flux on the sheath will be greater than 4.0 at 100 percent power. This criterion will prevent local overheating of the sheath.

(3) The minimum ratio of the critical heat flux in an internal flow channel to the maximum heat flux on the poison rod cladding will be greater than 2.0 at 100 percent power. The term "critical heat flux" is used instead of "burnout heat flux" for the heat flux at which a transition from nucleate to film boiling occurs, since burnout (i.e., melting of the cladding) would not occur even if there were steam blanketing in an internal channel (see Sec. 4.6.7.2.2). This criterion, however, will ensure that the B_4C cladding temperature is only a few degrees above water temperature. Control-rod lifetime is based on cladding strength, and lower cladding temperatures result in longer lifetime.

(4) The maximum B_4C temperature will be less than the melting point at 100 percent power. This criterion is based on considerations of control-rod lifetime.

4.6.7.2.1. External Coolant Channels. Figure 4.63 shows coolant flow paths for a fully inserted control rod. Coolant for a control rod enters the control-rod shroud in the core inlet plenum through the clearance between the scabbard (attached to the bottom grid plate) and the control-rod shaft. The subcooled water flows through the control-rod shrouds and into the control-rod channel in the active core section. The flow is not appreciably affected by the position of the control rod in the core, since the friction loss in the control-rod channel, compared with that in the entry flow restriction, is small.

A clearance between the vertical posts and the fuel element shrouds provides flow communication between control-rod channels in the active core section and between control-rod channels and the side reflector region. The friction loss in the active core section of the control-rod channels and in the reflectors is small, so the pressure gradient in the control-rod channels in the active core region is about the same as in the side reflector. The flow in the active section of control-rod channels is determined by the difference between the average density of the two-phase mixture and the average density of other control-rod channels and the side reflector.

The heat generation in a fully inserted control rod is highest for a rod at the edge of the blade and lowest for a rod nearest the center. For 100 percent power operation, the maximum heating in the edge rod (in the highest region of core flux) is calculated to be 4×10^4 Mev/cm³-sec in the boron carbide and 1.33×10^{14} Mev/cm³-sec in the stainless-steel cladding and sheath. The corresponding maximum heat fluxes on the poison rod and sheath are given in Table 4-10. The axial distribution of heat flux in the active core region is the same as that for the fuel elements (Fig. 4.35).

The control-rod channel pressure gradient for various flows in the active core section was calculated by the digital computer code used in fuel channel calculations, that accounts for two-phase friction, steam-to-water velocity ratio, and buoyance effects. The flow required to obtain an average density slightly greater than side reflector density is 20.3 gpm for each control rod. The pressure gradient for this flow, assuming no leakage to or from the control-rod channel, is compared with that in the side reflector (Fig. 4.64). The side reflector vertical pressure gradient is obtained as in Fig. 4.55, with steam voids conservatively calculated to be about 40 percent down to the bottom of the separator inlet. Figure 4.55 also shows the fuel channel pressure gradient.

Coolant conditions within the channel for the minimum flowrate of 20.3 gpm are given in Table 4-10. The average void fraction in the hottest control rod is 24 percent, and the exit void fraction is 41 percent. These void fractions are well below the criteria of 0.60 for 100 percent power.

The burnout heat flux in the external coolant channel is evaluated from Weatherhead's correlation⁽⁴⁻¹²⁾ for low-velocity water. For low-velocity water, burnout heat flux does not occur until the exit steam quality is greater than 70 percent. For the external

coolant channel, the burnout heat flux is $7 \times 10^5 \text{ Btu}/(\text{hr})(\text{ft}^2)$. The maximum heat flux on the sheath for normal conditions ($12,500 \text{ Btu}/(\text{hr})(\text{ft}^2)$) is that from heat generation in the sheath, since poison rod heat is removed by internal coolant channels (see Sec. 4.6.7.2.2). However, for conservatism in computing minimum burnout ratio, all the heat from the half-thickness of the control-rod blade ($60,000 \text{ Btu}/(\text{hr})(\text{ft}^2)$) is used. The minimum burnout ratio at 100 percent power is thus 11, which well satisfies the criteria. Exit steam voids and minimum burnout ratio at 120 percent power even satisfy the criteria for 100 percent power.

4.6.7.2.2 Internal Coolant Channels. The control rod consists of B_4C -filled Inconel tubes inside perforated stainless-steel sheaths of cruciform shape. The holes in the sheaths permit coolant flow in the spaces between the rods and the sheath (the internal coolant channels). The flow area of the hole in the sheath is much larger (0.44 in.^2) than the flow area of the internal channels it serves (0.0415 in.^2), so flow through the holes is unrestricted (Fig. 4.63). The internal flow channel areas are sufficiently large to allow nucleate boiling, ensuring cladding temperatures of only a few degrees above the water temperature.

The minimum ratio of critical heat flux to the actual heat flux on the poison rod is 2.75 at 100 percent power. This ratio occurs at the region of peak heat flux in the reactor. It was assumed that heat from the B_4C cladding and half of the sheath goes into the internal channel. The maximum heat flux on the poison rod is $30,000 \text{ Btu}/(\text{hr})(\text{ft}^2)$.

The maximum cladding temperature was also calculated, assuming that an internal channel was blocked so that steam formed in the channel. For this condition, about 10 percent of the heat would be conducted through the steam, and the rest would be conducted in the cladding to the area in contact with the sheath. On the basis of calculational results given in Table 4-10, it is concluded that steam blanketing of an internal channel would not cause failure of a poison rod.

Control-rod temperatures are summarized in Table 4-10. It is seen that the maximum B_4C temperature, even assuming steam in an internal channel, is considerably less than the melting point (4900 F). The maximum B_4C temperature at 120 percent of full power would be 2120 F.

4.6.7.3 Steam Separators. The performance of a prototype LACBWR separator was determined by the steam-water tests described in Amendment No. 6 to Application for Construction Authorization (ACNP-64572). The size of prototype separator inlets is 1.1 in. x 78 in., compared with 1.1 in. x 72 in. for the LACBWR separators. The steam carryunder and pressure drop as a function of water flowrate are shown in Figs. 4.65 and 4.66. The flow at which steam carryunder increases significantly is about 1850 gpm. Below that point, steam carryunder is less than 2 v/o. Pressure losses through the separator are due primarily to inlet losses. The total pressure is less than the design criterion of 2 ft of water that is used to determine system head loss.

TABLE 4-10

SUMMARY OF CONTROL-ROD FLOW AND TEMPERATURE FOR
100 PERCENT POWER AND 30,000 GPM FLOW

heat flux on poison rod cladding, Btu/(hr)(ft ²):	
average in reactor	7,150
maximum	30,000
core radial peak-to-average	1.49
core axial peak-to-average	1.87
peak-to-average across blade	1.5
minimum ratio: $\frac{\text{critical heat flux in internal channel}}{\text{maximum heat flux on poison rod}}$	2.75
heat flux on sheath, Btu/(hr)(ft ²):	
maximum (normal conditions)	12,500
maximum (internal flow channel blocked)	60,000
minimum ratio: $\frac{\text{burnout heat flux in external channel}}{\text{maximum heat flux on sheath}}$	11
minimum coolant flow per rod, gpm	20.3
void fraction for inserted rod, percent:	
average	24
exit	41
steam quality at exit, fraction	0.09
reactor water temperature, F	577.5
maximum sheath temperature, F:	
normal conditions	580
internal flow channel blocked	600
maximum cladding temperature, F:	
normal conditions	585
internal flow channel blocked	718
maximum B ₄ C temperature, F:	
normal conditions	1,865
internal flow channel blocked	1,998

The LACBWR separator inlet length was reduced 6 in. to increase the flow required for a significant increase in steam carryunder. For the LACBWR separator, the calculated flow at which carryunder begins to increase significantly is 2200 gpm, so there is a satisfactory margin above the maximum flow capability (1850 gpm). The pressure drop for the LACBWR separator at a given flow rate is about 20 percent higher than the measured pressure drop for the prototype separator.

These tests also demonstrated that not all of the steam generated in the LACBWR core (220 cfm per separator) would be carried into the separators. With a perforated baffle over the riser, the maximum flow into each separator was 90 cfm (a total of about 40 percent of the generated steam). Even with a solid baffle over the riser, there was a flow of only 180 cfm into each separator. For LACBWR operation with a perforated baffle over the core, less than 40 percent of the generated steam is expected to be carried into the separators. The LACBWR path from the center of the riser to the separator inlets is longer than in the tests, and steam will have a longer time to be separated by gravity in the core outlet plenum. For the maximum flow of steam into the prototype separators (180 cfm), the quality of the exhaust steam was greater than 0.96. The remaining moisture in the exhaust steam is separated in the steam dome. The splash plate on the exhaust prevents the exhaust steam from impinging on the steam dryers.

In order to ensure satisfactory steam separator performance, the separator inlet submergence should be within specified limits. Inlet submergence is defined as the height of water (expressed as equivalent saturated water) above the top of the separator inlet. The upper limit is imposed to prevent entraining excessive water with the steam. The lower limit is imposed to prevent the top of the vortex from dropping below the top of the separator inlet, which would increase the pressure drop across the inlet. These phenomena have been observed in Allis-Chalmers tests. For the LACBWR performance tests, the inlet submergence was varied from 0 to 4 ft of saturated water (Fig. 4.65). The limits specified for LACBWR design are:

(1) For 100 percent flow, maximum submergence is 4.0 ft, and the minimum is 2.0 ft.

(2) For 50 percent flow, maximum submergence is 3.0 ft, and the minimum is 1.0 ft.

4.6.7.4. Steam Dryers. The steam dryers are designed to discharge steam of 99.9 percent quality at 720,000 lb/hr and 1300 psia, provided the quality of the entering steam is 95 percent or higher. The moisture separated in the dryers is drained back to the water in the reactor (Fig. 4.14). The pressure drop across the dryers is less than 4.0 in. of water for full power operation. Details of the steam dryer design are given in Sec. 4.3.7.

The steam velocity in the steam dome is about 1.3 ft/sec. It has been experimentally determined that, if the steam-water interface is 8 in. or more below the bottom of the steam dryers, the steam entering the dryers will have a quality of at least 98 percent. The actual operating water level is 7 ft below this upper limit.

The upper nozzle for water level measurement is on the downstream side of the dryers (Fig. 4.53). Therefore, the indicated water level will be corrected for the pressure drop through the steam dryers. The maximum correction in water level is 4 in. of water (at full power).

4.6.7.5 Reactor Vessel. The reactor vessel is designed for a temperature of 650 F. For full power operation, the water and steam temperature at the cylindrical section and the top head is 577.5 F (the saturation temperature at 1300 psia). The lower head is in contact with subcooled water (565 F). The small amount of heat generated in the vessel by gamma rays from the core is removed from the inside surface. The outside of the vessel is covered with 2 in. of insulation ($k = 0.03 \text{ Btu}/(\text{hr})(\text{ft})(^\circ\text{F})$).

The gamma heat in the cylindrical portion of the vessel above the separator support plate is removed by natural circulation in the annulus formed by the internal thermal shield and the vessel. The cylindrical portion of the vessel below the separator support plate is cooled by water from the side reflector, which flows downward in the annulus and out through a flow restriction around each of the four forced-circulation outlet nozzles. The lower vessel head is kept at a uniform temperature by a large forced-flow of recirculating water between the bottom grid plate and the head. Cold control-rod seal water (80 F) is removed from the control-rod nozzles, and some 565 F water is removed from the reactor inlet plenum. The resultant ratio of inlet plenum flow to seal water flow is 2 to 1, ensuring hot-water contact with the control-rod nozzles in the vicinity of the vessel head. There is very little gamma heating in the upper portion of the cylindrical vessel and the upper head, which are in contact with steam and are adequately cooled by this steam.

The following are thermal and hydraulic design criteria to preclude temperature and over-temperature oscillations (from hydraulic oscillation) in the vessel wall:

- (1) The maximum vessel wall temperature will be less than 650 F at 195 Mwt and 1300 psia. The higher power (195 Mwt) is used because of the expected higher power output (60 Mwe) for future cores.
- (2) Steam void fraction at the outlet of the annulus will be less than 70 v/o at 195 Mwt and 1300 psia.
- (3) Critical heat flux shall be a factor of 4.0 times the maximum flux at 195 Mwt.

The maximum temperature in the vessel will occur at the core elevation of the neutron flux peak (Fig. 4.35). The volumetric gamma heating rate is calculated to be 33,000 Btu/(hr)(ft³); the maximum heat flux on the inside surface of the vessel is 3350 Btu/(hr)(ft²), and the heat transfer coefficient is 1200 Btu/(hr)(ft²)(°F). The temperature difference across the wall is 12 F, and the film temperature drop on the inside surface is 3 F. Thus, the maximum vessel wall temperature is 592 F, whereas the design temperature is 650 F. The wall thermal stress caused by the maximum temperature drop (12 F) across the wall has been shown to be acceptable (Appendix A).

A sketch of the thermal shield and side reflector region is shown in Fig. 4.67. Saturated water enters the annulus through the holes near the core support plate and leaves through the holes near the top of the thermal shield. The flow from the annulus into the forced-circulation outlet nozzle that results from the pressure drop through the steam separators is restricted to 400 gpm by means of a 0.1-in. clearance at the nozzle inlet.

Natural convection in the annulus depends on the pressure difference between the lower and upper holes in the thermal shield in the reflector region, which is equivalent to 8.0 ft of saturated water at rated operating conditions. A velocity of 0.125 ft/sec in the thermal-shield-to-vessel annulus results in a pressure drop of 8.0 ft, for a total natural-circulation flow of 60 gpm. At 195 Mwt, the maximum gamma heating at the inside surfaces of the thermal shield and vessel wall is 1.02 w/cm³ and 0.34 w/cm³. These values correspond to a maximum heat flux of 4000 Btu/(hr)(ft²) from the thermal shield and 3350 Btu/(hr)(ft²) from the pressure vessel. For a conservative calculation of natural-circulation flow, the value of total heat entering the annulus is doubled. The two-phase friction loss is negligible, relative to the static head. For a slip ratio of 3.0, the void fraction at the exit is 0.41 and the steam quality is 0.12. Calculations were made using 15 axial segments in the annulus. The void distribution is shown in Fig. 4.54.

It is concluded from the calculations and from critical heat flux experimental data (e.g., Ref. 4-13) that the ratio of the critical heat flux in the annulus to the maximum heat flux at 195 Mwt is greater than 10.0. The maximum exit void at 195 Mwt is 0.41, well below the criterion limit of 0.70. Therefore, hydraulic oscillations in the annulus are extremely unlikely.

4.6.8. Operation at Various Power Levels

For automatic operation between 60 percent and 100 percent power, the design method of control is to vary the recirculation flowrate without moving the control rods. The relationship of power, flow, voids, and subcooling were calculated by the digital computer methods described in Sec. 4.6.7.1.1. A discussion of the calculations in determining expected flowrates at various power levels appears in Amendment No. 6 to the LACBWR Application for Construction Authorization (ACNP-64572). It was assumed that the same proportion (93.2 percent) of the total flow would pass through the fuel elements both for partial-power and for full-power conditions. Amendments Nos. 6 and 7 (ACNP-64604) give the transient variations of power and flow during plant maneuvers.

Figure 4.68 gives the minimum flowrates that satisfy the thermal and hydraulic core criteria for a given power level. The criteria for minimum flowrates are a maximum exit steam volume fraction of less than 0.70 and a minimum burnout ratio greater than 1.5. The burnout heat flux ratios and the maximum exit voids for several flowrates were computed for the 40 to 120 percent power range and for feedwater return temperatures of 292 F and 577 F (saturation temperature). Minimum flowrates were then selected in accordance with the maximum exit void and minimum burnout ratio criteria. The normal operating curve is shown for comparison in Fig. 4.68.

The primary limitation for minimum flow is the exit steam volume fraction for power levels of less than 100 percent. For normal feedwater temperature, the maximum exit steam volume fraction is about 65 percent (Table 4-11), while the minimum burnout ratio increases from a minimum of 1.55 at 120 percent power and 100 percent flow to a value of 3.6 at 40 percent power and 15 percent flow.

TABLE 4-11

LIMITING CONDITIONS AT VARIOUS POWER LEVELS

power level, %	<u>40</u>		<u>60</u>		<u>80</u>		<u>100</u>		<u>120</u>	
feedwater temperature, F	<u>577</u>	<u>292</u>	<u>577</u>	<u>292</u>	<u>577</u>	<u>292</u>	<u>577</u>	<u>292</u>	<u>577</u>	<u>292</u>
maximum exit void, fraction	62	60	65	61	66	62	68	64	68*	65
minimum burnout ratio	3.5	3.6	2.5	2.4	2.0	2.0	1.6	1.7	1.5*	1.55
minimum flow, %	19	15	31	25	49	43	74	67	110*	100

*Reactor power = 117 percent

4.6.9 Water Level

The operating range (0 to 100 percent power) for water level is selected to meet the limitations given in Sec. 4.6.9.1 with a fixed setting on the indicated water levels. The lower nozzle for water level measurement (Fig. 4.53) is below the regions in the reactor that contain significant steam voids (Fig. 4.54). Thus, the indicated water level is a measure of water inventory in the reactor. During initial operation, the two-phase level will be measured, and the indicated level will be correlated with it as a function of power level and flowrate.

4.6.9.1. Limitations to Water Level (see Table 4-12)

(1) Containment vessel pressure. The maximum water level at 577.5 F (with no steam voids) in the reactor vessel must be at least 122 in. below the vessel flange so that containment shell pressure would be below the design pressure if there is a primary system rupture.

(2) Moisture carryover. The maximum two-phase level for satisfactory steam dryer performance is 8 in. below the bottom of the steam dryers.

(3) Water recirculation after scram. In order to assure water recirculation after scram, the water level after a scram should be above the lowest openings in the peripheral shroud. This requirement is satisfied if the vessel water content is equivalent to a saturated water level of at least 2 ft above the top of the active core.

(4) Steam separator performance. For satisfactory steam separator performance at full power and full flow, the equivalent saturated water level above the top of the separator inlet (i.e., the submergence) must be between 2 and 4 ft (see Sec. 4.6.7.3). An additional requirement is that, for recirculation rates of 50 percent or lower, the submergence should be between 1 and 3 ft of water. The calculated pressure drop of 0.2 ft in the downcomer must be added to these submergence requirements. The permissible operating range at full flow is for the equivalent of 2.2 to 4.2 ft of saturated water above the separator inlet. The permissible operating range for flows of 50 percent or less is 1 to 3 ft of saturated water, since pressure drop in the downcomer is less than 0.05 ft. For flowrates of 20 percent or less, the inlet does not have to be submerged, since steam will not be entrained in the recirculating water.

4.6.9.2. Operating Water Levels. Steam voids in the reactor for 100 percent power are given in Sec. 4.6.5. The two-phase levels given in Table 4-12 are based on void distribution for 100 percent power even for the limitations that apply to 50 percent flow. This basis was used because the anticipated method of operation is to control power by varying recirculation flow and keeping the control rods fixed over the 60 to 100 percent power range. For this method of operation, the core voids for 50 percent flow will be about the same as for 100 percent flow. The selected operating water level range is from 666 ft - 0 in. to 666 ft - 10 in. (equivalent saturated water elevation), which corresponds to a two-phase operating level of 669 ft - 9.5 in. to 671 ft - 1.5 in. The description of water level control and measurement is given in Sec. 5.1.1.2.2.

TABLE 4-12

CALCULATED WATER LEVELS FOR LIMITING CRITERIA OF SEC. 4.6.9.1

limitation	two-phase elevation	equivalent saturated water elevation*
(1) maximum level as a result of containment shell pressure requirement (100 percent power)	675 ft 3 in.	670 ft 7 in.
(2) maximum level as a result of steam dryer requirements (100 percent power)	678 ft 6 in.	672 ft 3 in.
(3) minimum level as a result of recirculation requirements after scram	668 ft 4 in.	665 ft 8 in.
(4) steam separator requirements:		
(a) maximum level for 100 percent flow	673 ft 1.5 in.	668 ft 0 in.
(b) minimum level for 100 percent flow	669 ft 9.5 in.	666 ft 0 in.
(c) maximum level for 50 percent flow**	671 ft 1.5 in.	666 ft 10 in.
(d) minimum level for 50 percent flow**	667 ft 9.5 in.	664 ft 10 in.

* Represents water elevation in reactor if steam voids were not present. This elevation is the level in a saturated water column, since the lower nozzle on the column is below all steam voids in the reactor.

** Represents levels calculated for steam voids equivalent to the steam voids at full power and full flow.

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41-503-090

LACBWR REACTOR VESSEL AND INTERNAL COMPONENTS

ALLIS-CHALMERS
MANUFACTURING COMPANY
ATOMIC ENERGY DIVISION
BETHESDA, MARYLAND

**LACBWR
REACTOR VESSEL AND
INTERNAL COMPONENTS**

DETAILED DESIGN-CONTRACT NO. AT (11-1) 850

DWG. NO. 41-503-090

SCALE 1/4" = 1'-0" SHEET OF 1

REV. 1

FIG. 4.1

LACBWR REACTOR VESSEL OUTLINE

FIG. 4.2

LACBWR CORE INTERNALS INNER THERMAL SHIELD ELEVATION

FIG. 4.3

LACBWR INNER THERMAL SHIELD CORE INTERNALS-SECTIONS

FIG. 4.4

41-503-124
SHEET 1 OF 1

MATERIAL - INCONEL

ALLIS-CHALMERS MANUFACTURING COMPANY ATOMIC ENERGY DIVISION NUCLEAR POWER DEPARTMENT MORGANTHAU, W. VA.	
LACBWR CORE INTERNALS PLENUM SEPARATOR PLATE DETAILED DESIGN - CONTRACT NO. AT (11-1) 590	
DWG. NO. 41-503-124	REV. 02
SHEET 1 OF 1	

LACBWR CORE INTERNALS PLENUM SEPARATOR PLATE

FIG. 4.6

41-503-107

ALLIS-CHALMERS MANUFACTURING COMPANY STEAM ENGINE DIVISION NUCLEAR POWER DEPARTMENT ROCKFORD, ILL.	
LACBWR CORE INTERNALS BOTTOM GRID ASSEMBLY	
REPLACES DESIG. CONTRACT NO. AT 11-1500	
REV. NO. 41-503-107	04
SHEET	OF

LACBWR CORE INTERNALS BOTTOM GRID ASSEMBLY

FIG. 4.7

FIG. 4.8

CONTROL ROD GUIDE

FIG. 4.9

LACBWR CORE INTERNALS LOWER GRID PLATE

FIG. 4.10

41-503-109

LACBWR STEAM SEPARATOR

FIG. 4.12

ALLIS-CHALMERS
MANUFACTURING COMPANY
ATOMIC ENERGY DIVISION
PITTSBURGH, PENNSYLVANIA
LACBWR
STEAM SEPARATOR
DESIGNED BY: CONTRACT NO. AT (11-1) 880
DWG. NO. 41-503-109
SCALE: SHEET 1 OF 1

LACBWR CORE INTERNALS SHOCK SHIELD

FIG. 4.13

41-503-113

LACBWR CORE INTERNALS STEAM DRYER ASSEMBLY

FIG. 4.14

CORE INTERNALS FAILED FUEL DETECTION AND EMERGENCY COOLING SYSTEM ASSEMBLY

FIG. 4.15

PAGES 4.16 to 4.21
WERE INTENTIONALLY
LEFT BLANK

41-503-031
SHEET 1 OF 1

41-503-031

ALLIS-CHALMERS
MANUFACTURING COMPANY
ATOMIC ENERGY DIVISION
NUCLEAR POWER DEPARTMENT
WASHINGTON, D. C.

**LACBWR
FUEL ELEMENT
DETAILS**

DETAILS DESIGN CONTRACT NO. AF 33-11-1000

DRW. NO. 41-503-031

SHEET 1 OF 1

Technical drawings of the reactor core showing View A-A and Section B-B.

VIEW A-A

GRID
SST TYPE 304

5.46
IN.

5.46
IN.

0.06

SECTION B-B

3 EQUAL SPACES @ .365" ± 0.008

3 EQUAL SPACES @ .512" ± 0.008

0.365

INTERMEDIATE GRID
INCONEL 600

3 SST TYPE 304 PLACES

LACBWR FUEL ELEMENT

FIG. 4.23

ALLIS-CHALMERS
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ATOMIC ENERGY DIVISION
NUCLEAR POWER DEPARTMENT
WASHINGTON, D. C.

LACBWR

FUEL ELEMENT

DETAILED DESIGN—CONTRACT NO. AT (11-1) 99

DRWG. NO. **41-503-032**

SHEET 1 OF 1

MAILED DESIGN—CONTRACT NO. AT ()

NO. 41-503-102

NOTES:

1. EACH POST SHALL BE IDENTIFIED WITH A TWO DIGIT NUMBER STARTING WITH 01, MARKING TO CONFORM TO SPEC 43-101-12L-401.
2. ALL DIMENSIONS SHOWN ARE AFTER PICKLING AND CORROSION TESTING.
3. GEOMETRIC FORM DIMENSIONING ARE TYPICAL FOR ALL GROUPS.
4. CLEAN ASSEMBLY PER 43-101-765-401.

NUMBERS SHOWN THUS 3 ARE PART NUMBERS
AND ARE PREFIXED THUS 41-503-10;
FOR BILL OF MATERIAL SEE BM 41-200-128

ALLIS-CHALMERS
MANUFACTURING COMPANY
ATOMIC ENERGY DIVISION
NUCLEAR POWER DEPARTMENT
WASHINGTON, D. C.

LACBWR
VERTICAL POSTS

DETAILED DESIGN-CONTRACT NO. AT (11-1) 850		REV.
GEO. NO. 41-503-101		
SHEET	OF	04

VERTICAL POSTS

FIG. 4.25

41-503-097
SHEET 1 OF 4

LACBWR CORE LATERAL SUPPORT STRUCTURE

ALLIS-CHALMERS
MANUFACTURING COMPANY
ATOMIC ENERGY DIVISION
NUCLEAR POWER DEPARTMENT
WASHINGTON, D. C.

**LACBWR
CORE LATERAL
SUPPORT STRUCTURE
DETAILS**

DETAILED DESIGN—CONTRACT NO. AT (11-1) 850
DWG. NO. 41-503-097
SHEET 2 OF 4

REV.
04

LACBWR CORE LATERAL SUPPORT STRUCTURE DETAILS

FIG. 4.27

41-503-097 SH. 3 OF 4

41-503-097

ALLIS-CHALMERS
MANUFACTURING COMPANY
ATOMIC ENERGY DIVISION
NUCLEAR POWER DEPARTMENT
MILWAUKEE, WIS.
LACBWR
PERIPHERAL SHROUD
DESIGN NUMBER - CONTRACT NO. AT (11-0-000)
DRAWING NO. 41-503-097
SHEET 3 OF 4

LACBWR PERIPHERAL SHROUD

FIG. 4.28

41-503-097
SHEET 4 OF 4

ALLIS-CHALMERS MANUFACTURING COMPANY ATOMIC ENERGY DIVISION NUCLEAR POWER DEPARTMENT WASHINGTON, D. C.	
LACBWR CORE LATERAL SUPPORT STRUCTURE DETAILS	
DETAILED DESIGN—CONTRACT NO. AT (11-11)888	
DWG. NO. 41-503-097	REV. 01
SHEET 4 OF 4	

LACBWR CORE LATERAL SUPPORT STRUCTURE DETAILS

FIG. 4.29

LACBWR CONTROL ROD ASSEMBLY

FIG. 4.30

FIG. 4.31

INITIAL QUARTER CORE MOCKUP, 7 CONTROL RODS,
 28 STAINLESS-STEEL SHROUDS.

- ①. H_2O
- ②. Zirconium posts
- ③. Shroud plus 10% H_2O
- ④. Fuel region
- ⑤. Fuel region
- ⑥. Fuel region

PDQ CELL MOCKUP USED IN LOCAL PEAKING CALCULATIONS

FIG. 4.33

$$P = \frac{\text{Average Power in Fuel Element}}{\text{Average Power in Core}}$$

Fully Inserted Control Rods

Fully Withdrawn Control Rods

calculated power based on uniform steam voids
in each assembly equal to average in core (0.20)*

*Based on core water volume only.

RADIAL POWER DISTRIBUTION,

FIG. 4.34

AXIAL POWER DISTRIBUTION NORMALIZED TO CORE AVERAGE OF 1.0 (LACBWR core, hot, operating, and clean) FIG. 4.35

T = 0 7 rods in

T = 2000 Mwd/MT
7 rods rotated 90°

T = 4000 Mwd/MT
5 rods in

T = 5500 Mwd/MT
3 rods in

T = 7000 Mwd/MT s.s. shrouds re-
moved - 5 rods in

T = 10,000 Mwd/MT
center rod in

beginning of fuel cycle 5 rods
in

Notes

- (a) inserted rod denoted by large cross
- (b) T = average core exposure

£

£	x	a	a	x	y
	9455	12558	11894	10851	8084
	a	a	x	y	
	11940	11416	10049	9795	
	a	x	y	x	
	11488	10319	9371	8282	
	a	y	x	y	
	11262	10139	8486	6166	
	y				
	8566

ⓐ denotes elements removed from core
 (x) (y) denote groups of lesser exposed elements
 elemental values = exposures in Mwd/MT
 group ⓐ avg. exposure = 11,760 Mwd/MT

LACBWR ELEMENTAL EXPOSURE AT 10,000 Mwd/MT
 AVERAGE CORE EXPOSURE IN X-Y GEOMETRY

FIG. 4.37

L

€

$\begin{matrix} x \\ .856 \\ 10,851 \end{matrix}$	$\begin{matrix} x \\ 1.271 \\ 10,049 \end{matrix}$	$\begin{matrix} x \\ 1.413 \\ 8486 \end{matrix}$	$\begin{matrix} b \\ 1.376 \\ 0 \end{matrix}$	$\begin{matrix} y \\ .851 \\ 8084 \end{matrix}$
$\begin{matrix} x \\ 1.265 \\ 10,319 \end{matrix}$	$\begin{matrix} x \\ .895 \\ 9455 \end{matrix}$	$\begin{matrix} b \\ .916 \\ 0 \end{matrix}$	$\begin{matrix} y \\ 1.001 \\ 9795 \end{matrix}$	
$\begin{matrix} x \\ 1.409 \\ 8282 \end{matrix}$	$\begin{matrix} b \\ .912 \\ 0 \end{matrix}$	$\begin{matrix} y \\ .612 \\ 9371 \end{matrix}$	$\begin{matrix} b \\ .752 \\ 0 \end{matrix}$	
$\begin{matrix} b \\ 1.369 \\ 0 \end{matrix}$	$\begin{matrix} y \\ .991 \\ 10,139 \end{matrix}$	$\begin{matrix} b \\ .750 \\ 0 \end{matrix}$	$\begin{matrix} y \\ .509 \\ 6166 \end{matrix}$	
$\begin{matrix} y \\ .853 \\ 8566 \end{matrix}$

LACBWR ELEMENTAL POWER DISTRIBUTION AND EXPOSURE
 WITH FUEL MANAGEMENT IN X-Y GEOMETRY AT 0 MWD/MT
 AFTER FIRST FUEL REPLACEMENT

FIG. 4.38

£

£	x	x	x	b	y
1.352	1.292	1.182	.971	.562	
14,275	15,133	14,138	5504	11488	
x	x	b	y		
1.288	1.315	1.276	.860		
15379	13035	3664	13799		
x	b	y	b		
1.181	1.271	1.022	.768		
13918	3648	11819	3008		
b	y	b	y		
.967	.854	.768	.508		
5476	14103	3000	8202		
y					
.563					
11978

group (x) avg exposure = 14,313 Mwd/MT

LACBWR ELEMENTAL POWER DISTRIBUTION AND EXPOSURE WITH FUEL MANAGEMENT IN X-Y GEOMETRY AT 4000 Mwd/MT AFTER FIRST FUEL REPLACEMENT

FIG. 4.39

£

£

^y .708 14103	^y 1.072 11978	^y 1.221 11488	^b 1.269 5504	^c .983 0
^y 1.082 13799	^y .778 11819	^b .857 3664	^c 1.134 0	
^y 1.342 8202	^b .895 3648	^c .735 0	^c .834 3008	
^b 1.360 5476	^c 1.192 0	^b .853 3000	^b .625 0	
^c 1.065 0

© denotes fresh fuel group

LACBWR ELEMENTAL POWER DISTRIBUTION AND EXPOSURE
WITH FUEL MANAGEMENT IN X-Y GEOMETRY AT 0 Mwd/MT
AFTER SECOND FUEL REPLACEMENT

FIG. 4.40

€

€	y	y	y	b	c
	1.161 16935	1.166 16266	1.098 16372	.934 10580	.659 3932
	y	y	b	c	
	1.153 18127	1.210 14931	1.229 7092	.989 4536	
	y	b	c	b	
	1.146 13570	1.235 7228	1.164 2940	.821 6344	
	b	c	b	c	
	.949 10916	.996 4768	.824 6412	.589 2500	
	c				
	.675 4260

group (y) avg exposure = 16,034 Mwd/MT

LACBWR ELEMENTAL POWER DISTRIBUTION AND EXPOSURE WITH FUEL MANAGEMENT IN X-Y GEOMETRY AT 4000 Mwd/MT AFTER SECOND FUEL REPLACEMENT

FIG. 4.41

↑

b	b	b	c	d
.703 10916	1.083 7092	.679 6412	.417 3932	.430 0
b	b	c	d	
.814 7228	1.212 10580	.876 4768	.561 0	
b	c	d	c	
1.059 6344	1.532 4536	1.388 0	.910 2500	
c	d	c	d	
1.516 4260	1.555 0	1.216 2940	.811 0	
d				
1.239 0

(d) denotes fresh fuel group

LACBWR ELEMENTAL POWER DISTRIBUTION AND EXPOSURE WITH
FUEL MANAGEMENT IN X-Y GEOMETRY AT 0 Mwd/MT AFTER
THIRD FUEL REPLACEMENT

FIG. 4.42

LACBWR MULTIPLICATION CONSTANT VS. AVG CORE EXPOSURE IN AXIAL GEOMETRY

FIG. 4.43

XENON POISONING AFTER SHUTDOWN-LACBWR

FIG. 4.44

CHANGE IN MULTIPLICATION CONSTANT VS FUEL EXPOSURE FOR UNRODDED CORE
(assuming uniform burnup)

FIG. 4.45

station	fluid	temperature, °F	flow rate, lb/hr
1	saturated steam	577.5	611,500
2	saturated water	577.5	10,109,850
3	subcooled water	563.5	20,000
4	subcooled water	422.0	13,000
5	subcooled water	291.6	602,150
6	subcooled water	563.5	10,750,000
7	subcooled water	450.0	20,000
8	subcooled water	79.0	4,350
9	subcooled water	79.0	5,000

