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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 II

ALLIS-CHALMERS
ATOMIC ENERGY DIVISION

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SAFEGUARDS REPORT
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prepared for
THE UNITED STATES ATOMIC ENERGY COMMISSION
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VOLUME II

ALLIS-CHALMERS
ATOMIC ENERGY DIVISION
BETHESDA, MARYLAND

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8. INSTRUMENTATION AND CONTROL

This section supersedes Secs. 7 and 11 of ACNP-62574, "Hazards Summary Report for Construction Authorization of LACBWR."

8.1 GENERAL DESCRIPTION

Instrumentation and control for the LACBWR consists of seven basic systems:

- (1) the plant control system (Sec. 8.2),
- (2) the nuclear instrumentation system (Sec. 8.3),
- (3) the process instrumentation and control system (Sec. 8.4),
- (4) the rod control and safety system (Sec. 8.5),
- (5) the in-core flux monitoring system (Sec. 8.6),
- (6) the failed-fuel element location system (Sec. 8.7), and
- (7) the radiation monitoring system (Sec. 8.8).

Each system consists mainly of analog equipment, with some digital devices for a few miscellaneous applications. Critical plant variables are monitored and then transmitted by signals to readout and control equipment in the main control room. Local indication and alarm is provided for parameters nonessential to plant safety and operation.

The main control room in the turbine building contains all the indicating, recording, and control equipment necessary for plant operation. Indicating and control devices that must be readily accessible to the operator are mounted on the operator's console. This console is located in the center of the control room to provide the best view of the main vertical panel, which contains most of the readout equipment. Equipment mounted on the console includes rod control, nuclear instrumentation readout and pump control equipment, and process control equipment with variable setpoint stations. On the main panel, the instrumentation for the nuclear portion of the plant is in the center and right sections, the generator plant and electrical distribution equipment is in the center and left sections, and the remote annunciators extend across the top of the panel.

The electrical distribution equipment on the panel consists of the necessary indicating ammeters, voltmeters, etc., for continuous determination of the condition of the entire electrical plant. A mimic bus layout for the electrical distribution system is outlined on the main panel with lights to indicate breaker positions. Control switches are mounted on the console to operate breakers, motor starters, and valves.

Instrumentation for the reactor safety system is independent of control instrumentation and is grouped in the center of the control panel. The primary control-rod position indicators on the panel are arranged in the core layout pattern for easy identification of control-rod locations. The secondary control-rod position indicators are mounted above this pattern.

Anticipatory alarms for plant variables warn of abnormal plant conditions so that the operator can take corrective action. Audible and visual alarms are given by 240 annunciator stations in the control room. Alarms that initiate safety system action are grouped together, and the other alarms are in a symmetrical pattern to aid operator interpretation. The annunciator panel alarm groupings are shown in Fig. 8.1.

Plant communication is provided by local telephone stations throughout the reactor and turbine buildings. Public address speakers commanded by the local telephone stations are located throughout the plant. Sound-powered telephone jacks are also provided throughout the plant on a common parallel circuit. The LACBWR area, building and site evacuation is accomplished by use of the communications system described in Sec. 7.8. The containment building has, in addition to the communications system described in Sec. 7.8, an electric siren which can be activated from either the control room or from inside the containment building (see Sec. 15 for Evacuation Procedures).

8.2 PLANT CONTROL SYSTEM

The plant control system coordinates reactor output with turbine-generator demand, without control-rod movement between 60 and 100 percent of full power. The reactor power level is controlled by varying the speed of the two forced-circulation pumps. Increased pump speed increases the coolant flow rate through the core, which increases reactor power level. A decrease in coolant flow rate causes a reduction in reactor power level.

The primary signal to the plant control system is from turbine inlet pressure (see Sec. 5.1.1.2). Pressure is sensed and a proportional signal is transmitted to the master controller. Any deviation from the normal signal causes a corrective signal to be transmitted to the forced-circulation pump speed controller. Reactor power follows the recirculation flow rate until the turbine inlet pressure returns to the set pressure of the master controller.

The speed of the forced-circulation pumps is regulated by varying the liquid level of the fluid coupling located between the impeller and motor of each forced-circulation pump. Liquid level is controlled by the position of a scoop tube within the coupling (see Sec. 5.1.2.5.2). The scoop tube is positioned by a servomotor, designed for a maximum speed that limits the rate of scoop-tube movement to 2 deg/sec or less.

The pump speed controller can adjust reactor power to meet load demand between 60 and 100 percent of full power. However, a main steam bypass valve, which permits steam to flow directly to the turbine condenser, accommodates any abnormal increases in pressure. This valve begins to open when the turbine inlet pressure exceeds the operating pressure by a preset amount. For continued rises in turbine inlet pressure, the input signal to the servo control system regulates the valve until the valve is fully open. The valve accommodates up to 100 percent of full steam flow when fully open.

An initial pressure regulator (IPR) system assumes control of turbine inlet pressure if pressure is abnormally low. If primary system pressure continues to drop below the IPR control setpoint, the IPR system maintains constant setpoint pressure at the turbine stop valve by regulating the turbine inlet valves to close with decreasing pressure. This action prevents rapid system depressurization from a "nuisance" scram or if there is malfunction of the main steam bypass valve in the open position.

A three-element feedwater control system (Sec. 5.1.1.2.2) operates in conjunction with the plant control system. The system measures mass steam flow, feedwater flow, and reactor water level, compares the signals, and regulates the speed of the feedwater pumps accordingly. The feedwater pump speed is regulated by positioning a fluid coupling scoop tube in the same manner as for the forced-circulation pumps.

An analog-computer simulation of the LACBWR control system has been performed to investigate the safety aspects of varying the coolant flow rate through the core, and to evaluate the system with respect to plant performance requirements. The safety studies established the rate limit of forced-circulation pump fluid-coupling scoop tube movement, based on the criteria that the centerline temperature of the fuel pin of maximum power shall not exceed the melting temperature and that the burnout safety factor shall not decrease below 1.5 at any point in the core. These criteria are met over the entire normal operating range with a flow rate corresponding to the design point for a rate of change of scoop tube position equal to 2 deg/sec, assuming continuous flow rate increase from the minimum flow for two pump operation to the full flow capability. It has also been demonstrated that, for abnormal initial operating conditions with a rate of change of scoop tube position less than 2 deg/sec, the severity of the transient is dependent only on the total flow increase. The maximum reactivity insertion rate associated with forced-convection flow changes is never greater than 30 c/sec .

The performance evaluation indicates that the reactor plant, by use of the variable speed forced-circulation pumps, can automatically follow turbogenerator steam demands from 60 to 100 percent of full power (50 Mwe), with no control-rod movement. Automatic load-following can be accomplished for the following turbine load demand design criteria:

- (1) an instantaneous load increase of 5 Mwe
- (2) a uniform rate of load increase or decrease of 1.5 Mwe/min
- (3) an instantaneous load drop of 25 Mwe without reactor scram

The analog simulator also provided design information for the main steam bypass valve and the feedwater temperature control system. It was determined that a bypass valve time constant of 1 sec assures that main steam relief valves do not open after a turbine trip, and provides the pressure relief needed to avoid high-pressure scram during a 50 percent drop in turbine power demand. Feedwater temperature control was found to be

nonessential for satisfactory plant performance, since system response to required plant maneuvers is not significantly altered by its elimination. Steady-state power operation was found to be relatively insensitive to fluctuations in feedwater enthalpy.

A detailed description of the analog simulator is included in ACNP-64572 (Amendment No. 6 to Application for Construction Authorization Docket No. 115-5). The initial analyses and their results are reported in ACNP-64604 (Amendment No. 7). The final results of the LACBWR dynamic analysis, including some revision and extension of the initial results, are reported in ACNP-65511, which is incorporated in Amendment No. 9 to the construction application (ACNP-65517). The analyses include stability studies and accident investigations.

8.2.1 Circuitry for Scoop-Tube-Position Control

Control system components for a single forced-circulation pump are represented in a simplified manner in Fig. 8.2. The figure shows that the output of the turbine inlet pressure transducer is compared with the desired turbine inlet pressure (P_{set}) to obtain an error signal (ϵ_1). The error signal is then fed to a P.I.D. controller, which has proportional, rate, and reset action. The controller output is compared with a signal from a tachometer, which indicates actual pump speed (N) to give a final error signal (ϵ_2). If the turbine inlet pressure drops (indicating an increase in turbine power demand), a positive pressure error signal (ϵ_1) is generated. This signal increases the P.I.D. controller output, making the desired pump speed greater than the actual speed (N). Thus, the error signal (ϵ_2) indicates that the pumps should speed up to meet the new power demand, i.e., the scoop-tube position (X) should be increased. The circled components will normally be set to have no effect on the system response, so that the error signal (ϵ_2) can be considered to feed directly to the scoop-tube positioner. Scoop-tube position is changed by a constant-speed positioning motor. Thus, if the error signal (ϵ_2) is positive and greater than the positioner dead band, the scoop tube is withdrawn at a constant rate ($\dot{X}$). A negative value of ϵ_2 causes a decrease in scoop-tube position, i.e., insertion of the scoop tube at a constant rate ($\dot{X}$).

The components shown in Fig. 8.2 were simulated on the computer, with and without the bypassed components. The bypassed components consist of a controller having proportional and reset action and a servoamplifier that compares the controller output with a scoop-tube position feedback signal and feeds the error signal (difference between desired scoop-tube position and actual scoop-tube position) to the scoop-tube positioner. A simple lag with a 1/2-sec time constant is shown at the output of the scoop-tube positioner; it simulates, for computer purposes, the time required for the positioning motor to reach full speed after voltage is applied.

Results of the computer investigation indicate that the circled components in Fig. 8.2 do not improve overall plant response to the changes in turbine power demand. However, the components will be left in the system, though normally bypassed, so control adjustments can be made to meet any unanticipated response characteristics.

8.2.2 Circuitry for Bypass Valve Control

The bypass valve is normally closed. When turbine inlet pressure rises to a given amount above the set pressure, the bypass valve opens over a range of pressure until fully open. Attempts will be made during plant operation to maintain turbine inlet pressure within a range of ± 15 psi. The bypass valve is therefore set to start opening at a turbine inlet pressure + 15 psi above set pressure and to be fully open at +30 psi above set pressure. Bypass-valve actuator characteristics are represented by a simple lag. Bypass valve control circuitry as simulated for the analysis is shown in Fig. 8.3, in which $\bar{O}$ represents the desired valve position and S'Bp the actual valve position. S'Bp is an input to the steam system model.

8.2.3 Control System Performance

Using a controller proportional gain of approximately 0.2, a reset gain of 0.02, a rate gain of 2.0, and a scoop-tube movement rate of 2 deg/sec, the ramp and step changes in turbine load demand specified in the contract were imposed on the system. Figure 8.4 shows system response to a decrease in turbine load demand from 100 percent power at a rate of 1.5 Mwe/min. Figure 8.5 shows response to an increase of 1.5 Mwe/min in turbine load demand from an initial power level of 60 percent. These figures show the plant control system satisfactorily matches reactor power level to turbine power demand throughout the entire load change, with a deviation in turbine inlet pressure of only 2.5 psi from its setpoint. The power-to-flow ratio during the transient is always very close to that for steady-state operation. It is thus seen that the plant control system provides good load-following during the specified ramp changes in turbine load demand.

System response to a step increase in turbine load demand of 5 Mwe (10 percent of full power) was obtained for initial power levels of 60 percent and 90 percent. These responses are shown in Figs. 8.6 and 8.7. In both cases, the step change in turbine power level is accommodated without severe perturbation of the reactor system. Turbine inlet pressure decreases by a maximum of 12 psi for a step change from 60 percent power, and by 13 psi for a change from 90 percent. Some overshoot in reactor power level occurs during the transient, to a power level of 112 percent for the 90 percent case and to a power level of 92 percent for an initial power of 60 percent, although in both cases recirculation flow rate changes to its new value smoothly and without oscillation. The results show the reactor system can follow step changes of 5 Mwe in turbine power level over a control range of 60 to 100 percent power.

The most severe control requirement for the plant is an instantaneous drop in turbine power demand of 25 Mwe (50 percent of full power). The results of this transient are shown in Fig. 8.8. Reduction in recirculation flow rate alone cannot reduce reactor power level to below 60 percent because of the speed interlock on the forced-circulation pumps. Thus, power level excess of 10 percent over the turbine power demand that exists at the end of the transient must be relieved by opening the turbine bypass

valve and dumping steam to the turbine condenser. This transient must be accomplished without scram. The initial rise in reactor power level to 108 percent of full power is well below the scram point of 115 percent. The reactor pressure rises to a maximum of 1318 psia, which is well below the pressure scram setpoint of 1340 psia. An analysis of the power-to-flow ratio as a function of time during the transient reveals that the power level at any given flow rate never exceeds the normal power level for that flow by more than 10 percent of full power (Fig. 8.9). The limiting power-vs.-flow curve in Fig. 8.9 shows a spread of ~ 20 percent of full power between the normal power and the limiting power at any given flow rate. Thus, there should be no unacceptable power-to-flow ratios during the transient. The maximum heat-flux deviation from the normal power-to-flow curve was about 5 percent of full power, even less than the deviation in reactor power level. The results show that the control system requirements, including that for a 50 percent drop in turbine load, are adequately met.