FLOW DIAGRAM OF EXTERNAL FLUID STREAMS AT FULL POWER

FIG. 4.46

FLOW DIAGRAM OF INTERNAL FLUID STREAMS

reactor pressure - 1300 psia
reactor power - 165 Mwt

HEAD VERSUS FLOW FOR FORCED CIRCULATION SYSTEM

FIG. 4.48

LEGEND

- STEAM FLOW
- UNRESTRICTED WATER FLOW
- RESTRICTED WATER FLOW
- LEAKAGE WATER FLOW

REACTOR INTERNAL FLOWPATHS

FIG. 4.49

FLOW HOLES IN BOTTOM OF GRID PLATE

FIG. 4.50

FLOW DISTRIBUTION IN INLET PLENUM

CONTROL ROD WATER INLET FLOW RESTRICTION

FIG. 4.52

REACTOR INTERNAL COMPONENT ELEVATIONS

FIG. 4.53

STEAM VOID DISTRIBUTION IN REACTOR AT FULL POWER

FIG. 4.54

PRESSURE DISTRIBUTION FOR FULL POWER OPERATION

FIG. 4.55

$P = 0.838$ (0.845) $V = 6.57$ $\bar{\alpha} = 0.165$ $\alpha_e = 0.324$ $X_e = 0.044$	$P = 1.291$ (1.27) $V = 6.21$ $\bar{\alpha} = 0.303$ $\alpha_e = 0.509$ $X_e = 0.091$	$P = 0.838$ (0.740) $V = 6.57$ $\bar{\alpha} = 0.165$ $\alpha_e = 0.324$ $X_e = 0.044$	$P = 0.391$ (0.330) $V = 6.61$ $\bar{\alpha} = 0.006$ $\alpha_e = 0.033$ $X_e = 0.003$	$P = 0.391$ (0.360) $V = 6.61$ $\bar{\alpha} = 0.006$ $\alpha_e = 0.033$ $X_e = 0.003$
$P = 0.838$ (0.890) $V = 6.57$ $\bar{\alpha} = 0.165$ $\alpha_e = 0.324$ $X_e = 0.044$	$P = 1.465$ (1.43) $V = 6.00$ $\bar{\alpha} = 0.350$ $\alpha_e = 0.565$ $X_e = 0.112$	$P = 0.838$ (0.910) $V = 6.57$ $\bar{\alpha} = 0.165$ $\alpha_e = 0.324$ $X_e = 0.044$	$P = 0.391$ (0.480) $V = 6.61$ $\bar{\alpha} = 0.006$ $\alpha_e = 0.033$ $X_e = 0.003$	
$P = 1.12$ (1.03) $V = 6.39$ $\bar{\alpha} = 0.254$ $\alpha_e = 0.446$ $X_e = 0.0716$	$P = 1.465$ (1.49) $V = 6.00$ $\bar{\alpha} = 0.350$ $\alpha_e = 0.565$ $X_e = 0.112$	$P = 1.291$ (1.31) $V = 6.21$ $\bar{\alpha} = 0.303$ $\alpha_e = 0.509$ $X_e = 0.091$	$P = 0.838$ (0.87) $V = 6.57$ $\bar{\alpha} = 0.165$ $\alpha_e = 0.324$ $X_e = 0.044$	
$P = 1.465$ (1.47) $V = 6.00$ $\bar{\alpha} = 0.350$ $\alpha_e = 0.565$ $X_e = 0.112$	$P = 1.465$ (1.47) $V = 6.00$ $\bar{\alpha} = 0.350$ $\alpha_e = 0.565$ $X_e = 0.112$	$P = 1.12$ (1.17) $V = 6.39$ $\bar{\alpha} = 0.254$ $\alpha_e = 0.446$ $X_e = 0.0716$	$P = 0.838$ (0.771) $V = 6.57$ $\bar{\alpha} = 0.165$ $\alpha_e = 0.324$ $X_e = 0.044$	
$P = 1.12$ (1.16) $V = 6.39$ $\bar{\alpha} = 0.254$ $\alpha_e = 0.446$ $X_e = 0.0716$				

reactor power 165 Mwt
 reactor pressure 1300 psia
 recirculation flow 30,000 gpm
 fuel channel coolant average velocity 6.38 ft/sec
 overall fuel channel average void 0.210
 overall fuel channel exit void 0.370
 overall fuel channel exit quality 0.061
 fuel channel flow 28,000 gpm
 core pressure drop 10.4-ft H₂O (including static-head)

$P = \frac{\text{average power in fuel element}}{\text{average power in core}}$ approximated for void calculation
 (number in parenthesis is from physics calculation)
 $V =$ inlet water velocity, ft/sec
 $\bar{\alpha} =$ average steam void fraction in fuel element
 $\alpha_e =$ fuel element exit steam void fraction
 $X_e =$ fuel element exit steam quality fraction

AXIAL STEAM VOID DISTRIBUTION IN FUEL ELEMENTS

FIG. 4.57

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RADIAL TEMPERATURE DISTRIBUTION IN FUEL ROD EXPOSED TO MAXIMUM FLUX

FIG. 4.60

COMPARISON OF DESIGN BURNOUT HEAT FLUX CORRELATIONS WITH EXPERIMENTAL DATA

FIG. 4.61

local peak to average heat flux 1.43
 radial peak to average power 1.49
 axial peak to average heat flux 1.87
 water inlet velocity 6.0 ft/sec
 water inlet subcooling 18.5 Btu/lb
 reactor pressure 1300 psia
 flow through fuel assemblies 10,029,800 lb/hr (28,000 gpm)

BURNOUT HEAT FLUX SAFETY MARGIN FOR HOTTEST FUEL ROD

FIG. 4.62

SCHEMATIC OF FULLY INSERTED CONTROL ROD

FIG. 4.63

COOLANT PRESSURE DISTRIBUTION WITHIN CONTROL ROD CHANNEL IN ACTIVE CORE
(assuming no leakage)

FIG. 4.64

STEAM CARRYUNDER LACBWR PROTOTYPE SEPARATOR TEST
(pressure 1285 - 1300 psig)

FIG. 4.65

SEPARATOR PRESSURE LOSS, LACBWR PROTOTYPE SEPARATOR TEST
(pressure 1285 - 1300 psig)

FIG. 4.66

SCHEMATIC OF THERMAL SHIELD-TO-VESSEL ANNULUS

FIG. 4.67

Basis:

- (1) maximum exit steam void < 70 percent
- (2) burnout safety margin ≥ 1.5
- (3) reactor pressure = 1300 psi

MINIMUM RECIRCULATION FLOW AS A FUNCTION OF POWER AND FEEDWATER TEMPERATURE

FIG. 4.68

5. SYSTEM DESCRIPTIONS

5.1 PRIMARY COOLANT SYSTEM

This section supersedes the preliminary description given in Sec. 4.2 of ACNP-62574, LACBWR Hazards Summary Report for Construction Authorization.

5.1.1 Main Steam and Feedwater Loop

5.1.1.1 General Description (Figs. 5.1 and 5.2). Steam is withdrawn from the reactor vessel through two 8-in. steam lines leading to a single 10-in. steam line that passes from the containment shell to the turbine building. Condensate returns from the turbine building through an 8-in. feedwater line, which connects to the 16-in. forced-circulation pump suction manifold where the condensate is mixed with the recirculating coolant before entering the recirculation pumps.

Within the biological shield the 10-in. steam line is opened to the safety valves by a connecting 10-in. steam line, which leads to the 6-in. line to the shutdown condenser. The condensate from the shutdown condenser is returned to the forced-recirculation pump suction manifold by a 4-in. line.

There are provisions for emergencies involving a primary system rupture within either the containment shell or turbine building. A rotoport reactor building steam isolation valve is installed in the 10-in. steam line within the containment shell, and a turbine building steam isolation gate valve is installed in the turbine building.

The reactor building steam isolation valve is hydraulically operated and can be closed in ~ 10 sec. This valve can be controlled manually from the control room and is closed automatically upon signals for low reactor level, low steam pressure at the turbine stop valve, and low main condenser vacuum.

The turbine building steam isolation valve is motor-operated and is controlled from the control room.

Both valves initiate a full reactor scram and automatically start the shutdown condenser if not fully open. The main feed return line has a check valve inside the containment building and a remotely-operated shutoff valve in the turbine building to prevent activity release from the building during an emergency. The shutoff valve is operated from the control room.

Flow through a 1-1/2 in. bypass line around the reactor building steam isolation valve is controlled by a remotely-operated bypass valve. Any leakage through the lantern

ring packing on the valve stem is directed through a 1/2-in. drain line to the retention tanks. Drain from the main steam line is passed through a 2-in. line to the sump, or through a trap to the retention tanks.

The main steam piping from the reactor vessel up to the main steam isolation valve in the reactor building is of Type-304 stainless steel, Spec. ASTM A376, as is the steam piping to the first elbow upstream of the safety valves. The rest of the main steam piping is of chromium-molybdenum steel, Spec. ASTM 335, Gr. P-11.

The body and plug of the main steam isolation valve are chromium-molybdenum steel, Spec. ASTM A217, Gr. Wc-6. The plug seating surface and the valve body seat surface are of Colmonoy No. 4.

The main feedwater pipe from the high pressure heater to the reactor is of Type-304 stainless steel, ASTM Seamless Steel Pipe Spec. A-312 TP304. All piping and valves in both the steam and main feedwater sections are designed for 1600 psig and 650 F, according to the ASME Code.

To ensure the safety of the reactor plant, reactor steam can be dissipated in several ways. A main-steam bypass valve between the turbine building steam isolation valve and the turbine stop valve can pass a steam flow of 100 percent. The valve is set to operate at 1265 psig and is opened fully at 1280 psig. The shutdown condenser in the containment building operates automatically when the reactor high pressure partial scram set point (1325 psig) is reached, and will condense 10 percent of full steam flow. Three safety valves on the 10-in. steam line within the containment shell will pass ~ 120 percent of full load steam flow. The discharge of these valves is retained within the containment shell. One safety valve is set for 1390 psig and the other two are set for 1426 psig.

Turbine pressure is maintained at 1250 psig by automatic speed control of the forced-circulation pumps. Abnormal pressure conditions actuate the safety system.

The main steam bypass valve control system is primarily to relieve reactor pressure by passing main steam flow directly to the main condenser during a pressure transient. The full stroke time of the valve is 1 sec. The valve can be controlled to dump small amounts of steam during warmup and turbine startup and loading, and may be used during shutdown periods and for handling of steam during tests.

5.1.1.2 Instrumentation and Control. Main steam and feedwater pressure and flow and temperature parameters are monitored for control and safety system purposes. The safety system transmitters are discussed fully in Sec. 8.5. Two narrow range (1200 - 1400 psig) and one full range (0 - 1500 psig) pressure transmitters are located in the

10-in. main steam line before the turbine stop valve. Signals from the narrow range transmitters go to a high signal selector; the output of the high signal selector goes to the reactor master pressure controller (see Sec. 5.1.2.5), the narrow range controller on the main steam bypass valve, and a recorder in the control room. The full-range transmitter indicates pressure in the control room, provides a pressure correcting signal for the steam flow element of the three-element feedwater controller, gives a "Steam to Turbine Pressure Low" alarm, and closes the reactor building steam isolation valve. An additional pressure transmitter (500 - 1300 psig) at the turbine stop valve provides a signal to the turbine initial pressure regulator controller, and a signal to the wide range main steam bypass valve controller.

A 0 - 1600 psig pressure transmitter in the 10-in. steam line to the shutdown condenser supplies a full scram signal to the safety system, and input to a pressure recorder in the control room. A 0 - 100 psig pressure transmitter in the same line is used to allow overhead storage tank water to flow directly to the core spray header if low reactor pressure is combined with low reactor water level. Another 0 - 1600 psig pressure transmitter in the main steam line between the reactor and the reactor building steam isolation valve supplies a high pressure partial-scram signal to the safety system and provides pressure indication in the control room. Both of these transmitters also provide signals to control valves which place the shutdown condenser system in operation.

Main-steam temperature is monitored between the pressure vessel and the reactor building steam isolation valve, and recorded in the control room. Feedwater temperature is monitored in the containment vessel and indicated in the control room.

5.1.1.2.1 Turbine Condenser. Two control schemes are used to control the main steam bypass valve over the full range of reactor pressure during shutdown and test periods and to maintain the accuracies required at normal operating pressure. For wide range control, a pressure transmitter between the turbine building steam isolation valve and the turbine stop valve feeds a 10-50 ma output (500 - 1300 psig) to a set-point controller (this pressure transmitter is the one that feeds the initial pressure regulator of the turbine). For narrow range control, a 10-50 ma output signal from the high signal selector of the pressure control system is fed to another set-point controller. The high signal selector transmits the highest signal sensed by two 1200-1400 psig pressure transmitters.

The set-point controllers can operate either automatically or manually. They feed a 10-50 ma output signal to a high signal selector, where the highest input signal is transmitted to a servo control system to operate the main steam bypass valve. The valve position is indicated in the control room over the 0 to 100 percent range. System accuracy is ± 1 percent of range.

The turbine condenser is provided with local pressure and temperature readout and with level readout and control. A transmitter provides a signal to the safety system and trips the turbine if condenser vacuum is low.

5.1.1.2.2 Feedwater Control System. The feedwater control system regulates the unvoided water level in the reactor vessel by varying the speed of either feedwater pump. The unvoided water level is defined as the level for no heat generation and for no water voids. A three-element control system measures mass steam flow, feedwater flow, and reactor water level.

Reactor water level is monitored by a differential pressure transmitter that feeds a 10-50 ma output signal (corresponding to a water level of 0-60 in.) to a recorder on the main panel and a level controller on the control console. The controller has proportional band, reset, and rate actions. Steam and feedwater flow are monitored in the turbine building by differential pressure transmitters. The transmitters are across two flow nozzles — one between the reactor building steam isolation valve and the turbine building steam isolation valve and the other at the discharge of the feedwater pumps. A 10-50 ma signal from these transmitters feeds two independent square root converters, where linear signals are computed for control and indication. The steam flow signal is compensated for pressure by a signal from the 0-1500 psi transmitter.

To provide control, the indicated difference between the linear steam flow and feedwater signals is compared with the difference signaled between the actual level and the level set-point. The controller has proportional band, reset, and rate actions, and feeds the resultant error signal to two manual stations, which can provide either automatic or manual control. The output from each manual station consists of a 10-50 ma signal that is converted to a pneumatic signal to drive the respective pump actuators and reposition the fluid coupling scoop tube of the variable-speed feedwater pump. The scoop tube regulates the pump speed. The range of level control is between -30 and +30 in., with the zero level set at a reactor elevation of 666 ft 5 in. The operational range is ± 5 in. System accuracy is ± 1 percent of range. Two additional differential pressure transmitters provide an abnormal level scram signal to the safety system, as described in Sec. 8.5.

If the control air supply should fail, the scoop tube actuator on the boiler feed pump drive provides for maximum flow output to the reactor. Flow can be stopped by cutting off the power to the pump or closing the control valve in the feedwater line downstream of the pump discharge. This is a diaphragm-operated air-to-close valve, manually controlled in the control room. This arrangement minimizes the possibility of losing reactor feedwater supply.

5.1.2 Forced-Circulation System (Figs. 5.3 and 5.4)

The operating principles for this system are discussed in Sec. 11.1.

5.1.2.1 General Description. The reactor core is cooled by the forced circulation of as much as 30,000 gpm of water. The recirculation flowrate automatically regulates reactor power according to plant load demand, between power levels of 60 and 100 percent.

Primary water passes upward through the core, and then down through the steam separators to the recirculating water outlet plenum. The water then flows to the 16-in. forced-circulation pump suction manifold through four 16-in. nozzles and is mixed with reactor feedwater that enters the manifold through four 4-in. connections. From the manifold, the water flows through 20-in. suction lines to the two 15,000 gpm variable-speed forced-circulation pumps. The pumps are above the basement floor, within their own shielded cubicles. Hydraulically-operated rotoport valves are at the suction and discharge of each pump. The 20-in. pump discharge lines return the water to the 16-in. forced-circulation pump discharge manifold. From the manifold, the water flows through four equally spaced 16-in. reactor inlet nozzles to the annular inlet plenum, and then downward along the bottom vessel head to the core inlet plenum.

The system piping is designed for a maximum working pressure of 1450 psig at 650 F (a pressure above the maximum reactor working pressure to allow for the static head and the pump head).

Since the piping from the reactor to the rotoport valves is within the biological shield and is not accessible, the valves and piping are clad with stainless steel. The piping between the rotoport valves and the pumps is low-alloy steel. Provisions have been made for determining the rate and type of any corrosion, and the low-alloy piping can be replaced if the corrosion rate is excessive. To facilitate repair or replacement, decontamination solutions can be circulated to remove radioactive particles.

5.1.2.2 Forced-Circulation Pumps. The reactor forced-circulation pumps are Allis-Chalmers pumps of the vertical, single stage, mixed-flow type, rated for an 85-ft head at 15,000 gpm. These pumps have a fluid piston bearing and a buffer-type shaft seal and are driven through variable speed fluid couplings by 350 hp, 2400-v, 1180 rpm motors.

The shaft sealing arrangement consists of four sections: 1) two lower seal rings, 2) a buffer-type seal, 3) a mechanical seal in series with the buffer seal as a backup to the buffer seal, and 4) a labyrinth bushing above the mechanical seal as a backup to the mechanical seal. The buffer seal contains a floating sleeve, four seal rings, and a face

seal, which limit buffer seal leakage. Most of the high pressure seal water normally supplied to the annulus between the buffer seal and the lower seal rings (see Sec. 5.2.4.2) leaks past the two lower seal rings and enters the pump casing to prevent primary water from leaking out of the pump casing. The remainder of the seal water leaks past the buffer seal to an annulus between the buffer seal and the mechanical seal. The normal leakoff piping returns the water to the seal-water-injection-pumps reservoir.

Mechanical seal failure would allow leakoff water to leak to the annulus between the seal and the labyrinth bushing, but the backup seal leakoff piping would carry the water to the reactor building sump. A slinger ring within the annulus centrifugally throws the water toward the annulus periphery and limits labyrinth bushing leakage.

The buffer-type shaft seal is mounted directly above the pump casing cover. A spacer shaft between the pump shaft and the variable-speed coupling-driven shaft permits maintenance of the seal without disturbing the pump cover or the variable speed coupling.

The fluid piston bearing consists of four pockets formed by the journal and an extension of the impeller (Fig. 5.13A). Each pocket is pressurized by water supplied by the pump volute. This water leaves the pockets through the clearance between the bearing and the journal. As a radial load is applied, the journal moves off center, increasing the pressure in one pocket and decreasing the pressure in the opposite pocket. This pressure differential centers the journal in the bearing.

The shaft has a balance piston below the buffer-type seal. The two lower seal rings cause a pressure drop across the piston, thus absorbing part of the upward thrust. The remainder of the thrust is absorbed by the angular contact roller bearings on the fluid coupling output shaft.

The fluid coupling is an Allis-Chalmers unit rated for 500 hp at 1140 rpm. Basically, it consists of an impeller, a runner, a scoop tube, and a rotating casing that encloses the impeller and runner. These components are in a stationary outer housing that also serves as a water reservoir. The impeller and rotating casing are connected to the input shaft, which is driven at a constant speed by the pump motor. The runner is connected to the output shaft and rotates at a speed governed by the quantity of water in the rotating casing. The fluid coupling water system supplies water at a uniform rate through the inlet pipe and manifold. Water enters the impeller through holes in the impeller hub and flows freely through communication holes in the outer periphery of the impeller, into the rotating casing where the scoop tube is located. The water flow tends to fill the rotating elements, but the degree of filling is limited by the position of the scoop tube. The scoop tube removes the excess water from the rotating casing and discharges it into the stationary outer housing. The water is drained from the outer housing by gravity.

The variable speed coupling output shaft has an angular-contact roller-type thrust bearing and a radial ball bearing, which are lubricated by the forced-circulation pumps lube oil system (see Fig. 5.4), and serve primarily as thrust and radial bearings for the pump, as there is a solid coupling between the pump and fluid coupling.

5.1.2.3 Valves. There are two 20-in. rotovalves for each of the two forced-circulation pumps — one in the pump suction line and the other in the pump discharge line. These valves isolate the pumps for maintenance, without draining the reactor. In addition, the pump discharge valve is closed during pump startup to provide shutoff head.

The Allis-Chalmers Rotovalves are essentially plug cocks with raised stellite seat rings that seat tightly to cut off the flow when the valve is closed. For opening or closure, the valve is lifted from its seat, rotated 90 degrees, and then resealed. The valve operator is a hydraulic cylinder with a double acting piston. It drives the valve shaft through a special crosshead arrangement which provides both rotary and axial movement. The hydraulic fluid is water supplied by the hydraulic valve accumulator system.

The body of the rotovalves is ASTM A216 WCB carbon steel, with an overlay of Type-308 stainless steel. The plugs are stainless steel with a seating surface of Colmonoy No. 4 (a hard surfacing material). The seating surfaces on the valve body are stainless steel.

The valve shaft is sealed by a packing gland, with five rings of Inconel and asbestos packing, and a lantern ring. Any leakage through the three inner packing rings is collected in the lantern ring and drained to the retention tank.

There is a 2-in. diaphragm-operated control valve in the bypass line around the main discharge rotovalve in each of the two pump discharge lines. These valves are solenoid-piloted pneumatic valves, operated from the control room. They are opened by air pressure and fail shut.

The basic control philosophy for the forced-circulation system valves is to:

- (1) provide maximum circulation through the core when core cooling is required,
- (2) prevent the formation of a cold water leg in the forced-circulation loops,
- (3) prevent the discharge of cold water into the reactor core,
- (4) prevent the simultaneous opening of a pump discharge valve and withdrawal of a control rod, both of which would contribute an increase in reactivity,

- (5) provide a means of overriding automatic controls to permit pump startup,
- (6) prevent closure of a suction valve while the associated pump is running, and
- (7) prevent reverse rotation of a forced-circulation pump.

5.1.2.4 Instrumentation. Temperature is measured at the discharge of each forced-circulation pump, and indicated in the control room. The two pump temperatures are automatically compared, and the difference displayed on a temperature difference indicator. High temperature differential between the two loops automatically closes the pump discharge rotovalve in the cold loop and thermowells in the discharge line of each pump are used in the measurement of local temperatures until the condition is corrected. Temperature is also measured at the suction of the forced-circulation pumps and on the reactor side of the rotovalve discharge, and recorded on a multipoint recorder in the control room. Other temperature monitors include the lube oil return from each forced-circulation pump coupling, seal leakoff temperature from the pumps, pump motor and hydraulic coupling bearings, and the seal water supply header temperatures.

The pressure differential across each forced-circulation pump is measured by a pressure differential transmitter and indicated in the control room. A local pressure gauge is installed in the lube oil supply header for the forced-circulation pumps. The pressure differential between the seal water inlet and the pump suction is indicated locally for each forced-circulation pump on a differential pressure gauge.

Flow through each forced-circulation loop is measured by an elbow flowmeter on the suction side of the pump. Two flow transmitters at each elbow provide independent signals for the flow in each loop to the reactor power-to-flow relation safety system channels (Sec. 8.5.2.5.3). In addition, the flowrates are recorded on a two-pen flow recorder in the control room.

The position of each rotovalve is measured by a position transmitter and indicated in the control room.

The shaft speed of each forced-circulation pump is measured with a tachometer generator. These shaft speed signals are also used in the automatic pump speed control circuits, and transmitted to speed indicators in the control room.

5.1.2.5 Controls.

5.1.2.5.1 Forced-Circulation Pump Motor Control. The reactor forced-circulation pump motors can be started and tripped with individual control switches in the control room. The pump motor cannot be started until conditions are as follows:

- (1) Lube oil pressure is satisfactory.
- (2) Hydraulic coupling is set for zero speed.
- (3) Speed control selector switch is on "Manual."
- (4) Suction valve for pump is fully open.
- (5) Discharge valve for pump is fully closed.

The pumps will be automatically tripped by the malfunction signals described in Sec. 11.1.5.3.

5.1.2.5.2 Forced-Circulation Pump Speed Control. Forced-circulation pump speed is controlled by positioning the scoop tube in the variable speed fluid coupling to control the amount of fluid in the coupling. A MAN-AUTO selector switch controls the mode of operation of the two pumps.

Interlocks prevent any increase in pump speed during control-rod withdrawal, and vice versa. Simultaneous addition of positive reactivity by both means is thus prevented.

When the selector switch is turned to AUTO, the fluid coupling scoop tubes are automatically positioned to maintain the turbine inlet steam pressure at its set-point. The pump speed can be varied from 40 to 100 percent of full speed to regulate reactor power between 60 and 100 percent of full power. The automatic control system cannot reduce pump speed below 40 percent.

The cascade-type control system consists of a master pressure control station and two individual pump speed control stations. The master pressure control station compares steam pressure with the set-point, and the resulting signal is used as a speed set-point signal in the individual pump speed control stations. The individual stations compare pump speed with the set-point and position the scoop tube to obtain the required speed.

On reactor scram, the pumps are automatically transferred to manual control; a scram interlock in the manual control circuit causes them to run at a preset lower speed to prevent pump cavitation.

When the selector switch is turned to MAN, pump speed is controlled by the two individual manual pump speed control switches and the manual gang speed control switch.

The manual pump speed control switches have four positions: 1) DECREASE, 2) NEUTRAL, 3) INCREASE, and 4) a pull-out position on DECREASE. They are spring-returned to neutral except from the pull-out position. The gang speed control switch is spring-returned to neutral and has three positions: 1) DECREASE, 2) NEUTRAL, and 3) INCREASE.

The individual pump speed control switches and the gang speed control switch vary pump speed between 40 and 100 percent. The scoop tube moves only while a switch is held in the DECREASE or INCREASE position, and stops moving when the switch is released. An interlock prevents a pump speed below 40 percent, unless the individual pump speed control switch is in the pull-out position. For the pull-out position the interlock and the seals in the decrease circuit are bypassed and the pump speed decreases to zero. When the switch is moved from the pull-out position, the pump speed automatically increases to 40 percent, ensuring that the pump passes through the 0 to 40 percent speed rapidly to avoid damage to the fluid piston bearing.

The gang speed control switch cannot vary the pump speed unless the individual speed control switches are in the neutral position. If the pumps are at different speeds, they are synchronized by the individual pump speed control switches.

Upon reactor scram, the pumps are automatically run back to a speed of 80 percent (unless their speed is below 80 percent), thus preventing pump cavitation. The interlock also prevents an increase in pump speed while the reactor is scrambled.

5.1.2.5.3 Rotovalve Controls and Interlocks. The rotovalve control components are as follows:

- (1) control switch in control room (key lock in base),
- (2) various interlocks (described below),
- (3) solenoid valve No. 1 (3-way direct-operated solenoid valve),
- (4) solenoid valve No. 2 (4-way pilot-operated solenoid valve),
- (5) manual valve (4-way manual valve with AUTO, OPEN, and CLOSE positions),
- (6) pilot-operated control valve (4-way pilot-operated control valve with auxiliary manual stops for locking in the open or closed positions),
- (7) a flow control valve in the line to the rotovalve operator cylinder "open" port, to control the rotovalve closing rate, and
- (8) a flow control valve in the line to the rotovalve operator cylinder "close" port, to control the rotovalve opening rate.

A rotovalve is on automatic control when its manual valve is on AUTO and the pilot-operated control valve stops are turned out. The rotovalve is closed when the solenoid valves are energized by closure of the control switch or interlock contacts. The solenoid valves direct high-pressure hydraulic fluid through the manual valve to the pilot-operated control valve. The solenoid valves position the control valve to supply high-

pressure hydraulic fluid to the "close" port of the rotovalve operator cylinder and to drain fluid from the "open" port. The rotovalve is opened when the solenoid valves are de-energized to direct high-pressure hydraulic fluid to the "open" port and drain fluid from the "close" port.

The suction rotovalves are interlocked with the shutdown condenser to open automatically when the shutdown condenser operates. This interlock overrides the valve control switch. By an interlock with the pump motor control circuit the suction rotovalve prevents valve closure while the associated pump is operating. Other interlocks prohibit startup of the associated forced-circulation pump if the suction valve is not fully open or if the discharge valve is not fully closed.

The discharge valve closes automatically when the associated pump is tripped or operating at zero speed and when the other pump is above zero speed. This action can be overridden only by turning the manual valve to OPEN or by screwing in the OPEN stop on the pilot-operated control valve.

The discharge valve opens automatically when the associated pump is operating above 40 percent speed or when neither pump shaft is rotating. This action can be overridden by:

- (1) turning the control switch to CLOSE, unless the shutdown condenser is operating (an interlock prevents closure of the discharge valve during shutdown condenser operation);
- (2) high temperature differential between forced-circulation loops, unless the shutdown condenser is operating, or
- (3) an interlock that prevents discharge valve opening during control-rod withdrawal (this interlock has no effect when the discharge valve is fully open).

Manual control always overrides automatic control. The flow of hydraulic fluid to and from the pilot on the pilot-operated control valve is then controlled by the manual valve instead of the solenoid valve.

The handwheels on the pilot-operated control valve can block this valve either in the OPEN or CLOSE position and override both automatic and manual control. The rotovalve operating speeds for opening and closure are controlled independently by the flow control system for the rotovalve operator cylinder ports hydraulic fluid line.

5.1.2.5.4 Discharge Bypass Valve Control. The pump discharge bypass valve control components are a keylock selector switch in the control room with OPEN, CLOSE, and AUTO positions, and an interlock with the associated discharge rotovalve. With the

selector switch on AUTO, the bypass valve automatically opens when the associated discharge rotovalve is not fully open, and closes when the rotovalve is fully open. When the selector switch is turned to OPEN or CLOSE, the interlock with the discharge rotovalve is bypassed.

5.2 AUXILIARY SYSTEMS

This section describes the auxiliary systems of LACBWR and supersedes the preliminary descriptions given in Sec. 4.3 of ACNP-62574.

5.2.1 Decay Heat Cooling System (Fig. 5.5)

The functions of the system are:

- (1) to cool the reactor from 470 F to 120 F after reactor shutdown,
- (2) to maintain reactor water temperature at not more than 120 F while the vessel is open for refueling or alterations,
- (3) to heat the reactor water to 60 F above the reactor Nil Ductility Transition Temperature (NDT) if core decay heat is not sufficient for the purpose, prior to cold hydrostatic testing, and
- (4) to blow down reactor water to the main condenser.

5.2.1.1 General Description. The decay heat cooling system consists of a single closed loop, with a pump and heat exchanger. The inlet of the 6-in. loop is connected to the suction line of forced-circulation pump 1A between the suction manifold and the forced-circulation pump suction valve. Reactor water flows through the decay heat pump and the decay heat cooler, and is returned to the reactor through a discharge line connection for pump 1A between the forced-circulation pump discharge valve and the discharge manifold. Both connections are outside the biological shield.

The system piping and valves are designed for operation at 1500 psig and 650 F. The piping is 6 in. Schedule 120, ASTM Spec. A376 and is of TP-304 stainless-steel. The material is selected for resistance to corrosion and erosion in water with a pH of 7.

A 2-in. blowdown line to the main condenser hotwell is connected to the decay heat system on the discharge side of the decay heat cooler. When the reactor is shut down, the reactor water level is controlled by blowing down reactor water from the decay heat system to the main condenser. A 2-in. line connecting the blowdown line to the reactor vent line is used during reactor cooldown to promote the water circulation within the vessel head.

Inlet and outlet stop valves isolate the system from the reactor. Additional valves isolate the pump and cooler from the system for maintenance. The decay heat cooler is equipped with shell and tube relief valves (set for 150 psig and 1500 psig) as protection from overpressure, and the system blowdown valve fails closed upon loss of air to prevent inadvertent draining of the reactor.

When not in use, the decay heat system outlet valve is locked closed to prevent water circulation through the system. Prior to using the system after a reactor shutdown, the system is warmed up by allowing ~ 10 gpm of reactor water to flow backwards through it. The necessary head is provided by the reactor forced-circulation pumps. A bypass line and valve around the decay heat pump discharge check valve permits backward flow and regulation of the flowrate. Component cooling water is circulated through passages in the decay heat pump to prevent high temperature damage to the pump pedestal, bearings, and mechanical seal.

Immediately after plant shutdown, steam is passed through the main steam bypass valve to the main condenser until enough heat is removed to lower the primary system temperature from 577 F to 470 F. After the decay heat cooling system has been warmed to within 120 F of reactor water temperature, it may be brought into operation to remove the heat. At reactor water temperatures above 165 F the decay heat cooler has sufficient capacity to remove the maximum decay heat and cool the reactor water at a rate of 50 F/hr. The cooler maintains the reactor water temperature at 120 F or less, using 90 F component cooling water with a decay heat rate of 5.7×10^6 Btu/hr. This is a safety factor of 25 percent more than the calculated decay heat rate 14 hr after shutdown, following infinite core exposure at full power. The line connecting the reactor head vent line and the decay heat system blowdown line permits water circulation within the vessel head during reactor cooldown if accelerated reactor cooldown is desired. The line can be used in the following ways:

- (1) to promote circulation, water can be pumped to the top of the reactor vessel by the decay heat pump or forced-circulation pump, and
- (2) to blow down steam or water from the top of the reactor vessel to the main condenser.

During operation of the system, flowrate is controlled by throttling the decay heat cooler outlet valve and the bypass valve to maintain the minimum allowable pump flow of 100 gpm. To obtain this flowrate through the pump and a lower flowrate through the cooler, some water must be bypassed around the cooler.

Prior to cold hydrostatic testing, the vessel must be warmed to a temperature 60 F above its NDT temperature. Steam supply and condensate drain connections on the shell side of the decay heat cooler allow the cooler to be used as a warmup heater for this purpose if required.

5.2.1.2 Decay Heat Cooler. The decay heat cooler is a horizontal shell and U-tube heat exchanger of all-welded construction. It is designed in accordance with ASME Code, Section VIII, and Nuclear Code Case 1270-N. The tube side of the cooler is designed for 1500 psig and 470 F. The shell side is designed to accommodate saturated water (338 F) at the component cooling water system design pressure of 150 psig.

The cooler has 6-in. inlet and outlet connections on the shell and tube sides. Reactor water flows through the tubes. For cooling the reactor water and removing decay heat, component cooling water is circulated through the shell. For heating the reactor water prior to cold hydrostatic testing, the shell side is supplied with saturated steam from the heating boiler and the condensate is returned to the heating boiler system. A vent connection is provided on the shell side of the cooler.

5.2.1.3 Decay Heat Pump. The decay heat pump is a single-stage, horizontal, motor-driven, centrifugal pump rated for a total discharge head of 65 ft at 885 gpm. It is designed for operation at 1500 psig and 585 F. The rated capacity of 885 gpm is the design flowrate for the decay heat cooler.

The pump has one radial-type ball bearing and three thrust-type ball bearings. An oil sump and oil rings lubricate the bearings.

The pump has a mechanical seal with an internal pumping ring and an external cooler. The pumping ring removes seal water from the mechanical seal annulus and circulates it through the cooler and back to the seal. The cooling water flowrate to the cooler must be regulated to maintain a seal water temperature of 160 F or less at the seal outlet.

The radial and thrust bearing jackets, the mechanical seal chamber jacket, and the pump pedestal are cooled by circulation of cooling water.

5.2.1.4 Instrumentation. Local temperature indicators are provided at the decay heat cooler primary water inlet and outlet and at the cooling water outlet. A low cooler outlet temperature sounds an alarm in the control room (see Sec. 11.5.3). The system inlet and outlet temperatures are indicated in the control room. Local temperature indicators are also provided for the pump mechanical seal water outlet, seal and bearing cooling water outlet, mechanical seal cooling water outlet, and the pedestal cooling water outlet. A local pressure differential indicator is provided for the decay heat pump.