8.3 NUCLEAR INSTRUMENTATION

The nuclear instrumentation system monitors thermal neutron flux, transmits the signal to indicators and recorders in the control room, and provides signals to alarms and the reactor safety system upon abnormal flux conditions.

8.3.1 General Description

The system consists of eight information channels in three functional divisions: source, intermediate, and power ranges. Two source range channels (1 and 2) monitor the first five decades of reactor operation; two intermediate range channels (3 and 4) cover seven decades of operation to ~ 10 percent of full power; two wide-range power channels (5 and 6) cover eight decades of operation to ~ 150 percent of full power; and two standard power range channels (7 and 8) monitor the last two decades of power operation (see Fig. 8.10) to ~ 150 percent of full power.

Reactor flux is measured between 10^{-1} to 4.3×10^{10} nv by thermal neutron detectors in horizontal instrument tubes, tangent to the outside of the reactor vessel on the reactor side of the biological shield. These detectors, which use the salient features of both vacuum tube and solid state circuitry to provide an optimum combination, transmit signals received in computing equipment in the main control room. The computing equipment provides signals to indicators at the operator's console, recorders in the main panel, and to the safety system.

The thermal neutron level and the rate of neutron level change are indicated as follows:

(1) source range:	level - counts/sec (logarithmic)
	period - sec

(2) intermediate range: level - current (logarithmic)
period - sec

(3) power range: percent power (linear)

8.3.2 System Design Characteristics

The basic nuclear instrumentation design philosophy is one of high performance and reliability. A minimum of two channels are provided for each range. A component failure in any channel causes a safety system signal from that channel. Failure of equipment in any one channel does not affect the operation of any other channel. Down-scale alarm trips signal loss of detector power in any of the source, intermediate, and power range channels or indicate that the flux level is below the scale range of the intermediate or power range channels. All trip circuits that provide alarm and safety system actuation are normally energized in case the main power supply should fail. However, input power is supplied from the vital noninterruptible bus. Overload protection is provided for all power supplies.

Any one of the channels can be tested and calibrated with test signals from a test chassis. Front panel controls, when moved from the "operate" position, provide a scram signal from the channel under test; this does not affect the operation of any other channel.

All switches are of the rotary type, with positive detents, and are located in the front of each drawer.

8.3.3 Detector Housing Assemblies and Positioning Mechanism

The nuclear instrumentation detectors are contained in housing assemblies provided with a positioning mechanism to facilitate installation, positioning, and removal. The detectors are encased in polyethylene shrink tubing, which insulates the detectors from the positioning mechanism and ground and provides additional moisture sealing at the connectors. The assemblies rest on wheels to provide ease of movement in the horizontal plane of the instrument tubes. A locking screw is provided to prevent inadvertent position changes from the desired location.

The positioning mechanism itself is a simple, manually controlled device consisting of a 1-1/4 x 1-1/4 x 1/8-in. channel that is attached to the housing assembly and extends to the outside of the biological shield. The channel assembly is fabricated in four 4-ft sections to facilitate installation and dismantling, and high voltage cable and signal cable are routed through the channel. Excess cable is stored on individual wrap reels at the face of the biological shield. Detector position is indicated by a scale attached on one side of the channel. The mechanism is designed to prevent any mechanical load on the signal or high voltage cables. The equipment allows the operator to locate and position the detectors within ± 2 in. of the optimum location

in the instrument tubes. The optimum location for each detector is that location which provides the range and overlap requirements for its channel; this location will be determined during the power calibration of the core.

Eight horizontal carbon steel instrument tubes are embedded in the biological shield in vertical rows of two, tangent to the outside of the reactor vessel (see Fig. 8.11). Since only two of these tubes are through-type, there is capacity for an array of ten detector locations. The detector and cable assignments for the numbered instrument tubes are given in Table 8-1. Each instrument tube is 8 in. in diameter for the first 24 in. from the face of the biological shield, and 6 in. in diameter the rest of its length.

The source range detectors (channels 1 and 2) are positioned on the vessel OD approximately 90 deg in each direction from the source, which is in one of the spare fuel assembly shroud cans on the storage well side of the reactor. This shroud can is of zirconium; the other spares are of stainless steel.

The end of each instrument tube is sealed with a single-unit 45-in. heavy-concrete instrument plug. The plug is designed to limit streaming to the unrestricted area outside of the plug. Each plug is provided with a 1-1/4-in.² hole at the centerline to allow passage of the positioning mechanism. A permanent monorail directly above each vertical row of instrument tubes facilitates plug removal. A portable carriage device is used in this operation.

To insure that the system is as noise-free as possible, four separate conduit runs are installed. In addition, the signal leads for the remote meters, recorders, and the range-changing switches are in similar conduits. The high voltage cables for the detectors penetrate the containment vessel independently and are not grouped with any other power or control cables.

8.3.4 Source Range

The source range instrumentation consists of two identical channels (Fig. 8.12) that monitor reactor power level and compute startup rate from source level over a flux range of 10^{-1} nv to 10^4 nv at the detector locations.

Source range level is monitored by B-10 proportional counters having a sensitivity of 12 counts/sec/nv at an operating voltage of approximately 750-v d-c. The detector has an overall length of 17-1/2 in., a sensitive length of 10-1/2 in., a 3-1/2-in. dia, and is mounted in a movable housing assembly for positioning in the guide tube. The two source range detectors are energized from the same dual high voltage power supply chassis; however, the power supplies in this chassis are independent of each other. Individual front-mounted controls are provided for each power supply.

TABLE 8-1

NUCLEAR INSTRUMENTATION DETECTOR AND CABLE ASSIGNMENTS

<u>instrument tube number</u>	<u>N.I. channel</u>	<u>number of cables assigned</u>		<u>conduit assigned</u>	<u>type of cable</u>
		<u>operating</u>	<u>spare</u>		
1	N-5	3	1	C	RG-59B/U
2	N-2	1	1	B	RG-6A/U
3	N-6	3	1	D	RG-59B/U
4	N-1	1	1	A	RG-6A/U
5	N-4	3	0	D	RG-59B/U
6	N-8	4	1	D	RG-59B/U
7	Spare	None	2	C	RG-59B/U
8	Spare	None	2	D	RG-59B/U
9	N-3	3	0	C	RG-59B/U
10	N-7	4	1	C	RG-59B/U

<u>conduit no.</u>	<u>total number cables assigned</u>	<u>N.I. channels</u>
A	2	1
B	2	2
C	14	3, 5, 7
D	14	4, 6, 8

Each section of the dual power supply has a time delay relay circuit of 30 sec that prevents excessive voltage at the output connector before the regulation circuit begins to function. Voltage calibration adjustments and individual power supply fuses are on the back of the chassis. Each high voltage power supply contains its own regulation and filtering circuits; however, the high voltage output is further filtered in a separate high voltage filter to minimize noise. Pulse signals from the proportional counter are coupled through a low noise cable (which has a ferromagnetic shield insulated from ground) to the current pulse amplifier on the rear of the instrument panel in the control room. The gain of this unit is 4.4 v/pico-coulomb with a rise time of 100 nanosec and an adjustable attenuation factor of 1:1 to 8:1.

The amplified pulses are coupled through a cable to the active terminating element, which is connected directly to the input of the log count ratemeter. The active terminating element provides an impedance match between the cable and the log count ratemeter. No switches are associated with these units, though a power-on light and fuse are mounted externally on the current pulse amplifier. The voltage output of the current pulse amplifier is 10 v maximum without saturation, as measured at the output of the terminating element. This voltage drives the linear pulse amplifier stage and pulse height discriminator of the log count ratemeter where lower level pulses caused by gamma radiation and noise from the amplifier stages are eliminated.

The pulses from the discriminator are amplified, shaped, and converted into a current signal (proportioned to the pulse input rate) in the log countrate circuit. The log countrate circuit further converts this signal to an output proportional to the logarithm of the current, and the resultant countrate signal feeds to the period amplifier, a downscale trip stage, a local and remote meter (utilizing a five decade logarithmic scale from 1 to 10^5 counts/sec) and recorder jacks on the front of the panel. The trip stage is set at 2 counts/sec and provides contact outputs as follows:

- (1) local alarm (trip lamp)
- (2) remote alarm (annunciator)
- (3) reactor start circuit (rod control part of safety system)

Linear accuracy is ± 2 percent or better on the upper and lower decades, and is better than ± 1 percent for the center decades. Response time for the count rate output does not exceed 15 sec for a step function of 1 count/sec, and 3 sec for a step function greater than 10 cps.

Source range period is computed in the period amplifier stage of the log-n count ratemeter. The output feeds to a period trip stage, local and remote meters, and recorder jacks mounted on the front of the panel.

The period trip stage is set at 7 sec and provides contact outputs as follows:

- (1) local alarm (trip lamp)
- (2) remote alarm (annunciator)
- (3) rod withdraw prohibit (rod control part of safety system)
- (4) rod test circuit (rod control part of safety system)

The period accuracy is within 2 percent of the equivalent full scale output voltage. Period response time is less than 5 sec for a step change from infinity to full-scale period.

Individual front-mounted controls are provided for each log count ratemeter as follows:

- (1) power indication lamp (W)
- (2) blown fuse indication lamp (R)
- (3) fuse
- (4) a discriminator control potentiometer to adjust the input sensitivity of the LCRM for both positive and negative pulses
- (5) count ratemeter
- (6) period meter
- (7) one count rate alarm light and one period alarm light

The following controls and adjustments are accessible from the front of the chassis through a hinged protective cover:

- (1) operate/calibrate switch
- (2) period input test jack
- (3) countrate calibration and trip adjustments
- (4) period calibration and trip adjustments

Internal circuits provide a signal that simulates detector pulse, for checking count-rate calibration. Period test is provided by insertion of a ramp input from the test

chassis into the test jack on the chassis. Either placing the operate/calibrate switch in "calibrate" or inserting the period test jack results in a period scram unless the period bypass switch (Table 8-3) of the channel under test is in "bypass" position. Since a "calibrate" position trips only the contact output to the scram circuit, the alarms serve to verify proper operation of individual trip stages.

For accurate determination of pulse rate during startup and/or during special testing, a scaler accepts pulses from the discriminator of either source range channel. A three-position (Ch. 1, Off, Ch. 2) selector switch determines what channel is monitored. The display covers seven decades in a time or count mode, with a display duration of 1 to 50 sec (adjustable) or infinity. Manual controls (Stop, Start, Reset) are on the front of the scales.

8.3.5 Intermediate Range

The intermediate range instrumentation consists of two identical channels (Fig. 8.12) that monitor reactor power level and compute startup rate from approximately eight decades below full power to ~ 10 percent of full power. This range corresponds to a flux range at the detector of 2.8×10^2 to 2.8×10^9 nv.

Intermediate range level is monitored by gamma compensated ionization chambers having a neutron sensitivity of 4×10^{-14} amps/nv, an uncompensated gamma sensitivity of 2×10^{-11} amp/r/hr, and a compensated gamma sensitivity of 1×10^{-12} amp/r/hr. Operating voltage is 100 - 1200-v d-c. The detector has an overall length of 20-5/8 in., a sensitive length of 16-1/4 in., a 3-1/16-in. dia, and is mounted in a movable housing assembly for positioning in the guide tube.

The intermediate-range detectors are energized from separate dual high voltage power supplies, each of which supplies a positive and negative voltage to its respective compensated ionization chamber. Front-mounted controls, rear-mounted calibration adjustments and fuses, and the high-voltage time delay feature are identical to those of the source range power supplies.

Current output from the detector feeds to the logarithmic current amplifier stage of the log-n and period amplifier, where it is converted to a current proportional to the logarithm of the detector current. The resultant log-n output is fed to the following: the period amplifier stage, a downscale trip stage providing contact outputs to local and remote alarms, a two-pen recorder that monitors both intermediate range channels, and a local and remote log-n meter that utilizes a seven decade logarithmic scale from 10^{-11} to 10^{-4} amps. Log-n accuracy is the same as source range count rate accuracy, and response time does not exceed 2 sec. for levels of 10^{-10} amp or above.

Intermediate range period is computed in the period amplifier, and the resultant signal feeds to two independently adjustable trip stages, recorder output jacks on the front of the panel, and local and remote meters.