The system flowrate is monitored, and indicated in the control room. A level gauge mounted directly on the decay heat cooler indicates the level locally.

5.2.1.5 Controls. The decay heat cooling system is designed for local control with the reactor shut down. Control-room indications and alarms for the important operating parameters immediately signal any malfunction.

Two manual control switches — one mounted locally and the other in the main control room — start or trip the decay heat pump.

A manual indicating controller in the main control room actuates an electric-to-pneumatic converter, which operates a valve for decay heat cooling system blowdown to the main condenser. A valve position transmitter feeds information to the manual indicating controller. A low-level electric signal from the reactor safety system to the normally vented solenoid valve in the air supply line overrides manual control to maintain the solenoid in the vented position, which ensures that the blowdown valve is closed. The blowdown rate is controlled locally by a manual globe valve.

5.2.2 Shutdown Condenser System (Fig. 5.6)

Section 4.3.3 of ACNP-62574 states that the primary function of the shutdown condenser is to dissipate decay heat during the first 5 hr after shutdown. This statement is no longer true. Normally the main condenser will be used after shutdown until the reactor water temperature drops to 470 F, at which time the decay heat cooling system begins to operate (Sec. 5.2.1).

The shutdown condenser system will be used to condense reactor steam when the reactor is isolated from the main condenser by closure of the reactor building steam isolation valve or the turbine building steam isolation valve. In addition, the shutdown condenser steam inlet valve opens to relieve high pressure in the main steam outlet line before the primary system relief valves open to the containment building atmosphere.

5.2.2.1 General Description. The shutdown condenser is on a platform 10 ft above the top floor of the containment building. Steam flows in a 6-in. line from the main steam line to the tube side, where it is condensed. The condensate flows by gravity through a 4-in. line to the feedwater line, and noncondensables and vapor are vented back through the outlet channel of the shutdown condenser to the waste gas system, thus preventing buildup of vapor and noncondensables that could hinder the condensate flow. The 1-in. vent line from the water box to the waste gas system contains a control valve that opens automatically upon shutdown condenser operation signals.

The shell side is normally supplied with demineralized water through a 3-in. line. This line contains a control valve, automatically positioned by a level controller to maintain a constant shell-side water level. The water is evaporated and exhausted to the atmosphere through a 14-in. line penetrating the reactor building. If the normal water

supply fails, the shell side is automatically supplied with water through a 3-in. line from the high-pressure service water system. The tube and shell sides each have drain lines to one of the retention tanks. The shell-side drain is also used as blowdown line, and has a connection for taking water samples.

The system piping is designed for a maximum working pressure of 1400 psig (at 650 F), the design pressure of the reactor vessel. The piping and valves are sized to permit a flow of 75,000 lb/hr without excessive pressure drop. The condensate piping and the steam piping within the biological shield are of stainless steel. The rest of the steam piping is of low-alloy steel. The following have air-operated control valves: the 6-in. pipeline between the main steam pipeline and the shutdown condenser; the 4-in. pipeline from the condenser to the feedwater line; the vent from the outlet channel of the condenser; and the 3-in. demineralized water and high-pressure service water lines to the shell side of the condenser.

During normal plant operation, the shutdown condenser is isolated from the reactor and other systems by the control valves. A check valve in the condensate return line prevents reverse feedwater flow while the condensate-return control valve is open.

5.2.2.2 Shutdown Condenser. The shutdown condenser is a horizontal U-tube heat exchanger. The reactor steam is condensed in the tubes, and the cooling water is evaporated in the shell. It is designed in accordance with Section VIII of the ASME Code for Unfired Pressure Vessels, as modified by Code Case 1270-N. Tube-side design pressure and temperature is 1415 psig and 650 F. Shell-side design pressure is 40 psig internally and 75 psig externally, and design temperature is 275 F. The shell is made of carbon steel, the tubes are of copper-nickel alloy, and the inlet and outlet channels are carbon steel, clad with monel. The shutdown condenser can absorb without damage the thermal and physical shock of going in 5 sec from cold shutdown to operation at design conditions, for 50 cycles during a 20 year unit lifetime. The thermal shock is equivalent to a temperature transient of 500 F in 5 sec.

The tubes are seal-welded into the tube sheet, and the tube sheet is welded to the shell; however, a cutting ring is provided so that the tube bundle can be removed. Handholes are provided so that any leaking tubes may be plugged. The steam condensate nozzle is sized and shaped for a liquid velocity at the two-phase interface of less than 1 ft/sec, which precludes carryunder.

Because of the large temperature differential between the reactor steam and the shell-side water, a thermal barrier is provided to prevent overstress in the tube sheet. This barrier insulates the steam inlet channel and the tube sheet from the cold water. The large temperature differential also had led to the use of finned tubes to provide good heat transfer.

The shutdown condenser design capacity of 3.89×10^7 Btu/hr (6 percent of 190 Mwt) for saturated steam entering at 1285 psig and for saturated water leaving the condenser provides an adequate heat sink for emergency shutdown cooling and for cooling the reactor water to 300 F, at a rate of 50 F/hr. However, the decay heat cooler is normally used for cooling below 470 F.

5.2.2.3 Instrumentation. Temperatures in the steam line and condensate return lines are monitored with thermocouples and recorded in the control room. Thermowells are also provided for testing. The shutdown condenser shell-side temperature is measured with a thermocouple and recorded in the control room. A local pressure gauge is provided on the shutdown condenser outlet channel vent line.

A flow nozzle and flow transmitter in the shutdown condenser condensate return line measure the system flow. The signal is transmitted to a 0-250 gpm indicator in the control room.

A gauge glass in the shutdown condenser outlet water box locally indicates the tube-side level. This glass is rated to withstand the thermal shock conditions of condenser operation. The shell-side level is indicated locally by a gauge glass.

To maintain liquid level, the shell-side level controller regulates the control valve in the demineralized water supply line between high and low level signals and regulates the control valve in the high pressure service line between low level and low-low level signals. The controller provides an output signal to a level indicator that indicates the shutdown condenser cooling water level in the control room. Two relays actuate high and low level alarms (Sec. 11.6.2.6).

The position of the shutdown condenser steam inlet control valve is indicated on a manual indicator-controller and by red and green indicating lights in the control room.

The positions of the shutdown condenser condensate return control valve and the vent valve are indicated by red and green lights next to the manual control switches in the control room.

5.2.2.4 Controls. The system is designed for automatic startup and operation. The shutdown condenser controls are supplied by the 125-v d-c bus.

5.2.2.4.1 Automatic Controls. The shutdown condenser is automatically started when any of the following conditions exist:

- (1) reactor building steam isolation valve is not fully open,
- (2) turbine building steam isolation valve is not fully open,

(3) reactor pressure is above 1325 psig (partial-scram signal from reactor safety system), and

(4) reactor pressure is above 1350 psig (full-scram signal from reactor safety system-backup to item (3)).

When system operation is initiated automatically, the steam inlet valve and the channel vent valve open immediately. The condensate return valve opens 10 sec later, providing sufficient time to pressurize the tube side and purge it of noncondensables.

The tube-side vent valve is interlocked with the waste gas system to prevent the compression of an explosive mixture. This interlock automatically closes the vent valve after 2 min of operation if the gas storage tank pressure exceeds 3 psig and there is an explosive mixture at the recombiner condenser outlet. The 2-min time delay is sufficient to purge the shutdown condenser of noncondensables if these conditions exist at startup of the shutdown condenser.

A key-lock-type OPERATE-BYPASS switch allows bypass of the shutdown condenser interlocks for:

- (1) reactor building steam isolation valve not fully open, or
- (2) turbine building steam isolation valve not fully open.

The bypass switch can be used to prevent shutdown condenser operation if the valves are tested during normal plant operation, and when the valves are closed during normal plant shutdown and startup.

An operate-test mode switch allows bypass of the high reactor pressure interlock so that the shutdown condenser will not be placed in operation during testing of a high reactor pressure channel.

5.2.2.4.2 Manual Controls. The shutdown condenser steam inlet control valve has a manual indicator-controller in the control room. This controller is used to control the valve between the open and closed positions. However, if the system is operated automatically, the steam valve is fully opened, and the indicator-controller is bypassed and does not control the valve.

The shutdown condenser condensate outlet control valve has an OPEN-AUTO control switch in the control room. In the AUTO position, the valve is controlled by the automatic controls. In the OPEN position, the automatic controls are bypassed and the valve is locked open.

The shutdown condenser tube-side vent control valve has an OPEN-AUTO control switch in the control room that functions like the condensate outlet control valve switch.

5.2.3 Water Purification Systems

5.2.3.1 Condensate Demineralization System (Fig. 5.7). The condensate demineralization system removes soluble and insoluble impurities from the turbine condensate, to assure high purity of reactor feedwater. The system is designed for regeneration of ion exchange resins after depletion.

5.2.3.1.1 General Description. The condensate demineralizer portion of the system consists of three ion exchangers in parallel, each of which can handle half of full power flow. Each ion exchanger contains a mixture of 33 ft³ of cation resin and 17 ft³ of anion resin. A recirculation pump is provided to maintain optimum flow through the ion exchangers during partial load conditions.

The condensate effluent from the demineralizers is continuously monitored for conductivity before it passes through a wire mesh resin trap, which will retain the ion exchange resin if a resin retention element fails.

The regeneration portion of the system consists of a cation regeneration and resin separation tank, an anion regeneration and resin mix tank, a sluice water tank, an acid tank and pump, and a caustic tank and pump. The entire regeneration operation is conducted separately from demineralization and proceeds automatically by means of a timer. However, the operation is initiated by operator decision after the resin is sluiced manually from the demineralizers to the cation regeneration tank. The mixed resins may then be backwashed and returned to the demineralizer, or put through the full regeneration cycle.

Backwashing is accomplished by pumping water from the sluice water tank into the bottom of the cation regeneration and resin separation tank. The water flows up through the mixed bed and carries the insoluble material out the backwash overflow line to the 4500-gal waste water storage tank via the facility drain line. If chemical regeneration is not to be performed, the resin is then sluiced back to the empty demineralizer, using fresh demineralized water as the carrier and air pressure applied to the regeneration tank as the driving force; the demineralizer underdrain system is open to pass the carrier water during the transfer. Condensate is then admitted to the demineralizer to cover the resin bed, and the resin is mixed by admitting first an air-water mixture, and then air only, into the bottom of the demineralizer.

Complete regeneration, when required, is accomplished automatically by actuating the switch on the local control panel while the resin is in the cation regeneration and resin separation tank. The automatic regeneration cycle performs the following:

(1) The resin is backwashed and segregated, transferring the less dense anion resin to the top of the bed. The anion resin is then sluiced to the anion regeneration and resin mix tank with sluice water provided by the sluice water resin transfer pump.

(2) After the cation and anion resins are in their respective regeneration tanks, the anion resin is regenerated by injection of dilute sodium hydroxide, followed by a slow rinse with sluice water and a fast rinse with demineralized water; the cation resin is regenerated by injection of dilute sulfuric acid, followed by slow and fast rinses. The cation resin is then transferred to the anion regenerating and resin mix tank by demineralized water and air pressure. The resins are mixed by admission of air to the bottom of the resin mix tank. The mixed bed is thoroughly rinsed by recirculation of sluice water, until the conductivity cell indicates satisfactory completion of the rinse.

(3) After a successful regeneration cycle is completed, the mixed resin is stored under water in a service tank until required.

The automatic regeneration cycle is designed to detect faulty regeneration and, in some instances, to abort the cycle.

5.2.3.1.2 Demineralizers. The demineralizers are cylindrical-carbon-steel rubber-lined pressure vessels designed in accordance with ASME Code, Section VIII. These tanks have 100 percent joint efficiency and 100 percent X-ray of all pressure vessel welds, and were stress relieved after welding. Each ion exchanger has a capacity of 550 gal and is equipped with the following connections, internal components, and accessories:

- (1) 6-in. inlet connection (used for resin fill and water inlet)
- (2) 10-in. outlet connection
- (3) 16-in. access
- (4) 2-in. resin outlet connection
- (5) 1-1/2 in. relief valve connection
- (6) 1-in. vent connection
- (7) 1-1/2 in. sluice water inlet

(8) Underdrain system - "Johnson Well Screen" stainless steel manifold with openings of 0.005 in.

(9) Inlet distribution - a baffle plate suspended from the 6-in. inlet flanged connection.

(10) Exchanger - 33 ft³ of cation resin (Amberlite 200) and 17 ft³ of anion resin (Amberlite IRA 400C).

Stainless-steel filter elements are installed in the condensate outlet piping from each demineralizer.

5.2.3.1.3. Regeneration Tanks. The cation regeneration and resin separation tank is a cylindrical stainless steel (Type-316) pressure vessel designed in accordance with ASME Code, Section VIII.

The tank has the following connections, internal components, and accessories:

(1) 2-in. resin inlet connection

(2) 1-1/2 in. rinse water inlet connection to a PVC inlet distributor manifold

(3) 1-in. acid inlet connection to a stainless-steel acid distributor manifold

(4) 3-in. anion resin removal connection to a stainless-steel distributor manifold

(5) 2-in. cation resin removal connection

(6) 1-1/2-in. drain and back flush connection

(7) 11-in. by 15-in. manhole (ring-type)

(8) The resin inlet distributor - a flat, horizontal baffle plate suspended from the 2-in. resin inlet flanged connection

(9) Underdrain system - a flat plate pierced with holes which are covered with vertical "Johnson Well Screen" segments (0.010-in. mesh)

The anion regeneration and resin mix tank is a cylindrical stainless steel (Type-316) pressure vessel designed in accordance with ASME Code, Section VIII.

The tank has the following connections, internal components, and accessories:

(1) 2-in. resin inlet connection

(2) 11-in. by 15-in. manhole (ring-type)

(3) 2-in. fast rinse inlet and overflow connection to a stainless-steel manifolded distributor with "V" slots

- (4) 3/4 in. caustic inlet connection to a stainless-steel manifolded distributor with laterally facing orifices
- (5) 2-in. resin outlet connection
- (6) 1-1/2-in. drain and air-water mixing inlet connection
- (7) Underdrain system is a flat plate pierced with holes which are covered with "Johnson Well Screen" segments (0.010-in. mesh)

5.2.3.1.4 Other System Components. The sluice water tank is a vertical 3/8-in. carbon steel tank of 1100-gal capacity. It is 60 in. OD, with 108 in. straight sides and a flat top and bottom.

The acid day tank is a flat-bottom vertical tank with a loose-fitting flat cover. It has a capacity of 31 gal and is constructed of 1/8-in. carbon steel with a 3/16-in. lead lining.

The caustic day tank is a flat-bottom vertical tank with a loose-fitting flat cover. It has a capacity of 375 gal and is constructed of 3/8-in. carbon steel. A bracket mounted on the tank holds a mixer propeller mounted on an extended shaft, which connects to a 1/4-hp motor.

The acid pump is a Milton-Roy Simplex Milroyal positive displacement pump. It is rated for 2 gpm and a maximum discharge pressure of 60 psig. The body is of cast iron, and the 1-1/2-in.-dia. plunger is of ceramic material. The packing is Teflon.

The caustic pump is a horizontal motor-driven turbine pump rated to pump 310 gal of a solution containing 136 lb of sodium hydroxide in 36 min. The body is cast iron and is bronze fitted.

The sluice water pump is a single-stage horizontal motor-driven centrifugal pump rated for 40 gpm at a discharge head of 50 ft. This pump has a mechanical buffer seal provided with demineralized water as a buffer to prevent seal leakage.

The recirculating water pump is a single-stage horizontal motor-driven centrifugal pump rated for 600 gpm at a 140-ft total developed head in a 400 psig system. The pump seal has controlled leakoff, piped to the waste water storage tank.

Sluice-resin piping is of saran-lined carbon steel. Piping that contains caustic solution is of black steel. All other piping in the regeneration complex is Schedule No. 80 PVC pipe with PVC cemented joint fittings.

Relief valves are provided for each demineralizer, the cation and anion regeneration tanks, and at the discharge of the acid pump.

5.2.3.1.5 Instrumentation. The following pressures are indicated on the local control board:

- (1) condensate outlet from each demineralizer (a total of three)
- (2) demineralizer condensate influent header

The following local flow indicators are provided:

- (1) air inlet to regeneration complex
- (2) demineralized water inlet to regeneration complex
- (3) inlet condensate stream to each demineralizer (a total of three)

These level controls are provided:

- (1) caustic day tank
- (2) acid day tank
- (3) sluice water tank

These conductivity measurements are indicated on the local control board:

- (1) condensate outlet from each of the demineralizers (a total of three)
- (2) condensate inlet header to the demineralizer
- (3) anion regenerating and resin mixing tank outlet

Automatic temperature compensation is provided for all conductivity instruments.

5.2.3.1.6 Controls. A program controller is in the local condensate demineralizer control panel. The controller has 26 preprogrammed steps which comprise the resin separation, backwash, sluice, chemical treatment, mix, and test functions necessary to regenerate a mixed bed ion exchange resin.

The entire regeneration cycle is begun by pressing a start switch and is accomplished automatically by the controller. Some steps are controlled as a function of time, others are based on liquid level, and one is based on a conductivity indication signal. The individual steps operate valves, pumps, inquiry signals to instrumentation and indicator lights.

The controller is manufactured by Directron Corporation.

Controls are provided on the local control board for the following functions:

- (1) for start of the regeneration cycle controlled by the program controller
- (2) to lock out the acid pump, caustic pump, and sluice water pump
- (3) to operate the valves that sluice resin from the demineralizer to the regeneration complex and return it
- (4) to operate the valves which will accomplish a mid-run backwash (this backwash removes suspended solids from the resin without complete chemical regeneration and returns the resin to the demineralizers)
- (5) to place the selected demineralizers on stream in the turbine condensate system.

An alarm on the main control panel will alert the operator if the conductivity indicators for the condensate outlet of any of the three demineralizers exceeds a preset value.

The program controller for the regeneration complex is on the local condensate demineralizer control panel and is equipped with a light for each step of the program which is lighted during that step. If the level in either the acid day tank or the caustic day tank is improper, the program aborts on the step previous to the injection of the solution and the light will remain on as an alarm to the operator.

A red light is lighted on the program controller if the rinse water effluent from the mixed bed of regenerated resins does not have satisfactory conductivity after completion of the rinse cycle.

5.2.3.2 Primary Purification System (Fig. 5.8). The functions of the system are:

- (1) to maintain optimum reactor water quality (pH and conductivity) to minimize equipment corrosion,
- (2) to minimize fouling of heat transfer surfaces and plating out of corrosion products on primary system pipes and vessels, by removing dissolved and suspended solids, and
- (3) to provide an alternate means of removing excess reactor water during reactor startup and shutdown.

5.2.3.2.1 General Description. The primary purification system is a high-pressure closed loop with two coolers, a pump, and two ion exchangers. Reactor water (25 to

40 gpm) enters the system through a 2-in. nozzle on the bottom head of the reactor vessel and flows through the system components in the following sequence:

- (1) to the tube side of the regenerative cooler, where the water is cooled to 271 F or less by the cold purification system flow on the shell side,
- (2) to the tube side of the purification cooler, where the water is cooled to 120 F (maximum allowable resin temperature) or less by the component cooling water on the shell side,
- (3) to the purification pump,
- (4) to the cation exchanger, which filters the suspended solids and removes most of the dissolved solids (cations),
- (5) to the mixed-bed exchanger, which removes any anions, neutralizes the pH, and removes remaining cations,
- (6) to the pleated-screen-type filter, which prevents resin fines from leaking into the reactor and removes any suspended solids larger than 40 μ , and then
- (7) to the shell side of the regenerative cooler, where the water is heated to not more than 450 F by the hot purification system flow on the tube side.

The water is returned to the reactor through a 2-in. nozzle on the pump 1B suction line between the reactor and the pump suction valve.

The system can be used to blow down excess reactor water to control reactor level during startup and shutdown if the water cannot be blown down through the main condenser blowdown line in the decay heat cooling system (Sec. 5.2.1.1). For this purpose, a blowdown line to the overhead storage tank is connected to the system between the filter and the regenerative cooler. With the reactor at atmospheric pressure, the purification pump has sufficient head to pump 40 gpm of water to the overhead storage tank, while discharging through both ion exchangers.

The system also removes soluble poison after emergency reactor shutdown or control-rod calibration (Sec. 11.13.2). Initial reduction of the poison concentration is by flooding and draining, but the remainder is removed by ion exchange. Unlike normal reactor water conditions, where the solubles are predominantly cations, the poison, by weight, is predominantly anions. Therefore the cation exchanger is loaded with anion resin during removal of poison. The cation resin in the mixed bed exchanger removes cations.

Stop valves can isolate the entire system from the reactor or isolate the individual components from the rest of the system. Bypass lines and valves around the tube side of the regenerative cooler, the cation exchanger, and the mixed bed exchanger allow the

system to be operated when these components are isolated. A pneumatic control valve, whose main purpose is to close automatically upon boron injection, is operable from the control room. This valve is located just outside the biological shield at the inlet side of the regenerative cooler. This valve, together with the check valve on the discharge side of the pump, serves to isolate the tube sides of the coolers in case of failure. The two filters are installed in parallel so that the system can be operated while one filter is isolated for cleaning. Relief valves set at 1500 psig protect the ion exchanger vessels from overpressure.

The system piping and valves are of TP-304 stainless steel, Schedule 80, ASTM Spec. A376, and are 1-1/2 in., 2 in., and 2-1/2 in. in diameter. The system piping and valves from the reactor to the purification cooler and from the shell of the regenerative cooler to the forced-circulation line are designed for a maximum working pressure of 1400 psig at 595 F. The rest of the piping and valves are designed for a maximum working pressure of 1400 psig at 200 F. The piping and valves are sized to permit a flow of 40 gpm without excessive pressure drop, and the material is resistant to water corrosion and erosion.

5.2.3.2.2 Regenerative Cooler. The regenerative cooler is a horizontal, two-pass tube, U-tube heat exchanger of all-welded construction. It is designed in accordance with ASME Code, Section VIII, and Nuclear Code Case 1270-N. All material is AISI Type-304 stainless steel. The vessel shell is designed for 1500 psig at 500 F, and tubes are designed for 1500 psig at 625 F. Shell and tube side relief valves releasing to the containment building protect against overpressure.

5.2.3.2.3 Purification Cooler. The purification cooler is a horizontal, one-pass shell, two-pass tube, U-tube heat exchanger of all-welded construction. It is designed in accordance with ASME Code, Section VIII, and Nuclear Code Case 1270-N.

The tubes are of AISI Type-304 stainless steel and designed for 1500 psig at 325 F; the shell is of carbon steel and designed for 100 psig at 200 F. Relief valves are incorporated in both shell and tube sides and release to the containment building to protect against overpressure. The connections for primary water flow are of AISI Type-304 stainless steel; the connections for cooling water flow are of carbon steel.

5.2.3.2.4 Purification Pump. The purification pump is a single stage, horizontal, motor-driven centrifugal pump, rated for 260 ft total discharge head at 40 gpm.

The pump has one radial-type ball bearing and three thrust-type ball bearings. An oil sump and oil rings lubricate the bearings. The pump has a mechanical seal with an internal pumping ring and an external cooler. The pumping ring removes seal water from the mechanical seal annulus and circulates it through the cooler and back to the

seal. The cooling water flowrate to the cooler is regulated to maintain a seal water temperature of not more than 160 F at the outlet from the seal. Component cooling water circulation cools the radial and thrust bearing jackets and the mechanical seal chamber jacket.

5.2.3.2.5 Ion Exchangers. The ion exchangers are within cylindrical AISI Type-304 stainless steel pressure vessels, designed in accordance with ASME Code, Section VIII. Both vessels are surrounded by 4 in. of poured lead and 2 ft of concrete shielding.

The cation exchanger contains 9 ft³ of Nuclear Grade resin. The mixed bed exchanger contains 6 ft³ of Nuclear Grade cation resin and 3 ft³ of highly basic Nuclear Grade anion resin. The bottom screens are designed to withstand a pressure differential of 150 psi.

The resins are sluiced to the spent resin storage tank and allowed to decay prior to enclosure in drums for off-site shipment (Sec. 11.13.1.5). The exchanger is recharged with new, fully generated resin sludge in through connections at the top of the exchangers.

5.2.3.2.6 Filters. The filters are in cylindrical AISI Type-304 stainless steel pressure vessels of all-welded construction. Each filter has a removable pleated screen filter cartridge to filter particles 40 μ or larger. The filter cartridge is designed to withstand a pressure differential of 150 psi.

5.2.3.2.7 Instrumentation. Local temperature indicators are provided for the following:

- (1) purification system inlet,
 - (2) regenerative cooler tube-side outlet,
 - (3) purification cooler reactor water outlet,
 - (4) regenerative cooler shell-side outlet,
 - (5) purification cooler cooling water outlet,
 - (6) purification pump seal outlet,
 - (7) purification pump seal chamber and bearing jackets cooling water outlet,
- and
- (8) purification system outlet.

Temperatures for the following are indicated in the control room:

- (1) purification system inlet,

- (2) purification cooler reactor water outlet (also recorded in control room),
- (3) purification system outlet, and
- (4) cation exchanger inlet temperature high(annunciates an alarm).

Local pressure indicators are provided for the following:

- (1) purification pump pressure differential,
- (2) cation exchanger pressure differential,
- (3) mixed bed exchanger pressure differential,
- (4) purification filters pressure differential, and
- (5) purification filters inlet pressure.

The flowrate at the purification filters outlet is indicated in the control room. The flow element is upstream of the blowdown connection to indicate the total system flow. A flow switch actuates a control-room alarm for a low mixed bed exchanger to regenerative cooler flow ratio.

The conductivity at the following points is recorded in the control room:

- (1) cation exchanger inlet,
- (2) cation exchanger outlet, and
- (3) mixed bed exchanger outlet.

A high conductivity at any of these points actuates an alarm in the control room (Sec. 11.7.3). Three thermal elements are provided to compensate the conductivity cell signals for temperature.

5.2.3.2.8 Controls. Sufficient controls are provided in the control room so that the system can be started and shut down as required from the control room. An interlock in the purification pump start circuit prevents starting of the pump or turns it off, if operating, when the two-position pneumatic diaphragm-operated valve, just before the regenerative cooler, is closed. This interlock is provided to prevent pump cavitation when the water supply is cut off.

The purification pump and the pneumatic valve are controlled from the control room. The system flowrate is adjusted by local throttling of the purification pump discharge valve.

The blowdown flowrate of the purification system to the overhead storage tank is adjusted locally by throttling of one of the two manual globe valves that are in series in the blowdown line. The purification system pneumatic control valve is interlocked to close automatically upon boron injection and must be manually reopened.

5.2.4 Seal Injection System (Figs. 5.9 and 5.12)

5.2.4.1 General Description. The seal injection system provides cooling and sealing water for the forced-circulation pump seals and the reactor control-rod drive mechanisms. The system water supply is contained in a 150-gal reservoir, which is supplied with makeup water from the full flow condensate demineralizer system. Two motor-driven positive-displacement type seal injection pumps, each with a rated capacity of 52 gpm, take water from the seal injection reservoir. One pump is normally in operation, with the other on ready standby. Electrical power is supplied to the pump motors from two 480-v busses. If one 480-v bus should fail, power would still be supplied to the other pump. Upon failure of the operating pump, the standby is started automatically.

The pulsating flow from the pump is dampened by a bladder-type accumulator of 5-gal capacity, rated for 3000 psig, to maintain stable system conditions. A differential pressure controller and bypass valve maintain seal injection system pressure at a set amount above reactor pressure. Flow controllers in three branches of the system supply the seals.

Three piston-type accumulators pressurized with nitrogen bottles are connected to the header that supplies seal water to the forced-circulation pumps. The accumulators contain sufficient water to supply the seals for approximately 1-1/2 min, while the pumps coast to zero rpm, in case of pump failure or other malfunctions that cut off the normal water flow. The forced-circulation pump seals are supplied with cooling water whenever the pumps are rotating. Controls in the system maintain the flowrates within specified limits. The leakoff water from the forced-circulation pump seals is returned to the seal injection reservoir at a controlled back pressure of 5-10 psig.

Two full-flow pleated-screen type 10 μ filters, arranged in parallel, are in each of the supply headers to the pumps and the control-rod seals. One of the filters is on the line and the other on standby. The piping system is of carbon steel upstream of the filters, and of stainless steel downstream.

Water is also supplied to 29 control-rod drive seals. Controls in this branch of the system maintain the required rate of flow to the seal manifold. A throttle valve and a flowmeter are provided in each of the lines to the control-rod drives to equalize flow. Leakoff water from the control-rod drive seals is returned to the seal injection reservoir.

Continuous blowdown of water from the control-rod drive upper housing minimizes the temperature differential between the nozzles and the reactor vessel, to preclude excessive thermal stresses. A mixture of reactor water and seal water (~ 1 gpm) is blown down through a connection on each control-rod drive upper housing flange. The designs for seal injection and blowdown are illustrated in Figs. 5.10 - 5.12. Each blowdown line contains a flowmeter and a throttle valve to equalize flow. The lines connect to a manifold, from which suction is taken by two pumps. One of the pumps is normally running and the other on standby. The pumps are rated at 30 gpm and 50 ft TDH. Upon failure of the operating pump, a flow switch actuates a low flow alarm in the main control room and starts the standby pump.

The control-rod nozzle effluent pumps discharge to the decay heat cooling system upstream of the decay heat pumps. This piping also connects directly to the suction side of the reactor forced-circulation pumps.

Since the blowdown piping is connected between the suction and discharge of the forced-circulation pumps, sufficient differential pressure probably exists to provide the 29 gpm blowdown flow without operation of a control-rod nozzle effluent pump. Therefore, a bypass has been placed around the control-rod nozzle effluent pumps. This mode of operation will be investigated during preoperational plant testing.

Provisions for flushing of the control-rod drive housings are shown in Fig. 5.12. When appropriate valving has been made, flushing water is admitted to the housing through a special connection via a hose connected to the seal injection system. Water from the housing passes through the normal seal injection port, to the retention tanks of the drain and liquid waste system via the floor drain and sump.

5.2.4.2 Forced-Circulation Pump Seals. The shaft sealing arrangements for the forced-circulation pumps consist of a lower seal, an upper seal, and a mechanical backup seal (Fig. 5.13). The lower seal portion consists of two lower seal rings. These rings maintain a minimum pressure differential of 60 psi between the seal lower injection point and the pump casing, preventing reactor water from flowing upward past the rings.

The upper portion of the pump seal consists of four floating bushing rings, a floating sleeve, and a face seal. The four floating bushing rings provide resistance to leakage flow and a corresponding pressure drop between the injection point and the normal seal leakoff. This pressure drop causes unbalanced forces on the bottom and topmost surfaces of the floating sleeve. The floating sleeve is thereby moved upward with a force proportional to the leakage flow and the resulting pressure drop. A ring in the top of the floating sleeve forms the lower surface of the face seal. As the floating sleeve moves upward, the gap in the face seal is reduced, and leakage flow is further

restricted until the forces at the top and bottom of the floating sleeve are balanced and a constant leakage flowrate is reached. Any attempt to force more leakage through the seal causes the floating sleeve to close the face seal gap and restrict that leakage. The minimum design leakage flowrate for each pump is 0.2 gpm, which is required to cool and lubricate the rotating seal members.

A pressure of about 200 psig is required at the seal injection point to force the required 0.2 gpm leakage through the upper seal. When reactor pressure is zero this pressure would ordinarily be attained by forcing a high rate of seal flow through the two lower seal rings. As reactor pressure is increased to 200 psig, the seal flowrate can be reduced until the pressure drop across the lower seal meets the 60 psi differential requirement. The second seal injection point allows a constant seal injection flowrate of about 5 gpm to be maintained over the whole reactor pressure range. The upper injection point adds another floating bushing ring in the flow path of seal water to the pump casing, thus causing more pressure drop at a lower seal flowrate and providing the required leakage flow at low reactor pressure. When reactor pressure has increased to above 500 psig, the leakage flow will be more than 2 gpm, which is more than the required leakage flow, and flow to the pump casing will be less by a like amount. The seal flow can then be switched to the lower injection point. Pressure at the injection point will be reduced, since there is one less seal ring in the flow path to the casing. The leakage flow therefore decreases. The recommended reactor pressure range for switching injection points is between 500 and 600 psig.

A John Crane mechanical seal is located above the face seal on the pump shaft to prevent leakage from flowing along the shaft to the atmosphere, and to direct leakage flow out the normal leakoff port. A nominal back pressure of 10 psi is maintained on the seal leakoff to ensure enough leakage through the mechanical seal for lubrication. Since the mechanical seal is designed to withstand a 700 psi differential, the expected leakage to the atmosphere is on the order of 5 cc/hr with the existing 10 psi differential.

5.2.4.3 Control-Rod Drive Seals. The control-rod drive shaft seal is a spring-loaded mechanical seal in the lower rod drive housing, as shown in Fig. 5.10. Demineralized water injected into each of the 29 housings at a point just above the seal prevents the seals from contacting the hot reactor water. The normal seal leakoff water (0.15 gpm or less) is piped to the seal injection reservoir. The control-rod drives require periodic flushing in the area just above the lower housing. A special hose connection and tubing are provided on the housing for this purpose. The rod drives are flushed individually during reactor shutdown. The flushing leakoff water flows from the normal seal injection inlet to the reactor building sump when the proper valving has been made. A seal water flow of 10-20 gpm is provided for this purpose.

5.2.4.4 Seal Injection Pumps. There are two vertical, motor-driven, positive-displacement-type seal injection pumps in the system. The pumps are triplex plunger pumps of 52 gpm capacity and can deliver seal flow continuously at 1540 psig. The motor is rated for 60 hp. The calculated BHP at 1540 psig is 58.3 hp. The pumps are manufactured by the Aldrich Pump Company (a subsidiary of Ingersoll Rand), model number 17/8X3V-3.

5.2.4.5 Seal Injection Reservoir. The seal injection reservoir is a 150-gal all-welded carbon steel vessel built to the 1962 ASME Code, Section VIII, per Par. UW 12-b with Code Stamp required. The tank is designed for an external pressure of 60 psi at 200 F, and is mounted horizontally on mild steel saddles. The tank is manufactured by Alpha Tank and Metals Manufacturing Company.

5.2.4.6 Seal Injection Piston-type Accumulators. These three accumulators provide a backup supply of seal water to the forced-circulation pump while transfer is made to the standby seal injection pump after failure of the operating pump. The accumulators are each of 7.7 gal capacity and are mounted vertically, with inlets and outlets manifolded together in parallel. The accumulators are of carbon steel, with screwed end caps, and plated inside with chrome. They are rated for 3000 psig between minus 40 F and plus 200 F, and are hydrostatically proof-tested at 6000 psig. Teflon rings on the plungers contact the inside surface of the cylinder.

5.2.4.7 Control-Rod Nozzle Effluent Pumps. The pumps in the control-rod drive blowdown portion of the system are rated for 30 gpm and a total developed head of 50 ft and are driven by 440-v, 3-phase, 1-hp motors. The pumps are designed for a working pressure of 1400 psig and normally pump water at about 450 F, but are rated to pump reactor water at 575 F. A minimum net positive suction head of 27.3 ft occurs when both forced-circulation pumps are shut down under saturated reactor conditions, with no feedwater or control-rod drive seal water flow available. The normal NPSH is approximately 875 psia. One of the pumps is normally operating and the other on standby.