Outputs from the trip stages are as follows:

Trip Stage 1 - contact outputs to a local trip lamp, remote annunciator, and the rod withdraw prohibit circuit set at 15 sec

Trip Stage 2 - contact outputs to a local trip lamp, remote annunciator, and the scram prohibit circuit set at 3 sec

The period accuracy is identical to that of the source range channels. The period response is less than 5 sec in the lowest decade of log-n operation and less than 3 sec in the upper decades.

The following individual front-mounted controls are provided for each log-n and period amplifier:

- (1) power indication lamp (W)
- (2) blown fuse indication lamp (R)
- (3) fuse
- (4) log-n meter
- (5) period meter
- (6) operate/calibrate switch
- (7) one log-n and two period alarm lights

The following controls and adjustments are accessible from the front of the chassis through a hinged protective cover:

- (1) period input test jack
- (2) log-n calibration and trip adjustments
- (3) period calibration and trip adjustments

A signal that simulates detector output current is provided by internal circuits for checking log-n calibration. Period test is provided by inserting a ramp input signal from the test chassis into the test jack on the chassis. Placing the operate/calibrate switch in "calibrate" or inserting the period test jack results in a period scram unless the period bypass switch (Table 8-3) of the channel under test is in "bypass" position. Since a calibrate position trips only the contact output to the scram circuit, the alarms serve to verify proper operation of individual trip stages.

8.3.6 Power Range

The power range nuclear instrumentation consists of four channels (Fig. 8.12). Two channels monitor reactor power level from eight decades below full power to ~ 150 percent of full power, and two channels monitor the last two decades of power operation to ~ 150 percent of power (these levels correspond to neutron flux ranges of 2.8×10^2 to 4.3×10^{10} nv and 2.8×10^8 to 4.3×10^{10} at the detectors).

The power range channels initiate safety system action by providing outputs in coincidence to the rod control part of the safety system. Two separate coincidence circuits are used to provide adequate flux level protection for all ranges of operation. The output of each channel consists of normally open contacts that de-energize the full-scam relay through an auxiliary relay. At a power level of less than 5 percent of full power, either extended power range channel scrams the reactor at 115 percent of the particular range selected. At a power level above 5 percent the low-trip stage relays of either standard power range channel trip to de-energize an auxiliary relay on the safety system panel; the output contacts of this relay change the safety system power range input to a two out of four coincidence arrangement.

8.3.6.1 Extended Range Channels. Wide-range level is monitored by gamma compensated ionization chambers identical with those described for the intermediate range channels. The detectors are energized from separate dual high voltage power supplies, each of which provides a positive and negative voltage to its respective compensated ionization chamber. Front-mounted controls, rear-mounted calibration adjustments and fuses, and the high voltage time delay features are identical to those for the source and intermediate-range power supplies.

Current output from the detector feeds to a linear current amplifier (picoammeter), and the resultant signal feeds to three level trip stages, a two-pen recorder and local and remote meters calibrated in two major linear scales; one is from 0 to 60 percent, the other from 0 to 150 percent. The actual full-scale value of the picoammeter is dependent on the setting of the remote range switch on the operator's console. The range switch provides linear display of reactor power over the wide range requirements.

Outputs from the trip stages are as follows:

Trip Stage No. 1 - downscale trip providing contact outputs to local and remote alarms set at 5 percent for the 150 percent multiplier, or 2 percent for the 60 percent multiplier.

Trip Stage No. 2 - upscale trip providing contact outputs to local and remote alarms set at 110 percent for the 150 percent multiplier, or 44 percent for the 60 percent multiplier.

Trip Stage No. 3 - upscale trip providing contact outputs to local and remote alarms and the coincidence circuit of the safety system set at 115 percent of the 150 percent multiplier, and 46 percent for the 60 percent multiplier.

The accuracy of linear level indication is ± 5 percent of range. The response time does not exceed 100 msec except in the lowest decade, where the response time does not exceed 1000 msec for a one-decade step input.

Individual front-mounted controls for each picoammeter consist of:

- (1) power indication lamp (W)
- (2) blown fuse indication lamp (R)
- (3) fuse
- (4) percent power level meter
- (5) three alarm lights

The linear power calibration and trip adjustments are accessible from the front of the chassis through the hinged protective cover. One three-position test switch (Ch. 5 calibrate, operate, Ch. 6 calibrate) is on the panel between the two picoammeters. This switch allows only one of the picoammeters to be tested at a time. Separate input test jacks for each picoammeter are on either side of the operate/calibrate switch. Placing the switch in one of the calibrate positions does not affect the other channel, but does result in a high-flux scram for the channel under test. The scram can be bypassed, however, when operating on a one-out-of-two scram circuit (switch #S-24, Table 8-3). When operating on the two-out-of-four coincidence safety circuit, a scram signal from one of the other power channels will scram the reactor. Since placing the switch in the calibrate position trips only the contact output to the scram circuit, the alarms serve to verify proper operation of individual trip stages. An adjustable signal that simulates detector current is fed from the test chassis for check of circuit calibration and/or for trip adjustments.

The last position of the remote range change switch is also a test position. A screw-driver adjustment on the face of the remote range switch inserts a signal simulating detector current in parallel with any existing detector current for checking trip stages. Placing the switch in test position causes the same trip action as that described for the operate/calibrate switch.

8.3.6.2 Standard (Narrow) Range Channels. Standard range level is monitored by ionization chambers having a neutron sensitivity of 2.2×10^{-14} amp/nv and a gamma sensitivity of 1.7×10^{-11} amp/r/hr. Operating voltage is from 100 to 1200-v d.c. The detector has an overall length of 13-1/2 in., a sensitive length of 9-1/4 in., a

3-1/16-in. dia, and is mounted in a movable housing assembly that allows positioning on the guide tube.

Each ionization chamber is energized from a self-contained solid-state power supply that is an integral part of the respective flux amplifier. Current output from the detector feeds to a linear current amplifier or flux amplifier, and the resultant signal feeds to three trip stages, a two-pen recorder, one of two flow-power circuits of the safety system, and two meters calibrated linearly from 0 to 150 percent. The trip stages are mounted in a separate auxiliary trip module. Outputs from the trip stages are as follows:

Trip Stage No. 1 - downscale trip providing outputs to local and remote alarms and the safety system coincidence switching circuit set at 5 percent.

Trip Stage No. 2 - upscale trip providing contact outputs to local and remote alarms set at 110 percent.

Trip Stage No. 3 - upscale trip providing contact outputs to local and remote alarms and the safety system coincidence circuit set at 115 percent.

The flux amplifier is continuously self-tested by a pulse generator that sends a 70-msec pulse of scram level amplitude through the amplifier once every second. The pulse is registered in a monitor circuit at the trip output. If the test pulse train ceases, the monitor operates a trouble warning circuit annunciator relay and operates the input of the respective channel to the safety system coincidence circuit. Since the self-testing circuit obtains its power from the high voltage supply, a "safe" test signal indicates high voltage on the ion chamber; therefore, a malfunction or a discontinuity in the loop formed by the high voltage power supply, detector, and flux amplifier loop is indicated. The accuracy of the linear power level indication is ± 5 percent of full range, and the response time constant is 100 msec.

A 24-v d-c power supply in a separate module provides power for the auxiliary trip unit and the ion chamber test circuit. A relay module is mounted in the same chassis that contains the extra relays required for the entire system. The chassis contains a total of four relays; two provide additional contact outputs for the wide power range high flux trip stage (115 percent) to the safety system coincidence circuit; the other two relays provide additional output contacts for the wide power range calibrate circuit (coax relay is tripped when the respective channel is placed in "calibrate") and the source range countrate trip stage (rod test). The relay module and the three other modules associated with the standard range channels are of pullout construction.

An ion chamber calibration box at the rear of the panel allows measurement of the actual ion chamber current during reactor operation. A "push to test switch" is depressed on the calibration box, which diverts the signal from the ion chamber through

a precision resistor to ground. By connecting a high impedance voltmeter across this resistor, the input current is measured directly. Test jacks are provided for this purpose. Actuation of the "push to test switch" also de-energizes the relay controlled by the high flux trip stage (115 percent) to the safety system coincidence circuit.

Front-mounted controls for each flux amplifier consist of:

- (1) operate lamp
- (2) calibrate lamp
- (3) test pulse light (pulsating)
- (4) trouble monitor light
- (5) linear meter
- (6) operate/calibrate switch
- (7) test jack
- (8) calibration adjustments and trip

The three local trip lamps and the respective calibration adjustments are on the auxiliary trip stage module. When the flux amplifier operate/calibrate switch is placed in the calibrate position, the amplifier is connected to the test jack to introduce a calibration current from an external calibration unit. This action also trips the input to the safety system coincidence circuit; however, the alarm outputs do not trip, so that the alarms may be used to verify proper operation of individual trip stages.

8.3.7 Test Chassis

The test chassis is an independent unit built into the nuclear instrumentation system and mounted on the nuclear instrumentation panel. This unit permits complete functional testing of the nuclear instruments. The test chassis obviates the need for additional test equipment during normal operation and routine tests by providing for all tests required to check the operability and calibration of the nuclear instruments.

8.4 PROCESS INSTRUMENTATION AND CONTROL

The process instrumentation and control is described in the individual system descriptions of Sec. 5. Instrumentation is shown on the various system flow diagrams. Process instrumentation functions of the plant safety system are discussed in Sec. 8.5.2.

8.5 ROD CONTROL AND SAFETY SYSTEM

The rod control and safety system permits control-rod motion and automatically shuts down the reactor and initiates other vital actions if an unsafe condition occurs. The system consists essentially of the instrumentation and control system components that perform the required reactor safety functions. The system protects against conditions that could cause permanent damage to the core and protects plant personnel and the general public against accidental release of fission products.

8.5.1 Control-Rod Drive Mechanism

The development and production unit testing programs for the rod drive mechanism are defined in Amendment 2 to ACNP-62574. The control-rod drive mechanism is described in detail in Amendment 5, and subsequent mechanism revision is reported in Amendment 8. The preliminary reliability report was submitted as part of Amendment 6. Amendment 11 describes the final rod drive mechanism in detail, for operating authorization. That amendment also includes the final reliability report, results of prototype testing, and the rod-drop accident analysis.

A summary description of the control-rod drive mechanism is provided here so that the integrated control system may be understood without reference to the separate amendment for the rod drive mechanism.

Reactor power and fuel consumption are controlled by positioning 29 bottom-entry control rods within the reactor. The control rods are manipulated by 29 identical control-rod drive mechanisms (Fig. 8.13) that operate independently of each other. Two independent power sources - an electric motor and a hydraulic scram motor - are mechanically linked through power transmission to the output shaft for each rod drive. The motive force for the hydraulic motor is supplied by gas pressure acting on a piston in a scram accumulator. Slow-speed vertical motion (~ 20 in./min in both directions), and high-speed upward motion for reactor scram are accomplished by rotation of a leadscrew, which moves a roller nut assembly along the length of the leadscrew. A fixed key mates with the roller nut assembly to guide it vertically and prevent it from rotating. Thus, the leadscrew torque is converted to vertical thrust, which acts through a push rod to position the control rod.

The control rod is joined to the push rod by a positive latch, which consists of six Stellite balls; a retainer, which threads into the top of the push rod; and a spring-loaded plunger and sleeve. The latch is actuated remotely by the latch rod assembly,

which consists of a cylindrical hollow tube with a slotted fitting attached to the top end and a solid cylindrical tool-adaptor fitting attached to the lower end. The latch-rod assembly is located inside the leadscrew, and its axis is coincident with that of the leadscrew.

The push rod must be in the fully withdrawn position and the lower mechanism must be removed in order to open the latch for latching or unlatching. A special latch tool is used to rotate the latch rod to the latch actuating position, which is 90 deg from the position at which the slot in the end of the latch rod coincides with the latch pin. The latch rod is moved vertically upward 0.7 in. to open the latch. To close the latch, the latch rod is pulled downward and the tool is removed. If the control rod extension is properly latched, the latch rod will contact the extension after moving upward 1.0 in. It is apparent when the latch rod contacts the extension, since a force greater than the weight of the control rod must be applied to effect further vertical movement. If the latch rod can be moved vertically 1.8 in. before meeting positive resistance, the control rod extension is not latched. Once the control rod extension is properly latched, the compression spring will maintain the latched position.

The unit-design concept of the system provides overall system reliability, since any malfunction or failure of a drive mechanism in no way affects the capability or operability of the other drive mechanisms. Each control-rod drive mechanism (Fig. 8.13) consists of the following:

- (1) upper drive assembly
- (2) lower drive assembly
- (3) hydraulic charging system
- (4) gas charging system

The upper and lower assemblies are bolted together in line, flanged to the reactor bottom nozzle, and extend downward into the cavity below the reactor vessel. The hydraulic and gas charging systems are located below the reactor vessel.

The upper assembly consists of the components required to convert the rotary motion developed by the lower mechanism into vertical thrust to be applied to the control rod. In addition, the upper assembly contains a positive latch that engages the control-rod extension, and a mechanism to provide continuous signals to the secondary control-rod position-indicating system.