5.2.4.8 Instrumentation and Control.

5.2.4.8.1 Hand Control Switches. The seal injection pumps are operated by hand control switches in the control room. The hand control switches for control-rod nozzle effluent pumps are also in the control room.

The pneumatic 3-way valves in each pump seal injection supply line are operated remotely from the control room to route seal water to the desired forced-circulation pump seals injection point by solenoid valve switches that regulate the pneumatic valve air supply. The pneumatic valves are designed to ensure that the port to one

injection point opens as soon as the port to the other point begins to close, thus avoiding interruption of seal flow.

5.2.4.8.2 Flow. Total flow from the seal injection pumps is measured by a local flow indicator. This indicator also has contacts for a low flow alarm and auto-start of the backup pump, has an accuracy of 2 percent deviation from true flow, and has a range of 0 to 60 gpm.

Flow from the pump discharge header branches into three main lines; one line to each of the two forced-circulation pump seals and a third line to the 29 control-rod drive seals. Each of these lines contains flow indicator controllers that may be operated from the control room. The controllers, upon a signal from the flow-measuring orifice and transmitter in the associated supply lines, operate associated needle valves to regulate the supply flow at the set-point value. The flow controller in each of the pump seal supply lines has a range of 3 to 10 gpm, and the controller in the control-rod drive seal supply line has a range of 0 to 15 gpm.

The flowmeter in the seal water inlet line to each of the forced-circulation pumps actuates an alarm upon a low flow (~ 3 gpm) signal and closes a pneumatic valve in the associated normal leakoff line. An additional flowmeter downstream of the pneumatic valve in each of the leakoff lines actuates another alarm for low leakoff flow (~ 0.5 gpm) and trips the forced-circulation pump at a sub-alarm set-point (~ 0.2 gpm). The leakoff flowmeters are rotameter-type units with a scale range of 0 to 5 gpm, and are rated for 150 psig.

The flowmeter in the seal water inlet line to the control-rod drive seals actuates an alarm upon a low flow (~ 6 gpm) signal. This flow is manifolded into 29 separate lines, one to each control-rod drive. A small local flowmeter in each of these lines, upstream of a throttling valve, allows adjustment to equal flow for each drive seal. These rotameters each have a range from 0.1 to 1 gpm and are rated for 1600 psig.

A small local flowmeter is also located in each of the 29 blowdown lines from the control-rod drive housings. These rotameters have a range from 0.3 to 3.0 gpm and are rated for a pressure of 1400 psig at 575 F. The normal temperature of the blowdown water is approximately 450 F.

Blowdown flow is also monitored beyond the junction of the 29 blowdown lines to ensure a reactor water flow adequate to avoid thermal stresses at the junction of the control-rod nozzles and the reactor vessel. An alarm is actuated if total blowdown flow drops below the set value (~ 25 gpm).

5.2.4.8.3 Pressure. A differential pressure transmitter, accurate to within 2 percent and with a scale range of 0 to 400 psi, is connected between the suction of one forced-circulation pump and a point downstream from the discharge header of the seal injection pumps. This transmitter maintains the pressure in the seal injection pump discharge header at a set amount (300 psi) above reactor pressure, by regulation of a control valve through which water bleeds back to the pump reservoir. For a low differential pressure (180 psi), the transmitter actuates the seal-injection accumulators to maintain seal water flow; a check valve prevents back flow from the accumulators to the seal injection pump discharge header. A second lower differential pressure (160 psi) trips the forced-circulation pumps. Another set of contacts in the pump-start circuit prevents pump startup below a differential pressure of 160 psi.

Differential pressure indicators across the forced-circulation pump seal water filters and the control-rod drive seal water filters actuate alarms in the control room upon a high differential pressure signal. The instruments are rated for 1600 psig and have an accuracy of 1 percent and scale range of 0 to 30 psi.

Seal injection pump discharge pressure (between 0-2000 psig) is read in the control room and pressure in the seal leakoff lines (between 0-50 psig) is measured by local gauges.

Differential pressure instruments indicate locally the pressure difference from the seal injection point to the pump suction, for each of the forced-circulation pumps. These instruments have a range of 300 psi and an accuracy of 1 percent of full scale, and are designed for 1600 psi pressure.

5.2.4.8.4 Temperature. Fast response thermocouples, grounded in wells for 150 psig, monitor temperature in the leakoff lines from each of the forced-circulation pump seals. The temperature indicators actuate high temperature alarms in the main control room, and trip the associated pump. A local temperature indicating switch in the seal injection pump suction line supplies the signal to a high temperature alarm in the control room. The discharge line temperature of the seal injection pumps is measured with a thermocouple and recorded in the control room.

5.2.4.8.5 Level. A liquid level gauge and controller monitors water level locally in the seal injection pump reservoir and operates a control valve in the normal supply line from the condensate demineralizers. The level is normally maintained at the tank centerline, and a drop of 4 in. will cause the controller to open a second control valve in the emergency supply line from the overhead storage tank. Level switches actuate high and low level alarms (i.e., 9 in. above the tank centerline and 4 in. below the tank centerline).

Level switches in the leakoff line from the mechanical backup seal of the forced-circulation pump detect leakage and actuate an alarm in the control room if there is excessive leakage.

5.2.5 Boron Injection System (Fig. 5.14)

The boron injection system is partially in common with the emergency core spray system discussed in Sec. 5.2.6. The system injects a sodium pentaborate solution to provide an alternate means of manual reactor shutdown in the event of stuck control rods.

5.2.5.1 General Description. The boron injection system consists of a boron tank with the necessary control valves and piping to connect the tank to the core spray pumps suction header and to connect the core spray pumps discharge header to the boron injection nozzle on the forced-circulation pump discharge header. Normally, the boron tank is isolated from the emergency core spray system by a control valve in the boron supply line to the core spray pumps suction header, and the control valve in the boron discharge line to the reactor is closed to isolate the spray system from the forced-circulation pump discharge header. A manual stop valve can also be used to isolate the boron tank from the emergency core spray pumps suction header.

The control valve in the boron supply line to the core spray pumps is provided with a bypass so that boron can still be injected if the control valve fails to open. Gate valves on each side of the control valve permit isolation for maintenance and testing. Sample connections at the discharge of the boron tank and on the inlet side of the control valve allow periodic inspection and flushing of the valves in contact with sodium pentaborate solution to ensure that the boron supply line is not clogged.

A check valve is provided beyond the bypass line discharge tee connection to prevent backflow of water from the overhead storage tank or the high pressure service water supply line. The valve in the boron tank drain line to the retention tanks is locked closed to prevent inadvertent draining of the boron tank. The system also contains two core spray pump relief valves, each of 50 gpm capacity. There are additional valves used to isolate the individual system components for maintenance. These valves are locked open to prevent the inadvertent closing of a valve and consequent system inoperability.

The system piping and valves on the suction side of the core spray pumps are designed for 60 psig and 160 F. The piping and valves on the discharge side of the pumps to the injection control valves are designed for 1400 psig and 160 F; the control valves and piping beyond the control valves are designed for 1400 psig and 595 F. The piping and valves are made of AISI Type-304 stainless steel and are sized for a flowrate of 100 gpm.

The boron injection line and the boron tank drain line are insulated and traced with electric heating cables from the boron tank to the first normally closed valves. The heating cables maintain the piping at 100-110 F.

The system contains sufficient sodium pentaborate to render the reactor subcritical in the cold clean condition with all control rods fully out. Based on a maximum excess reactivity of 20 percent, ~ 10.2 lb of natural boron uniformly distributed in the core provide a 5 percent shutdown margin. To obtain the necessary concentration after dilution by primary system water, 640 gal of 7.8 w/o sodium pentaborate decahydrate ($\text{Na}_2\text{B}_{10}\text{O}_{16} \cdot 10 \text{H}_2\text{O}$) must be injected into the reactor. Since the boron tank contains 800 gal of the sodium pentaborate solution, an excess capacity of approximately 25 percent is present.

The combined capacity (100 gpm) of the core spray pumps (Sec. 5.2.6.2) provides sufficient sodium pentaborate within 1 min to overcome the voids and Doppler fuel temperature effects and to provide 0.4 percent additional negative reactivity.

5.2.5.2 Boron Tank. An 800-gal tank contains the sodium pentaborate solution. The solution must be maintained at 100-110 F to prevent precipitation of the sodium pentaborate; it will be less than 90 percent saturated at 100 F. The tank is maintained at this temperature by two electric heating elements and a temperature switch, although one heating element by itself could maintain the temperature. The tank is insulated to reduce heat losses.

The boron tank is fabricated with 3/16-in.-thick AISI Type-304 stainless steel, according to the American Water Works Association (AWWA) Standard Specification D-100. It is a cylindrical tank with flanged and dished end caps.

The tank penetrations are as follows:

- (1) 18-in. manhole
- (2) 12-in. chemical fill connection with blind flange
- (3) 3-in. outlet and drain connection
- (4) 2-in. vent
- (5) two 2-in. pipe taps for heating elements
- (6) 1.5-in. demineralized water inlet
- (7) 1-in. threaded water inlet

- (8) 1-in. pipe tap for temperature controller
- (9) two 0.5-in. connections for heated gauge glass

5.2.5.3 Instrumentation. The boron tank temperature is recorded in the control room. The recorder provides an alarm signal for low temperature. In addition, the controller provides a signal to the "Reactor Auxiliaries Temperature High" alarm for high boron tank temperature.

Local pressure gauges are provided on the core spray pumps suction header and at the discharge of each pump.

The boron tank has a heated gauge glass for local level indication and two level switches that activate control room alarms. One switch activates a low level alarm when there is less than 650 gal sodium pentaborate in the tank. The other switch activates a "Lo-Lo" level alarm when the tank level drops below the heating elements.

5.2.5.4 Controls. Boron injection is initiated by a pushbutton, which starts both core spray pumps and energizes two solenoid valves in the parallel piping from the nitrogen supply header to position the control valves for boron injection. The button also closes the primary purification system inlet control valve, to prevent that system from removing the boron, and cuts off the purification pump. If any rods fail to insert fully upon a full scram, a control room alarm is given. This alerts the operator to the possible need of pushing the boron injection button.

A three-position boron test keyswitch permits bypass of the full scram interlock to operate the control valves and start the pumps. A second position on the test switch operates the alarm that signals when all rods are not inserted 3 sec after a full scram signal. The third position is normal. The core spray pumps can also be controlled from the control room.

The boron tank temperature switch controls the two electric heating elements to automatically maintain a temperature of 100-110 F. A boron tank level switch shuts off the heaters on low-low tank level. The boron piping temperature switch controls the electric heating cable that traces the pipes to automatically maintain a temperature of 100-110 F.

5.2.6 Emergency Core Spray System (Fig. 5.14)

This system minimizes fuel damage due to residual and decay heat if the reactor water level accidentally drops below the top of the core.

5.2.6.1 General Description. The emergency core spray system includes two pumps which normally take suction from the 42,000-gal overhead storage tank and discharge

to the core spray header (ring) at pressures up to 1400 psig. The spray header supplies the lines leading to a spray nozzle just above each fuel assembly.

The emergency core spray pumps are installed in parallel. Each can supply 52 gpm of water to the core spray header, a flow sufficient to remove the latent and decay heat. The standby pump can be started from the control room to supply ~50 gpm of additional flow over the core if desired. The emergency core spray pumps are also used in the boron injection system (Sec. 5.2.5). Control valves are provided for this purpose in the core spray suction and discharge lines. A demineralized water supply line to the pumps provides for lubrication of the pump pistons. The pumps are used for core spray when the reactor remains pressurized, as in the case of a small leak below the core. When the reactor and containment building pressures are equalized, as after a major system leak or rupture, a low pressure supply line bypassing the emergency core spray pumps allows water to flow directly from the overhead storage tank (or service water line) to the core spray header.

The overhead storage tank contains 42,000 gal of water above the core spray supply nozzle. Of this, 27,000 gal is above the building spray supply nozzle. Thus there is always at least 15,000 gal of water available for core spray except during refueling operations even after the building spray system has been used. In addition, make-up water is supplied automatically from the 30,000-gal demineralized water storage tank in the event of low level in the overhead storage tank. When all the overhead storage tank water has been used, high pressure service water can be used. Normally, 2650 gpm of service water can be supplied for the emergency core spray cooling system, the shield coolers, and the shutdown condenser. Upon a complete loss of electrical power, the supply is reduced to 200 gpm.

The core spray header above the top of the failed fuel assembly detection tube support grid supplies the 72 spray lines. An individual 3/8-in. spray line is provided for each fuel assembly. The spray lines are installed concentrically within the failed fuel element detection tubes. Each spray line contains a needle valve used to set the flow for each fuel element assembly location. The valves are tack-welded after being set to lock them in position. The required flow per assembly varies between 0.40 to 0.77 gpm, depending on the assembly location.

The system piping and fittings from the overhead storage tank to the core spray pumps are designed for a maximum working pressure of 150 psig at 160 F. The pump discharge piping and fittings to the injection control valves are designed for a maximum working pressure of 1400 psig at 160 F, and that from the control valves to the reactor is designed for 1400 psig at 595 F. All piping and valves are of AISI Type-304 stainless steel.

Stop valves permit isolation of the emergency core spray system from the overhead storage tank and service water system, and isolation of the core spray pumps from the rest of the system. The stop valve in the water supply line to the pumps must be closed when the reactor is not pressurized to prevent overhead storage tank water and/or service water from flowing to the reactor under any of the following conditions:

(1) The head from the overhead storage tank is sufficient to open the core spray pump internal inlet and outlet valves.

(2) Inadvertent opening of the service water system control valve when reactor pressure is less than service water system pressure.

(3) Loss of air and/or control power causes the low pressure core spray control valve to open and reactor pressure is insufficient to prevent flow from the overhead storage tank.

Check valves in the service water line and the line from the overhead storage tank prevent water from flowing between systems. The check valve in the low pressure core spray supply line prevents a backflow of reactor water when the control valve is open.

Two relief valves on the core spray pumps protect against overpressure.

5.2.6.2 Emergency Core Spray Pumps. The two horizontal, motor-driven, positive-displacement-type emergency core spray pumps are rated for 50 gpm and a maximum operating pressure of 1450 psig. They are driven by 50 hp 880 rpm motors. Both pumps are connected to the emergency power supply.

5.2.6.3 Overhead Storage Tank. The 42,000-gal overhead storage tank in the dome of the reactor building is supplied with demineralized water. The tank supplies water for several plant functions and systems, primarily for the emergency core spray cooling system and the building spray system. The emergency core spray cooling system is connected to the bottom of the tank, and the building spray system is connected to a standpipe within the tank. The top of the standpipe is located so that 15,000 gal are reserved for the emergency core spray cooling system.

The tank also supplies water for the following:

- (1) seal injection reservoir emergency supply
- (2) reactor water supply during refueling
- (3) water supply during loading of the fuel element shipping cask

5.2.6.4 Instrumentation. Local pressure gauges are provided on the suction header of the emergency core spray pumps and at the discharge of each pump.

Excessively low reactor level is alarmed (see Sec. 11.9.4) and starts one of the emergency core spray pumps. In case of low reactor pressure combined with a low reactor level, a solenoid pilot valve is actuated to open the low pressure core spray valve so that water flows directly from the overhead storage tank to the core spray header.

The overhead storage tank level is indicated in the reactor building. Level switches (see Fig. 5.18) on the tank actuate high and low level alarms, and control the demineralized water supply control valve for the tank. Low level opens the pneumatically operated valve; normal level closes it.

Water flow in the line to the spray header is sensed by a flow switch that actuates an alarm in the control room when low flow (see Sec. 11.9.4) and low reactor water level exist simultaneously.

5.2.6.5 Controls. The emergency core spray pumps are started and tripped with individual control switches in the control room. However, an interlock prevents manual starting of the pumps unless the reactor is scrammed. A selector switch is positioned to place one pump on automatic control and the other on standby. If the reactor water level drops to the low level switch set-point, the reactor scrams and the core spray pump selected for automatic operation starts. If this pump trips because of overcurrent, the standby pump starts automatically. Low emergency core spray flow is annunciated in the control room to indicate that the standby pump should be manually started if it is not already in use. A high reactor water level trips the pumps except when they are being used for boron injection.

The low pressure core spray control valve opens automatically when the reactor is at low pressure and the level drops to the low-low level alarm point. This valve also has a hand controller in the control room.

When boron injection is initiated, both core spray pumps are started and the core spray control valves at the pumps suction and discharge are closed.

The core spray system service water supply control valve is controlled manually in the control room.

5.2.7 Failed Fuel Element Location System (Fig. 5.15)

The failed fuel element location system is part of the LACBWR research and development program. The laboratory tests and development of the system are discussed in the amendments to ACNP-62574.

Failed fuel is detected through information from the vent and waste-gas disposal system radiation monitors. Failed fuel elements are identified by means of the failed fuel element location system.

Sampling tubes take steam and water samples from the top of each fuel assembly. Another sample is collected from one of the spare fuel assembly shrouds as a clean reference. The part of the failed fuel element location system within the reactor vessel is discussed in Sec. 4.3.8.

The tubes penetrate the reactor vessel in the upper head flange. Each line, equipped with a cut-off valve, is manifolded into one of two headers. Figure 5.15 shows the flow path, control equipment, pressure, temperature, and flow measuring instruments beyond the headers. There are two identical systems, largely independent of each other, each connected to a header. The sample piping arrangement is such that the flow from any line can be directed to either system. Design conditions for the stream entering the gas stripping equipment are 577 F, 1300 psi, and 1 gpm (500 lb/hr). The sample may be as much as 50 percent steam by volume, but cooling enroute condenses most of the steam. The sample first passes through a heat exchanger where its temperature is reduced to below 150 F as measured at the entrance to the gas stripping column. The temperature is set by manually adjusting the volume of cooling water passing through the heat exchanger shell. The rate of flow of cooling water is indicated on a rotameter of the sight flow indicating type. Because the column is designed for 95 percent stripping efficiency (of xenon) at 120 F, small variations of temperature around this point have only minor effects on the stripping efficiency. A rupture disc in the cooling water outlet bursts at 150 psi, a condition that might occur if upstream and downstream lines are shut off, or if a tube of either heat exchanger ruptures. If the cooling water flow becomes insufficient to maintain the proper temperature, an over-temperature switch downstream of the heat exchanger closes the control valve by deenergizing a solenoid valve, which releases the diaphragm air pressure. When the control valve is closed, high-temperature water cannot flow to the stripping column. The over-temperature switch is adjusted to only slightly above the normal operating point so that it acts promptly. The sample stream pressure is reduced by a normally-closed, pressure-to-open, diaphragm-operated control valve. This valve with its pressure controller controls the outlet pressure to approximately 600 psig. The controller also indicates the outlet pressure. The rate flow of the sample is adjusted by a manual needle valve just upstream of the gas stripping column. Since the pressure both upstream and downstream of this valve is held constant, the flowrate is constant regardless of reactor pressure changes.

The reactor water sample enters the gas stripping column through a tee, where a thermometer indicates sample temperature. The sample then passes through a solid cone spray nozzle in the top of the stripping column, down through the packing and into a sump at the bottom of the column. From there the water flows into a continuous drainer

in the outlet line of the column. Approximately 6 in. of water is maintained in the column to prevent the escape of gas through the drain line. An equalizing line between the continuous drainer and the column ensures that the fluid level in both devices is the same. The water flows out through a rotameter, where the sample flowrate is measured. The discharge from the rotameter joins with the stream from the second system and passes through a check valve into the reactor retention tank system. The check valve prevents an accidental backflow into the equipment, which might be caused by a pressure surge in the purification system line. A drain valve at the low point of the water discharge line allows any accumulation of solids to be drawn off periodically and permits the gas stripping column and the continuous drainer to be emptied for servicing.

Stripping gas at a pressure of 50-250 psig enters both systems through a common shutoff valve and a pressure reducing regulator. Its flow into the stripping column is regulated by a needle valve and measured by a rotameter. It flows countercurrent up through the packing and exits through a line at the top of the column, where a gauge indicates the operating pressure. A rupture disc designed to burst at less than 15 psig protects the low pressure portion of the system from excessive pressure if the control valve fails. Approximately 95 percent of the total xenon is removed in the stripping column. The gas leaving the stripping column is saturated with moisture, and consists largely of the stripping gas, with only trace amounts of fission gases and radiolytic gases. The moisture is removed to a dew point of 42 F or less so the gas can be introduced into the electrostatic precipitation monitor. Excessive moisture can cause failure of the monitor insulation because of the high voltage. The dehydration system consists of a regenerative heat exchanger, where the gas is precooled, and a refrigeration gas dryer, to cool the gas below 42 F. Water that condenses drains out through a trap and joins the water from the gas stripping column. This total flow is measured by the rotameter. The gas again passes through the heat exchanger, where it is warmed by the incoming wet gas to a temperature not above 86 F, the maximum temperature at which the plastic scintillation phosphor should be operated. Heating improves the chamber operating conditions by decreasing the relative humidity of the gas. The temperature of gas leaving the dehydration system is controlled by use of a bypass around the heat exchanger. There is a small filter ahead of the precipitator. A purge stream from the gas supply is introduced into the precipitator to prevent the migration of fission gases down the precipitation wire passage to the scintillation detector. This stream is controlled with a purge rotameter that has a built-in regulating valve. The operating pressure of the entire system, from the sample flowrate adjusting valve through the precipitator, is maintained by a back pressure regulating valve downstream of the precipitator. This valve is set to hold a pressure of ~5 psig. The off-gas passes through a check valve to the vent and waste-gas disposal system.

The precipitation monitors consist of:

- (1) a chamber through which the sample gas is drawn

(2) a wire onto which the fission products (mainly solid daughters of xenon and krypton) are deposited

(3) a phosphor and scintillation counter detector

The wire passes through the chamber that carries the gas sample. The wire is at ground potential; the chamber walls are maintained at a high positive potential. As the sample passes through the chamber, some of the gaseous fission products decay by beta emission, and the ionized solid daughter atoms precipitate onto the wire. After a period of deposition, the wire is moved into the phosphor. The phosphor is a thin-walled tube, housed in a thick plastic light guide. Scintillations in the phosphor, caused by beta particles from the active daughter deposits, are detected by the scintillation detector and counted by apparatus external to the precipitator.

5.2.8 Component Cooling System (Fig. 5.16)

The component cooling system provides demineralized cooling water to serve as a heat sink for the following reactor systems equipment: shield cooler, decay heat cooler and pump, purification cooler and pump, fuel element storage well cooler, forced-circulation pump liquid coupling and lube oil coolers, sample cooling coils, and the air-conditioners for the containment building. The system also provides an additional barrier against the release of radioactive material from the reactor system to the plant environment.

5.2.8.1 General Description. The system consists of a water surge tank, two water circulation pumps, two coolers, and a loop monitor. The surge tank provides the excess volume for thermal expansion and contraction of the component cooling water. The tank is connected to the suction side of the component cooling pumps to provide a constant head. The two pumps can circulate system water at 1500 gpm, but the use of only one pump supplies the normal required flowrate. The approximate cooling flow distribution to individual components and the heat load at full power operation are indicated in Table 5-1.

TABLE 5-1
FULL-POWER FLOW DISTRIBUTION AND HEAT LOAD

<u>use</u>	<u>flow, gpm</u>	<u>heat load, Btu/hr</u>
fuel element storage well cooler (shell side)	240	2.6×10^6
shield cooler (shell side)	70	7.5×10^5
purification cooler (shell side)	260	3.0×10^6
air-conditioning units	70	6.8×10^5
sample coolers (shell sides)	220	2.7×10^6
forced-circulation pumps, liquid coupling coolers and lube oil coolers	110	1.1×10^5

The decay heat cooler (shell-side) requires a flow of 530 gpm to maintain the reactor water at 120 F when the decay heat rate is 5.7×10^6 Btu/hr.

The component coolers operate in parallel and use river water to absorb the heat load from the component cooling water passing through the system. Normally, 2400 gpm of river water cool the component cooling water from 110 F down to 90 F. The loop monitor is installed in a sampling line connected to the main stream piping, and indicates system leakage by recording the activity level of the component cooling water. Any losses in the closed loop component cooling cycle are replaced by demineralized water.

The piping and valves are made of carbon steel. All joints 2-1/2 in. or larger are butt welded and those 2 in. or smaller are socket welded. The piping and valves are designed for 125 psig at 120 F.

5.2.8.2 System Components. The surge tank is a 150-gal carbon steel pressure vessel designed in accordance with the ASME Code for Unfired Pressure Vessels. It is provided with a 2-in. overflow to the retention tanks and a 2-in. vent to the waste gas system.

The two 750-gpm component cooling pumps are single-stage centrifugal pumps designed for 100 psig. The pump and driver are mounted on a common bedplate, and each pump is driven by a 50-hp motor (Type "G", open, drip-proof).

The two component cooling water coolers are of a shell and tube design in accordance with Section VIII of the ASME Code, each with a rated heat duty of 12×10^6 Btu/hr and an effective tube surface of 5600 ft². The shell is carbon steel and the tubes are Arsenic-Admiralty.

Shell and tubes are designed for 100 psig. Relief valves are on both the shell and tube sides to protect against overpressure. A floating tube sheet provides for expansion.

5.2.8.3 Instrumentation and Control. There are local pressure indicators in both cooling pump outlets and one in the common component cooling water return header for the pumps.

Local temperature indicators are installed in the exit cooling water lines from each of the heat exchangers serviced by the component cooling water. Local temperature indicators are also located in the component cooling water lines from each of the component coolers. A thermowell in the manifold line from these coolers is provided for indicator calibration. The temperature of the component cooling water in the cooling water pump discharge manifold is recorded in the control room. The temperature in the component cooling water supply header for the reactor plant is recorded and annunciated (high) in

the control room. A temperature element in the cooling water supply header to the reactor plant equipment regulates the flow of service water to the tube side of the component coolers to assure adequate cooling.

The water level in the surge tank is regulated by a level controller, which operates a pneumatic control valve in the demineralized water supply line to the surge tank. The level is checked locally with the level glass attached to the tank. A level switch on a line from the tank actuates a low level alarm in the control room. A differential pressure indicating switch with a low flow alarm in the control room is installed at the pumps discharge manifold.

In addition, orifice plates are used to measure flow at the following points:

- (1) inlet to shell side of fuel element storage well cooler
- (2) inlet to shell side of shield cooler
- (3) inlet to shell side of purification system cooler
- (4) inlet to shell side of decay heat cooler

A portable differential pressure indicator monitors flow at these locations by connection to upstream and downstream valved-off pressure taps.

The component cooling water circulation pumps are started and tripped by individual manual switches in the control room.

A radiation monitor is installed in a bypass circuit connected in parallel with the main component cooling discharge header from the component coolers. The activity level of the component cooling water is monitored to detect any primary water leakage into the system and recorded on a continuous strip chart in the control room. The radiation element activates a control room alarm upon a high activity level in the component cooling water (Sec. 11.10.3).

5.2.9 Shield Cooling System (Fig. 5.17)

5.2.9.1 General Description. The shield cooling system removes heat transferred to and generated within the vessel external thermal shield and supports and the cavity liners surrounding the vessel. The system includes the various sections of cooling coils in the thermal shield and attached to the vessel supports and cavity liners, a heat exchanger, two circulating pumps, a filter, a surge tank, and the necessary instrumentation.

Two 75-gpm centrifugal pumps circulate the coolant through the heat exchanger and a filter, and into the various coils. The heat absorbed from the shielding and vessel supports by the required 70 gpm flow is transferred to the component cooling water in the shield cooler. The surge tank performs as a head tank and accommodates water expansion or contraction caused by temperature changes in the system. Any make-up water is added to the system through this tank. The shield cooling filter which has a capacity of 75 gpm removes any insoluble material in the closed circuit.

A portable demineralizer can be installed in the system directly behind the filter for use if the coolant exceeds conductivity limits.

The cooling coils are located around the shields and vessel supports. Six 53-ft cooling circuits are in the 119-in.-long, lead-and-steel thermal shield, and a 122-ft cooling coil circuit removes heat generated in the vessel support skirt. Nine 105-ft cooling circuits remove the heat transferred to the walls of the cavity in which the forced-circulation header connections to the vessel are located.

Additional coils in the biological concrete shielding include the coils above the thermal shield and the steam pipe chase coils. These two additional circuits consist, respectively, of nine 98-ft coils and one 262-ft coil.

The support skirt cooling coils are attached to the surface of the skirt. The coils for the lower piping are attached to the ceiling and walls of the cavity recess at the 644-ft level, where the coolant enters. The coils in the thermal shielding are embedded in the lead. The coils above the thermal shield are attached to the steel liner of the pressure vessel and the coil for the concrete surrounding the main steam outlet pipes is embedded in the concrete.

The total heat load to the water cooling circuits is 3.1×10^5 Btu/hr at full power operation. This heat load, which includes gamma heating and the thermal radiation from the vessel and the forced-circulation piping, is distributed as shown in Table 5-2.

TABLE 5-2
COOLING CIRCUIT HEAT REMOVAL

<u>area</u>	<u>heat removed, Btu/hr</u>
thermal shield	5.61×10^4
concrete adjacent to thermal shielding	5.17×10^4
concrete around upper vessel head	3.67×10^5
concrete around recirculation piping cavity	1.56×10^4
vessel support skirt	1.14×10^4

The shield cooling heat exchanger has a capacity of 7.5×10^5 Btu/hr and provides water to the shield cooling circuits at 105 F, a temperature sufficient to remove the heat loads given in Table 5-2.

The maximum temperature of the lead thermal shield and the concrete below is ~ 140 F. The maximum temperature for the concrete outside the steel liner is ~ 150 F. Above the upper vessel head the cooling coils maintain a concrete temperature of < 150 F. The average temperature for concrete surrounded by 70 F air is 105 F. The average concrete temperature between the storage well and the reactor is 130 F, assuming a maximum fuel element storage well water temperature of 120 F. The maximum support surface temperature is ~ 160 F.

Dual, parallel coils ensure reliability of biological shield cooling if a cooling coil should rupture at an inaccessible location.

Supply and return lines within the reactor biological shield (a short portion of each line is just outside the biological shield) are designed for 100 psig at 120 F. This piping is Schedule-40 seamless stainless steel and conforms to ASTM Spec. A312, TP-304. The remaining piping up to the biological shield is of carbon steel.

All joints 2-1/2 in. or larger are butt welded and all joints 2 in. or smaller are socket welded.

Table 5-3 gives data for the individual coils.

TABLE 5-3
COIL DATA

<u>coil</u>	<u>function</u>	$\Delta t, ^\circ\text{F}$	$\Delta P, \text{psi}$	<u>gpm</u>	<u>length, ft</u>	<u>size & mat.</u>
thermal shield cooling coils	to cool the thermal shield	90	5.0	2.4	53	Sch. 40 0.5 in.
recirculation piping cavity coils	to cool the lower head and recirculating piping	11	7.0	2.9	105	Sch. 40 0.5 in.
vessel support skirt coil	to cool the vessel support skirt	21	9.3	3.0	122	Sch. 40 0.5 in.
coils above the thermal shield	to cool the coils above the thermal shield	11	10.3	2.6	98	Sch. 40 0.5 in.
steam pipe chase coil	to cool the shielding about the main steam line	21	8.0	1.9	262	Sch. 40 0.5 in.

The return header from the cooling coils to the shield cooler heat exchanger contains a 2-in. relief valve. This relief valve allows the water to flow around the outlet manifold valve if the water temperature is raised while the water is isolated in the coils. Two 1-in. relief valves protect against overpressure in the shield water cooler, one shell-side and the other tube-side.

5.2.9.2 System Components. The system components are described briefly in the following paragraphs.

The shield cooling heat exchanger is of shell and U-tube design, with 111 ft² of effective tube surface. The maximum pressure drop on shell side and tube side is 8.7 psi.

The two 75-gpm shield cooling pumps are of the centrifugal, heavy duty type, with pump and driver mounted on a common bedplate. The pumps are designed for 100 psig and are driven by two 5-hp 3500-rpm motors of the open drip-proof type.

The shield cooling filter is a "fulflo" type of 75 gpm capacity, designed according to Section VIII of the 1962 ASME Code. The filter material is graded-density cellulose, designed for removing particles 10 μ and larger. The filter design conditions are 150 psig at 120 F. The filter has a clean flow pressure drop of 3 psi.

The 100-gal carbon steel surge tank is equipped with a low level alarm and level indicator, and is provided with a 1-in. overflow to the retention tanks and a 1-in. vent to the waste gas system. Gases in the surge tank are continuously ventilated to the waste gas system to avoid any accumulation of an explosive gas mixture from radiolytic decomposition of shield cooling water.

5.2.9.3 Instrumentation and Control. Local pressure gauges are in each pump outlet, and one is in the common suction header for the pumps. A differential pressure indicator is provided across the filter to warn of excessive crud buildup on the filter elements. A system pressure indicator is provided upstream of the cooling coils.

Temperatures are measured at the inlet and outlet of the shield cooler tube side and transmitted to a multipoint temperature recorder in the control room. There are also three local temperature gauges, at the inlet and outlet of the shield cooler tube-side and at the outlet of the shield cooler shell side. Temperature wells are provided for the cooling coils in the thermal shielding.

The demineralized water level in the surge tank is maintained by a level controller that positions a control valve in the demineralized water supply line. The water height can be visually checked with the level glass attached to the tank. A level switch attached to the tank actuates a low level alarm in the control room.

To ensure adequate coolant flow to the various coils, a flow assembly, consisting of an element, indicator, transmitter, and alarm, monitors the total flow of cooling water from the filter. The flow of coolant through the various coils is monitored by flow indicators provided in each individual coil. The flow indicator switches for the thermal shielding coils operate a common low flow alarm in the control room.

The shield water circulating pumps are started by individual manual switches in the control room. Water flow out of the circuits is controlled by adjusting valves locally in the individual cooling circuits.

5.2.10 Storage Well Cooling System (Fig. 5.18)

The storage well cooling system cools the water used in underwater storage of the spent fuel, control rods, irradiated shrouds, and fuel channel plugs. The system also provides for transfer of demineralized water to and from the overhead storage tank, purification of the storage well water, and drainage of system water to the retention tanks.

5.2.10.1 General Description. The system consists of the storage well and a fluid system that includes pumps, a heat exchanger, filter, piping, valves, and instrumentation. The storage well is designed to store 84 spent fuel elements and 29 control rods. The system heat removal capacity is designed to remove the decay heat of a full core one week after shutdown; this capacity is more than adequate for the decay heat of one-third of a core 12 hr after shutdown. Component cooling water at 90 F is available to remove 2.6×10^6 Btu/hr in the cooler and to keep the storage well water below 120 F. Below this temperature convective currents in the water do not interfere with the view of the storage racks. Water temperatures below 120 F also permit use of the portable demineralizer to remove soluble radioactive material from the storage well water.

One pump draws the water from the storage well through a filter and discharges it through the tube side of a heat exchanger (cooler), from which it flows back to the storage well. Another identical pump serves as standby. Either pump can develop sufficient head to pump water from the storage well to the overhead storage tank. Water is pumped from the well to the overhead storage tank or drained to the retention tanks, depending on water purity and radioactivity level.