The lower assembly consists of the components required to provide the following:

- (1) rotary shim (electric-motor powered) and scram (hydraulic-motor powered) motion to the upper assembly leadscrew.
- (2) continuous signals to the primary control-rod position-indicating system.

(3) visual control room alarms and partial-scram signals to the safety system in the event of either low oil level or low gas pressure in any control-rod drive accumulator.

The hydraulic-charging system consists of the components, equipment, and instrumentation required to charge oil and to maintain an oil pressure of 3200 psig to each of the 29 scram accumulators. The gas-charging system consists of the components, equipment and instrumentation required to charge nitrogen gas and to maintain a gas pressure of 3000 psig to each scram accumulator. Higher pressure is maintained on the oil side of the accumulator piston to assure that the piston is against the piston stop that is provided to fix the charged gas volume.

During rod insert shimming operations, torque is transmitted from the electric motor to the leadscrew through an overrunning clutch. During rod withdrawal shimming operations, torque is transmitted from the electric motor to the leadscrew through a magnetic clutch. The overrunning clutch transmits torque in the rod-insert direction only. The magnetic clutch is energized to transmit torque in the rod-withdraw direction only. The electric motor is controlled manually.

During normal scrambling operations, torque is transmitted from the hydraulic motor to the leadscrew through gearing. When a scram condition exists, simultaneous signals are transmitted to a dual solenoid-operated hydraulic valve and the electric drive motor. The hydraulic valve is de-energized, permitting high-pressure oil to flow from the scram accumulator to the hydraulic motor. The magnetic clutch is interlocked with the scram circuit so that it cannot be energized during a scram stroke; thus torque cannot be transmitted from the hydraulic motor to the electric motor. However, the electric motor is energized to drive in the insert direction through the overrunning clutch. The electric motor actually exerts torque only if the hydraulic motor does not drive faster than the electric motor; this arrangement ensures power to the leadscrew if one power source should fail. The pressure-compensating flow-control valve between the accumulator and the hydraulic valve maintains constant oil flow to the hydraulic motor. The oil pressure drop across the hydraulic motor produces the torque required for rod scram. The oil flow to the hydraulic motor is sustained by expansion of nitrogen gas acting against the scram accumulator piston.

The control rod is decelerated near the end of the scram stroke by a combination of dashpot and mechanical contact action. Dashpot action from buffer piston travel in the water-filled buffer near the end of scram stroke (Fig. 8.14) reduces the rod velocity by two-thirds. The remaining kinetic energy is absorbed by a constant torque disc-type mechanical brake splined to the leadscrew, and by the leadscrew as the nonjamming stops of the roller nut assembly contact the blind grooves in the leadscrew. Upon completion of shimming or scrambling motion, the mechanical brake positively prevents further vertical rod motion.

During rod shimming and scrambling, and during stationary periods, rod position is continuously indicated through signals generated by two independent control-rod position-indicating systems - a primary indicating system, and a secondary (backup) indicating system. The two systems operate independently of each other and independently of the other mechanisms. The primary indicating mechanism is a synchro-torque transmitter that transmits its angular position, corresponding to the travel position of the rod drive, to a synchro-torque receiver at the control panel. The secondary indicating mechanism indicates the travel position of the roller nut assembly as sensed by a series of reed switches which are actuated by an ALNICO magnet that moves with the roller nut assembly.

Each of the lower assemblies has six limit switches, two of which are spares. These switches are fastened to the bottom side of the gear housing and are actuated by the shaft that operates the torque transmitter. One rod-in limit switch in the rod-insert circuit and one rod-out switch in the rod-withdraw circuit open the circuits that cut off the motor when the rod is fully inserted or fully withdrawn. The remaining rod-in switch, which is actuated at a point of rod travel $1/2$ in. before the rod-in motor cutoff limit switch actuation point, interlocks the reactor start circuit, the boron-inject alarm circuit, the rod insert test timing circuit, the rod-under-test withdraw permit circuit, and lights a position indicator light. The remaining rod-out switch, actuated $1/2$ in. before the rod-out motor cutoff limit switch actuation point, lights a position indicator light.

Prior to reactor startup, all scram accumulators are fully charged by the gas and hydraulic charging systems; the hydraulic system is operable remotely from the control room. Nitrogen gas is first bled into the accumulator until the accumulator piston is forced to the top end of the accumulator and the gas pressure is approximately three-fourths that required for the full charge. The gas is then compressed by pumping oil to the hydraulic side of the accumulator at a pressure sufficient to force the piston back to the piston stop. The gas pressure is thus raised to the fully charged level. Sufficient loss of either accumulator gas pressure or oil level gives a warning signal. A further decrease in gas pressure or oil level initiates a partial scram. The values selected for these scram points are high enough to enable a malfunctioning unit to complete a full-stroke scram within the required 2.5 sec.

In the upper portion of the lower housing a mechanical seal is provided for the high pressure water within the pressure housing. During normal operation, the mechanical seal is cooled by continuous injection of 0.3 gpm of seal cooling water (at 125 F max) into the pressure housing. The coolant then flows upward to a buffer cylinder, where it mixes with 0.7 gpm of reactor water (at 577 F) flowing downward into the buffer cylinder through the reactor bottom nozzle. The mixed flows exit the pressure housing through a penetration approximately 3 in. below the pressure housing top flange (see Fig. 8.14); the maximum temperature of the 1 gpm is 475 F. This flow arrangement permits continuous reactor blowdown, cooling of the mechanical seal, and flushing of the pressure housing, and maintains an acceptable temperature difference between the reactor bottom nozzle and pressure housing flanges. The

restricted flow path from the reactor nozzle through the buffer assembly also limits transfer of solid particles from the control-rod nozzle to the drive mechanism during the scram stroke.

Control rods are latched and unlatched from the drive mechanism by a remotely operated latch mechanism after the push rod is fully retracted. During mechanism removal, leakage from the reactor is restricted by a poppet that forms metal-to-metal seals in the reactor nozzles.

8.5.2 Safety System

8.5.2.1 General Description. The safety system consists of the protective circuitry and equipment that initiates or performs the required safety actions in the plant.

This section describes the following:

- (1) the rod control and test circuits that restrict the conditions of rod withdrawal (Fig. 8.15),
- (2) the scram input channels that sense nonpermissible operating conditions and transmit signals which cause full or partial scram and associated safety actions (Figs. 8.16 and 8.17), and
- (3) the bypass key switches for safety system signal inputs.

Engineered plant safeguards that give automatic safety actions not related directly to reactor scram (containment building isolation, control of gaseous waste release, fail-safe design of valves, etc.) are discussed elsewhere.

A full or all-rod scram signal causes rapid insertion of all control rods not already fully inserted and results in a complete reactor shutdown. A partial scram signal transmits only to 13 control rods of a pattern pre-selected to ensure that the reactor will be subcritical to a hot, non-boiling condition at any stage of fuel lifetime, when those rods are fully inserted. The partial scram allows the reactor to be more quickly brought to power from a hot condition by withdrawing only those control rods (of the 13 partial-scram rods) which are programmed for a withdrawn operating position at the existing stage of the core fuel exposure.

The system monitors and initiates safety action under preset conditions for the following parameters:

- (1) neutron flux
- (2) rate of change of neutron flux (period)

- (3) primary system pressure
- (4) reactor water level
- (5) reactor power to recirculation flow relation
- (6) condenser vacuum
- (7) bus voltages
- (8) rod drive accumulator oil level and gas pressure
- (9) reactor building steam isolation, turbine building steam isolation, and turbine stop valve positions

In addition there are manual scram buttons on the control panel and in the containment building.

The reactor flux inputs (Items 1 and 2 above) are analyzed and converted to safety system signals in the nuclear instrumentation system; pressure, water level and flow-power signals (Items 3 through 5) are received and analyzed by equipment located on the safety system panel. The remaining variables generate signals that are transmitted directly or through auxiliary relays to the safety system actuating device.

All of the plant parameters providing inputs to the safety system actuate individual alarms at the main console when a scram signal is initiated. Where feasible (see Sec. 11.3), anticipatory alarms are also provided to warn the operator of an abnormal condition prior to initiation of scram action. Redundancy and/or backup are provided throughout the system by duplicate channels which monitor the same variable, by alternate inputs to actuate the safety system, and by a full scram back-up for some partial scram signals to act if the scram condition worsens.

Prior to startup of the safety system equipment from a completely shutdown condition, all operate/calibrate switches on the process and nuclear instrumentation panels are placed in the operate position and all bypass switches are placed in the normal position. The equipment is energized by closing the respective feeder breakers from the noninterruptible a-c and d-c buses. All partial and full scram signals are then cleared and the reactor start and withdraw permit circuits are energized.

The safety system remains energized at all times during reactor operation. The system provides operating information to the plant operator and automatically initiates emergency action for any unsafe condition.

The reactor start circuit must be re-energized after a shutdown by full scram, but not after partial scram, since the reactor start circuit "seal in" is only de-energized by a

full scram. Both the full and partial scram circuits are interlocked so that once de-energized they must be manually reset. This arrangement ensures complete partial scram or full scram action for any transient signals.

8.5.2.2 System Design Characteristics. The safety system protects the reactor at all times and for all operating ranges.

The readout equipment is designed for operation at atmospheric pressure, background activity level, a temperature range of 40 - 110 F and a relative humidity range of 20 - 90 percent. The temperature and relative humidity parameters are extreme conditions since the equipment is in air-conditioned space.

System response time is defined as the time from initiation of a scram signal until the control rods have reached their fully inserted position. This response time does not exceed 3 sec for any scram input, including a maximum 2.5 sec for a full scram stroke. The scram stroke has been measured in prototype control rod drive tests. It has been found to be 90 percent complete within 2 sec after de-energization of the scram solenoid.

All relays are quick acting and highly reliable. They are provided with solderless terminals and have a contact rating of 2 amps for 120-v a-c and 125-v d.c.

Fail-safe design features have been incorporated in safety system equipment. Channel component failure that prevents proper scram channel operation actuates channel trip stages. Equipment failure in any channel does not affect the operation of any other channel. Each channel has a separate power supply (single phase, 120-v a-c at 60 cycles) incorporating independent overload protection.

Safety system indicating meters are standard long-scale indicating instruments accurate to ± 1 percent. Switches are of the rotary type with positive detents.

8.5.2.3 Rod Control. Reactor startup is prohibited by the reactor start and rod withdraw circuits, until certain plant conditions are met. However, a rod test circuit permits testing of individual control rods to ascertain that the rod scram mechanism is functioning properly. All rods will be individually tested and timed to ensure that they will scram in the required 2.5 sec.

The test withdraw permit is obtained when the "Rod Under Test" key switch is turned to the "Test On" position, one of the 29 rods has been selected for testing, and all other rods are in the "Full-In" position. However, an interlock will open the circuit if source channels (1 or 2) have a period less than 7 sec.

The "Rod Under Test" switch will energize an indicating light and the motor of the "Rod-Insert Timer." After a rod is withdrawn for testing, the "All Rods Scram" push-button is depressed to start the test. When the "All Rods Scram" relay drops out, the

clutch of the test timer is engaged to initiate timing. When the rod under test reaches the full-in position, the timer clutch is disengaged so that the test timer indicates the elapsed time for the full scram stroke. Period interlocks from source channels (1 and 2) will cause a scram in the event of a period less than 7 sec during the rod test.

A reactor start permit is obtained when the following conditions are satisfied:

- (1) All rods are fully in.
- (2) The building locks are secured.
- (3) The nuclear instrumentation intermediate range channels (3 and 4) period trips are not bypassed.
- (4) The nuclear instrumentation source range count rate is greater than 2 counts/sec.
- (5) Partial scrams are cleared or bypassed.
- (6) All-rod scrams are cleared or bypassed.

Once these conditions are met, the reactor start pushbutton is depressed, the reactor start relay is energized, and the circuit "seals in," providing a signal to the withdraw permit circuit. The start permit also starts the "Reactor-On Warning Timer," and energizes the reactor building lights and horn to warn personnel. The timer will open the horn circuit after sufficient warning.

To establish a withdraw permit, the following conditions must be met:

- (1) The reactor start is "sealed in."
- (2) The nuclear instrumentation source range period is greater than 7 sec.
- (3) The nuclear instrumentation intermediate range period is greater than 15 sec.
- (4) The partial scram signal is clear.
- (5) The rod test key switch is not in "Test On" position.
- (6) The scram accumulators low oil level and/or low gass pressure partial scram protection is not bypassed (switches S-15 and S-16, Table 8-3).

Bypass interlocks in the "seal in" portion of the withdraw permit circuit prevent rod withdrawal after a partial or full scram until the period bypass switches of the source and intermediate range channels are returned to normal (open). The operating procedures (Sec. 13.4) govern the manual actions that close period channel bypass circuits during increase to high power levels and that reopen the bypass circuits to put the period scram channels back "in circuit" after a manual reduction to a low power level.