Normal water level in the storage well is maintained at el 679 ft - 11 in. An overflow nozzle leading to the retention tanks is at el 680 ft - 2-1/2 in. A valve in the overflow line is closed prior to flooding the storage well. A gutter at el 700 ft - 9-1/2 in. leading to the retention tanks is provided for overflow in the flooded condition.

The filter removes from the system water insoluble impurities caused by corrosion or fuel assembly failure. It removes particles 10μ and larger. This radioactive particulate

matter is collected on removable cartridges, which are disposed of as solid radioactive waste. Temporary shielding can be applied to the filter housing, if required to maintain low local radiation levels. If it becomes necessary also to remove soluble radioactive material, the water can be recirculated through a portable demineralizer.

The storage well, described in Sec. 5.2.11.2, is a stainless-steel-lined cavity in the biological shielding. It will contain enough demineralized water to cover the stored elements such that the dose rate in the normal working areas around the storage well is less than 2.5 mr/hr.

5.2.10.2 System Components. The two centrifugal circulating pumps are of the horizontal single-stage type, with a design pressure of 100 psig. The pump is designed for 260 gpm at a total developed head of 65 ft. The pumps are driven by 7-1/2 hp single-speed induction motors operating at 3500 rpm.

The storage well cooler is a shell and U-tube heat exchanger designed for two passes on the shell side and four passes on the tube side. All surfaces of the heat exchanger that contact the demineralized water are austenitic stainless steel. The tube side pressure drop at full flow conditions is 4.3 psi. Both the shell and tube sides of the cooler are designed for 100 psig and 200 F.

The storage well water filter is of the "fulflo" type and contains removable cartridges of graded density cellulose in a stainless steel tube. The filter housing is a stainless-steel vessel made according to ASME Code, Section VIII. The pressure drop at full flow conditions is 3 psi when the filter is clean. Design pressure and temperature are 150 psig and 150 F.

The storage well piping system is of AISI Type-304 seamless stainless steel and includes the complete recirculation circuit, interconnecting piping with the overhead storage tank, piping connections for the portable demineralizer, and the various vents and drains. Design pressure and temperature for this piping are 100 psig and 120 F. All piping joints 2-1/2 in. or larger are butt welded; joints 2 in. or smaller are socket welded.

Isolation valves for each major system component permit maintenance without complete system drainage. Bypass valves at the filter and cooler permit water circulation with either of these units out of service.

5.2.10.3 Instrumentation and Control. A differential pressure indicator across the filter indicates filter fouling. Local pressure gauges are installed in the discharge lines from each pump. A low level alarm in the control room warns against excessive loss of water in the fuel storage well.

Water losses from evaporation, system leakage, and deliberate drainage are made up from the overhead storage tank by manual operation of the makeup line valves.

The flowrate of the clean water returning to the storage well is measured downstream of the cooler and indicated in the control room.

Local temperature indicators for measuring storage well water loop temperature are provided upstream and downstream of the cooler. A temperature recorder in the control room records these temperatures. Another local indicator is provided to measure the temperature of the component cooling water leaving the well cooler.

The pump motors can be started locally or from the control room by individual manual switches.

5.2.11 Fuel and Control Rod Handling System

5.2.11.1. General Description. The primary functions of the system are to load, unload, transfer, and store new and used fuel assemblies and control rods. In addition, the system is used to install and remove the fuel shroud cans and the spare fuel channel grid plate plugs. Figure 5.19 shows the fuel transfer system elevation, and Fig. 5.20 shows the fuel transfer system plan.

The failed-fuel location tube bundle must be removed in order to gain access to the fuel (see Fig. 5.21). The tube bundle assembly is removed from the reactor by first unscrewing captive bolts and disconnecting the failed-fuel element location system piping. The building crane is then positioned above the reactor, and a hookup using the crane hook is made to the three lifting lugs on the top flange of the failed-fuel location assembly. The assembly is lifted clear of the reactor cavity to a position above the main floor elevation, positioned over the fuel storage well, and lowered into the failed-fuel location assembly support rack located at the bottom of the well. The total radiation dose received during transfer is less than 100 mrem.

All handling operations of irradiated fuel, control-rods, shrouds, and spare fuel channel plugs are performed under water. For these operations, the reactor top shield plugs, the vessel head, the canal shield plug, and the canal gate are removed and the storage well, the canal, and the upper vessel cavity are then filled by opening valves in the pipe from the overhead storage tank to the well. Spent fuel and irradiated shrouds and plugs are transferred through the canal from the reactor with a minimum water cover of 10 ft. Used control rods are transferred with the poison section under a minimum water cover of 7 ft - 8 in. The dose rates during transfer are given in Sec. 10.2.2.2. The rods are tilted while traversing through the canal so that the required water cover is maintained. On completion

of the transfer operation, the excess water can be drained to the retention tanks in batches or pumped to the overhead storage tank using the alternate pump of the storage well cooling system. This operation is accomplished without interrupting the circulation and cooling of the well water.

To remove spent fuel from the reactor, the canal gate retaining bolts are backed away from the gate to release the gasket seal and permit withdrawal of the gate from the transfer canal. The building crane removes the gate and stores it in the cavity provided in the floor east of, and adjacent to, the spent fuel storage well. The steel-lined concrete canal shield plug is removed with the auxiliary hoist of the crane and stored in the same cavity as the canal gate. The fuel assembly grapple and grapple actuator are attached to the fuel transfer bridge hoist, and the bridge is positioned over the selected fuel position in the reactor core. The grapple is lowered and the operator makes final alignment of the grapple with the appropriate fuel shroud can by twisting the hoist cable.

Red signal lights on the control pendant indicate whether the grapple is in the engaged or released position. The fuel assembly is "grappled" and raised to a position where it clears the bottom of the transfer canal. The bridge carries the assembly to the selected position above the spent storage racks in the spent fuel storage well and lowers it into position. The grapple releases the load, and is raised to the stored position on the bridge hoist. The transfer of the spent fuel from the reactor is shown schematically in Fig. 5.22.

Used control rods are moved from the reactor to the rod rack in the spent fuel storage well similar to the way that the fuel assembly is transferred. While the rod is carried through the canal, it is tilted, as in Fig. 5.23, to maintain the minimum water cover. In some cases, a horizontal grid bar must be removed to obtain access to the control rod. In this case, the four captive bolts holding the grid bar to the center posts are backed out with a special remote tool. The tool is then up-ended and is screwed into the center of the grid bar for removal.

For the installation of new fuel assemblies (Fig. 5.24), the fuel transfer bridge is positioned above the new fuel storage rack, and the grapple is lowered and aligned with a new fuel assembly. When engagement is complete and the fuel assembly is grappled, the load is raised, carried to a position over the reactor core, and lowered into the fuel shroud.

To install a new control rod, the rod is lifted from the shipping container by the auxiliary hoist of the building crane using a special hook. The control rod is transported to a service area on the main floor, cleaned, and removed to the transfer canal where it is placed in a portable service fixture, shown in Fig. 5.25. The portable service fixture maintains the control rod in an upright position for the purpose of disengaging the special hook and permitting engagement with the control rod grapple attached to the fuel transfer bridge hoist. The holding fixture is designed to receive or release the vertically held control rod using

lateral travel. The side loading feature of the fixture reduces the head room requirements when the control rod is handled with the bridge hoist. The mode of control and the operations required when handling the control rod with the grapple and transfer bridge are identical with those employed when installing new fuel elements. During the sequence, signal lights on the control pendant indicate whether the grapple is in engaged or released position.

To handle the fuel channel plugs, the fuel assembly grapple and actuator are installed on the bridge hoist assembly. Control of the loading or removal sequence between the spare channels of the reactor core and the spent fuel storage racks is the same as that employed in handling fuel assemblies and control rods.

The fuel assembly shrouds are handled using the building crane and the shroud installation and removal tool. There is no special rack where new shrouds are stored. Used shrouds are stored in the spent fuel storage well; no special rack is provided for storage.

Spent fuel and control rods are removed from the storage well in a lead shielded shipping cask. The shipping cask is brought into the containment building through the personnel airlock. The building crane is used to place the cask onto the bottom center of the storage well. The top of the cask is removed and the fuel and control rods are inserted using the bridge hoist. During this operation, the water level of the well is raised so that a minimum water coverage of 10 ft is provided between a moved component and the top of the well. After the cask has been loaded, it is covered and removed from the storage well using the building crane. The cask is removed from the containment building through the personnel airlock.

5.2.11.2 Storage Well. The spent fuel storage well is a cavity in the concrete biological shield surrounding the reactor vessel. The cavity has a square cross section and is connected to the reactor cavity via the transfer canal. The well is lined with 16 gauge AISI Type-316 stainless steel.

Normally, the water level is 20 ft - 5-5/8 in. above the bottom elevation of 659 ft - 5-3/8 in., with the well containing approximately 18,500 gal. The water level is sufficient to reduce the dose level from 72 fuel assemblies, cooled for 24 hr, to 1 mr/hr at the surface of the water. The temperature of the water is kept below 120 F by the storage well cooling system (Secs. 5.2.10 and 11.12). An overflow nozzle at el 680 ft - 2-1/2 in. controls the normal water level, while an overflow gutter at el 700 ft - 9-1/2 in. controls the water level when the well is flooded.

A corrugated aluminum cover is provided to cover the pool when access is not required. This cover is in three separate sections.

5.2.11.3 Fuel Transfer Canal and Gate. The fuel transfer canal is a 20 ft - 0 in. deep, stainless-steel lined cavity in the biological shield. The canal provides an underwater access between the reactor cavity and the spent fuel storage well. During reactor operations, a concrete shield plug fills the canal cavity. During refueling operations, the concrete shield plug is stored in a specially provided cavity east of the spent fuel well.

A watertight canal gate at the spent fuel storage well (Fig. 5.26) isolates the reactor cavity and the well. The gate, when installed and sealed, permits the water level in the spent fuel storage well to be raised above the level of the bottom of the canal, so that additional shielding is provided when radioactive components are handled in the pool while the reactor is in operation. In the event of minor leakage at the canal gate, a drain in the canal floor drains the water off before it can enter the reactor cavity.

The canal gate is a removable, gasketed, aluminum plate which fits into a stainless steel casing anchored into the wall of the spent fuel storage well at the south end of the fuel transfer canal. Leak-tightness is obtained by tightening bolts in the stainless-steel casing.

5.2.11.4 Underwater Lights. Two underwater lights are installed in the reactor cavity and two in the spent fuel storage well. The brackets which support the lights extend from a retracted stored position to any position required to obtain greatest visibility. The brackets remain in the reactor cavity and spent fuel storage well at all times. When not in use, the removable lights are stored.

The lights are designed for operation under water and should be turned on only when completely submerged.

5.2.11.5 Fuel Transfer Bridge and Drive. The fuel transfer bridge is a specially designed structure driven by an electric motor north and south on rails recessed in the floor of the containment building at el 701 ft - 0 in. It travels at speeds between 2.6 and 18.2 ft/min. The bridge assembly is shown in Fig. 5.27.

The bridge traverses over the areas where service operations are performed at the reactor cavity, transfer canal, spent fuel storage pool, and the new fuel storage racks. The bridge serves as the structural support for the fuel transfer hoist and the fuel and control-rod transfer tools.

In conjunction with the hoist and the fuel and control-rod transfer tools, the bridge performs the following service functions:

- (1) transfers new fuel assemblies from the shipping container to the new fuel storage racks;

(2) moves fuel assemblies from stations in the fuel storage racks to operating positions in the reactor during initial fueling of the reactor;

(3) removes spent fuel assemblies from the reactor and transfers and positions them in the spent fuel storage racks;

(4) transfers spent fuel assemblies from the spent fuel storage racks to the shipping cask;

(5) moves the control rods from their storage positions in the storage racks to their operating position in the reactor during initial assembly of the reactor;

(6) removes used control rods from the reactor and places them in the storage well racks; and

(7) transfers used control rods from the storage well racks to the shipping cask.

The bridge drive consists of a 1/4 hp variable speed gearmotor and roller chain arrangement. An SCR power supply is provided.

A 115-v, single-phase, 60-cycle power supply is furnished to the bridge by a three-conductor cable and reel. The cable is connected by a locking plug to the power supply receptacle located on the south wall of the containment building. The reel is mounted on the bridge and unwinds and winds the cable as the bridge travels back and forth during refueling operations. The cable is wired from the reel to the bridge motor controller, for bridge travel, and to a feedrail mounted on the hoist monorail. A collecting shoe mounted on the hoist provides power for operating the hoist motor.

The bridge is equipped with a monorail which permits east and west movement of the bridge hoist.

Two adjustable 300 w flood lamps on the monorail vertical supports provide concentrated light at the bridge area.

5.2.11.6 Fuel Transfer Bridge Hoist. The hoist for the transfer of fuel and control rods traverses on the monorail of the fuel transfer bridge. The operator manually positions the trolley of the hoist on the monorail. Manual traverse of the hoist trolley, in conjunction with power drive traverse of the fuel transfer bridge, positions the hoist directly over any loading or unloading station in the transfer system.

The hoist was specially designed for this application. It consists of a 3/4 hp, variable speed gearmotor, coupled through a torque limiter to a grooved stainless-steel drum. The hoist cable is 7/32 in., 7 x 19 stainless steel with a 50 ft lift, and it is provided with a special connector to couple the grapple actuator.

The hoist gearmotor is SCR-driven with a hoisting speed range that is continuously variable from 8 in./min to 23 ft/min.

The trolley of the hoist is carried by the monorail of the transfer bridge. The trolley is manually moved along the monorail by means of a hand chain.

A reel is mounted on the frame of the hoist to pay out a 5-conductor electric cable for operation of the grapple actuator.

The hoist is remotely controlled by a pendant suspended from the hoist frame. The following units are mounted on the pendant:

- (1) On-Off switch for power to hoist
- (2) Momentary contact pushbuttons for Up and Down hoist travel
- (3) Potentiometer for control of hoist speed
- (4) 4-position key switch for
 - (a) Off
 - (b) New Component Handling (signal light present)
 - (c) Fuel Handling (signal light)
 - (d) Control Rod Handling (signal light)
- (5) 2-position selector switch for
 - (a) Grapple (signal light)
 - (b) Release (signal light)

A rotary limit switch actuator automatically stops the hoist at the upper and lower limits of cable travel. The rotary actuator automatically stops upward travel of the hoist at elevations where adequate heights of water shield both the fuel handling and control rod handling operations.

5.2.11.7 Grapple Actuator Assembly. An electrically operated grapple tool actuator (Fig. 5.28) operates the fuel assembly grapple and the control rod grapple. The grapple actuator is a cylindrical, watertight, welded stainless-steel housing, attached to the free end of the bridge hoist cable. Within the housing, a miniature electrically powered linear actuator is installed. The housing is designed to be quickly and easily attached and detached from either the fuel grapple assembly or the control-rod grapple assembly by two self-locking, quick-release pins. A third, smaller quick-release pin connects the output shaft of the linear actuator to the pivoted rocker arm of the fuel grapple assembly or the actuator sleeve of the control rod grapple assembly. The linear actuator electrical leads

are brought into the grapple actuator housing through a watertight compression fitting. An electric cable reel on the bridge hoist winds, unwinds, and stores the cable when the grapple actuator is raised or lowered by the hoist.

When the GRAPPLE position of the selector switch on the hoist pendant is pressed, the linear actuator extends 1/4 in. to move the locking fingers of the fuel grapple or control rod grapple to the HOLD position. At the end of the stroke, limit switches interrupt power to the actuator motor, and a red light on the hoist pendant indicates GRAPPLED. In like manner, when the selector switch is turned to RELEASE, the linear actuator retracts, releasing the fuel assembly or control rod. At the end of the stroke, limit switches interrupt power to the actuator motor, and a red light on the pendant indicates RELEASED.

The linear actuator motor circuit includes a dynamic braking feature. When power is interrupted to the motor, the brush leads are shorted together, and the motor acts as a generator, sending current around a closed circuit. The rotating armature energy is converted into electrical energy, quickly dissipated as heat, thus eliminating most of the armature coast.

5.2.11.8 Fuel Grapple Assembly. The fuel grapple assembly (Fig. 5.29) consists of a welded stainless-steel housing containing a pivoting rocker arm mounted upon a horizontal shaft. When the fuel grapple assembly is attached to the grapple actuator, this rocker arm is driven by the linear actuator. Two tapered slots, opposite each other in the lower rectangular skirt of the housing, guide the adapter assembly into proper position on top of the fuel assembly. These slots straddle the handle of the fuel assembly when the grapple bottoms during a lowering operation to pick up fuel. In this position, the pivoting rocker arm may be driven by the linear actuator below and under the fuel assembly handle. A projection on the lifting surface of the rocker arm locks the rocker arm to the fuel assembly handle as an additional safeguard during fuel hoisting. The design of the grapple prevents release of the fuel from the grapple by any means other than electrical operation of the grapple actuator.

5.2.11.9 Control Rod Grapple Assembly. The control rod grapple assembly (Fig. 5.30) consists of a relatively long hollow housing containing two sleeves or plates. The inner, or center, sleeve is constrained within a slot machined down the center of the outer sleeve; it is free to move axially with respect to the outer sleeve and the housing. The outer sleeve is fixed to the housing. The whole assembly constitutes a cruciform-shaped cross section designed to enter the similarly shaped apertures between the fuel shroud cans in which the control rods are installed within the reactor.

The heavy outer sleeve and the assembly housing are attached at their upper extremities to the grapple actuator by the quick-release pin. The center sleeve is connected by the smaller quick-release pin to the spring-loaded clevis in the grapple actuator, which is driven by the linear actuator.

A recessed pocket near the lower end of the outer sleeve accommodates two identical, opposed, notched, pivoting pickup fingers, which, in their open position, clear the control-rod handle. In their closed position, their notched faces abut one another and capture the control-rod handle. A pin, installed at the centerline of the fingers pivot point, keeps the fingers separated and in proper position. During hoisting, all resisting forces of the control-rod handle are transmitted through the lifting fingers to the heavy outer sleeve and thence to the hoist cable.

The lifting fingers are positioned into open or closed attitudes by linear movement of the center sleeve which, by means of circular ears projecting into recesses machined into the lifting fingers, translates the axial movement of the sleeve into radial movement of the lifting fingers. During a control-rod grapple operation, the opposing faces of the pickup fingers contact each other to ensure the best grip on the control rod handle. A limit switch, which stops the motor of the linear actuator, is adjusted to stop the motor just after the pickup fingers have been driven into contact with each other. The motor can then compress the clevis spring and stop.

5.2.11.10 New Fuel Storage Racks. Three new fuel storage racks (Figs. 5.31 and 5.32) are located in pits in the biological shield. They are designed for an inventory of 84 new fuel assemblies (28 assemblies per rack). The covers for the racks are flush with the floor at el 701 ft - 0 in. and are gasketed. Locks are mounted in the covers to prevent unauthorized access to the fuel.

The stored fuel arrangement ensures that criticality cannot occur. Drains below the cover gaskets prevent leakage into the storage pit past the gasket. As a further safeguard, drains have been permanently installed in the bottom of each pit to prevent inadvertent flooding of the storage cavity. However, even if the storage cavity is flooded, subcriticality is maintained. The k_{eff} when the racks are filled with new fuel of the initial core design and under water is 0.903.

Each storage position for each new fuel assembly is identified numerically to assist in fuel inventory control.

5.2.11.11 Spent Fuel and Control Rod Storage Racks. Spent fuel storage racks have been designed to store 84 spent fuel assemblies underwater at the bottom of the spent fuel storage pool. There are a total of six independent racks. These racks, fabricated from AISI Type-304 stainless steel, are located around the periphery of the well. Stainless-steel plates, 1/4 in. thick, containing 2 percent boron by weight, isolate the stored fuel assemblies from each other. This material extends the whole length of the active fuel. The minimum water distance between spent fuel racks is 13 in. Stainless-steel racks are also

provided for the storage of 30 control rods. These racks are located in the four corners of the well bottom. The arrangement of the storage racks within the pool is shown in Fig. 5.33. The k_{eff} for the spent fuel storage racks, completely filled with spent fuel of initial core design and under water, is 0.82.

5.2.11.12 Failed-Fuel Location Tube Bundle Storage Rack. The failed-fuel location storage rack shown in Fig. 5.34 is a stainless-steel structure designed to rest on the floor of the spent fuel storage well. The rack is installed in, or removed from, the storage well with the building crane auxiliary hoist. The rack is furnished with lifting eyes for easy rigging.

During refueling operation, the rack is positioned in the center of the storage pool floor. Except when fuel is being transferred from the storage racks to the shipping cask, the rack is stored in the pool.

5.2.11.13 Fuel Assembly Shroud Installation and Removal Tool. The fuel assembly shroud installation and removal tool, shown in Fig. 5.35, is designed to perform two functions:

- (1) to install or remove a fuel assembly shroud, and
- (2) to lock or unlock the fuel assembly shroud to the core grid plate.

The tool is manually operated from the fuel transfer bridge.

The tool consists of two concentric tubes with the outer tube serving as the housing. The inner tube transmits torque to the linkage, which engages the lifting holes at the top of the shroud assembly, and to the locking ring at the bottom of the shroud assembly.

For convenience in handling, the tool is made in two parts, an upper and a lower assembly. The two assemblies are coupled together by a pipe union fitting.

In assembling the tool, the lower part is supported in the control rod receiving fixture (Fig. 5.25). The upper part is then transported by the building crane to a position where the union fitting can be assembled. When the halves of the union are screwed together, the spliced tool is ready for installing or removing shroud assemblies.

Actuation of the tool is manual. A hand crank at the top of the tool is rotated between two stops to lock or unlock a shroud assembly. Rotation of the crank from one stop to the other simultaneously actuates the fingers which engage the lift holes of the shroud and actuates the locking ring at the bottom of the shroud.

5.2.12 Hydraulic Valve Accumulator System (Fig. 5.36)

5.2.12.1 General Description. The hydraulic valve accumulator system is an automatically controlled system that supplies pressurized water to operate piston-type valve actuators, which operate rotary cone-type valves (rotovalves) in the forced-circulation system and the main steam system. A system water accumulator normally containing ~300 gal of water is air-pressurized to 150 psig. Air pressure above the water is maintained automatically by an air compressor. There are two air compressors, one normally on standby.

A 4-in. supply header from the accumulator supplies demineralized water to the individual valve actuators through 2-in. supply lines. The water is returned via a 4-in. return header to the 300-gal water return sump. The water level in the accumulator is maintained automatically by a pumping system and appropriate controls. Two hydraulic pumps (one on standby) pump water from the water return sump to the accumulator.

The five rotovalves served by this system are in the suction and discharge of each of the forced-circulation pumps and in the main steam line to the turbine. All these valves are inside the reactor building. If power is lost to the hydraulic pump and compressor motors, the system capacity is sufficient to allow operation of all rotovalve actuators for three complete piston strokes. Administrative control will not permit continued plant operation if the hydraulic system is inoperable (Sec. 11.1.5.7). All the major components of the system are mounted on a common base, forming one package.

The water piping is of 1-1/4 in. Type L hard copper tubing and is bolted to the package plate. The pump takes suction from the sump and discharges into a header, from which the water is routed to the accumulator or returned to the sump through gate valves in the individual lines. Both pumps discharge into the same header. Backflow of water to the standby pump is prevented by check valves.

Each pump has its own suction line with an isolation gate valve and strainer. There is a 1-1/2 in. relief valve in the discharge line of each pump, upstream of the check valve, which discharges to the sump.

The compressed air piping consists of 3/8-in. Type L hard copper tubing. Each compressor has a discharge line containing a brass check valve, an isolation globe valve, a globe type drain valve from the bottom of the vertical leg, a 3/8-in. relief valve, and a union for easy disassembly. The two discharge lines are combined into a common line discharging into the accumulator. The common line contains an isolation globe valve.

The 4-in. water supply header from the water accumulator, the 2-in. piping to and from the valve actuators, and the 4-in. return header are of seamless carbon steel piping per ASTM A106. The piping joints are socket-welded in the 2-in. lines and are butt-welded in the 4-in. lines.

5.2.12.2 Accumulator. The accumulator is a horizontal steel pressure vessel designed, fabricated, and tested according to the ASME Boiler Code for Unfired Pressure Vessels; the tank is spot X-rayed according to paragraph UW52 of the ASME Code. The steel used for the tank is per ASTM A285, Grade C.

The tank is 88-5/8 in. long and 48 in. in OD. It has an 11-in. x 15-in. manway for access to the inside of the tank. The tank is supported on a saddle which is part of the support structure for the accumulator package. The centerline of the tank is 66 in. above the base line of the package.

Pressure data for the tank are:

(1) working pressure	150 psig
(2) design pressure	165 psig
(3) hydraulic test pressure	248 psig

5.2.12.3 Air Compressors. There are two motor-driven Ingersoll Rand Type-30, Model-232, two-stage air compressors. The drivers are induction-type, drip-proof motors, rated at 1 hp. Voltage rating is 440-v, 3-phase, 60 cycle. The compressors are designed for continuous duty with a 60 C rise. They are mounted on the common bedplate of the hydraulic accumulator package.

The air compressor has a maximum discharge pressure of 200 psig and a piston displacement of 4.5 cfm. The unit is air cooled and has a centrifugal unloader with outside exhaust pilot valves and automatic start and stop controls. An intercooler cools the air between stages. A belt wheel fan passes air over the finned surfaces of the intercooler to dissipate the heat from the air. Safety valves are provided in the intercooler and in the compressor discharge.

5.2.12.4 Hydraulic Pumps. There are two hydraulic pumps, one on standby. The pumps are turbine-type, vertically-split, horizontal motor-driven centrifugal pumps made by the Kennedy Pump Co. Pump capacity is 15 gpm at a 150 psi differential head. They are driven by an induction-type drip-proof electric motor rated at 7.5 hp. The motors use 440-v 3-phase 60 cycle power.

The pump shafts are supported on both ends of the pump by ball bearings, which are lubricated from a grease cup. The coupling is of the flexible-type.

5.2.12.5 Water Return Sump. The water return sump is an atmospheric tank that forms a part of the accumulator support structure. It is made of carbon steel per ASTM A-283-A and has a capacity of ~300 gal. The tank has a 17-in. x 11-in. manway for cleaning access.

The 4-in. water return header enters the tank 12 in. above the bottom line of the accumulator package. The nozzle is a 4-in. steel pipe with a 150-lb slipon weld flange. The tank bottom is sloped to promote drainage.

5.2.12.6 Instrumentation and Controls. There are three Type-DA-33 range-8 120-v flanged-case Mercoid pressure switches mounted on the accumulator tank. One of these switches is used for automatic compressor control, one for high pressure alarm, and one for low pressure alarm. The switch settings are adjustable; abnormal pressure is annunciated in the control room (see Table 11-1).

A pressure gauge of 0 - 300 psi range is provided to measure accumulator pressure.

There are four motor starters, for the two compressors and the two pumps. These are Allis-Chalmers CL 5756 size-1 starters, for 400 line-volts and 110 coil-volts. The control circuitry uses 110-v power, obtained through 150-v transformer.

The motors have selector switches for "manual-off-auto" operation. When the compressors are in Auto a pressure switch is used for on-off control, and the compressors cannot operate unless the hydraulic pumps are shut off. With the hydraulic pumps in Auto, on-off control of the pumps is by the upper float switch in the accumulator.

Under automatic control both pumps start or stop together since the switches are in the feeder common to both pumps. The compressors operate in the same manner. To use one compressor and one pump during operation the selector switch for the standby pump and compressor must be turned to Off.

Two float switches permit automatic level control in the accumulator. The lower switch is a Magnetrol (Model TF-63) with a 2-1/2-in.-dia float, a 6-in. float stem, and two DPST switches. The upper float switch is a Magnetrol (Model TF-63x) tandem unit with 6-in.-long float stem and two DPS-1 switches (DPST) with a maximum differential of 4 in.

Automatic accumulator control is by the upper float switch. This float switch has four sets of contacts. One set of contacts is used for automatic pump control, one set for automatic compressor control, and one set for a high level alarm. The contacts for the automatic compressor and pump control are ganged so compressors and pumps cannot operate simultaneously. The lower float switch also has four sets of contacts. One set provides a low level alarm with the remaining contacts as spares. Abnormal accumulator water level is annunciated in the control room (Table 11-1).

The accumulator is equipped with a gauge glass for local level indication. Normal water level is at the half-full mark. The sump tank has a float switch identical to the lower float switch of the accumulator. This float switch energizes a remote low level alarm for low sump water level (Table 11-1). The sump has a gauge glass for local level indication.

5.2.13 Sampling System

The sampling system provides means to sample various reactor and generator plant auxiliary systems. The sampling program is designed to supplement plant operating instrumentation.

Routine samplings of the reactor and generator plant auxiliary systems enable the plant chemist to:

- (1) Maintain chemical control within specified limits, and to control water chemistry to achieve optimum water quality for protection of plant equipment.
- (2) Establish normal water chemistry and radiochemistry data for use in the detection and interpretation of abnormal conditions.
- (3) Calibrate and check station operating instruments.

In addition to the normal control procedures, such as those for pH, conductivity, and chloride determinations, which are used to determine if chemical control is as designed, the following chemical tests provide an operational history of the plant on which to base the detection and interpretation of abnormal conditions:

- (1) Total Iron and Copper - The level of copper in the reactor coolant, feedwater, and full flow condensate systems is determined to establish a material balance and to detect abnormal corrosion in system components.
- (2) Dissolved Oxygen - The dissolved oxygen concentration in the reactor coolant, feedwater, and full flow condensate systems is determined to provide corrosion information and information to assist in the detection of air leakage.
- (3) Total Solids - The total solids concentration of the reactor coolant, feedwater, and full flow condensate systems is determined for considerations of overall plant corrosion behavior and for the detection of condenser leakage.
- (4) Silica - The silica concentration of the feedwater, main steam, and the virgin and condensate storage tanks' water is determined to establish a material balance, to detect condenser leakage, and to ensure proper quality of makeup water.
- (5) Radionuclide Analysis - Identify the radionuclides and measure the quantity to establish the concentration and types of corrosion-erosion products present in the primary coolant.

The frequency of collection from specific sampling points, for various chemical and radio-chemical tests, is given in the LACBWR Operating Manual. Sample connections consist of small-diameter (1-in. or less) piping and globe valves of stainless steel or carbon steel composition (depending on sample location) and terminate at convenient working locations. Sample coolers are provided for samples drawn from high temperature regions of the plant system. Table 5-4 lists the sample connections in the reactor and the generator plant auxiliary systems.

TABLE 5-4
SAMPLE CONNECTIONS IN AUXILIARY SYSTEMS

<u>reactor plant auxiliary systems</u>	<u>location</u>
(1) primary purification system (Fig. 5.8)	two points: influent and effluent streams
(2) overhead storage tank (Fig. 5.18)	one point: drain line upstream of drain valve
(3) boron injection system (Fig. 5.14)	two points: effluent from sodium pentaborate tank, and injection line upstream of initial control valve
(4) shield cooling system (Fig. 5.17)	one point: shield cooling pump discharge header upstream of the filter
(5) fuel storage well (Fig. 5.18)	two points: upstream of the filter, and return line upstream of the storage well
(6) component cooling system (Fig. 5.16)	one point: component cooling pump discharge header
(7) drain and liquid waste disposal (Fig. 5.37)	nine points: bottom of the two retention tanks, two waste storage tanks, evaporator feed tank, spent resin tank, water collection tank, concentrated waste storage tank, and the waste evaporator

Table 5-4 - Sample Connections in Auxiliary Systems (cont'd)

<u>reactor plant auxiliary systems</u>	<u>location</u>
(8) vent and waste gas disposal (Fig. 5.38)	four points: bottom (for condensed moisture) and top (for gas) of the two gas storage tanks
(9) shutdown condenser (Fig. 5.6)	two points: shell side coolant and condensate return
(10) main steam and feedwater (Fig. 5.1)	two points: feedwater downstream of final valve in feedwater line; steam sample from a throttling calorimeter off the 10-in. main steam line
<u>generator plant auxiliary systems</u>	<u>location</u>
(1) condensate demineralization (Fig. 5.7)	four points: one influent sample from the condensate pump discharge header, and effluent samples from each of the three demineralizers
(2) demineralized makeup and well water	five points: deep well water supply to cation exchanger, effluent from cation exchanger, effluent from anion exchanger, and sample points for the 30,000-gal virgin water and 20,000-gal condensate storage tanks

5.2.14 Demineralized Makeup and Well Water System

5.2.14.1 General Description. Two 120-gpm deep well pumps (one on standby) are used to pump water at 40 gpm to a demineralization system.

The well water is treated in a three-step demineralization process:

- (1) Cationic constituents (such as iron, calcium, and magnesium) are removed in a cation exchanger
- (2) CO₂ is removed in a decarbonator

(3) Anion constituents (such as chlorides, nitrates, sulphates, silica, and organic matter) are removed in an anion exchanger.

The well water to the cation exchanger is first routed through a strainer, which removes insoluble impurities > 0.005 in. From the cation exchanger the water is routed to the top of the decarbonator tower. The water flows over the Styron packing, counter to air flow introduced in the bottom of the tower by a blower. Exhaust gases containing the CO_2 leave the tower top through the 8-in. exhaust duct. A water storage section below the tower receives the degassed water. The demineralized water booster pump draws water from the storage section of the decarbonator and discharges into the anion exchanger. The demineralized water from this exchanger is pumped to the demineralized water storage tank on the turbine building roof. Resin retention screens on the underdrain system of the anion and cation exchangers retain resin fines. The demineralized water storage tank has two compartments. The top compartment contains the virgin demineralized water, the bottom compartment the condensate from the main condenser hotwell. The virgin water is not contaminated by radioactive condensate.

A resin regeneration system regenerates resins after depletion. Sulphuric acid is the regenerant for the cation exchanger, caustic for the anion exchanger. An 8000-gal acid storage tank, equipped with a vertical acid transfer pump, is used to store and transfer acid for cation resin regeneration. A smaller acid-measure-tank and acid pump are near the cation exchanger for immediate service. Similarly, a caustic measure tank and caustic pump are near the anion exchanger for service. Resins are automatically regenerated after initiation of a regeneration cycle.

A heater in the caustic pump discharge line to the anion exchanger heats the caustic before the regeneration process, promoting silica removal from resin during regeneration, thus increasing the exchange capacity of the regenerated resin.

5.2.14.2 Major System Components. The cation exchange demineralizer is complete with tank, water, distribution pipes, underdrain system, retention screen, resins, and appropriate piping and instrumentation. The tank is 36 in. in diameter, 78 in. high, rubber lined, and has a 16-in. diameter manhole. Design is to ASME specifications. The resin is Amberlite IR-120A, a strongly acidic sulphonic-type resin with a high exchange capacity ($\sim 30 \text{ kg/ft}^3$), unusual stability at high temperature, and good chemical resistance over a wide pH range. The loaded weight of the cation exchanger is 5300 lb. It is supported on three legs. The retention screen is made by Johnson Well Screen and has a slot spacing of 0.010 in.

The decarbonator consists of a tower and a storage section. The tower is 18 in. in diameter and the storage section is 42 in. in diameter. The tower is packed with MASPAC FN-90 (15 ft^3) Styron.