A withdraw permit energizes the withdraw permit relay, allowing rod withdrawal. The system is interlocked so that rods can be withdrawn only one at a time.

The forced-circulation pump and safety system rod control are interlocked to prevent simultaneous reactivity increase from both controls. Auxiliary contacts are opened upon withdrawal operation of the rod drive control to prevent automatic or manual increase of pump speed. Auxiliary contacts on the pump speed control circuits open to prevent rod withdrawal while a signal for pump speed increase exists. An interlock prevents either pumps' discharge rotovalve from being opened, if they are not already open, while a rod is being withdrawn.

The full and partial scram circuitry of the rod control system each consists of a "string" of contact inputs that trip the full or partial scram relays to actuate a scram. (See Figs. 8.16 and 8.17.) Partial scram is used for plant conditions that require rapid power reduction to a subcritical condition, but not complete shutdown. Full scram is used for plant conditions that require complete and immediate reactor shutdown. In both cases, the required reactor power reduction rate is too rapid to be obtained by shimming action. A tabulation of scram conditions and other safety action initiated under scram conditions is given in Table 8-2.

Signals for safety action are also provided when nuclear or process instrumentation channels are rendered inoperable by failure of a channel component or by closure of an instrument calibration circuit. There are operate/test switches in all nuclear channels (Sec. 8.3) and in the water level, pressure and power-flow channels (Sec. 8.5.2.4) of the safety system.

Key switches permit bypass of safety system inputs to allow testing of safety system instrumentation during plant operation without scrambling the reactor. Backup signals that monitor the same parameter provide continuous operating protection when safety system circuits are being tested. The design characteristics of bypass circuits and administrative controls ensure that safety system input signals are not bypassed in a manner that would result in inadequate protection. Separate keys for each bypass switch are kept in a locked cabinet under control of the shift supervisor. All bypass switches for safety switches for safety system inputs are tabulated in Table 8-3, and the design philosophy for each is indicated. Actuation of any bypass switch will trip a remote annunciator.

TABLE 8-2

SAFETY ACTION INITIATED AT SCRAM SET POINTS*

scram condition	partial scram	full scram	action to open shutdown condenser	**action to close reactor building steam isolation valve	***cont. building isolation	operation of emergency core spray	prevention of reactor blowdown	reference section
low voltage on 2400 v bus 1A	X							8.5.2.8
low voltage on 2400 v bus 1B	X							8.5.2.8
turbine stop valve not fully open	X							8.5.2.12
reactor pressure above 1325 psig - pressure Channel 2	X		X					8.5.2.5.2
unsafe power-flow relation - power-flow Channel 2	X							8.5.2.5.3
low oil level in any of 29 control-rod drive accumulators	X							8.5.2.6
low gas pressure in any of 29 control-rod drive accumulators	X							8.5.2.13
pressing of reactor console partial-scram button	X							8.5.2.13
low voltage on both 2400 v busses (1A and 1B)		X						8.5.2.8
low voltage in reactor building motor control Center 1A		X						8.5.2.7
turbine building steam isolation valve not fully open		X	X					8.5.2.12
reactor building steam isolation valve not fully open		X	X					8.5.2.12
main condenser vacuum low		X		X				8.5.2.10
reactor water level high - water level Channel 1		X				****		8.5.2.5.1
reactor water level low - water level Channel 1		X		X		X	X	8.5.2.5.1
reactor water level high - water level Channel 2		X						8.5.2.5.1
reactor water level low - water level Channel 2		X		X		X	X	8.5.2.5.1
reactor pressure above 1350 psig - pressure Channel 1		X	X		X			8.5.2.5.2
unsafe power-flow relation - power-flow Channel 1		X						8.5.2.5.3
period less than 3 sec - log-n Channel 3		X						8.5.2.4.1
period less than 3 sec - log-n Channel 4		X						8.5.2.4.1
two of four nuclear level channels at above 115 percent on selected scale if power level is ≥ 5 percent of full power		X						8.5.2.4.2
signal above 115 percent on selected scale for either linear flux channel (nuclear Ch. 5 & 6) while power level is < 5 percent		X						8.5.2.4.2
pressing the all-rod scram button on reactor console		X						8.5.2.13
pressing the all-rod scram button in the containment building		X						8.5.2.13
manual operation of valves from the control room			X	X	X	X	X	5

*All scrams actuate separate main console alarms and cause transfer to the reserve feeder.

**Reactor building steam isolation valve is also closed automatically because of abnormally low main steam pressure (Sec. 8.5.2.11).

***High containment building pressure will automatically isolate the containment atmosphere; high radiation detected by the ventilation system monitors will cause the ventilation dampers to close.

****Emergency core spray pumps are tripped if they are operating (except for boron injection). (See Sec. 8.5.2.11 for explanation of gravity feed to core spray).

TABLE 8-3
SAFETY SYSTEM KEY BYPASS SWITCHES

switch no.	description	position	function	remarks
S-6	Log-n, Ch. 3	(1) bypass	to bypass withdraw prohibit and full scram outputs of this channel	Log-n Channels 3 & 4 bypassed when boiling occurs interlocks reactor start circuit and withdraw permit circuit so that switches S-6 & S-7 must be in normal position for start and withdraw permits initially and after partial and all-rod scrams
		(2) normal	period protection for reactor scram prohibit and withdraw permit circuits	
S-7	Log-n, Ch. 4	same as S-6	same as S-6	
S-8	reactor water level	(1) Ch. 1 bypass	to bypass Ch. 1 low and high full scram	prohibits Channels 1 & 2 from being bypassed simultaneously
		(2) normal	to duplicate full scram protection	
		(3) Ch. 2 bypass	to bypass Ch. 2 low and high full scram	
S-9	boron inject test switch	(1) test valves	to allow testing of boron valves without a scram	
		(2) normal	to prevent boron alarm and boron inject without scram	
		(3) test alarm	to allow testing of alarm and timer without a scram	
S-10	reactor pressure high	(1) Ch. 1 bypass	to bypass Ch. 1 high pressure full scram	prohibits Channels 1 & 2 from being bypassed simultaneously
		(2) normal	partial scram and full scram protection	
		(3) Ch. 2 bypass	to bypass Ch. 2 high pressure partial scram	
S-11	flow-power	(1) Ch. 1 bypass	to bypass Ch. 1 full scram	prohibits Channels 1 & 2 from being bypassed simultaneously
		(2) normal	partial scram and full scram protection	
		(3) Ch. 2 bypass	to bypass Ch. 2 partial scram	
S-12	Startup Ch. 1	(1) bypass	to bypass withdraw permit	Startup Channels 1 & 2 bypassed to prevent detector burnup after neutron flux goes above startup range interlocks withdraw permit circuit so that switches S-12 & S-13 must be in normal position to obtain initial withdraw permit
		(2) normal	period protection on withdraw permit	
S-13	Startup Ch. 2	same as S-12	same as S-12	
S-14	turbine stop valve	(1) bypass	to bypass partial scram	permits bypass of turbine stop valve position scram high pressure scrams provide backup
		(2) normal	partial scram protection	
S-15	accumulator oil level	(1) bypass	to bypass partial scram	permits bypass of all 29 control rod drive accumulator oil level scrams interlocks withdraw permit circuit
		(2) normal	partial scram protection	
S-16	accumulator gas pressure	(1) bypass	to bypass partial scram	permits bypass of all 29 control rod drive accumulator gas pressure scrams interlocks withdraw permit circuit
		(2) normal	partial scram protection	
S-17	bldg lock secured	(1) bypass	to bypass reactor start	bypasses both the personnel air lock and emergency air lock
		(2) normal	to require locks secured for reactor start permit	
S-18	turbine bldg isolation valve	(1) bypass	to bypass full scram	permits bypass of turbine building isolation valve position scram; high pressure scrams provide backup
		(2) normal	partial scram protection	
S-19	reactor bldg isolation valve	(1) bypass	to bypass full scram	permits bypass of reactor building isolation valve position scram; high pressure scrams provide backup
		(2) normal	partial scram protection	
S-20	condenser vacuum low	(1) bypass	to bypass full scram	permits bypass of low condenser vacuum scram; turbine trip from separate vacuum switches provides backup scram from turbine stop valve closure
		(2) normal	partial scram protection	
S-24	linear flux level Ch. 5 & 6	(1) Ch. 5 bypass	to bypass Ch. 5 full scram with power < 5 percent	prohibits Channels 5 & 6 from being bypassed at the same time while operating on 1 out of 2 scram circuits above 5 percent power, Ch. 5 & 6 are in 2 out of 4 coincidence with Ch. 7 & 8, and there is no bypass capability
		(2) normal	full scram protection from Ch. 5 & 6	
		(3) Ch. 6 bypass	to bypass Ch. 6 full scram with power < 5 percent	

8.5.2.4 Nuclear Instrumentation. The nuclear instrumentation system provides overlapped protection through the source, intermediate, and power ranges of operation. The two source and two intermediate range channels feed directly to the rod control system to provide period protection, and the four power range channels feed through auxiliary relays to provide linear flux level protection. Local and remote alarms are provided as an integral part of this system.

8.5.2.4.1 Period Protection. Reactor period protection is provided by:

(1) control rod withdraw prohibit for short periods on any of the four source and intermediate range channels, and

(2) full scram upon abnormally short periods, as monitored by either intermediate range channel.

An interlock in the reactor start circuit prohibits reactor startup unless there is a count rate of ≥ 2 counts/sec in both source range channels.

8.5.2.4.2 Flux Level Protection. Power range channels (5 through 8) each contain normally open contacts that de-energize the full scram relay through an auxiliary relay. Two circuits are used to provide adequate flux level protection for all ranges of operation. At a power level below 5 percent of full power, either extended power range channel scrams the reactor at 115 percent of the range selected. At a power level ≥ 5 percent of full power, the low trip stage relays of either standard power range channel trip to de-energize an auxiliary relay on the safety system panel. The output contacts of this relay change the power range input to the safety system to a two of four coincidence arrangement, so that trip output from a single channel will not cause scram. Since the response time of this relay is much shorter than that of the full scram relay, a scram does not occur during coincidence change. The coincidence circuitry avoids some of the false scrams that would otherwise result from fail-safe design.

Each of the narrow power range channels provides a signal to the power-to-flow channels of the safety system. When either of these nuclear instrumentation channels is calibrated, however, the respective bypass switches of the flow-power channels must also be bypassed or inadvertent reactor scram will result.

8.5.2.5 Process Instrumentation Analog Channels. Three of the process safety parameters -- reactor water level, reactor pressure, and reactor power-to-flow relation -- require analog channels which include amplifiers and trip stages to convert transmitter output to input for the rod control and other systems. The outputs from measuring devices are fed to voltage amplifiers remote from the high temperature environment to ensure reliable operation of solid-state components. These amplifiers provide outputs to trip stages and local indicating meters. Trip stage

outputs go to local and remote console-mounted alarms and to rod control and process instrumentation actuating devices. There are provisions for the testing and calibration of each amplifier, trip stage, and meter. There are six completely independent analog channels, each constructed in building-block fashion through the application of standard solid-state function modules.

8.5.2.5.1 Reactor Water Level (Two Channels). Input to each reactor water level channel (Fig. 8.17) is obtained from one of the two level transmitters, each of which supplies an output signal of 10 to 50 ma. The ma signal is adapted to the circuit through a ma input module that provides a 0 to 4-v d-c signal to a direct reading indicator (0 - 150 in.) and several trip stages each of which consists of a comparator and a relay driver module. There are four trip stages in Channel 1 and two trip stages in Channel 2. The comparator unit compares the 0 to 4-v d-c input with a setpoint manually established by a potentiometer, to produce a voltage signal output. Under normal level conditions, the comparator feeds a relay driver to energize the trip stage relays. Upon abnormal level, the trip stage relays are de-energized; for low level, the comparator feeds the relay driver directly, and for high level the comparator feeds another comparator, which then feeds the relay driver. The high-level arrangement is required to keep the output relays energized for fail-safe operation. The comparator modules are constructed as dual units; however, each section performs as an independent unit.

Upon high water level, Channel 1 gives local and control room anticipatory alarms and trips the emergency core spray pumps if they are operating (except if they are being used for boron injection) (Sec. 5.2.6.5). If the water level continues to rise to a preset value, both channels initiate a full scram and local and control room alarms.

Upon low water level, Channel 1 gives local and control room anticipatory alarms. If the water level continues to fall to a preset value, both channels initiate a full scram and local and control room alarms. Each channel also initiates automatic operation of the emergency core spray system, closure of the reactor building steam isolation valve and its bypass, and prevents an open blowdown valve in the 2-in. blowdown line from the decay heat system to the main condenser hotwell (Sec. 11.5.2). If the low level is combined with low reactor pressure, a signal from either level channel causes the low pressure core spray valve to open, which allows flow of water by gravity from the overhead storage tank to the core spray ring (Sec. 5.2.6.4).

The components for each channel are mounted in a drawer, along with the associated output trip stage relays and the range and trip setting potentiometers. The channels are tested by switching an equivalent analog signal into the ma input module, in place of the process transmitter output. In this test mode the channel circuit and trip stage settings are exercised through manual simulation of the signal input. This signal is provided from a separate test chassis.

During circuit test operation, the action of some output relays of the channel under test must be inhibited to prevent operation of critical plant equipment. This is accomplished by interlocking the output contacts with each operate/test switch. The scram outputs, however, are tripped when the test switch is placed in "test" position, and they must first be bypassed in order to avoid unintentional reactor scram.

8.5.2.5.2 Reactor Pressure (Two Channels). Input to each reactor pressure channel (Fig. 8.16) is obtained from one of the two 0-1600 psig pressure transmitters, which supplies an output signal of 10 to 50 ma. The sensing element of one transmitter is in the main steam line between the reactor outlet and the reactor building steam isolation valve. The other is in the steam line to the shutdown condenser. The ma signal is adapted to the circuit through a ma input module that provides a 0 to 4-v d-c signal to a direct-reading process indicator (0-1600 psig) and two trip stages; each trip stage consists of a comparator and a relay driver module. The comparator unit compares the 0 - 4-v d-c signal with a setpoint manually established by a potentiometer to produce a voltage signal output, and feeds another comparator unit that supplies voltage to de-energize the trip stage output relays. The second comparator is required to keep the output relays energized for fail-safe operation.

The trip stage relays provide inputs to a local alarm lamp, remote annunciators, the rod control part of the system, and the process instrumentation system. Upon high reactor pressure, Channel 2 gives local and control room anticipatory alarms. If the pressure increases to 1325 psig, a partial scram is initiated, along with local and control room alarms; the steam inlet valve to the shutdown condenser immediately opens, and the shutdown condenser system is automatically placed in operation (Sec. 5.2.2.4.1). If the pressure increases to 1350 psig, Channel 1 initiates a full scram and local and control room alarms. The same signal automatically places the shutdown condenser system in operation and closes the ventilation dampers and the 4-in. vent header valve from the containment building.

Both channels are mounted and tested with an equivalent analog signal, as described for the reactor water level equipment. During circuit testing, the action of the output relays of Channel 1 must be inhibited to prevent operation of critical plant equipment. Channel circuitry is provided to block these actions by interlocking the output contacts with the test-operate switch. Since Channel 2 initiates a partial scram rather than a full scram, the inputs to the containment building vent header valve and the ventilation system dampers are not actuated under normal trip conditions. The system, however, is interlocked so that channel testing of Channel 1 causes Channel 2 automatically to back up these outputs in case an actual scram occurs during test. This backup is accomplished through the test operate switches of both channels. The scram outputs are interlocked to trip and must be bypassed to avoid unintentional reactor scram.

8.5.2.5.3 Flow-Power Relation (Two Channels). If the total recirculation loop flow is not sufficient to meet thermal-hydraulic criteria for core cooling at the existing power level, safety system action occurs. There are two independent channels, one to initiate a partial scram and the other a full scram. Each channel receives signals from two recirculating loop flow square root converters and one power range nuclear instrumentation channel (separate monitors are used for the two channels). The flow inputs to each channel are 10 to 50 ma signals adapted to the circuit through two independent ma input modules that provide a signal to an adder/subtractor module. The 0 - 10-v d-c output of this unit represents total recirculation loop flow which is indicated locally (0 - 100 percent), and feeds to a second adder/subtractor unit. Here the total flow is compared with a 0 - 20-v d-c signal representing a 0 - 150 percent of plant power. This signal is derived from the nuclear instrumentation standard power range channels; Channel 7 provides signal to flow-power Channel 1, and Channel 8 supplies Channel 2. The adder/subtractor assembly produces a -5 to 0 to +5 v d-c signal representing the difference between actual power and 75 percent of actual flow; the $3/4$ factor, which is introduced by coefficient resistors, matches the slope of the linear scram limit curve to that of the normal power vs. flow operating curve (Fig. 8.9). The significance of the limit curve is given in Sec. 4.6.8 and in Appendix B.

The resultant flow-power signal feeds to an indicator and to two trip stages. Each trip stage consists of a comparator and a relay driver module. The comparator unit compares the d-c input with a setpoint manually established by a potentiometer to produce a voltage output signal, which directly feeds a relay driver to de-energize the trip stage relay when the flow is below that required for the existing power level. The comparator modules are constructed as dual units; but each unit operates independently. The trip stage relays provide inputs to a local alarm lamp, remote annunciator, and the partial scram (Channel 2) and full scram (Channel 1) circuits of the rod control part of the system.

Mounting of flow-power channel components and testing of the circuits are as described for the reactor water level channels (Sec. 8.5.2.5.1). The operate/test mode switches are interlocked with their full scram or partial scram outputs so that they must be bypassed during testing to avoid an unintentional reactor scram.

8.5.2.6 Control-Rod Drive Accumulator Oil Level and Gas Pressure. In the event of significant loss of oil or of gas pressure in any of the 29 control rod drive accumulators, the associated control rod will lose capability for scram, although its capability for electric motor driven insertion will not be affected. The safety system therefore incorporates a partial scram for low oil level or low gas pressure in any accumulator. (Visual control room alarms are given before the scram levels are reached.)

Two transmitters monitor control-rod drive oil level in each accumulator. These units are essentially travel limit switches consisting of a plunger arm that actuates

two normally closed contacts at a preset point corresponding to low oil level. A contact of one switch opens to de-energize an auxiliary relay, which de-energizes the partial scram relay of the rod control part of the system, and a contact of the other switch opens to actuate an annunciator on the main control console. Similarly, two pressure switches monitor accumulator gas pressure in each accumulator. At a preset point, low accumulator gas pressure actuates the pressure switches, opening the single normally-closed contact. The output of one pressure switch opens to de-energize an auxiliary relay that, in turn, de-energizes the partial scram relay, and the output of the other switch opens to actuate an annunciator on the main console. Gas pressure is also monitored independently by individual pressure transmitters.

8.5.2.7 Undervoltage - Reactor 480-v Motor Control Center 1A. Loss of voltage on reactor building motor control center (1A) means loss of electrical power to the control-rod drive hydraulic pumps as well as to the drive motors.

Loss of voltage or undervoltage on this control center for a time longer than that required for the reserve feed breaker (Sec. 9.1) to operate initiates a full scram. A time delay undervoltage relay is designed so that its two normally open contacts de-energize the full scram relay in the rod control part of the system and cause an annunciator on the main console to be actuated. The time delay is set for approximately 1 sec longer than the time that is normally required for the reserve feed breakers to operate automatically.

8.5.2.8 Main Bus Voltage. The electrical distribution system is fed from two 2400-v main buses. Loss of one bus causes the loss of various vital motor-driven auxiliaries. Since the loads are divided to provide sufficient equipment for the resultant hot shutdown condition, a partial scram is initiated if one bus is lost for a time longer than that required for automatic transfer to reserve feed. Loss of both buses for that same length of time, however, causes a full scram.

A time delay undervoltage relay (to allow transfer of feeder breakers) is connected to each 2400-v bus to initiate the partial scram and a console mounted alarm if either relay is de-energized. If neither bus has sufficient voltage, the full scram relay in the rod control system is also de-energized by the same undervoltage relays.

8.5.2.9 Building Locks. Containment vessel building locks actuate the rod control circuit directly. If these locks are not secured, the reactor-start relay in the rod control part of the system does not energize.

8.5.2.10 Main Condenser Vacuum. Loss of main condenser vacuum actuates an anticipatory alarm and then trips the turbine; turbine stop valve closure actuates a partial scram (Sec. 8.5.2.12). Since the reactor has lost its main heat sink under these conditions, a backup full scram signal is also provided from a separate main condenser vacuum transmitter. An auxiliary relay on the safety system panel is

de-energized when the scram set-point is reached; this relay initiates the scram and a console alarm, and closes the reactor building steam isolation valve. This last action provides a second full scram signal and starts the shutdown condenser system to provide a reactor heat sink.

8.5.2.11 Low Main Steam Pressure. A pressure transmitter in the 10-in. steam line to the shutdown condenser (Fig. 5.1) has a calibrated range of 0 to 100 psig with an output signal of 10 to 50 ma. Its output feeds an alarm unit relay that has a normally open contact, which is closed when the pressure is above the preset value. This normally open contact is connected in parallel with the normally open contacts from the reactor level transmitters in the safety system. A simultaneous low reactor pressure with a low reactor water level condition de-energizes a solenoid pilot valve, which automatically opens a pneumatic diaphragm valve, allowing water to flow from the overhead storage tank to the core spray nozzles (see Sec. 5.2.6.4).

A full range main steam pressure transmitter, located close to the turbine stop valve, initiates a signal at a preset low steam pressure to close the reactor building steam isolation valve. The purpose of this action is to isolate the primary system within the containment building if a primary system rupture occurs. The valve is closed by an output contact that de-energizes the tripping coil of the valve. This action leads to a full scram (Sec. 8.5.2.12).

Other main steam pressure transmitters are discussed in Sec. 5.1.1.2.

8.5.2.12 Reactor Building Steam Isolation, Turbine Building Steam Isolation, and Turbine Stop Valves. The reactor building steam isolation and turbine building steam isolation valves actuate a full scram when either is not in the full-open position. Travel limit switches that are an integral part of each valve have output contacts which open when the valves are not fully open. The contacts de-energize separate auxiliary relays that de-energize the full-scram relay in the rod control system and actuate separate annunciators on the main control panel. In addition to actuating a full scram, closure of either the reactor building or turbine building steam isolation valves starts the shutdown condenser by signals to control valves.

The turbine stop valve actuates a partial scram if not fully open; a travel limit switch opens to de-energize an auxiliary relay on the safety system panel. The normally open contacts of this relay de-energize the partial scram relay of the rod control part of the system and annunciate an alarm on the main console.

8.5.2.13 Manual Scram Switches. Two manual switches on the operator's bench actuate a full and partial scram by directly de-energizing a full or partial scram relay in the rod control part of the system. A third manual switch located in the containment building de-energizes the full-scram relay and actuates an annunciator on the main control panel.

8.6 IN-CORE FLUX MONITORING

The in-core flux monitoring system monitors the thermal neutron flux distribution in the reactor core by means of a traversing system that utilizes miniature detectors. The radial and axial flux profiles obtained by this method are used for fuel and rod programming and to locate hot spots. The system was developed as part of the LACBWR R&D program, and development progress and tests are reported in the amendments to the construction application (ACNP-62574).

8.6.1 General Description

The in-core flux monitoring system consists of nine information channels, with detectors that are positioned in preassigned guide tubes. These channels map the neutron flux over a range from 10^{12} to 5×10^{13} nv at full power, or from 10^{11} to 5×10^{12} nv at 10 percent of full power, by displaying the flux level and the detector position while traversing the core. The guide tubes are placed strategically within the reactor (Fig. 8.18) to obtain the flux distribution patterns.

Each channel (Fig. 8.19) consists of a miniature uncompensated ionization chamber, a drive unit with position transmitter, control equipment, and readout instrumentation. The detectors and drive cable assemblies are driven into the reactor core by individual drive units that also provide cable storage when the chambers are fully retracted. The detectors may be inserted either singly or in groups, with any number of detectors being assigned as a group, and travel may be stopped at any point during traverse to analyze a particular flux pattern. All of the required controls, power supplies, indicating lights, and readout instrumentation are on an instrument panel in the main control room. The drive mechanisms and associated devices are in the containment building.

The nine ion chambers have been calibrated in a known flux field to obtain the highest accuracy of indication. Since the core dwell-time of a traversing system detector is negligible compared to that of a fixed-location system detector, the reduced rate of boron depletion extends detector sensitivity and permits longer periods between detector calibrations. Recalibration is accomplished by comparing the flux reading of the operation detector to a flux reading from a spare calibrated ion chamber. The spare chamber is driven into the core through any of nine calibration guide tubes, by a portable hand-driven mechanism. These guide tubes connect directly to the operational guide tubes in a "Y" as shown in Fig. 8.20.

8.6.2 Detectors and Drive Cable

The detectors that measure thermal neutron flux are miniature uncompensated ionization chambers with a nominal sensitivity of 4×10^{-18} amp/nv at an operating voltage of 100-v d-c. They have a sensitive length of 0.5 in., a diameter of 0.160 in., and are constructed of commercially pure titanium and a small amount of nickel. The

chamber design temperature range is from 70 F to 750 F. There are three separately adjustable 100-v d-c power supplies, each supplying three channels.