The anion exchange demineralizer is complete with tank, water distribution pipes, under-drain system, resin retention screen, resins, and appropriate piping and instrumentation. The anion resin is Amberlite IRA-402 resin (13 ft³), a strongly basic, high porosity, quaternary resin with a high exchange capacity and a good capacity for removal of organic compounds of high molecular weight. It can also remove silica. The tank is a vessel 30 in. in diameter and 60 in. high. The loaded weight of the demineralizer is 3700 lb.

The demineralized water storage tank is 20 ft in diameter and 23 ft high. It is made in accordance with the American Petroleum Institute Spec. 12G for welded aluminum alloy storage tanks. The tank is in two compartments, one above the other. The top compartment contains virgin demineralized water received from the well water demineralizers. The capacity of this compartment is ~30,000 gal. The bottom of the compartment is dished. The bottom compartment will contain any condensate that is radioactively contaminated. The capacity of this compartment is 19,000 gal. The cylindrical portion of the tank is insulated by 3 in. of insulation and a 5/8-in. insulation panel. The tank is equipped with four immersion type tank heaters (Chromalox Type MTS, Cat. 150 MTS-220) with stainless steel sheaths. Each has a 2-kw capacity for single-phase 60-cycle operation on 277 v.

Type-1 PVC piping is used generally throughout the system for unprocessed water, partially processed water, demineralized water, and acid and caustic. However, piping between the caustic measure tank outlet and the caustic pump, and between the caustic pump discharge line and the control valve inlet at the caustic heater inlet is 1/2-in. Sched. 40 black iron piping per ASTM-120.

A total of 19 air-operated on-off type control valves are used for automatic regeneration. These are Grinnell-Saunders valves of various models, sizes and materials.

5.3 RADIOACTIVE WASTE DISPOSAL SYSTEMS

This section supersedes Sec. 10 of ACNP-62574. However, the presentation and discussion of Tables 10-1, 10-2, and 10-3 of ACNP-62574 remain valid and are not repeated here. Table 10-1 lists isotopes and activities resulting from primary system cladding corrosion and erosion. Table 10-2 lists isotopes and activities resulting from uranium contamination on fuel rod surfaces and from cladding defects. Short-lived isotopes ($T_{1/2} < 1$ day) were not included in Tables 10-1 and 10-2 since, after holdup in the drain and liquid waste disposal system tanks, these isotopes are present only in very small quantities. Table 10-3 lists the gaseous isotopes and the stack release rates that will result if there are defects in rods producing a total power of 1 Mw. The production of the radioactive gases N-16, N-17 and O-19 is discussed in Sec. 10.1.3 of ACNP-62574. However, these isotopes have very short half-lives and would decay in the vent and waste gas disposal system before release to the stack.

5.3.1 Drain and Liquid Waste Disposal System (Fig. 5.37)

The drain and liquid waste disposal system is designed to:

- (1) collect and store radioactive liquid waste and resins generated in the plant during normal power operation, and
- (2) dispose of the collected waste safely and economically.

5.3.1.1 General Description. The process, floor, and overflow drainage is collected in two 6000-gal retention tanks in the containment building, or in two waste water storage tanks (3000 and 4500 gal) in the tunnel between the containment building and the turbine building.

Within the containment building, liquid waste accumulates from the following: primary system leakage, drainage from equipment during maintenance, washdown of building floors, in-place decontamination of equipment, personnel decontamination, sluice water from primary purification system resins, and building spray system operation in the event of a major accident. Turbine building liquid waste accumulates from turbine condensate leakage, hot change room and laboratory drains, excess water from full-flow demineralizer resin regeneration, and sluice water from full-flow demineralization system resins.

The lavatory and shower drains in the hot change rooms of the waste treatment and turbine buildings are piped to a 500-gal intermediate holdup tank. An alarm is annunciated when the tank contains approximately 200 gal or more. The contents can then be sampled and can be pumped: to the river through the main condenser circulating water line; to the turbine building sewer system; or into the other tanks or into the treatment facilities of the waste disposal system. Tank overflow will dump into the waste water storage tanks.

Other liquid waste is disposed by one of three methods:

- (1) discharge directly to the river after dilution by main condenser circulating water
- (2) ion exchange of waste having a low concentration of solids
- (3) evaporation of waste having a high concentration of solids

For each method, the system operates batch-wise. The various radioactive wastes are treated as distinct batches.

A storage tank is closed off after receiving a given amount of waste, and the contained waste is analyzed for activity levels. The waste may be discharged at a controlled rate to the circulating water discharge line from the main condenser, diluting the waste

discharged to the river to activity levels within the limits of 10 CFR 20. Continuous water monitor alarms signal any excessive levels of radioactivity and provide a record of waste discharge levels.

If the tank-contained waste is of an activity level that precludes direct discharge to the river, the solids content is determined. Liquid containing a moderate concentration of solids is processed through an ion exchanger to reduce activity levels to a safe point for discharge to the river or return to the power plant condensate system.

If the solids concentration is such that the ion exchanger treatment would be uneconomical, the waste is transferred to the evaporator feed tank, from which it is processed through the liquid waste evaporator. The vapor from the evaporator is collected in the water collection tank and analyzed after treatment. If activity levels are acceptable, the product is transferred either to the river or to the main condenser hotwell. The evaporator residue is enclosed directly in 55-gal drums and mixed with cement to form a solid mass. If direct enclosure in drums is not feasible, the residue is stored in a concentrated waste storage tank, from which the slurry is pumped to 55-gal drums and mixed with cement, in preparation for final disposal. The drumming facilities package waste in accordance with 10 CFR 20 and ICC requirements.

All depleted ion exchange resins are sluiced into a spent resin tank in the waste treatment building where they may be stored for decay or immediately drummed with cement for final disposal.

Tanks handling concentrated waste have heating devices and mixing spargers. The evaporator feed tank also has an air mixing sparger to ensure a homogeneous mixture.

The following portions of system piping are of stainless steel:

- (1) lines between the evaporator shell and the concentrated waste tank or the drumming station
- (2) line between the concentrated waste tank and the drumming station
- (3) vapor lines between the evaporator and evaporator condenser, and between the water collection tank and the evaporator condenser
- (4) lines between the evaporator feed tank and the following:
 - (a) evaporator
 - (b) spent resin tank
 - (c) neutralizer tank
 - (d) Radwaste ion exchanger (including return to the water collection tank)

- (5) line between the spent resin tank and the drumming station
- (6) miscellaneous sampling lines

All other lines are carbon steel.

The piping that transports the concentrated waste from the evaporator is electrically traced, and maintained at a temperature of ~ 150 F. Piping that transports concentrated evaporator waste and spent resin tank waste to the drumming station is shielded to prevent personnel exposure to high radiation.

Piping systems that carry concentrated liquid wastes or spent resins contain plug-type valves. Pipes carrying liquid wastes in the waste treatment building contain Saunders-type diaphragm valves.

The waste disposal processing unit is behind concrete walls to reduce background radiation to permissible levels.

Biological shielding for the equipment allows semirestricted access to the waste treatment building and areas adjacent to the storage tanks. Wall openings permit extended stem valve handwheel operation from a low radiation area.

Relief valves protect against excessive pressure. All pumps have automatic low-level cut-offs to preclude operation with no fluid.

The process vents are manifolded to the off-gas system to provide monitoring of all gaseous activity from waste disposal operations so such activity can be safely disposed of. Radiation monitors measure radiation and alarm at a predetermined μ /cc level.

5.3.1.2 6000-gal Retention Tanks. The two retention tanks are vertical cylindrical vessels with flange and dished heads on both ends, designed and fabricated in accordance with Section VIII of the ASME Code (1962), Para. UW 12-B.

The two tanks, which are in the containment building, collect and store various process water drainage. They are sized to handle one complete drainage of the reactor and connected primary piping systems. The retention tank effluent may be discharged directly to the river or pumped to the Radwaste ion exchanger unit or the evaporator for decontamination.

5.3.1.3 3000- and 4500-gal Waste Water Storage Tanks. These storage tanks are horizontal cylindrical vessels with flanges and dished heads on both ends. They are designed for 25 psig and 200 F and are fabricated in accordance with the 1962 ASME Code, Section VIII, Para. UW 12-B.

The tanks store and collect various process water drainage. The drainage may be stored in the tank for decay, pumped to the river as waste, or to the Radwaste exchanger or the evaporator for reclamation. The tanks are in the tunnel between the containment and turbine buildings.

The waste water storage tanks have been designed to receive the complete batch of liquid waste produced during one regeneration cycle of full flow condensate resin. Table 5-5 is a list of liquid waste sources in the plant and the anticipated quantities.

TABLE 5-5
LIQUID WASTE SOURCES

<u>source</u>	<u>max (gal/day)</u>	<u>avg (gal/day)</u>
reactor building		
floor drains and sump	1,000	100
process drains	12,000	15
turbine building		
floor drains and sump	1,000	100
process drains	3,000	15
regenerant solution	4,500	---
hot change room drains	150	---

5.3.1.4 Evaporator Feed Tank. The evaporator feed tank in the waste treatment building is a 1000-gal vertical cylindrical tank with flanged and dished heads and is fabricated in accordance with the 1962 ASME Code, Section VIII, Para. UW 12-B.

This tank can receive liquid-waste from the retention tanks, waste water storage tanks, concentrated waste tank, the water collection tank and the Radwaste ion exchanger. It discharges into the evaporator after the solution pH is adjusted.

5.3.1.5 Water Collection Tank. The water collection tank is a 1000-gal vertical cylindrical pressure vessel with flanged and dished heads, and is fabricated in accordance with the 1962 ASME Code, Section VIII, Para. UW 12-B.

The tank receives the tops from the evaporator and can also receive liquid from the Radwaste ion exchanger. It is located in the waste treatment building.

5.3.1.6 Spent Resin Tank. The spent resin tank is a 1000-gal vertical cylindrical pressure vessel with flanged and dished heads on both ends and is fabricated in accordance with the 1962 ASME Code, Section VIII, Para. UW 12-B. It is located in the waste treatment building and is sized to accept one complete condensate demineralizer charge of resin and the accompanying sluice water.

Depleted resin is sluiced to the spent resin tank from the Radwaste ion exchanger, the primary purification system cation exchanger, and the primary purification mixed bed ion exchanger. Condensate demineralizer resin that becomes fouled or otherwise not regenerable is also sluiced to the spent resin tank. The resins can be stored in the tank to allow short-lived isotopes to decay.

5.3.1.7 Radioactive Waste Evaporator. The waste evaporator in the waste treatment building is designed to process 500 lb/hr of waste containing a high concentration of radioactive solids.

The evaporator is a vertical cylindrical pressure vessel with hemispherical heads. The base material is stainless steel. The vessel is fabricated in accordance with the 1962 ASME Code, Section VIII. The evaporator consists of a shell, removable tube bundle, and an internal demister. The bottom head and tube side of the callandria contains the evaporator residue. The upper portion of the tube side and the upper head is the vapor area of the evaporator. An internal demister is in the upper head. Steam is fed to the middle section on the shell side of the tube bundle in the callandria. The upper head is connected to an external demister packed with stainless-steel mesh. The external demister is fitted with piping for periodic flushes to clean the mesh and improve the decontamination factor when necessary. The evaporator top head is fitted with a hi-foam switch that automatically secures the evaporator in case of excessive foam levels. The level control is separated from the evaporator bottoms by a diaphragm. Thus, the level control operating mechanism does not contact the evaporator bottoms and become fouled.

5.3.1.8 Radwaste Ion Exchanger. The ion exchanger in the waste treatment building processes waste that contains a moderate amount of solids.

The exchanger is a vertical cylindrical pressure vessel with a bottom flanged and dished head and a top flat plate cover. It is fabricated according to the 1962 ASME Code, Section VIII, Para. UW 12-A, and is code stamped. The vessel contains 6 ft³ of cation resin equivalent to Rohm and Haas IRA-120, and 3 ft³ of highly basic anion resin equivalent to Rohm and Haas IRA-400. The resins are premixed. This unit is nonregenerative. The resin retention element consists of six stainless-steel laterals wrapped in stainless-steel screen. The laterals are arranged like spokes on a wheel and are connected to a central 1-1/2 in. stainless-steel outlet.

The overflow and relief valve nozzle is covered with a layer of stainless-steel screen to prevent resin transport into this line. The inlet waste is distributed through six stainless-steel laterals arranged like spokes around a 1-1/2-in.-inlet stainless-steel pipe. Each lateral has twelve 3/16-in. holes that face upward. A 40 gpm/ft² flow loading on the resin provides effective filtering as well as ion exchange.

5.3.1.9 Concentrated Waste Tank. The 500-gal concentrated waste tank is a horizontal cylindrical pressure vessel with flanged and dished heads and is fabricated according to the 1962 ASME Code, Section VIII, Para. UW 12-B. The tank is equipped with a sparger connected to a steam supply for combined heating and mixing, and to an air supply when only mixing is required.

The tank is in the waste treatment building and stores the evaporator residue if the drumming station is not available for direct drumming from the evaporator.

5.3.1.10 Chemical Mixing Tank. To control pH and foam level in the evaporator feed tank waste solution, the adjusted contents of the chemical mixing tank in the waste treatment building are drawn into the waste solution through the eductor.

The mixing tank is a vertical cylinder with a flange and dished head on the lower end and a flat plate hinged cover on the top. The tank has a 50-gal capacity and is fabricated according to the AWWA Code.

5.3.1.11 Portable Demineralizer. Liquid waste can be passed from one tank to another through this demineralizer to decontaminate the retention tank. The demineralizer is also used in the shield cooling system (Sec. 5.2.9) and the storage well cooling system (Sec. 5.2.10).

The components of the demineralizer are mounted together on a portable skid and include the following:

- (1) demineralizer
- (2) afterfilter
- (3) pump
- (4) conductivity meter
- (5) total-flow water meter
- (6) pressure gauges
- (7) connecting pipe

The demineralizer consists of a vertical tank containing 3.6 ft³ of cation resin equivalent to Rohm and Haas IR-120, and 2.5 ft³ of anion resin equivalent to Rohm and Haas IRA-400. It is nonregenerative. Exhausted resins are sluiced to the spent resin tank via the same piping that sluices spent resins from the purification system ion exchangers. The 10- μ afterfilter retains all resins, to protect against failure of a bottom distributor resin retention element. The 50-gpm canned pump associated with this unit overcomes the line losses caused by the filter and the ion exchangers, so that the system connected to the unit is not overburdened by the increased pressure drop from the demineralizer.

5.3.1.12 System Pumps. Each of the pumps described below is a NEMA design-B class-A insulated pump, with a single-speed squirrel-cage induction motor, using a 440-v, a-c, 3-phase 60-cycle current; each has a drip-proof frame, except for the sump pumps, which have a submersible frame.

The concentrated liquid waste transfer pump transfers liquid waste from the concentrated waste tank to the drumming station. It is a horizontal-worm positive-displacement type pump, with the pump and driver mounted on a common bedplate. All wetted parts of the pump are of stainless steel. The pump has a revolution counter, calibrated so the total volume pumped can be determined. This counter connects to the pump controls to stop the pump automatically after a preset amount of waste has been pumped.

The waste water storage tank pump transfers liquid to the Radwaste ion exchanger, evaporator, or the line leading to the main circulating water discharge line. The waste water storage tank pump is a horizontal centrifugal-type pump, with pump and motor mounted on the same bedplate. The pump has a capacity of 50 gpm under the normal 100-ft back pressure conditions associated with water transfer. It has a capacity of 20 gpm for feeding the Radwaste ion exchanger with a net developed head of 140 ft. It is designed for 100 psig.

The 6000-gal retention tank pump is used to transfer wastes between retention tanks, to the line discharging into the main circulating water discharge line, or to the waste treatment facilities. The retention tank pump is a horizontal centrifugal-type pump, with pump and driver mounted on a common bedplate. The pump has a normal capacity of 50 gpm, with a net developed head of 100 ft. It has a 20 gpm capacity, with a 140 ft net developed head when pumping waste through the Radwaste ion exchanger during cleanup operation. It is designed for 100 psig.

The liquid waste demineralizer pump is used for the following:

- (1) to transfer liquid from the evaporator feed tank to the Radwaste ion exchanger,
- (2) to transfer liquid from the evaporator feed tank through the inductor (where chemicals are added from the chemical mixing tank) and back to the evaporator feed tank,

(3) to wash concentrated waste lines, and

(4) to transfer water from the water collection tank to the main circulating water discharge line or to the main condenser hotwell.

The liquid waste demineralizer pump is a horizontal centrifugal-type pump, with pump and driver mounted on the same bedplate. It is designed for 100 psig and has a 20 gpm capacity with a net developed head of 80 ft.

The sump pumps are as follows:

- (1) two reactor building sump pumps
- (2) two waste gas storage vault sump pumps
- (3) two waste disposal building sump pumps

All of these pumps are vertical-shaft, submerged, centrifugal, nonclogging-type pumps, supported from their motor base plates by suspension pipes and directly connected to their motors by suitable flexible coupling, being thus readily removable from the sump. Each pump has a 20 gpm capacity, with a net developed head of 30 ft. The motors for the reactor building and waste gas storage vault pumps are in the sump directly over the pump. The motor for the waste disposal building pump is on the sump cover plate and is connected to the pump by a vertical shaft with an intermediate bearing.

There are two pumps in each sump. Each pump has a copper float, float guide, and brass float rod, arranged to operate the float control switch to start and stop the pump automatically at preset levels. One pump starts automatically at a predetermined high level (to be set in the field), and stops when the water level falls to 8 in. above the bottom of the sump. The pumps are controlled by an alternator that starts and stops the pumps alternately. If the level continues to rise with one pump running, the second pump of the duplex unit automatically starts. If one pump should fail, the other pump starts automatically.

The evaporator feed pump normally transfers liquid from the evaporator feed tank to the evaporator. It is a horizontal centrifugal-type pump, with pump and driver mounted on a common bedplate. The pump has a capacity of 1 gpm, with a net developed head of 25 ft.

The process water transfer pump transfers water from the water collection tank to either the main circulating water discharge line or the main condenser hotwell. The process water transfer pump is a horizontal centrifugal-type pump, with pump and driver on a common bedplate. The motor has a NEMA design-B designation, with class-A insulation. The pump capacity is 5 gpm, with a net developed head of 30 ft. It is designed for continuous duty.

5.3.1.13 Drumming Station Mixer. Concentrated waste is mixed with dry cement and sawdust in shipping drums using a wall-mounted drilling machine which has been modified for this purpose.

The mixer motor is vertically positioned on a rack and pinion remotely by an electric drive. The rack and pinion assembly is permanently attached to the wall above the drumming station in the waste treatment building basement. The mixer blade consists of 1-in. bars stacked in T-shape. The cross bar length equals the diameter of the shipping drum. The vertical bar is long enough to keep the mixer chuck out of the cement mixture when the blade is fully inserted into the drum, but does not protrude above the drum. The mixer motor is connected by a gear train to the chuck that holds the mixing blade, so the chuck can turn at either 35 or 70 rpm. Speed is changed with a lever on the gear box, or by a change in mixer motor speed.

5.3.1.14 Hoist and Monorail System. This handling equipment is on the ceiling above the grade floor. It runs from above the hatch that opens above the drumming station, through the decontamination room, out the double doors in the decontamination room, and terminates at the loading dock. The hoist has sufficient travel to lower drums from the grade level floor down to the drumming station in the waste treatment building basement. Clamps are provided so that the standard 55-gal shipping drums can be lifted without modification of the drums.

The hoist is a manually operated Safeway-Army-type 10-ton chain hoist manufactured by Wright Hoist Co.

5.3.1.15 Instrumentation and Control. The retention tanks have been instrumented to indicate level locally and to annunciate a visual and audible high level alarm in the control room when tank level reaches a predetermined point. A low level condition actuates a level switch, which automatically stops the retention tank pump. The pump does not automatically start if the water level again rises past the pump cutoff level; it must be started manually by a hand control. This hand control also allows the pump to be tripped locally. A pressure gauge is on the discharge side of the pump.

The water collection tank has a sight glass to indicate water level. A level switch indicates level locally and actuates high and low local alarms. The process water transfer pump has a low level pump cutoff switch, which prevents pump startup when the tank liquid level is low. This pump also has a thermal overload cutoff. Pressure is measured and indicated locally at the discharge end of the pump. A local manual control switch controls pump operation.

A sight glass is used to indicate the evaporator feed tank level. A level switch indicates level locally and actuates high and low local alarms. A pump cutoff actuated by the limit

switch upon low level trips the liquid waste demineralizer pumps and the evaporator feed pump. There are two hand controls for pump operation. Pressure is measured at the discharge end of the pumps and indicated locally.

The spent resin tank contains a gauge glass located outside the shield wall with a 200-mesh screen at both ends of the gauge glass to prevent resin accumulation in the sight glass. Overflow from the spent resin tank is sensed by a level switch and is annunciated locally, and remotely in the control room.

The evaporator feed is controlled by a flow control valve operated by two level transmitters. If the flow control valve accidentally closes, the evaporator feed pump is tripped by a limit switch on the feed control valve. A direct-operated solenoid valve in the heating steam supply closes when the evaporator liquor level reaches a predetermined low level, preventing any caking of impurities onto the tubes.

Liquid level is measured in the evaporator with two diaphragm-type pressure transmitters. One transmitter is in the steam section of the evaporator and the other in the water section. The steam transmitter feeds a pneumatic signal, proportional to the static pressure in the evaporator, to a level controller. The other transmitter feeds a pneumatic signal, proportional to static plus liquid level pressure, to the same level controller. The controller compares both signals, and the resultant signal, proportional to the liquid level, is compared with a set-point signal. The difference is transmitted pneumatically to position the "Liquor Feed" valve to oppose the error signal and return it to zero. High and low evaporator levels actuate local alarms.

A foam cutoff switch on top of the evaporator automatically shuts down the evaporator upon foam levels that would reduce the attainable decontamination factor. For excessive foam, the switch closes a direct-operated solenoid valve in the heating steam supply line and stops feed flow to the evaporator by closing the solenoid-piloted diaphragm-operated valve.

Pressure in the steam section of the evaporator is measured by a pressure transmitter and indicated locally. Pressure in the heating steam section is measured and indicated on a local pressure indicator. Two local temperature gauges measure liquid and steam section temperature in the evaporator tank. Manual control switches are provided for the heating steam direct-operated solenoid valve and the solenoid pilot valve for the feed control valve.

The local control panel in the waste treatment building indicates the concentrated waste tank surry level measured by a level transmitter and annunciates a high level alarm on the annunciator panel. The concentrated waste transfer pump has a revolution counter for

accurate control of waste pumped to each barrel, by an automatic shutoff after a preset number of revolutions. There are local manual controls to operate the pump.

Conductivity cells at the inlet and outlet of the Radwaste ion exchanger continuously indicate conductivity on the local control panel. A gas bulb temperature gauge and a flow element in the influent line give continuous indications on the local control panel. A differential pressure gauge measures pressure difference across the exchanger and transmits a pneumatic signal to a local pressure differential indicator on the local control panel.

The waste water storage tanks indicate water level locally and annunciate an alarm for high level in either tank. Level glasses and local level indicators are provided. At low water level the pump is protected by a low level switch that interrupts the hold circuit for the motor, interrupting the power supply and stopping the pump. The discharge pressure of the storage tanks pump is indicated locally by a pressure gauge.

A manual control for pump operation is at the local control panel. There are radiation monitors for the following (see Sec. 8.8.1.1):

(1) Liquid Waste and Service Water. This monitor is in a sampling line connected to the side of the main stream liquid waste pipe inside the waste treatment building.

(2) Turbine-Condenser Cooling Water Discharge (main circulating water discharge). This monitor is inside the turbine building in a sampling line connected to the side of the main stream pipe downstream of the main condenser.

The liquid waste and service water monitor alarms when the activity of the liquid, after dilution with the turbine-condenser cooling water, equals ten MPC. The other monitor is an additional check, monitoring the radioactivity in the discharge line to the river.

There are other temperature, flow, and pressure instruments in the system for routine monitoring of system performance.

5.3.2 Solid Waste Disposal

Resins and filters from the primary purification system and the Radwaste ion exchanger are the major sources of solid wastes. Resins are sluiced to the 1000-gal spent-resin tank and packaged in drums for shipment to an authorized disposal site (Sec. 11.13.1.5). The primary purification system filters are removable from their container as cartridges. If the filters become highly radioactive (such as after fuel element failures) connections permit the removal and replacement of entire units. The units can be stored in a dry-storage pit until decay permits shipment to an authorized disposal site.

Items such as contaminated clothing are collected in drums and sent to a licensed contaminated laundry facility. Radioactive laboratory glassware and other contaminated solid wastes are packaged in drums for off-site disposal, or temporarily stored in a dry-storage pit.

A cement mixer and transfer pump are used for packaging the solid wastes (Sec. 11.13.1.3).

5.3.3. Vent and Waste Gas Disposal System (Fig. 5.38)

The vent and waste gas disposal system:

- (1) collects and contains radioactive waste gases generated or released in plant components,
- (2) transports collected waste gases to the stack-release point,
- (3) monitors waste gas activity before final release to the stack,
- (4) controls rate of activity release to the surrounding atmosphere,
- (5) stores radioactive waste gases for decay to safe release levels.

5.3.3.1 General Description. Waste gas is collected in all components that may contain radioactive gases. A 4-in. vent header routes gases from the air ejector vents in the main condenser and the shutdown condenser to the 150-ft³ holdup tank. A total gas flow of 15 scfm is expected at full power. The holdup tank allows decay of short-lived isotopes such as N-16. From the holdup tank the waste gas is routed directly to the base of the stack, or to the gas storage tanks if a monitor downstream of the holdup tank records excessive activity. In the latter case, the gases are drawn from the holdup tank and mixed with steam from a steam ejector, to reduce the hydrogen concentration to below the explosive limit. The inlet of the ejector receives 50-55 psig steam. Gases and steam flow through a recombiner, where hydrogen and oxygen in the waste gas are recombined. The hydrogen and oxygen thus removed lessen the storage load of the waste gas system and eliminate the problems of handling combustible gas mixture in a compressed gas system. In the recombiner condenser the steam is condensed, and gases are vented to the compressor, where they are compressed to a minimum of 300 psig and charged to the storage tanks. Since steam condensation on the cold vessel surface during warmup reduces recombiner efficiency, gases from the recombiner condenser are initially vented directly (at up to 3 psig) to the gas storage tanks, preventing the compression of a combustible gas mixture. After sufficient decay a controlled release of waste gas in the storage tanks can be allowed by a remotely operated control valve.

The stack is 350 ft high (240 ft higher than any adjacent structure). The two 35,000 cfm stack blowers normally operate together. The blowers operate through a common suction header, taking gas from a plenum in the base of the stack and discharging it into the stack. Motor operated dampers are in the suction and discharge ducts of the stack blowers. Outside air enters the stack blower suction header upstream of the dampers on the suction side of each stack blower. Stack monitors are in the stack, downstream of the outside air line tie-in.

In addition to the main vent lines, other vent headers collect off-gases from the other plant components. A 4-in. vent header collects gases from reactor plant equipment in the turbine building. This equipment includes the sluice water storage tank, the condensate demineralizer vessels, and the vessels of the resin regeneration system. Two 3-in. vent lines from the two waste water storage tanks connect to this 4-in. vent header. The 4-in. vent header goes directly to the plenum in the base of the stack. Before entering the stack it receives two other vent headers, a 3-in. vent header from the equipment in the waste treatment building and the 4-in. vent header from the containment building equipment. The vent header from the waste treatment building collects the gases from the evaporator feed tank, the spent resin tank, the water collection tank and the concentrated waste tank. The 4-in. header from the containment building normally collects gases from the purification cooler, the regenerative cooler, the shield cooling and component cooling surge tank, and the retention tanks. It can also collect gases from the reactor cavities and the fuel element well, but such gases normally are routed through a 4-in. vent header to the suction side of the reactor containment ventilation exhaust fans. The waste gas vent headers are routed to the stack through a tunnel that also serves as the ventilation exhaust duct for the turbine building and the reactor containment exhaust fans.

Two 1600-ft³ (12,000-gal) gas storage tanks are in the underground vault. This underground vault is connected to the plenum in the base of the stack. The waste treatment building exhaust fan discharges into the underground vault through a 20-in. duct.

The system piping is in two basic groups -- low pressure off-gas piping, and high pressure off-gas piping. The first group consists of seamless carbon steel piping (ASTM Type A-106, Grade A or B) designed for a maximum working pressure and temperature of 30 psig and 200 F. The second group consists of piping of the same material but designed for a maximum working pressure and temperature of 300 psig and 250 F. All piping has welded joints (butt-welded for pipe 2.5 in. and larger, and socket-welded for pipe 2 in. and smaller). The piping is designed for maximum system pressure, temperature, and flow conditions. All piping which may contain off-gas from the main condenser is designed for 300 psig and 250 F. Piping that may contain compressed gas at up to 300 psig is also designed for 300 psig and 250 F. All other vent piping is designed for 30 psig and 200 F.

There is a separate 6-in. line from the mechanical vacuum pump to the stack. The line is sized to transport the quantities of air expected during startup of the main condenser.

5.3.3.2 Stack and Stack Blowers. All collected gaseous wastes are routed to the stack. Here they are mixed with outside air to increase the effective stack height, which reduces the downwind activity level of gases discharged to the atmosphere. The gas is driven through the stack with two stack blowers.

The stack is 350 ft high and is of structural concrete, with an aluminum nozzle at the top. The nozzle tapers to 4 ft-6 in. at the discharge, providing a stack exit velocity of approximately 70 fps.

Each of the two 35,000-cfm stack blowers are centrifugal single-inlet, class-II motor-driven weatherproof units, with a multiple V-belt drive. The units are installed outdoors near the stack. The motors are rated for 40 hp. The blower bearings are air-cooled.

5.3.3.3 Holdup Tank. The holdup tank in the turbine building provides decay time for short-lived radioactive gases present in the main condenser off-gas. At full power the reactor is expected to produce ~ 10 scfm of noncondensable gases, primarily oxygen and hydrogen from radiolytic decomposition of water in the core. Main condenser "inleakage" of air adds ~ 5 scfm. Thus, a total of about 15 scfm of noncondensable gases (including radioactive gases) is removed to the holdup tank by two air ejectors. The 150-ft³ tank volume provides an average holdup time of ~ 10 min.

The holdup tank is a cylinder 5 ft in diameter and 6 ft long, with elliptical dished heads. It is made of carbon steel (ASTM A-212, Grade B). The tank is designed for 150 psig and 650 F, and is designed and constructed according to the ASME Code (Section VIII, Code Stamp required).

5.3.3.4 Recombiner. Hydrogen and oxygen in the main condenser off-gas are recombined in a catalytic recombinder, with a platinized Ni-Cr alloy base catalyst. Gas mixtures entering the recombinder must contain less than 4 percent hydrogen for safe recombination. For proper dilution, the gases are mixed with dry steam before entering the recombinder. Recombination efficiencies can approach 100 percent for hydrogen and oxygen in stoichiometric mixtures. The surplus oxygen goes through the recombinder unprocessed. The recombinder effluent is mostly nitrogen and steam.

If 100 percent recombination of the stoichiometric mixture is not achieved, the recombinder effluent may be combustible. Recombiner efficiencies significantly below 100 percent may occur if the catalyst becomes coated by impurities or moisture. The system design prevents moisture from entering the recombinder and warns the operator when the effluent contains combustible gas mixtures (Sec. 11.14.2.1).

The recombiner is a carbon steel vessel containing a platinized nickel-chrome alloy base catalyst. The catalyst bed thickness is 7.5 in. The vessel is an ASME Code vessel (Section VIII, Code Stamp required). It is 56 in. long and has a 24-in. OD. The total design inflow of gas is 20.6 scfm. The design steam flow is 12.6 lb/min at 50-55 psig. The temperature of the recombiner effluent is expected to be 600 F.

5.3.3.5 Recombiner Condenser. The recombiner condenser condenses the steam from the recombiner. It is a shell and U-tube type heat exchanger with Cupro-nickel tubes and a carbon steel shell. The tube sheet is made of forged steel with a monel overlay. The shell side is designed and constructed according to the ASME Code (Section VIII) and to Nuclear Code Case 1270-N.

The rated duty of the unit is 1.04×10^6 Btu/hr. Shell design pressure is 300 psig and tube design pressure is 100 psig. Both the tube (cooling water) and shell (gas) sides have relief valves. The shell diameter is 10 in. and the overall length of the unit is 11 ft.

5.3.3.6 Waste Gas Compressor. The waste gas compressor compresses the gases from the recombiner condenser to a maximum of 300 psig. It is a one-cylinder, two-stage, double-acting water-cooled motor-driven compressor, with V-belts and splash-lubrication.

The unit has a design capacity of 16.6 cfm of free gas at a discharge pressure of 300 psig. The compressor speed is 440 rpm. The motor is a drip-proof induction type with a 10 hp rating at 1800 rpm. The compressor has an intercooler and aftercooler, each requiring 1 gpm of cooling water at 90 F. The valve unloader is of the air-operated diaphragm type.

5.3.3.7 Gas Storage Tanks. The two 1600-ft³ gas storage tanks receive gases from the waste gas compressor at a maximum pressure of 300 psig. The tanks can store the gases produced by 72 hr of full power operation. The 72 hr gas storage capacity allows sufficient time to reduce or control the gas release upon high radiation levels, or to permit unfavorable atmospheric conditions to pass. It also allows sporadic bursts of high gaseous activity from the primary system to decay and short-lived radioactive gases to decay.

Each gas storage tank is a cylinder, 10 ft-6 in. in diameter and 18 ft-6 in. long, with elliptical dished heads. The material is carbon steel, ASTM Type A-212, Grade B. The vessels are ASME (Section VIII) vessels (Code Stamp required, Para. UW 12A, UW 2). They are designed for 350 psig and 212 F. The tanks are behind shielding in the underground vault.

5.3.3.8 Absolute Filters. The reactor containment building ventilation system and the waste disposal ventilation system are equipped with Cambridge absolute filters (Model 1A-990-1), each complete with a filter box and gaskets. Unit efficiency is above 99.9 percent for particles larger than 0.3μ . The pressure drop across the filters is less than 1 in. of water for a rated capacity of 1100 cfm of air at STP.

5.3.3.9 Instrumentation. Steam flow is sensed at the recombiner steam ejector inlet with an orifice plate flow element and is measured with a differential pressure transmitter. The transmitter sends an electrical signal, proportional to steam flow, to an indicator in the control room. Low flow actuates an alarm in the control room.

Flow of the noncondensable gases leaving the air ejector aftercooler is measured with an orifice plate and differential pressure transmitter and transmitted electrically to an indicator in the control room. Low flow actuates an alarm in the control room.

The containment building air flowrate is measured at the suction of the containment ventilation exhaust fan with a flow element, and indicated locally. Low flow initiates an alarm in the control room.

Gas temperature at the holdup tank inlet is indicated locally by a temperature gauge. A temperature well is also provided for use if needed. Temperatures of the recombiner inlet and outlet are measured with two iron-constantan thermocouples and transmitted to a 2-pen temperature recorder on the panel in the main control room. High temperature at the outlet of the recombiner actuates an alarm in the control room.

A temperature switch in the steam pipeline to the recombiner operates a 24-kw pipe heater when a high radiation condition exists at the holdup tank outlet. This control action aids recombination by heating the entering gas mixture.