All the detectors were previously calibrated in a known high flux field and are accurate to between +0 and -5 percent of the calibrated sensitivity, for integrated thermal fluxes below 1.4×10^{19} nvt. This nvt value corresponds to about 130 hr in a flux field of 3×10^{13} nv or 520 traverses of 15-min duration in the same flux field. A flux exposure greater than 1.4×10^{19} nvt requires ion chamber recalibration to maintain the specified accuracy. Detector operating life is such that, at 1.8×10^{20} nvt, less than 50 percent of the active boron coating has been depleted. This depletion results in a chamber sensitivity loss of less than 50 percent, so the detector signal output remains usable. The detectors are sized for insertion in the guide tubes and for complete traverse without binding or jamming. The chamber assembly is permanently attached to the end of a flexible cable, the core of which is the signal cable. The total reactivity worth of the drive cables with detectors is less than 0.05 percent when fully inserted.

The drive and signal cables are constructed of commercially pure titanium, with an inorganic insulating material in the signal cable portion to minimize the leakage of current under operating conditions. The cables are designed for 750 F. The outer wrap of the cables is a helically wound titanium wire that forms fixed pitch along the cable length. The helical wrap engages a hobbled wheel on the cable drive mechanisms to drive the cables in and out of the guide tubes. A connector at the lower end of each cable provides connection at the cable take-up reels. The cable insulation resistance between center conductor and either braid, or between braids, is 10^6 ohms at 700 F. The cable length for the operational units is approximately 50 ft, and approximately 70 ft is provided for the calibration unit.

The detectors and drive cables are stored on individual storage drums in a limited access area. The activity level of the cable is minimized by the choice of cable materials and its short dwell time in the core. However, entry to the limited access area near the storage drums is controlled and maintenance or calibration of drive units or cable is done only after a sufficient decay time from the last core insertion.

8.6.3 Drive Units

Nine operational drive units are used for detector positioning. Each drive unit consists of a drive motor with a motor-mounted friction-type brake, a gear reducer, a storage reel, a hobbled wheel and straight lead drive box, a position transmitter, and a rotary position indicating limit switch mechanism. The drive units are designed for 52 psig and 150 F. The drive motor is a 1/8-hp induction-type motor for use on 120-v a-c, single-phase, 60-cycle power. It is coupled directly to a right angle gear reducer with a planetary output stage, and does not put more than a 20-lb tension on the drive cable. The gear motor output speed is approximately 6 rpm, producing a detector travel rate of 5 ft/min. A 3 ft-lb, motor-mounted friction-type

120-v electric brake is included as part of the gear motor unit. A mechanical clutch, adjustable from 5 to 15 ft-lb, is provided to couple the gear motor drive to the hobbled drive wheel and lead drive box. The clutch limits the torque transmitted to the helical-wrapped drive cable. An extended motor shaft allows use of a hand crank.

Gleason-type takeup reels with continuous electrical contact are provided to store the drive cables. A rotary-type limit switch mechanism is coupled directly to the output shaft of the drive unit to stop the drive motor automatically at the full withdrawal position and to actuate a green indication light on the control panel. A Veeder Root position transmitter is installed on each of the drive mechanisms to provide signals for digital readout on the control panel. This unit transmits the number of drive wheel turns to a counter on the control panel. A contact signal is sent for each 1/10-revolution of the drive wheel. Two setscrews are included in the coupling to the drive wheel. The screws must be loosened to allow calibration of the counter in inches from the end of the run.

8.6.4 Guide Tubes

Guide tubes (Fig. 8.18) are installed to guide the detector and drive cables into the reactor core from the stored position. The tubes permit flux measurements through the complete 83-in. vertical core section. The guide tubes are in nine of the core vertical posts and are sealed at the upper end of the core. A high pressure coupling is provided at the reactor vessel nozzle since the guide tubes are exposed to full reactor pressure within the vessel. If any leakage develops in this in-core guide tubing, the tubing is capped at the coupling to the reactor nozzle and pressure is allowed to increase within the capped tubing.

The sections of guide tubing outside the reactor vessel are normally exposed to atmospheric pressure and a maximum temperature of 130 F. The tube bend radius is less than 30 in. for any bend, with a maximum of 180 deg in total angle bends for any one channel. The guide tubing is AISI Type-304 stainless steel, conforming to ASTM A269. The guide tubing inside the reactor vessel is of Inconel in accordance with ASTM B167 and is designed for 1400 psig and 750 F.

A static "Y" type switching unit is installed outside the reactor in each of the nine guide tubing sections. Additional guide tubing extends vertically approximately 18 ft through the biological shield from each "Y" connection and terminates at the face of the biological shield just above the gallery at el 656 ft. These tubing sections are used for calibration. The portable hand-driven mechanism for insertion of the calibration chamber is mounted on wheels and is moved along the gallery to any of these guide tubes.

Two thin sleeves or bands of silver coating are placed on the outside of each in-core guide-tubing at a fixed distance from each end of the core. The silver bands are

permanently attached to the guide tubes and are of sufficient size and thickness to perturb the flux so that detector position within the core is indicated on the recorder by a 25 percent reduction in flux measurement when the ion chamber goes past the silver bands. The bands do not interfere with guide tubes insertion into the reactor internals, and are covered with a protective jacket of Inconel to prevent silver redeposit. The bands are in the active core zone, one below the top of the active core and the other above the bottom.

8.6.5 Readout Instrumentation

Each flux monitoring channel has an independent flux amplifier, recorder, and position indicator for instrument readout. The readout instrumentation is designed for atmospheric pressure and 110 F, and all readout equipment is supplied with 120-v a-c, single-phase, 60-cycle power.

The output current from the ion chamber is amplified, and flux level is indicated on a meter and on a single-pen recorder fed from the flux amplifier. The amplifier full-scale current can be adjusted from 20 to 2000 μ a. This range corresponds to a flux reading of from 5×10^{12} to 5×10^{14} nv, based on a nominal chamber sensitivity of 4×10^{-18} amp/nv.

The system is primarily designed to allow complete flux mapping from 20 to 100 percent reactor power. However, flux indication is possible at lower levels. The maximum sensitivity of the flux amplifiers gives full scale deflection for a 20 μ a input. This input corresponds to about 5×10^{12} nv, based on the nominal detector sensitivity of 4×10^{-18} amp/nv. Therefore, flux level indication is available for about two decades below this level.

Detector position is displayed in a digital mode on separate indicators over the entire range of traverse. Indication covers three decades and is presented in three rows of ten miniature indicating lights, each unit representing 1/10-revolution of the drive wheel. An increasing count is indicated as the cable moves toward the takeup device, and readout is nonresetting since it adds and subtracts any drive-wheel motion. An adjustable limit switch is provided in each digital readout unit to de-energize the drive motor automatically at the fully inserted position, and to energize a red indication light at the control panel. Position is also indicated on the flux recorder as a function of flux measurement, since the flux level signal is perturbed as each chamber traverses the two silver sleeve locations in the in-core guide tube.

The system has a separate control panel that is an integral part of the main panel. This panel contains all the necessary controls, indicating lights and readout instrumentation. The "channel selected" indicating lights are arranged in a graphic core layout with the "control rod selected" indicating lights at the top of the panel. This assists the operator in relating the channel readout to the control rod configuration.

8.7 FAILED-FUEL-ELEMENT LOCATION ASSEMBLY

The failed-fuel-element location bundle is described in Sec. 4.3.8, and the system is discussed in Sec. 5.2.7. This system was developed in the LACBWR R&D program and development progress and the tests are reported in the amendments to ACNP-62574.

The system can be used periodically or whenever an increased activity level detected by the air ejector off-gas monitor indicates that failed fuel may be present. Defective core fuel assemblies presenting a radiation hazard are located and removed.

8.8 RADIATION MONITORING

This section completely supersedes Sec. 11 of ACNP-62574.

LACBWR radiation monitoring instrumentation consists of fixed plant surveillance equipment, portable survey meters, and personnel monitors (see Fig. 8.21). All instrumentation is calibrated prior to initial reactor startup and is recalibrated periodically. Check sources are included with each monitor for calibration checks; and the liquid, gas, particulate, and area monitors have control switches in the main control room which operate a solenoid, thus moving the check source in front of the detector. The equipment is maintained according to manufacturer recommendations, and routine preventive maintenance is performed periodically. The alarms are set using 10 CFR 20 as the guideline.

The monitoring systems are fail-safe. Local and remote alarms, visual and audible, indicate monitoring system malfunctions, including power supply failure. Lights on the remote alarm panel in the control room identify the specific cause of the particular channel alarm. Examples of trouble monitoring are given in Table 8-4. The circuitry is such that failure of one component or power supply in a multiple channel monitoring subsystem would not affect other power supplies or their channels.

TABLE 8-4

TROUBLE MONITORS

<u>monitor</u>	<u>trouble</u>
gas	low sample flow, torn filter paper, down-scale trip (except for the air ejector off-gas fission product monitor, which takes no flow samples and has no filter paper)
liquid	low sample flow and downscale trip
area	down-scale trip, high and low voltage trips

8.8.1 Fixed Monitors

8.8.1.1 Water Monitors. The liquid-waste and service water, the auxiliary component cooling water, and the plant circulating water discharge are monitored by three identical instrument systems. Each system includes the following Tracerlab equipment: an MW-2P liquid sampler (scintillation-type, gamma); an MD-5B gamma scintillation detector with a sensitivity of 250 cpm over a background of 250 cpm, for a concentration of $6 \times 10^{-6} \mu\text{c/cc}$ of Zn-65; a local flow indicator; a remote low flow alarm; a local control panel and cabinet assembly; and a remote RM-20B transistor precision log ratemeter and RM-40B power supply in the main control room, with an output to drive a strip chart recorder (same six-point recorder for three systems, including three spare points) and an adjustable level trip setting. A built-in $9 \mu\text{c}$, Cs-137 check source for calibration is also included.

The liquid waste and service water monitor is in a sampling line connected to the side of the service water line before it joins the plant circulating water discharge line. The component cooling water monitor is in a sampling line connected to the side of the main stream pipe. The plant circulating water discharge monitor is inside the turbine building in a sampling line connected to the side of the main stream pipe downstream from the turbine main condenser.

The output of each detector is fed to a log ratemeter, which has its own power supply and whose output is recorded in the control room. The ratemeter is equipped with an adjustable alarm contact that actuates a visual and audible alarm in the control room when the activity exceeds preset levels. Alarm actuation by the component cooling water alarm indicates primary water leakage into the component cooling system. Measures are then taken to locate the leak and correct the condition. If the liquid waste and service water monitor activates an alarm, manual action is taken to prevent release of liquid waste until the activity decreases below the alarm set point. This alarm set point corresponds to a 10 MPC concentration after dilution with main condenser circulating water. The plant circulating water discharge monitor serves as an additional check to monitor the radioactivity in the discharge line to the river.

8.8.1.2 Air Ejector Off-Gas Fission Product Monitor. This system includes the following Tracerlab equipment: a model MQ-6B low sensitivity gamma monitor, an MD-5B gamma scintillation detector with a range of $1 \times 10^0 \mu\text{c/cc}$ to $1 \times 10^3 \mu\text{c/cc}$, a local control panel and cabinet assembly, and a remote RM-20B transistor precision log ratemeter and RM-40B power supply in the main control room with an output to drive a single pen strip chart recorder. A built-in $9 \mu\text{c}$, Cs-137 check source is included for calibration. The level trip of the log ratemeter is adjustable over the entire range of the instrument.

The air ejector off-gas monitor is at the downstream exit of the 150-ft³ off-gas hold-up tank and upstream from the point where gas can be diverted to the storage tanks. The detector and the lead shielding are installed directly against the off-gas line (line gas is not sampled) in the cable vault area. The line is 12 in. from the floor and is in a straight horizontal run.

Since the monitor is downstream of the holdup tank, the gas is delayed long enough for N-16 to decay. From the holdup tank the gas passes the detector on its way to the stack.

The detector output is indicated on the ratemeter, and the ratemeter output is recorded in the control room. Adjustable alarm contacts permit visual and audible alarms in the control room to be set on the basis of allowable concentrations at the stack inlet. When ten times the maximum permissible concentration is reached, a signal closes the valve just beyond the holdup tank and prevents further release to the stack. The gas is then diverted to the recombiner along with steam, and stored in 12,000-gal storage tanks from which the gases can be released to the stack through a manually operated valve.