The temperatures of the two 12,000-gal storage tanks are measured with iron-constantan thermocouples and transmitted to a high precision multipushbutton-type indicator in the control room.

Local indicating pressure gauges are at the following locations:

- (1) gas influent to the holdup tank,
- (2) recombiner inlet, and
- (3) waste gas compressor discharge.

There is remote pressure indication in the control room from the following locations:

- (1) waste gas compressor discharge,
- (2) gas storage tank 1B,
- (3) gas storage tank 1A, and
- (4) containment vessel.

Differential pressure between the plenum of the stack and the outside air is measured with a differential transmitter and indicated in the control room. Differential pressure across the containment ventilation exhaust duct outlet filter is indicated locally with a differential pressure gauge. Differential pressure across the filter in the 20-in. ventilation exhaust duct in the waste treatment building is indicated locally with a differential pressure gauge.

Either pressure transmitter on the 12,000-gal gas storage tanks actuates a common alarm if pressure exceeds 3 psig when an explosive mixture condition exists in the recombiner condenser (> 4 v/o of hydrogen). There are also pressure alarms in the control room for high waste-gas compressor discharge pressure, high gas storage tank pressure, and high containment building pressure.

A combustible gas-analyzer analyzes off-gas from the recombiner condenser. Readings are recorded in the control room. An alarm in the control room warns of excessive hydrogen in the gas.

5.3.3.10 Radiation Monitors. There are radiation monitors for the following (Sec. 8.8.1):

- (1) the effluent from the holdup tank for gaseous activity
- (2) reactor containment building ventilation fan discharge for particulate and gaseous activity
- (3) stack effluent for particulate and gaseous activity.

Items 1, 2, and 3 are recorded in the control room. High radiation alarms are in the control room for all three items. The settings are based either on maximum permissible concentration (MPC) or ten times MPC, as discussed in Secs. 8.8.1.2 through 8.8.1.4.

5.3.3.11 Controls. Vent and waste gas disposal system operation is discussed fully in Sec. 11.14.

A high radioactivity alarm signal from the radiation monitor just downstream of the holdup tank actuates solenoid valves, which control the instrument air flow to position the four diaphragm on-off-type control valves to route holdup tank effluent to the recombiner, and from the recombiner condenser to the gas storage tanks (Sec. 11.14.2.1). The same signal also starts the heating element upstream of the recombiner. The heating element is an automatic on-off type, controlled with a temperature monitor. The gas analyzer, which analyzes for combustible mixture in the recombiner condenser effluent line, is interlocked with the startup circuit of the waste gas compressor, so that the compressor cannot operate while the hydrogen percentage in the effluent gas is above 4 percent. The compressor is

started manually as discussed in Sec. 11.14.2.1. When the holdup tank effluent activity level returns to normal, the control valves are manually returned to their normal positions. The compressor can be shut down manually by a hand control.

Two particulate monitors and a gaseous monitor in the reactor building ventilation exhaust line just before the exhaust fan produce signals to close the air operated ventilation system dampers upon excessive particulate or gaseous activity. These dampers also close automatically upon high vapor containment pressure or high reactor pressure. The 4-in. vent header from the containment building contains a diaphragm on-off-type control valve, operated by instrument air, which is also closed by signal to its pilot solenoid valve upon high reactor or containment building pressure. Isolation of the containment building atmosphere is thus assured.

Manual controls operable from the control room are as follows:

- (1) a hand indicating controller to adjust flow of gas from storage tanks to the stack
- (2) a switch to stop or start stack blower 1A
- (3) a switch to stop or start stack blower 1B
- (4) a switch to open or close the motor operated air louvers in the stack
- (5) a switch to stop or start the reactor building ventilation exhaust fan
- (6) a hand controller to stop or start reactor plant vent header flow to stack
- (7) a switch to open or close reactor building ventilation dampers and the air recirculation valve
- (8) a switch to stop or start waste gas compressor
- (9) a switch to route air ejector off-gas either to the stack or through the recombiner to storage tanks
- (10) a switch to route off-gas from the reactor plant vent header either to the stack or the compressor suction
- (11) a switch to route reactor cavity vent flow to either the ventilation blower suction or the vent header in the containment building.

FIG. 5.1

LACBWR GENERATOR PLANT MAIN STEAM AND FEEDWATER SYSTEMS

FIG. 5.2

REACTOR PLANT FORCED-CIRCULATION SYSTEM

FIG. 5.3

NOTES:

1. F.C. PUMP PERMIT START. NORMAL PRESS. 14 PSI & ABOVE.
2. STANDBY PUMP AUTO START. 10 PSI & BELOW.
3. F.C. PUMP TRIP. 5 PSI & BELOW.
4. INTERNAL FILTER BY PASS SET AT 20 PSI

FIG. 5.6

• TEMP. COMPENSATORS FOR CONDUCTIVITY ELEMENTS.

LACBWR REACTOR PLANT PURIFICATION SYSTEM

FIG. 5.8

LACBWR SEAL INJECTION SYSTEM

FIG. 5.9

LACBWR CONTROL ROD DRIVE MECHANISM SHOWING
SEAL INJECTION, FLUSHING, AND BLOWDOWN PORTS

FIG. 5.10

LACBWR SYSTEMS FOR CONTROL ROD DRIVE FLUSHING, SEALING AND EFFLUENT

FIG. 5.11

LACBWR FORCED-CIRCULATION PUMP SEAL

FIG. 5.13

LACBWR FORCED-CIRCULATION PUMP IMPELLER AND CASING

FIG. 5.13A

LACBWR REACTOR PLANT SHIELD COOLING SYSTEM

FIG. 5.17

ALLIS-CHALMERS MANUFACTURING COMPANY ATOMIC ENERGY DIVISION NUCLEAR POWER DEPARTMENT WASHINGTON, D. C.	
LACBWR FUEL TRANSFER SYS. ELEVATION	
DWG. NO. 41-503-132	
SHEET 1	OF 1
REV. 08	

FIG. 5.19

LACBWR FUEL TRANSFER SYSTEM PLAN

Fig. 5.20

FAILED FUEL DETECTION BUNDLE HANDLING PROCEDURE

FIG. 5.21

LACBWR SPENT FUEL TRANSFER DIAGRAM

FIG. 5.22

LACBWR USED CONTROL ROD TRANSFER DIAGRAM

FIG. 5.23

LACBWR NEW FUEL INSTALLATION DIAGRAM

FIG. 5.24

FIG. 5.25

- NOTES:
1. CEMENT GASKET TO GATE WITH MINN. MINING AND MFG. CO. NO. EC-1300 CLEAN ALUMINUM SEAL SURFACE WITH METHYL ETHYL KETONE BEFORE APPLYING CEMENT.
 2. ALL AUSTENITIC STAINLESS STEEL SHALL BE PICKLED AND PASSIVATED AFTER WELDING.
 3. ASSEMBLY SHALL BE ADEQUATELY BRACED FOR SHIPMENT TO MAINTAIN STRAIGHTNESS AND ALIGNMENT.

ALLIS-CHALMERS
MANUFACTURING COMPANY
ATOMIC ENERGY DIVISION
NUCLEAR POWER DEPARTMENT
WASHINGTON, D. C.

LACBWR
CANAL GATE AND HOUSE

DWG. NO. 41-503-143
SHEET 1 OF 2

FUEL TRANSFER BRIDGE

LACBWR GRAPPLE ACTUATOR

LACBWR
FUEL ASSEMBLY
GRAPPLE

DWG. NO. 41-503-156	REV.
SCALE FULL	SHEET 1 OF 2

CONTROL ROD GRAPPLE ASSEMBLY

SPENT FUEL AND CONTROL ROD STORAGE, PLAN

Fig. 5.33

LACBWR POOL STORAGE RACK FOR FAILED FUEL
LOCATION AND EMERGENCY COOLING BUNDLE

FIG. 5.34

- Fig. 5.36

FIG. 5.37

LACBWR VENT AND WASTE GAS DISPOSAL SYSTEM

6. CONTAINMENT BUILDING

This section presents the revised description of the containment building, and includes the information presented in the "LACBWR Hazards Summary Report for Construction Authorization" (ACNP-62574) and Amendment 1 to the Application (ACNP-62614), as revised by the design changes described in Amendment 3 to the Application (ACNP-63584). In addition, the description is expanded to include information developed during construction of the containment building.

6.1 GENERAL DESCRIPTION

The containment building (Figs. 6.1 to 6.5) is a right circular cylinder with a hemispherical dome and semi-ellipsoidal bottom. It has an overall internal height of 144 ft and an inside diameter of 60 ft, and it extends 26 ft 6 in. below grade level. The shell thickness is 1.16 in., except for the upper hemispherical dome, which is 0.60 in. thick.

The building contains most of the equipment associated with the nuclear steam supply system, including the reactor vessel and biological shielding, the fuel element storage well, the forced-circulation pumps, the shutdown condenser, and process equipment of the reactor water purification system, decay heat cooling system, shield cooling system, seal injection system, emergency core spray system, boron injection system, and storage well cooling system.

The containment building is designed to withstand the instantaneous release of all the energy of the primary system to the containment atmosphere at an initial temperature of 80 F, neglecting the heat losses from the building and heat absorption by internal structures.

The interior of the shell is lined with a 9-in.-thick layer of concrete, to an elevation of 727 ft 10 in., to limit direct radiation doses in the event of fission-product release within the containment building.

The external surface of the shell was spray-painted with one coat of Carbo-Zinc No. 11, applied to produce a dry film thickness of 2 to 3 mils. The internal surface below the crane rail was spray-painted with a one-coat Epoxy-Tar System consisting of Carbomastic No. 12, applied to produce a dry film thickness of 8 to 12 mils. Above the crane rail the interior surface was covered with a two-coat Epoxy System consisting of a prime coat of Plasite No. 444 and a second coat of Plasite 7122.

The containment building is supported on a foundation consisting of concrete-steel piles and a pile capping of concrete approximately 3 ft thick. This support runs from the bottom of the semi-ellipsoidal head at about el 612 ft 4 in. to an elevation of 621 ft 6 in. The 232 piles that support the containment structure are driven deep enough to support over 50 tons per pile.

The containment bottom head above el 621 ft 6 in. and the shell cylinder from the bottom head to approximately 9 in. above grade elevation (639 ft 9 in.) are enveloped by reinforced concrete laid over a 1/2 in. thickness of premolded expansion joint filler. The reinforced concrete consists of a lower ring, mating with the pile capping concrete. The ring is ~4-1/2 ft thick at its bottom and 2-1/2 ft thick at a point 1-1/2 ft below its top (owing to inner surface concavity). The ring then tapers externally to a thickness of 9 in. at the top (el 627 ft 6 in.), and the 9 in. thickness of concrete extends up the wall of the shell cylinder to el 639 ft 9 in. The filler and concrete are not used, however, where cavities containing piping and process equipment are immediately adjacent to the shell.

Except for areas of the shell adjacent to other enclosures, the exterior surface of the shell above el 639 ft 9 in. is covered with 1-1/2-in.-thick siliceous filter insulation, faced with aluminum. The insulation of the dome is Johns-Manville Spintex of 9 lb/ft³ density, faced with embossed aluminum sheet approximately 0.032 in. thick. The insulation of the vertical walls is Johns-Manville Spintex of 6 lb/ft³ density, faced with corrugated embossed aluminum sheet approximately 0.016 in. thick. The insulation minimizes heat losses from the building and maintains the required metal temperature during cold weather, and reduces the summer air-conditioning load.

The shell includes two airlocks. The principal access to the shell will be through the personnel airlock that connects the containment building to the turbine building. The airlock is 21 ft 6 in. long between its two doors, which are 5 ft 6 in. by 7 ft and are large enough to permit passage of a spent fuel element shipping cask. The containment building can also be evacuated, if necessary, through the emergency airlock, which is 7 ft long and 5 ft in diameter, with two circular doors of 32-1/2 in. diameter (with a 30-in. opening). Both airlocks are at el 642 ft 9 in. and lead to platform structures from which descent to grade level can be made. When the doors are closed, a clamp exerts a positive force, which is transmitted through the doors to live-rubber gaskets around the door frames to ensure gas tightness. Electrical interlocks on the airlock doors are connected in the reactor start circuit to prevent reactor startup if the airlocks are not closed. This condition is also annunciated in the control room.

An 8 ft-by-10 ft freight door opening in the containment building accommodates large pieces of equipment. It will be used only when the reactor is shut down and only if large pieces of equipment must be removed. During operation, 9-in.-thick concrete blocks will be placed on the outside of the door for shielding. The door is bolted internally to the door frame in the shell. Two rubber gaskets in parallel between the door and door frame ensure a pressure-tight seal.

Approximately 300 MI cables and 75 bulkhead conductors penetrate the containment shell. These are in the northwest quadrant of the shell adjacent to the electrical room under the control room. Pipe penetrations leave the containment vessel 1 to 10 ft below grade level and enter either at the northwest quadrant into the pipe tunnel that runs to the turbine building, or on the east side into the tunnel connecting the turbine building, reactor building, stack, and the waste treatment and waste gas storage areas.

Two 30-ton air-conditioning units keep the containment building air temperature at or below 80 F. The units also have steam heating coils to provide heat to the building, in case a prolonged shutdown is necessary during the winter, and provide air circulation throughout the building. Each unit circulates approximately 12,000 cfm of air, of which approximately 9500 cfm is recirculated air and 2500 cfm is ventilation air from outside the containment building.

Air is exhausted from the building by means of an exhaust blower, which discharges approximately 5000 cfm through a pair of isolation dampers and through the containment shell penetration into the suction plenum side of the stack blowers. The exhaust air is pulled from both of the main pump cubicles through two filters located outside the pump cubicles and just upstream of the exhaust blower. The first filter is a coarse filter to remove large particles, and the last is a high-efficiency type filter. An automatic bypass permits the exhaust blower to continue to circulate air from the pump cubicles whenever the containment isolation dampers are closed.

Ventilation air is admitted into the building through the containment penetration and a pair of isolation dampers, and then through duct work to the suction side of the blower of each air-conditioning unit.

Excessive external pressure on the building is prevented by two vacuum breakers which start to open when the negative pressure within the building exceeds 0.3 psig.

A 42,000-gal storage tank in the dome of the containment building supplies water for the emergency core spray system and the building spray system. The piping connection to the emergency core spray system is on the bottom of the tank. The connection to the spray headers of the building spray system is a standpipe within the tank; the top of the standpipe is sufficiently above the bottom of the tank to leave 15,000 gal of water for use in the emergency core spray system. The storage tank also provides water for use during refueling and during loading of the fuel element shipping cask.

A 50-ton traveling bridge crane with a 5-ton auxiliary hoist is located in the upper part of the containment building. The bridge completely spans the building and travels on circular tracks supported by a ring of concrete around the inside of the building just below the hemispherical upper head. A trolley containing all the lifting mechanisms travels on the bridge to within 4 ft of the crane rail, and permits crane access to any

position on the main floor under the trolley travel-diameter. The lifting cables of both the 50-ton and the 5-ton hoists are also long enough to reach down through hatchways into the basement area. Hatches at several positions in the main and intermediate floors may be opened to allow passage of the cables and equipment.

The crane is used to install various pieces of equipment, handle the upper shield plugs and reactor vessel head, transport the fuel element shipping cask within the containment building, and for many other operations.

The crane is operated from a pushbutton station suspended from the trolley, and the control cables are long enough to allow the crane to be operated from the main floor at a distance of 20 ft from the lifting hook. All movements of the bridge, trolley, and hoists can be controlled from this station.

6.2. DESIGN DATA AND STRUCTURAL DESIGN

6.2.1 General

The design and construction of the containment vessel conforms to the applicable requirements of the 1962 edition of the ASME Boiler and Pressure Vessel Code, Section VIII, Unfired Vessels, and applicable code cases. The containment vessel will receive the ASME code stamp.

6.2.2 Design Conditions for Code Calculations

The design conditions for the code calculations are as follows:

(1) maximum internal pressure	52 psig
(2) maximum negative pressure	0.5 psig
(3) maximum temperature	280 F
(4) minimum temperature	-20 F
(5) welded joint efficiency	100%
(6) basic wind pressure	20 lb/ft ²

6.2.3 Materials

Forgings conform to ASTM A 350, LF1, and piping conforms to ASTM A 333. Containment vessel plates and reinforcements and their welded attachments, conform to the applicable requirements of ASTM A 201 ("Standard Specification for Carbon-Silicon Steel Plates of Intermediate Tensile Ranges for Fusion-Welded Boilers and Other Pressure Vessels") for Grade B steel. Plates also meet the applicable test requirements for ASTM A 300 ("Standard Specification for Steel Plates for Pressure Vessels for Service at Low Temperatures").

Drop weight tests on the 31 heats of ASTM A 201 Grade B materials used in the containment vessel indicate a nil ductility temperature range of -30 F to -60 F. Thermocouples are provided on the vessel external surface at locations given in Table 6-1 in order to ensure that the shell temperature is always above 0 F (NDT + 30).

TABLE 6-1

CONTAINMENT VESSEL THERMOCOUPLE LOCATIONS

thermocouple number	elevation	location
1	751' - 0"	on dome north centerline - about 10 in. above tank water level
2	756' - 6"	top center of dome
3	748' - 6"	on dome south centerline - about 18 in. below water tank connection to dome
4	722' - 8"	west centerline of cylinder - just above shutdown condenser penetration
5	725' - 6"	north centerline of cylinder
6	725' - 6"	south centerline of cylinder
7	689' - 6"	west centerline of cylinder
8	689' - 6"	north centerline of cylinder
9	689' - 6"	south centerline of cylinder
10	675' - 8"	at penetration M-33
11	661' - 0"	west centerline of cylinder
12	661' - 9"	at vacuum breaker penetrations M-32 and M-36
13	652' - 0"	north centerline of cylinder - above main lock
14	640' - 0"	north centerline of cylinder - below main lock
15	649' - 0"	on cylinder at north side of freight door
16	643' - 5"	on cylinder just below emergency lock
17	640' - 6"	south centerline of cylinder
18	637' - 9"	on cylinder just above pipe penetration area located in northeast quadrant

6.2.4 Construction

The containment vessel is airtight, with a maximum leakage of less than 0.1 percent of the contained volume per 24 hr at design pressure (52 psig). All detectable leaks will be repaired and made tight even though total leakage is less than the maximum permitted. The out-of-roundness does not exceed 0.5 percent.

6.2.5 Radiography

All butt welds were fully radiographed in accordance with Paragraph UW-51, ASME Boiler and Pressure Vessel Code. All welds of doors, nozzles, and opening frames, and all welds that could not be radiographed, were examined for cracks by the magnetic-particle or fluid-penetrant methods of inspection.

6.2.6 Testing

Requirements for the initial testing of the containment vessel included the following:

- (1) visual inspection of welding
- (2) pressurizing the containment vessel to 5.0 psig and performing soap bubble tests on all welds
- (3) pressurizing the shell to 59.80 psig, then reducing the pressure psig for final soap bubble tests on all welds at a vessel pressure of 52 psig. The 52 psig pressure was maintained for a 3-day period, with leakage measurements taken hourly. The inner chamber method was used to measure leakage.

After completion of internal equipment installation and the electrical penetrations, a final leak-rate test will be performed at 52 psig for three days.

6.2.7 Vacuum Breakers

Vacuum breakers are set to start opening at a 0.3 psig-vacuum and to be fully open at a 0.5-psig vacuum. They will remain tight at the 52 psig design pressure without damage to the seat, gasket, diaphragm, or casing. Each vacuum breaker is mounted inside a test chamber, which permits periodic testing of the vacuum breaker for leak-tightness (Fig. 6.6). During normal operation the upper half of the test chamber is jacked up from the lower half to permit air to flow into the vessel whenever the negative pressure exceeds 0.3 psig.

6.2.8 Contained Energy

The containment building can contain all the steam and water released from the primary system in the event of a major system rupture. Table 6-2 gives the mass and energy

balance for the primary system, including the reactor vessel, the recirculation loop, the decay heat removal loop, the condensate return line, and the steam line.

TABLE 6-2

PRIMARY SYSTEM MASS AND ENERGY BALANCE

mass of saturated water at 1300 psia	62,500 lb
mass of saturated steam at 1300 psia	2,753 lb
mass of water at 292 F	920 lb
total mass	66,173 lb
total contained energy	39.54×10^6 Btu

The free-air volume of the containment building is $264,160\text{ft}^3$. If all the energy of the primary system is assumed to be instantaneously released to the containment atmosphere at an initial temperature of 80 F, and if there are no heat losses from the building or heat absorption by internal structures, the resulting pressure and temperature would be 48.5 psig and 273 F. These values compare favorably to the design values of 52 psig and 280 F.

6.2.9. Containment Building Loads

Table 6-3 lists the approximate weights of the containment building and its contained equipment.

TABLE 6-3

CONTAINMENT WEIGHTS

<u>item</u>	<u>weight, lb</u>
containment vessel top head	147,200
overhead storage tank	29,300
water for overhead storage tank	350,000
containment vessel shell	953,000
containment vessel bottom head	199,500

Table 6-3 - Containment Weights (Cont'd)

<u>item</u>	<u>weight, lb</u>
concrete (150 lb/ft ³)	10,500,000
concrete (230 lb/ft ³)	3,650,000
reinforcing steel	579,000
spent fuel and water in storage well	320,000
overhead crane	88,000
reactor	530,000
airlocks	16,000
shipping cask	70,000
6000-gal retention tanks and contents	130,000
800-gal boron tank and contents	8,000
rotovalves	40,000
recirculation pumps (2)	80,000
miscellaneous equipment and live loads	2,424,000
primary water	<u>66,000</u>
total	20,180,000

6.3 LEAK TESTS

6.3.1 General

The overload test procedure fulfilled the requirements of the ASME Code, Section VIII, as modified by Code Case 1272 N. The leakage rate test was conducted in a manner similar to previously successful tests of containment vessels at Vallecitos, California; Dayton, Ohio; Dresden, Illinois; Lagoon Beach, Michigan; Indian Point, New York; Charlevoix, Michigan; and Elk River, Minnesota.

The method used for the leakage test consisted basically of comparing the pressure in the containment vessel with that in an airtight inner chamber. Location of the inner chamber approximately at the center of the air mass within the containment vessel allowed the average temperature of the air in the inner and outer vessels to be reasonably close during the daylight hours and practically equal during the late night hours. Data obtained from previous tests show that during the midnight to dawn periods of normal atmospheric conditions the air temperature becomes relatively uniform throughout the containment vessel, and that the temperature at the geometric center represents the average air temperature throughout the vessel. With no difference in average air temperature between the inner chamber and the containment vessel, the possibility of a pressure differential caused by temperature can be discounted. The inner chamber was proved tight by thorough preliminary inspection methods, and any relative decrease in containment vessel pressure under this temperature condition must be regarded as from external leakage. A high degree of sensitivity to this pressure differential can be achieved by use of a water manometer to measure the pressure differential between the two air volumes.

6.3.2 Preliminary Check

Preliminary testing was performed in the shop and field before the overload and leakage-rate tests at the site. All airlocks were shop tested for tightness and for operation of door mechanism, including the equalizing valves, and were found to be adequate. All shop-welded manholes and nozzles were magnafluxed inside and outside after shop stress relief; there was no indication of cracks or other defects. The reference chamber was shop-tested with freon to 60 psig after assembly, and was found to be tight.

Following the final installation of leakage rate equipment inside and outside the vessel, the reference chamber system was pressurized to 60 psig with freon and was checked with a halide leak detector; no leaks were detected. A secondary check with temperature-pressure data was conducted from October 8 to October 11, 1963. The recorded data indicate that the system was tight.

6.3.3 Overload Test

After the successful checking of the test equipment the vessel was closed for the overload test. No water was introduced into the differential manometer; this approach made it possible to allow the containment vessel and the inner chamber system to be pumped up simultaneously to the same pressure.

The vessel was pressurized to 5 psig on October 11. All vessel connections and welds were checked by a soapsuds inspection. Starting at 5 p.m. on October 11, the air pressure was increased in increments to the test pressure of 59.8 psig, which was reached at about 7:45 a.m. on October 12. After a 1-hr holding period, vessel pressure was reduced,

to the design 52 psig. The second and final soapsuds inspection was conducted at design pressure on October 12th. The outer door of each lock was left open.

The soapsuds inspection at 5 psig found several minor leaks in connections but no leaks in welded seams or in gaskets. Three leaks were in pipe caps on the No. 8-A insert and several leaks were in plugs on the electrical connections. All leaks were repaired and were then found to be tight under soapsuds inspection.

The overload test to 59.8 psig on the vessel and the airlocks was successfully completed.

The final soapsuds inspection at 52 psig found no leaks in welded seams. Very small leaks were found in temporary openings located in the vessel bottom. Several leaks were observed in plugs on electrical penetrations but were repaired and then found to be airtight by soapsuds inspection. Leaks were found around each of the 3-in. pipe caps on the No. 8-A insert, and all but one were repaired. A very small leak in the lower No. 8-5 cap could not be repaired.

6.3.4 Leakage Rate Test

Prior to the start of the leakage rate test, the blowoff valves at the bottom of the reference system and the containment vessel were opened. Approximately 50 gal of condensate were drained from the containment vessel and the pressure was adjusted so that maximum containment pressure would be ~52 psig during the holding period for the leakage rate test. Water was introduced into the differential manometer to approximately the mid-height of the scale. Air was then pumped into the containment vessel until the water manometer indicated a pressure about 10 in. higher than that in the inner chamber system.

The pressure and temperature readings were recorded hourly, starting at 6 p.m. on October 12. The data obtained after midnight were used for the leakage rate test. The holding period continued until 7 a.m. on October 17, when the test was accepted as successful. The data from the first night (October 13) were not considered typical and thus were not used for calculation of the leakage.

The differential manometer readings, for the midnight-to-dawn hours used for calculation of the leakage are summarized in Table 6-4.

Using the manometer data in Table 6-4 for the four successive nights, the calculated percent leakage per 24-hr period is as follows:

$$\frac{\% \text{ leakage}}{24 \text{ hr}} = \frac{1}{3} \times \frac{0.11}{129.80 \times 13.6} \times 100 = 0.002$$

TABLE 6-4

DIFFERENTIAL MANOMETER READINGS (Inches of Water)

<u>time</u>	<u>Oct.14</u>	<u>Oct.15</u>	<u>Oct.16</u>	<u>Oct.17</u>
midnight	9.74	10.23	9.16	9.45
1 a.m.	9.85	10.12	8.97	9.59
2 a.m.	9.82	9.76	9.90	9.57
3 a.m.	9.80	9.26	10.01	9.73
4 a.m.	9.76	9.40	10.05	9.69
5 a.m.	9.71	9.48	10.03	9.81
6 a.m.	<u>9.69</u>	<u>9.76</u>	<u>9.45</u>	<u>9.78</u>
average	9.77	9.72	9.65	9.66
initial ΔP - final ΔP =		0.05	0.12	0.11

The calculated leakage is well within the allowable value of 0.1 percent per 24-hr period. The measured tightness of the vessel is consistent with the results of the soapsuds inspection, since the extremely small leaks on temporary plugs or caps are insignificant in relation to the total contained air volume. The check of the reference system by both temperature-pressure data and by the halide leak detector indicated that no leaks were present.

6.4 CLOSURES AND PENETRATIONS

6.4.1 Airlocks

The personnel and emergency airlocks are basically steel cylinders, with bulkheads containing hinged, gastight doors at each end. The airlocks and doors are designed to remain gastight under a pressure of 52 psig from inside the containment shell and 1 psig from outside of the shell.

The airlock doors are manually operated. The operating mechanisms have the following characteristics:

- (1) When the doors are closed, a positive clamping force is exerted.
- (2) Before either door can be opened, a valve opens to equalize pressure on the opposite sides of the door.

(3) The doors are mechanically interlocked so that neither door can be opened unless the other door is closed.

(4) The outer door can be closed from inside the shell, and the inner door from outside the shell.

Leakage around the doors is prevented by a tongue-and-groove gasket arrangement, with silicone rubber in the groove. The shafts of the operating mechanisms are sealed by mechanical seals.

6.4.1.1. Personnel Airlock. The personnel airlock is 30 ft long overall and 13 ft in diameter, except at the penetration of the containment shell, where it tapers to a 10-ft diameter. The doors are 5 ft 6 in. wide and 7 ft high. The internal distance between doors is 21 ft 6 in. Figure 6.7 shows the mechanical design of the personnel airlock. The personnel airlock door interlock operates as follows: Handwheel A rotates shaft AG and parts B, AS, AK, and G. Interlock wheel AK locks interlock wheel AL and shaft AI through gears AU and AV and shafts BJ and BI. Handwheels C and D are now locked through shafts AH and AF and sprockets AP and AR. Handwheel E-1 is locked through shaft AJ.

The personnel airlock pressure-equalizing-valve mechanism operates as follows: Gear G engages gear H and turns shaft AA and parts I and T. Index wheel T engages and turns fork plate V which opens valve F through valve arm V. The valve is fully open after 1-1/4 turns of handwheel A. After the valve is open, index wheel T is "free wheeling" through the remainder of the opening cycle.

The personnel airlock doors are unlocked as follows: Gear I engages gear J and gear rack K, thereby moving locking bar L. Locking bars L, M, N, and O move as a unit, being tied together by chains P. Two and one-half additional turns of the handwheel move the locking bars so that cam followers BD move along wedge bars BE and unseat the door.

The personnel airlock exterior door is opened as follows: As the locking bar moves, gear BM is engaged and turns shaft AD and index wheel W. Throughout the valve-opening and door-unlocking cycles index wheel W was "free wheeling," but it is now ready to engage roller plate X. As it engages, roller plate X rotates shaft AE and, through miter gears S, turns shaft AC. Hinge clevis BN rotates and swings the door, acting through linkage BF.

A person inside the containment shell therefore leaves the shell through the personnel airlock by the following procedure:

- (1) Handwheel A is rotated five turns to open the interior door. The lock is entered.
- (2) Handwheel B is rotated five turns to close the interior door.

(3) Handwheel C is rotated five turns to open the exterior door. The person leaves the lock.

(4) Handwheel D is rotated five turns to close the exterior door.

If no pressure differential exists across the airlock doors, handwheel operation will continue uninterrupted. If a pressure differential does exist, however, a force of approximately 30 lb should be exerted on the handwheel until the differential is reduced enough to allow the door to be unseated. The time required for reducing the pressure differential varies with the pressure differential.

If it should become necessary for both doors to be open at the same time, the interlock can be disabled. A padlock ensures that the interlock is not disabled without proper authorization.

Opening both personnel airlock doors prevents reactor startup, and opening either door actuates a visual alarm in the control room.

The procedure for disabling the personnel airlock interlock is as follows:

- (1) The key to the disabling mechanism is obtained.
- (2) The interior door is opened in the usual manner.
- (3) Padlock BQ on the inside of the interior door is removed by use of the key.
- (4) Pin BG is removed and link BF is disengaged. Handwheels A and B can now be operated without swinging the interior door.
- (5) With the interior door in the open position, handwheel A is moved to the fully closed position. The interlock is now in the neutral position and the exterior door can be opened in the normal manner. There must not be any attempt to shut the interior door with the mechanism in the fully closed position, because the locking ring will interfere with the door wedges.

When it is no longer necessary for both personnel airlock doors to be open the interlock should be reset immediately. The procedure for resetting the interlock is as follows:

- (1) The exterior door is returned to the closed and sealed position.
- (2) Handwheel A is moved to the open position.
- (3) The interior door operating link BF is connected to the door hinge clevis by inserting pin BG.
- (4) Pin BG is locked with padlock BQ.

(5) The normal operating procedure for the door is run through to make certain the doors and interlock are operating properly.

(6) If the doors and interlock operate satisfactorily the key is returned.

6.4.1.2. Emergency Airlock. The emergency airlock is 7 ft long and 5 ft in diameter. The doors are circular; their diameter is 32-1/2 in. (30-in. opening). Figure 6.8 shows the mechanical design of the emergency airlock.

The emergency airlock operates as follows:

(1) As handle A is pulled downward, shaft AE rotates. This shaft, acting through miter gears R, turns shaft AC. In turn, interlock wheel AJ rotates and locks interlock wheel AK, preventing rotation of shaft AA and, in turn, of shafts X, W, AN, Y, and Z. All these shafts act as one shaft through miter gears R or couplings AI. Locking of these shafts renders exterior door handles D, C, and E-2 inoperative, thereby ensuring that the exterior door remains closed and sealed.

(2) Shaft AC also turns spur gear U, engaging partial gear ring Q mounted on the locking ring N and causing locking ring N to rotate. Another spur gear, S, engaged to gear ring Q, turns shaft AF, which turns index wheel J. Index wheel J turns fork plate K, thereby actuating valve F through levers G and I and arm H. After 30 deg of handle travel the valve is fully opened and the locking ring has rotated to the proper point to unseat the door from its gasket. Index wheel J disengages from fork arm K and is "free wheeling" for the remainder of the opening operation.

(3) During the next 60 deg of handle travel, locking ring N unseats the door from the gasket as cam followers O move along the wedge bars AQ. Index wheel AO has been "free wheeling" since the start of the opening cycle, having turned as partial gear ring P (engaged to spur gear T) moved shaft V through miter gears R. Index wheel AO is now ready to engage fork plate AP.

(4) As the handle moves through the final 30 deg of travel, index wheel AO, by engaging and rotating fork plate AP, turns the door operating arms L, M, and AH and swings the door on its hinge to the open position.

A person inside the containment shell therefore follows this procedure to leave the shell through the emergency airlock:

(1) Handle A is pulled through an arc of 120 deg to open the interior door. The lock is entered.

(2) Handle B is pulled through an arc of 120 deg to close the interior door.

(3) Handle C is pulled through an arc of 120 deg to open the exterior door. The person leaves the lock.

(4) Handle D is pulled through an arc of 120 deg to close the exterior door.

If no pressure differential exists across the airlock doors, the handle operation will continue uninterrupted. If a pressure differential does exist, however, a force of approximately 30 lb should be applied to the handle until the differential is reduced enough to allow the door to be unseated. The time required for reducing the pressure differential varies with the magnitude of the pressure differential.

If it should become necessary to have both doors open at the same time, the interlock can be disabled. A padlock ensures that the interlock is not disabled without proper authorization.

The procedure for disabling the emergency airlock is as follows:

- (1) The key to the disabling mechanism is obtained.
- (2) The interior door is opened in the usual manner.
- (3) The key is used to remove padlock AS.
- (4) Pin AL is removed and arm M is disengaged. Handles A and B can now be operated without swinging the interior door.
- (5) With the interior door in the open position, handle A is moved upward to the fully closed position. The interlock is now in the neutral position and the exterior door can be opened in the normal manner. There must be no attempt to shut the interior door with the mechanism in the fully closed position, because the locking ring will interfere with the door wedges.

When both emergency doors no longer need to be open, the interlock should be reset immediately. The procedure for resetting the interlock is as follows:

- (1) The exterior door is returned to the closed and sealed position.
- (2) Handle A is pulled downward to the open position.
- (3) Door operating arm M is connected to door lever AH by inserting pin AL.
- (4) Pin AL is locked with padlock AS.
- (5) The normal operating procedure for the door is run through to make certain the doors and interlock are operating properly.
- (6) If the doors and interlock operate satisfactorily, return the key.

6.4.2 Freight Door

The freight door consists of a flanged rectangular frame extending through the containment building wall, and a plate door bolted to the frame flange on the inside of the building. The door opening is 8 ft wide and 10 ft high. The door is designed for an internal pressure of 52 psig and an external pressure of 1 psig.

The door frame forms a nozzle approximately 1-1/2 ft long, extending on either side of the building wall. The part of the frame extending outside the building is strengthened by triangular-shaped metal braces regularly spaced around the frame and welded to the building wall and to the frame.

The part of the frame extending inside the building is similarly braced. The inner braces are welded on three of their four edges; one edge is welded to the door frame, one to the building wall, and one to a support box frame that extends around the door 3 ft from the door opening. The support frame is welded to the inside of the containment shell.

The door is bolted to a machined flange that is welded around the lip of the frame nozzle inside the building. Two parallel grooves machined around the flange face accommodate silicone rubber gaskets which provide a sealing surface for mating parallel tongues on the door flange. Periodic testing for seal leak-tightness can be made by pressurizing between the two gaskets.

The door is a flat metal plate 11 ft high and 9 ft 4 in. wide, strengthened on the inward side by an arrangement of horizontal and vertical braces running the full width and height of the door. Welded to the outward side of the door is a machined flange with parallel tongues that mate with the grooves in the frame flange. Since the height of the tongues is less than the depth of the grooves, the door flange may be fully seated on the frame flange without damaging the gaskets. The door is secured by hinged bolts, attached to the frame, which mate with open-ended lugs welded to the door flange. When the door is in place, the hinged bolts are swung into the open ends of the lugs, and nuts are tightened in a prescribed order.

Since the door is flanged to the inside of the door frame, increased pressure in the building tends to tighten the seal.

The door swings open on an arm attached at one end to the center of the door and at the other to the door frame. The arm is hinged at both ends, and the hinges permit vertical and horizontal adjustments in the door position to ensure proper seating.

6.4.3 Electrical Penetrations

All electrical penetrations are in the northwest quadrant of the containment shell. The penetrations are of two types and are grouped into six sections or units. Units 3

through 6 are used for 120-v a-c and 125-v d-c power and control circuits, and for all instrumentation leads except those to thermocouples and those requiring coaxial cables. Unit 2 is used for the 480-v a-c and 2400-v a-c penetrations, and Unit 1 is used for thermocouple leads, coaxial leads, and special multiconductor penetrations.

Units 2 through 6 employ the MI cable-type penetration. This type has solid conductors, compacted magnesium oxide insulation, and a round copper sheath. For Units 3 through 6 the MI cable runs through an inner chamber formed by the containment shell and an inner chamber wall; the chamber for each unit is 8 ft 8 in. high with a cross section 3 in. deep and 8 in. wide. The penetration is made by passing the MI cable through brass glands which are screwed into 3/4-in. tapped holes in the containment shell and in the inner chamber wall. The copper sheath is sealed at the gland, and the inner chamber is periodically pressurized to leak test the penetrations. Units 3 through 6 each have 66 cable penetrations of 7-conductor 12-gauge cable which terminate at terminal boards mounted on racks inside and outside of the containment shell.

Unit 2 uses the MI cable-type penetration in a similar fashion, but with different details of chamber construction. Unit 2 has five small chambers, four of which are used for 480-v a-c power cable. There are eight cable penetrations in each of three of these chambers, and the fourth is a spare. The fifth chamber has six cable penetrations for the 2400-v power cable to the forced-circulation pump motors. These 2400-v penetrations do not terminate at terminal boards but are directly connected to the cable that runs to the forced-circulation pump motors inside the containment and to the 2400-v switchgear outside the containment.

Unit 1 uses the second type of penetration, which consists of hermetically sealed bulkhead connectors. The connectors are mounted on the outer and inner walls of the chamber. There are ten coaxial penetrations for the signal leads from the in-core flux monitoring system, 32 for the nuclear instrumentation system, and three for the containment building ventilation radiation monitor signal leads. The inner chamber wall and outer chamber wall for the coaxial penetrations are micarta-plate to insulate the shell of the connector from the ground. Hermetically sealed multipin connectors are used for six penetrations for thermocouple leads, nine penetrations for the in-core monitor digital position readout, and ten penetrations for the secondary position control-rod readout lights.

6.4.4 Piping and Mechanical Penetrations

Piping and mechanical penetrations (Figs. 6.9 and 6.10) leave the containment shell in the northwest quadrant (Area 1) 1 to 10 ft below ground level. The piping then goes to a tunnel that leads to the turbine building.

The penetrations of the east side (Area 2) of the containment shell leave 1 to 5 ft below ground level and enter a tunnel connecting the turbine building, reactor building, stack, and the waste treatment and waste gas storage areas.

Both Area 1 and Area 2 have spare penetrations with pipe caps welded on one end. Penetrations are also provided in the main airlock, the emergency airlock, and at other locations.

Containment penetrations are listed in Table 6-5.

TABLE 6-5

CONTAINMENT PENETRATIONS

Area 1

<u>no.</u>	<u>description</u>	<u>size (in.)</u>	<u>schedule</u>
1	main steam	20 OD	(1/2 in. wall)
2	feedwater	14 OD	(1/2 in. wall)
3	high pressure service water	3	40
4	component cooling supply	10	40
5	component cooling return	10	40
6	demineralized water	3	40
7	control air	1-1/2	80
8	station air	1-1/2	80
9	spare	1	80
10	spare	1	80
11	spare	1	80
12	startup water removal	3	40
13	condensate to seal injection reservoir	1-1/2	80
14	shutdown condenser vent	1	80
15	domestic water	1	80

Area 2

1	spare	1	80
2	vent air exhaust	20 OD	(1/2 in. wall)
3	retention tank discharge to circulating water discharge	1-1/2	80
4	resin sluice line	1-1/2	80
5	spare	1-1/2	80

Table 6-5 - Containment Penetrations (cont'd)

<u>no.</u>	<u>description</u>	<u>size (in.)</u>	<u>schedule</u>
6	sump pump discharge to retention tanks	1-1/2	80
7	heating steam and condensate return	12	40
8	retention tank discharge to evaporator feed tank	1	80
9	spare	1	80
10	vent header	3	40
11	spare	1	80

Western Quadrant (el 722 ft)

<u>no.</u>	<u>description</u>	<u>size (in.)</u>	<u>schedule</u>
1	shutdown condenser steam outlet	14 OD	(1/2 in. wall)

Northwest Quadrant (above ground level)

<u>no.</u>	<u>description</u>	<u>size (in.)</u>	<u>schedule</u>	<u>elevation</u>
1	vacuum breaker	6	40	661 ft 6 in.
2	vacuum breaker	6	40	661 ft 6 in.
3	vent air intake	24 OD	1/2 in. wall	646 ft 9 in.
4	spray water handwheel	1-1/2	80	675 ft 6 in.

Main Airlock

<u>no.</u>	<u>description</u>	<u>size (in.)</u>	<u>schedule</u>
1	door operators	1	80
2	door operators	1	80
3	door operators	1	80
4	door operators	1	80

Table 6-5 - Containment Penetrations (cont'd)

Emergency Airlock

<u>no.</u>	<u>description</u>	<u>size (in.)</u>	<u>schedule</u>
1	door operators	1	80
2	door operators	1	80
3	door operators	1	80
4	door operators	1	80

Piping penetrations in the overhead storage tank are listed in Table 6-6.

TABLE 6-6

OVERHEAD STORAGE TANK PENETRATIONS

<u>no.</u>	<u>description</u>	<u>size (in.)</u>	<u>schedule</u>
1	spray water tank outlet	8	40
2	tank overflow	3	40
3	fill	6	40
4	drain	3	40
5	fuel flooding	6	40

6.5 ISOLATION VALVES

Containment integrity must be established within a short time after a major system rupture to prevent escape of fission products from melted fuel elements to the outside atmosphere. Valves in the piping and duct work that penetrate the containment vessel assure containment integrity in the event of a rupture and permit isolation of the containment atmosphere when desired (Figs. 6.9 and 6.10). The criteria for the number, placement, and operation of the valves are as follows:

(1) Two valves in series are used at containment penetrations for piping connected directly to the primary system. At least one of these valves closes automatically following an accident, unless one is normally closed during plant operation. The second valve is operable either from the control room or from some other location which will be accessible after an accident. The valves are on different sides of the containment building wall.

(2) Two isolation valves in series are also used for any piping open to both the building atmosphere and the outside atmosphere. Unless one of the valves is normally closed, at least one of the valves must be a quick closing fail-safe valve interlocked to prevent reopening until conditions are safe. The second valve is operable either from the control room or some other location accessible after an accident.

(3) Piping that is open to the containment building atmosphere but that leaves and re-enters the containment building without being open to the outside atmosphere does not require isolation.

(4) Piping that is open to the outside atmosphere but that enters and leaves the containment building without being open to the containment building atmosphere or the primary system does not require isolation. Isolation is required for this type of penetration, however, if its chances for rupture are significantly increased by the conditions that require containment isolation.

Piping penetrations open to the primary system are the 10-in. main steam line, the 8-in. feedwater return line, the 2-in. blowdown line from the decay heat system to the main condenser hotwell, and the shutdown condenser off-gas vent line. The steam line has a rotoport valve in the containment building that can be manually controlled from the control room and that closes automatically within 10 sec after receiving a signal caused by low steam pressure at the turbine stop valve, low vacuum in the main condenser, or low reactor water level. A second main steam line isolation valve in the turbine building is motor operated by a signal from the control switch in the control room.

The 8-in. feedwater return line has both a check valve and a shutoff valve. The check valve is in the containment building and fulfills the requirement for automatic closure. The shutoff valve is in the turbine building and has a valve operator that is actuated from the control room.

The 2-in. blowdown line (3-in. at the containment building penetration) from the decay heat system has a control valve in the containment building and manual valves in the containment building and piping tunnel area. The control valve and the manual valve in the piping tunnel area are normally closed during plant operation. In addition, the control valve can be operated from the control room and is automatically closed upon a low reactor water level signal from the reactor safety system.

A 3/4-in. line connecting to the blowdown line in the piping tunnel area contains a check valve to prevent primary water from backing into the line.

The shutdown condenser system contains control valves which normally isolate the system from the primary system when the plant is operating. When the shutdown

condenser is operating and the off-gas vent control valve is open, the off-gas passes to the radioactive waste gas disposal system, which is designed for containment and automatic control of the radioactive gases. A manual globe valve in the off-gas vent line in the piping tunnel area permits manual isolation of the line.

Penetrations which are open to both the containment building and outside atmospheres include the ventilation inlet and exhaust ducts, the off-gas vent header from the containment building, and the two vacuum breaker lines. Each ventilation duct contains two quick-closing butterfly valves (Sec. 6.6.4) in series. All these valves can be manually controlled from the control room and will close automatically within 10 sec after receiving a signal caused by high-containment-building pressure or high reactor pressure, or upon a high-radiation signal from the gaseous or particulate monitors in the ventilation exhaust duct. These valves are spring-loaded and are held open by a pneumatic cylinder so that they fail closed on loss of air.

The off-gas vent header from the containment building contains a control valve on each side of the containment vessel wall. The control valve inside the containment building is automatically closed upon a signal caused by high reactor pressure or high pressure of the containment building atmosphere. The control valve in the piping tunnel area can be manually closed from the control room.

The piping penetrations for the vacuum breakers are an exception to the criteria (Sec. 6.5(2)) since shutoff valves are not placed in these lines. Valves in these lines would constitute a hazard to containment vessel integrity should an excessive external differential pressure exist during an inadvertent closing of a building isolation valve. In lieu of isolation valves, provision has been made for periodic leak testing of the vacuum breakers under pressure (see Sec. 6.2.7 and Fig. 6.6.)

Figures 6.9 and 6.10 show valving for other containment shell piping penetrations.

6.6 HEATING, VENTILATING, AND AIR CONDITIONING SYSTEMS

6.6.1 Air Conditioning Design Conditions

The air conditioning equipment for the containment building conforms to the following design conditions:

cooling capacity	two 30-ton units
heating capacity	675,000 Btu/hr (total)
air volume capacity	24,000 cfm (total)
recirculating air volume	~19,000 cfm (total)
outside winter temperature	-24 F
inside winter temperature	70 F

outside summer temperature	95 F
inside summer temperature	78 F
condensing water temperature	90 F (max)
condensing water pressure	70 psig (max)
steam supply pressure	125 psig (max)
operating steam supply pressure	25 psig

Refrigeration compressors are equipped with crankcase heaters. An automatic flow control valve is provided in the cooling water line to maintain the temperature of the water leaving the condensers at 115 F.

6.6.2 Automatic Temperature Control

The air conditioning systems have completely automatic temperature control. The design of the automatic temperature control systems meets the summer and winter requirements at the design recirculation flow.

The containment vessel air conditioning and heating system provides the following control features:

- (1) an electric interlock to prevent operation of the cooling compressor unless the supply fan is in operation and condensing water is available.
- (2) a manual "Summer-Winter" selector switch to change from cooling cycle to heating cycle
- (3) a low-temperature-protection thermostat in the air exhaust from the coils to stop the supply fan if the temperature falls below its setting
- (4) a low-limit thermostat in the air discharger to maintain discharge temperature in winter operation
- (5) a thermostat in the return air duct to maintain building air temperature
- (6) a fan and bypass damper motor
- (7) a two-position automatic steam valve

When the selector switch is turned to "summer" position, the steam valve is closed tight, the low-limit thermostat in the air discharge is eliminated from the control circuit, and the cooling equipment circuit is energized. The return-air thermostat modulates the fan and bypass dampers to maintain building air temperature. When the fan dampers are in full-open position, the cooling equipment is started.

When the selector switch is turned to "winter" position, the cooling equipment circuit is de-energized and the steam valve is open. The low-limit thermostat in the air discharge and the return air thermostat controls the fan and bypass dampers to maintain building air temperature.

6.6.3 Filters

The containment building exhaust air system is equipped with high-efficiency filters as well as conventional filters. The high-efficiency filters are individually tested and certified for an efficiency of not less than 99.97 percent under test with 0.3 micron dioctylphthalate smoke. The clean-filter static pressure drop will not be greater than 1-in. of water for operation at a rated capacity of 1100 cfm at STP. Since the exhaust air flowrate will be set at approximately 5000 cfm, the flowrate through each of the six high-efficiency filters will be approximately 833 cfm.

6.6.4 Ventilation System Isolation Dampers

The containment building ventilation system has five 20-in. flanged, viton-seated, butterfly-valve-type isolation dampers. Each damper has a spring-loaded air cylinder operator, two limit switches, and a three-way solenoid-actuated control air valve.

The dampers are located inside the containment vessel in the ventilation system air intake and exhaust ducts that pass through the containment vessel. Four of the dampers will be used in the event of an accident to shut off and seal the plant ventilating air system, preventing the release of radioactive fission products to the atmosphere. Two of these four dampers are in the inlet duct and two are in the exhaust duct; the second damper in each duct provides backup protection if one of the dampers fails. The fifth damper is located in the recirculation duct downstream of the exhaust fan and is normally closed. It automatically opens when the building isolation dampers are closed, to permit air recirculation in the pump cubicles.

The spring-loaded cylinder operators are designed for quick opening or closing of the damper when actuated. Each cylinder is operated by air from the control air system at 75 to 100 psig, through an electrically latched, manually reset, 3-way solenoid valve rated for 115-v d-c continuous service. The solenoid valves will normally be energized to air-load the butterfly-valve air cylinder operators and to keep open the butterfly valves, except for the air recirculation valve, which is closed under normal conditions. When the solenoid valves are de-energized, they act to vent the air cylinder operators and allow the spring loading quickly to close or open the butterfly valves as required. The design criteria for closing the four isolation dampers is 10 sec after receipt of signal. The procurement specification, however, provides for a valve and valve operator system that closes the damper within 2 sec.

The solenoid valves are de-energized to operate butterfly valves for any of the following conditions:

(1) high activity measurement by the gaseous monitor or either particulate monitor sampling the exhaust duct air leaving the building

(2) high reactor pressure (Sec. 8.5.2.5.2)*

(3) high pressure in the containment building*

(4) remote manual operation from the control room

(5) loss of electrical power supply to the solenoids

* Conditions (2) and (3) above also automatically close the 4-in. vent header.

Each valve has two limit switches, one to be operated when the damper is fully open and the other when the damper is fully closed. Contacts are rated at 125-v d-c service.

The system is designed so that loss of air supply, as well as loss of electrical power, will cause closure of the four containment isolation dampers and will open the ventilation bypass damper for the pump cubicles. The equipment is designed to withstand, and to operate at, the maximum containment vessel pressure of 52 psig. The dampers are capable of repeated sealing against a differential pressure of 60 psi from either side. The design will accommodate a maximum indoor temperature of 280 F and a simultaneous outdoor temperature of -27 F when dampers are closed. The isolation dampers in the intake and exhaust ducts can be periodically leak tested by pressurizing between the two dampers in each duct.

6.7 BUILDING SPRAY SYSTEM

A building spray system is provided to condense vapors which would result from a primary system rupture and to partially remove released halogens and solid fission products from the containment atmosphere. Water is supplied to the building spray system from a 42,000-gal storage tank located at the top of the containment vessel. The piping connection to the emergency core spray system is on the bottom of the tank. The connection to the spray headers of the building spray system is a standpipe within the tank. The bottom of the standpipe is at a sufficient elevation above the bottom of the tank so that 15,000 gal of water is available for the emergency core spray system at all times except during refueling. The storage tank provides water for use during the refueling operation and during loading of the fuel element shipping cask.

Table 6-7 summarizes water volumes in the overhead storage tank during the various operations.

TABLE 6-7
WATER IN OVERHEAD STORAGE TANK

<u>specific condition</u>	<u>reactor operating conditions</u>	<u>water available for emergency cooling, gal</u>	<u>water available for building spray, gal</u>
loading of fuel element shipping cask	full power	15,000	11,300
refueling	zero power, with head off	0	0
all other times	full power	15,000	27,000

Building spray is delivered by gravity feed at ~ 1000 gpm to the spray headers. Operation of the spray system after the MCA would reduce the containment pressure and temperature and would also wash out, at least partially, released halogens and solid fission products from the containment atmosphere.

The overhead storage tank is supplied with plant demineralized water. A level indicator is provided in the control room, and level switches actuate high and low level alarms. The level switches also open and close the valve in the makeup line from the demineralized water supply. Water for the emergency core spray system will be available at all times except during refueling, when it is not needed. During plant operation, the only time the full 27,000 gal amount of water is not available for building spray is during loading of the fuel element shipping cask. For this condition, approximately 15,700 gal of water is required to raise the level of the fuel element storage well.

The valve that permits water to flow to the building spray headers can be opened either from the control room or from inside the containment building.

6.8 CRANE AND HOIST

The bridge and trolley permits the 50-ton bridge crane to reach any position on the main floor under the trolley travel span. The crane hook can reach horizontally to any point beyond 4 ft of the crane rail and can be raised to within 2 ft 6 in. of the crane rail elevation. Design characteristics are given in Table 6-8.

TABLE 6-8

DESIGN CONDITIONS FOR BRIDGE CRANE

1.	span	54 ft 0 in.
2.	wall clearance to centerline of crane rails	1 ft 0 in.
3.	distance from main floor to top of rails	28 ft 6 in.
4.	distance from basement floor to top of rails	107 ft 6 in.
5.	lowest position of main hook below top of rails	92 ft 6 in.
6.	lowest position of auxiliary hook below top of rails	92 ft 6 in.
7.	bridge speed	50 ft/min (max)
8.	trolley speed	2 ft/min (max)
9.	main hoist speed	12 ft/min (max)
10.	auxiliary hoist speed	25 ft/min (max)
11.	main hoist rated capacity	50 tons
12.	auxiliary hoist rated capacity	5 tons
13.	size of runway rails	80 lb A.R.E.A.

The operation and control of the crane is through a pendant-type pushbutton station suspended from the trolley and connected to a torque reel so the crane can be operated from the main floor at distances up to 20 ft from the hook.

The weight of the pendant station is carried by a steel messenger wire, not by the conductors of the control cable. The pendant station is grounded to the crane by a separate conductor within the control cable.

General characteristics regarding the crane are as follows:

- (1) The crane is center-reeving to provide straight vertical lifts.
- (2) Access from the plant floor to the crane footwalks is by a ladder attached to the containment building wall.
- (3) A footwalk along one side of the girder is provided with stairs to a platform at the center of the girder. The platform is level with the top of the girder.
- (4) The main hoist has a sister hook with an eye for single sling center lifting. The auxiliary hoist has a single hook provided with a latch.
- (5) The main hoist brakes consist of two brakes: an automatic, electric-release, spring-set, shoe-type brake with a 150 percent torque rating, installed on the motor shaft; and an automatic mechanical-load, shoe-type brake interposed in the mechanism between the power source and the winding drum.

(6) Crane component design, materials, fabrication, testing, and inspection comply with the requirements of the American Standard Safety Code for cranes, derricks, and hoists.

6.8.1 Motor Design

Motors are of the horizontal, variable-speed, induction type, with wound rotor and slip rings, rated at 440-v, 3-phase, 60 cycle. The main hoist, auxiliary hoist, and trolley drive motors are rated respectively at 50, 10, and 3 horsepower. The motors deliver their rated horsepower continuously without damage when the terminal voltage is 10 percent above or below rated voltage.

The rated horsepower of the motors is 120 percent of the maximum brake horsepower required by the driven equipment. This relation exists for all operating speeds and conditions. The motors can develop a starting torque and a breakdown torque of at least 275 percent of the torque that corresponds to the rated horsepower at full-load speed when operating on normal voltage and frequency.

6.8.2 Control Equipment

A 3-pole air circuit breaker with a 15,000 amp interrupting capacity is provided for the main 440-v power feed. A 3-pole magnetic contactor on the load side of the circuit breaker provides overload and undervoltage protection.

There are magnetic reversing contactors for primary control of each motor and resistors, auxiliary relays, etc. for secondary control. Each hoist motor control is provided with speed control.

A dry-type transformer is provided for lighting and control power. The transformer is rated at 480 - 120 v and is fed from the load side of the main feed circuit breaker. Air circuit breakers are provided for the load circuits.

Limit switches are provided for the up and down travel limits of each hoist.

6.8.3 Lighting

Four 500-w prismatic low-boy fixtures illuminate the area beneath the crane. The fixtures have a shockproof mounting. The lamp voltage is 115 v.

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ENGINEERS
CHICAGO, ILLINOIS
FOR

ALLIS-CHALMERS
MANUFACTURING COMPANY
ATOMIC ENERGY DIVISION
NUCLEAR POWER DEPARTMENT
WASHINGTON, D. C.

TITLE
GENERAL ARRANGEMENT
PLAN-GRADE FLOOR
LACBWR PROJECT
U.S. ATOMIC ENERGY COMMISSION
GENOA, WISCONSIN

DWG. NO. 41-503302

REV.	6
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FIG. 6.2

 DWG. 41-503304

DRWG 41-503305

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GENOA, WISCONSIN**

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FIG. 6.3

41-503304

<p>SARGENT & LUNDY ENGINEERS CHICAGO, ILLINOIS [FOR]</p>	
<p>ALLIS-CHALMERS MANUFACTURING COMPANY ATOMIC ENERGY DIVISION NUCLEAR POWER DEPARTMENT WASHINGTON, D. C.</p>	
<p>TITLE GENERAL ARRANGEMENT CROSS SECTION "A-A" LACBWR PROJECT U.S. ATOMIC ENERGY COMMISSION GENOA, WISCONSIN</p>	
<p>DWG. NO. 41-503304</p>	<p>REV. 6</p>

FIG. 6.4

FIG. 6.5

VACUUM BREAKER AND TEST CHAMBER

FIG. 6.6

PERSONNEL AIRLOCK

FIG. 6.7

EMERGENCY AIRLOCK

FIG. 6.8

FIG. 6.9

ALLIS-CHALMERS MANUFACTURING COMPANY ATOMIC ENERGY DIVISION NUCLEAR POWER DEPARTMENT WASHINGTON, D. C.	
LACBWR REACTOR PLANT CONTAINMENT VESSEL PENETRATIONS DIAGRAM	
DWG. NO. 41-300-088 SHEET 2 OF 2	REV.

FIG. 6.10

7. POWER PLANT

7.1 GENERAL LAYOUT

The general arrangement of the reactor and turbine buildings is shown in Fig. 3.3.

The turbine building contains a major part of the power plant equipment. The turbine-generator is on the main floor. Other equipment is located below the main floor.

This equipment includes the feedwater heaters, reactor feedwater pumps, air ejector, vacuum pump, full flow demineralizers, off-gas compressor and cooler, condensate pumps, air compressors, air dryer, oil purifier, service and component cooling water coolers and pumps, make-up water demineralizer system, domestic water heater, turbine oil reservoir, oil tanks and pumps, turbine condenser, unit auxiliary transformer, 2400 volt and 480 volt switchgear, motor control centers, diesel engine-generator set, emergency storage batteries, inverter set and other electrical, pneumatic, mechanical and hydraulic systems and equipment required for a complete power plant. A 30/5-ton capacity, pendant-operated overhead electric traveling crane spans the turbine building. The crane has access to major equipment items located below the floor through numerous hatches in the main floor.

The turbine building also contains the main offices, the control room (for both turbine-generator and reactor, locker room facilities, laboratory, shops, counting room, personnel change and decontamination facilities, heating, ventilating and air conditioning equipment, rest rooms, storeroom, and space for other plant services. In general, these areas are separated from power plant equipment spaces. The control room is on the main floor on the side of the turbine building that is adjacent to the containment building.

The general arrangement of the turbine building is shown in Figs. 10.1, 10.2 and 10.3. All persons entering the turbine building from the containment building main air lock must proceed along a ramp and stairway to a check-out, decontamination, and change room facility on the turbine building mezzanine floor. Similarly, all persons leaving controlled areas of the turbine building must pass through the same station.

Miscellaneous structures which are associated with the power plant and are located adjacent to the turbine building include the electrical switchyard, crib house, oil pump house, warehouse, construction and contractor office buildings, outdoor fuel oil and acid storage tanks, underground septic tanks and condenser circulating water discharge seal well.

Miscellaneous on-site improvements include roads, walks, parking areas, yard lighting, fire loop hydrants required for plant protection, access to and use of rail siding facilities, fencing, landscaping, and communication services.

7.2 TURBINE

The steam turbine is an Allis-Chalmers high-pressure condensing, straight reaction, tandem compound machine. It consists of a 2600 rpm single flow high and intermediate pressure element with a double flow low pressure element arranged to provide opposed steam flow in the low pressure element.

The turbine rotor is connected to the generator by a rigid coupling. The generator is connected to the main exciter by reduction gearing. The turbine-generator is oriented so that the axis of rotation of the rotors would, if extended, pass through the containment building.

The turbine condenser is mounted on the grade floor directly under the turbine exhaust. The horizontal axis of the condenser is perpendicular to the horizontal axis of the turbine. The expansion between the condenser and the turbine is absorbed by a flexible rubber belt expansion joint between the condenser neck and the turbine exhaust flange.

The turbine is designed for dry saturated steam at a throttle pressure of 1265 psia with a steam flow of 703,919 lb/hr from the reactor and an exhaust pressure of 1 in. mercury absolute. The turbine generator is capable of producing a gross electrical output of 64,537 kw and a net electrical output of 60,000 kw.

7.3 GENERATOR

The generator is an Allis-Chalmers two-pole, sixty cycle, three-phase, 13.8 kv hydrogen-cooled machine, rated at 76,800 kva at an 85 percent power factor with a hydrogen pressure of 30 psig. A shaft-driven 1207 rpm main exciter rated at 300 kw supplies 250 volts d-c for field excitation. A reserve 2300 volt motor-driven 1190 rpm exciter, similarly rated, is provided. A rotating amplifier voltage control set is located in the excitation cubicles on the turbine hall main floor. Cooling for the generator is provided by the hydrogen cooling system. Fans on each end of the rotor shaft circulate the hydrogen through the stator and rotor windings. Water-cooled heat exchangers in the generator casing cool the hydrogen.

7.4 ELECTRICAL

The generator neutral is grounded through a 50 kva transformer with a resistor in its secondary.

An isolated phase bus duct of 3200 amp capacity, rated at 14.4 kv, connects the generator to the step-up main power transformer. The three-phase, Class FOA, main power transformer is rated at 70,000 kva and 13.2/69 kv. It is set on a concrete pier

in the electrical switchyard. Lightning arrestors rated at 60 kw are mounted on the transformer. Fire protection for the transformer consists of hydrant and hose connections.

7.5 CONDENSER AND AUXILIARIES

The condensing equipment includes a 70,000 ft² two pass divided water box surface condenser, rated at 4.23×10^8 Btu/hr with a circulating water flow of 62,600 gpm at 60 F inlet temperature; two 32,000 gpm circulating water pumps; two 15,000 gpm condensate pumps; full flow condensate demineralizers; a motor-driven dry vacuum pump; and a steam-jet air ejector. The vacuum pump is used during startup and the steam-jet air ejector is used during normal operation. The condenser is supplied with circulating water from two pumps in the crib house inlet at the river. The water is discharged downstream of the crib house. A steam bypass system is provided from the main steamline to the turbine condenser. This system is available if required during reactor startup and shut down or during turbine or generator trip-out. A hydraulically-operated dump valve controls the bypass. The bypass line is designed for 100 percent of full throttle flow. The condenser hotwell has a 4,000 gallon capacity for control of system water inventory. Non-condensable gases removed from the condenser by the air ejector are discharged through a holdup tank to the stack. The off-gases are monitored so that if the radioactivity exceeds permissible levels, the gases are pumped to storage tanks which provide for 72 hr holdup at full power operation (Sec. 5.3.3.1). The gases are passed through a recombiner to eliminate the dissociated hydrogen and oxygen before being pumped to the storage tanks.

7.6 FEEDWATER HEATERS

Three closed-type feedwater heaters are provided in the condensate feedwater system. Low pressure heaters No. 1 and 2 are located in the neck of the condenser. Extraction steam from the low pressure turbine heats the condensate. Steam and condensate from the flash tank and moisture separator heat the feedwater entering the No. 3 heater.

7.7 REACTOR FEEDWATER PUMPS

The reactor feedwater pumps are Allis-Chalmers pumps of the horizontal, 5-stage, centrifugal type rated at 3150 ft head at 1580 gpm. Each pump is driven through a variable speed fluid coupling by a 1500 hp 2400-v, 3600 rpm induction motor. The rated full load speed for the motor is 3575 rpm.

The fluid coupling is rated for 1500 hp at 3585 rpm. Basically it consists of an impeller, a runner, a scoop tube, and a rotating casing that encloses the impeller and runner. These components are in a stationary outer housing that also serves as a water reservoir. The impeller and rotating casing are connected to the motor shaft which is driven at a constant speed by the pump motor. The runner is connected to the pump shaft and rotates at a speed

governed by the quantity of water in the rotating casing. The fluid coupling water system supplies water at a uniform rate through the inlet pipe and manifold. Water enters the impeller through holes in the impeller hub and flows freely through communication holes in the outer casing where the scoop tube is located. The water flow tends to fill the rotating elements, but the degree of filling is limited by the position of the scoop tube. The scoop tube removes the excess water from the rotating casing and discharges it into the stationary outer housing. The water is drained from the outer housing by gravity. The position of the scoop tube is controlled by a three-element controller that senses reactor steam flow, reactor liquid level, and feedwater flow (Sec. 5.1.1.2.2). If the control air supply should fail, the scoop tube actuator on the pump drive provides for maximum flow output to the reactor.

During normal power plant operation, one feedwater pump is used to supply feedwater to the reactor. The second pump is on ready standby. Under special conditions, the two pumps may be used in parallel, each pump supplying one-half the required feedwater flow. The feedwater pump motors can be started and tripped with individual control switches in the control room.

7.8 COMMUNICATIONS

Direct two-way voice communications between designated plant spaces and areas is possible by means of telephone-type PABX (private automatic branch exchanges) equipment. PABX stations are located at strategic and readily accessible points inside and outside the turbine and containment buildings, at the crib house, and at the waste treatment building. Three independent PABX stations are assigned to the control room. Station-to-station dialing may be carried out between any of the stations. Up to six simultaneous station-to-station links are possible. As many as four inside parties may hold a local conference by dialing digits assigned to "conference." Three inside parties may arrange a conference in the same way with one outside party. Outside numbers can be dialed directly from designated inside phones. The control room stations and the shift supervisor's station are direct dialing phones for placing outside calls.

A paging system permits rapid voice communication to be addressed to all accessible areas of the nuclear power plant. Any PABX handset may be utilized to transmit announcements via loudspeakers installed throughout the plant. Page calls may be directed to either the reactor plant areas or the generator plant areas. Page loudspeaker circuits are grouped into two zones. Zone 1 includes the containment building, waste treatment building, and adjacent outside areas. Zone 2 includes the turbine building, office area, the crib house, and adjacent outside areas. Each zone is assigned a separate dial number. To permit priority or emergency page calls to be addressed to either zone, a page override feature is provided at each control room handset and at the shift supervisor handset. In the event either zone's paging circuit is occupied, the use of the page override feature blocks other

calls and allows exclusive paging to be carried out from the priority stations. A completed primary and standby amplifier installation is provided in the paging system. Normal amplification is achieved with the primary amplifier. Failure of this equipment automatically places the standby amplifier in service. Indicator lights at the control room and shift supervisors PABX stations signal primary amplifier failure.

Sound powered telephone circuits are provided to permit direct two-way voice communications within the containment building and the turbine building and at the base of the stack.

Three telephone trunk lines serve the plant. By means of an automatic rotary selector, incoming or outgoing calls seek the first available unoccupied trunk thus providing multi-channel communication to and from the plant. Incoming calls on a trunk line may be extended to any of the PABX telephone stations on the plant system by an attendant at a telephone turret located in the general office of the plant. The attendant may connect any PABX station to an unoccupied trunk to place outside calls. When the turret is unattended, incoming calls on the trunks may be received at any control room station or at the shift supervisor's station.

The following drawings illustrate the locations of telephone and PABX handsets, paging system loudspeakers, and sound-powered test telephone plug-in stations.

- (1) 503606 (Fig. 7.1) Communication System Turbine Room, Main and Mezzanine Floor - LACBWR Generator Plant
- (2) 503608 (Fig. 7.2) Communication System Turbine Room Grade Floor - LACBWR Generator Plant
- (3) 41-503609 (Fig. 7.3) Communication System Reactor and Waste Handling Buildings - LACBWR Project

A high-frequency base station radio transmitter and receiver which operates on the assigned frequency of 37.62 mc for the Dairyland Power Cooperative communication network is located in the control room. Radio contact may be established with other Dairyland base or mobile stations. Carrier wave communication signals are transmitted over Dairyland Power Cooperative electrical transmission lines to provide direct communication links to all power generating stations, major transmission substations, and the central dispatch office. A carrier communication handset and speaker is provided in the control room.

Two pairs of battery-powered portable transceivers are available in the plant for direct mobile communications. The transceivers have a working range up to 1-1/2 miles.

Two portable megaphone-type loudspeakers are available for short range amplification. These loudspeakers are battery powered.

COMMUNICATION SYSTEM TURBINE RM. MAIN AND MEZZ. FLR.

FIG. 7.1