8.8.1.3 Stack Gaseous and Particulate Monitor. The system includes the following Tracerlab equipment: two MD-4B beta scintillation detector assemblies for particulate monitoring, one with a range of 3×10^{-9} to $2 \times 10^{-3} \mu\text{c/cc}$ for measuring immediate activity, and the other with a range of 3×10^{-10} to $2 \times 10^{-4} \mu\text{c/cc}$ for measuring delayed activity; a Model MA-1A(V) filter-type transport mechanism, used along with the beta detectors for particulate monitoring; one MG-1A gas sampler (scintillation-type, gamma) and MD-5B gamma scintillation detector assembly for gaseous monitoring, with a range of 3×10^{-6} to $2 \times 10^0 \mu\text{c/cc}$; a local flow switch that actuates a visual alarm for low flow; a local control panel and cabinet assembly; and three remote RM-20B transistor precision log ratemeters (with RM-40B power supplies in the main control room), with adjustable level trip settings to actuate visual and audible alarms and with outputs to drive a strip chart recorder. The strip-chart recorder is an eight-point recorder used for monitoring the stack gas and immediate and delayed particulate matter, containment exhaust gas and immediate and delayed particulate matter, and the plant immediate and delayed particulate matter (through mobile monitors). Each detector has a built-in $9 \mu\text{c}$ Cs-137 check source for calibration.

The monitoring detectors are installed in a sampling line which takes suction through an isokinetic nozzle mounted in the stack 115 ft off the base.

The continuous particulate monitor has two beta detectors to indicate activity on a moving filter paper. The first is at the point of activity collection on the movable filter and warns of any immediate significant increase in particulate radioactivity; the second beta detector monitors the filter paper approximately 6 hr after sample

collection. This arrangement decreases the background and increases the sensitivity of the second detector. The particulate-activity detector signals are sent to the transistorized log count-ratemeter and then to the control-room recorder.

The gas monitor measures the radioactivity of the gaseous effluent to the stack. The gaseous-activity detector signal is sent to another transistorized log count-ratemeter and then to the recorder. If the level of particulate or gaseous activity in the exhaust air would cause the maximum downwind average yearly concentrations to exceed the maximum permissible concentration, the monitor actuates a visual and audible alarm in the control room. Corrective action to reduce the activity released from the stack can then be taken; release of gas from the storage tanks is not permitted while this condition exists.

If the particulate or gaseous activity is ten times the alarm level, a control valve in the reactor plant vent header is manually closed, and the gas flow is to the gas storage tanks instead of to the stack. Other contained gases that normally vent to the stack are automatically shut off from the stack upon high activity signals from individual monitors (Secs. 8.8.1.2 and 8.8.1.4) rather than signals from the stack monitor. However, it is undesirable to cut off ventilation flow to the stack from the turbine building or elsewhere outside the containment building and closed piping systems even when these places have high activity; release from the stack is preferable to collection of activity in these locations.

8.8.1.4 Containment Building Air Exhaust Gaseous and Particulate Monitor. The containment building air exhaust gaseous and particulate monitoring system is identical to the stack gaseous and particulate monitoring system (Sec. 8.8.1.3) except that no isokinetic nozzle is included and the ranges are different. One beta detector has a range of 6×10^{-6} to $6 \times 10^{-3} \mu\text{c/cc}$ for measuring immediate particulate activity. The other beta detector has a range of 6×10^{-9} to $6 \times 10^{-4} \mu\text{c/cc}$ for measuring delayed particulate activity, and the gamma detector has a range of 6×10^{-3} to $6 \times 10^0 \mu\text{c/cc}$ for measuring gaseous activity.

The monitoring detectors are in a sampling line at the inlet end of the containment building exhaust duct, immediately beyond the exhaust air filters. If the level of particulate or gaseous activity in the exhaust air would cause a maximum downwind average yearly concentration of more than ten times the maximum permissible concentration, a visual and audible alarm in the control room is actuated and the ventilation dampers are automatically closed.

8.8.1.5 Plant Mobile Particulate Monitor. The components for the plant mobile particulate monitor are the same as those for the continuous particulate monitor portion of the stack gaseous and particulate monitoring system (Sec. 8.8.1.3), except that there are local rather than remote (control room) ratemeters and power supplies. Outlet connectors are provided at five locations in the containment building for combined power supply and remote signal and alarm indications to the main control room.

The plant particulate monitor continuously samples the air. The output of the log count-ratemeter is recorded in the control room. Local visual and audible alarms and control-room visual and audible alarms are actuated upon high activity concentration.

8.8.1.6 Fixed-Location Area Monitors. The fixed-location area monitor system includes the following Tracerlab equipment: 13 Model TA-6A-10R remote detectors (Geiger-Mueller) and two TA-6A-100R remote detectors (Geiger-Mueller), local meter indications, visual and audible local alarms, remote visual and audible alarms in the main control room, and 15 TA-3 Tracerlab model log count-ratemeters (with adjustable level trip settings and electrometer amplifiers) to drive a 16-point strip chart recorder in the main control room. A 1- μ c check source of Sr-90 is in each detector.

Area monitors indicate radiation levels in areas normally accessible to personnel but which may present a hazard if levels are not known prior to entry or during periods of occupancy.

The ranges and locations of the fixed-location area monitors are given in Table 8-5.

The halogen-quenched Geiger-Mueller detectors supply signals to the count-ratemeters. At high radiation levels, current is sent directly to the readout device, which frees the system from cable capacitance, stray fields, temperature drifts, and other instabilities. The count rate is recorded in the control room.

8.8.1.7 Portal Monitor. This monitor is a Technical Associates Portal Monitor, Model PPM-8. The system includes an upright framework with six beta-gamma Geiger-Mueller detectors in the two upright sides (one overhead, and one in the bottom or base), a remote eight-channel control panel (with indicating radiation level meters and individual alarm set points for each channel), and visual and audible local alarms. The full-scale readings of the beta-gamma detectors are 1000 cpm for the foot monitor and 500 cpm for both the head monitor and the body monitor.

The portal monitor is located in the turbine building change room hallway. It is used to check all personnel leaving the containment building and turbine building control areas for radiation contamination. One of the linear count ratemeters mentioned in Sec. 8.8.2.5 is used in conjunction with the portal monitor subsystem.

TABLE 8-5

FIXED-LOCATION AREA MONITORS

<u>unit no.</u>	<u>range</u>	<u>detector location</u>	<u>indicator location</u>
1	0.01 mr/hr to 10 r/hr	west side of main airlock on containment bldg wall at el 648 ft	at entrance to main airlock
2	0.01 mr/hr to 10 r/hr	containment bldg south wall at el 707 ft	integral with detector
3	0.01 mr/hr to 10 r/hr	outside of shield wall for primary system ion ex- changer	integral with detector
4	0.01 mr/hr to 10 r/hr	north shield wall of turbine bldg at el 660 ft	integral with detector
5	0.01 mr/hr to 10 r/hr	southeast wall of contain- ment bldg near emergency airlock at el 650 ft	integral with detector
6	0.01 mr/hr to 10 r/hr	southwest wall of contain- ment bldg at el 673 ft	integral with detector
7	0.1 mr/hr to 100 r/hr	near in-core flux monitor- ing units at el 637 ft	northwest side of reactor shield, at el 647 ft
8	0.1 mr/hr to 100 r/hr	on the platform on the west side of reactor at el 632 ft	northwest side of reactor shield at el 647 ft
9	0.01 mr/hr to 10 r/hr	main work area of waste treatment bldg	integral with detector
10	0.01 mr/hr to 10 r/hr	basement area of waste treatment bldg	next to indicator no. 9
11	0.01 mr/hr to 10 r/hr	north turbine shield wall in turbine bldg at el 674 ft	integral with detector
12	0.01 mr/hr to 10 r/hr	control room south wall	integral with detector

Table 8-5 - Fixed-Location Area Monitors (cont'd)

unit no.	range	detector location	indicator location
13	0.01 mr/hr to 10 r/hr	east wall of turbine bldg at el 648 ft	integral with detector
14	0.01 mr/hr to 10 r/hr	west wall of turbine bldg at el 648 ft	integral with detector
15	0.01 mr/hr to 10 r/hr	pipng tunnel outside the waste water storage tank area at el 633 ft	head of stairs in turbine bldg at south side of feedwater pumps

8.8.2 Portable Survey Meters

8.8.2.1 Alpha-Beta-Gamma Ionization Survey Meters. These consist of three Technical Associates Juno Survey Meters, Model SRJ-7, with full-scale ranges of 50, 500, and 5000 mr/hr, and one Technical Associates Juno Survey Meter, Model HRJ-7, with full-scale ranges of 250, 2500, and 25,000 mr/hr.

8.8.2.2 Alpha-Beta-Gamma Thin-Window Geiger-Mueller Survey Meters. These are two Victoreen portable survey meters, Thyac 11, Model 489-35 with full-scale ranges of 0.2, 2, and 20 mr/hr.

8.8.2.3 Alpha Gas Flow Proportional Survey Meter. This is an Eberline portable gas proportional alpha counter, Model PAC-3G with full-scale ranges of 1000, 10,000, and 100,000 counts/min.

8.8.2.4 Neutron Survey Meter. This is an Eberline portable fast-slow neutron counter, Model PNC-1, with full-scale ranges of 10 to 10,000 n/cm²-sec.

8.8.2.5 Linear Glass-Wall Geiger-Mueller Countermeters. These are two Tracerlab alpha-beta-gamma survey meters, Model SU-21 with full-scale ranges of 500, 5000, and 50,000 counts/min or 0.5, 5, and 50 mr/hr.

8.8.3 Personnel Monitors

The operating staff and visitors carry film badges and standard self-read and charger-read pocket dosimeters to measure cumulative dose. The pocket dosimeters consist of:

- (1) Tracerlab pocket dosimeters, Model K-112, with a range of 0-200 mr (self-reading)

(2) Tracerlab pocket dosimeters, Model L-65, with a range of 0-200 mr (charger-read)

(3) A Tracerlab charger-reader, Model L-60

(4) A Tracerlab charger, Model K-135-C

8.8.4 Other Instrumentation

Additional laboratory and health physics instruments furnished by Dairyland Power Cooperative consist of the following:

(1) Laboratory Scales	4
(2) Laboratory Monitors	4
(3) Wide Spectrum Pulse Height Analyzer	1
(4) Dosimeters, Low Range	85
(5) Dosimeters, Intermediate Range	10
(6) Dosimeters, High Range	10
(7) Dosimeters Charger	2
(8) Remote Radiation Monitor	1
(9) Hand and Foot Counter	2
(10) Survey Meter, High Range C.P.	4
(11) Survey Meter, Low Range C.P.	8
(12) Survey Meter, G-M	8
(13) Survey Meter, Alpha	2
(14) Survey Meter, Neutron	1

8.8.5 Alarms

The radiation monitoring system alarms are listed in Table 8-6.

TABLE 8-6

RADIATION MONITORING SYSTEM ALARMS

<u>window</u>	<u>nameplate</u>	<u>actuating device</u>
C 5-1	stack gaseous radiation high	radiation monitor
C 5-2	stack particulate radiation high	radiation monitors
C 5-3	stack monitor trouble	radiation monitors
C 6-1	air ejector off-gas monitor high	radiation monitor
C 6-2	air ejector off-gas monitor trouble	radiation monitor
C 6-3	liquid waste & service water radiation high	radiation monitor
C 6-4	liquid waste & service water monitor trouble	flow switch
C 7-1	containment bldg. air exhaust gas radiation high	radiation monitor
C 7-2	containment bldg. air exhaust partic. radiation high	radiation monitors
C 7-3	containment bldg. air exhaust monitor trouble	radiation monitors
C 7-4	area monitors radiation high	area monitors
C 8-3	turbine-condenser cooling water radiation high	radiation monitor
C 8-4	turbine-condenser cooling water monitor trouble	flow switch
C 9-1	component cooling water radiation high	radiation monitor
C 9-2	component-cooling water monitor trouble	flow switch
C 9-3	mobile plant particulate mon. radiation high	radiation monitor
C 9-4	mobile plant particulate monitor trouble	radiation monitor

A1-1	A2-1	A3-1	A4-1	A5-1	A6-1	A7-1	A8-1	A9-1
GEN. OR AUX. BKRS. TRIPPED	69KV GEN. OCB AIR PRESS. LOW	69KV RES. AUX. TRANS. O.C.B. AIR PRESS. LOW	MAIN POWER TRANS. LIQUID LEVEL LOW	UNIT AUX. TRANS. HOT SPOT TEMP	VOLTAGE REGULATOR TRIPPED	GEN. ROTOR TEMP. HIGH	EMER. DIESEL AC GEN. RUNNING	REACTOR PLANT BATT. CHARGER FAILURE
A1-2	A2-2	A3-2	A4-2	A5-2	A6-2	A7-2	A8-2	A9-2
69KV TIE LINE AIR PRESS. LOW	GENERATOR LOCK-OUT RELAY TRIPPED	RES. AUX. TRANS. LOCK-OUT RELAY TRIPPED	MAIN POWER TRANS. TROUBLE	480V TRANS. 1A OR 1B HIGH TEMP.	VOLTAGE REGULATOR OUTPUT (HI-LO)	GEN. STATOR TEMP. HIGH	EMER. DIESEL GENERATOR TROUBLE	GEN. PLANT BATT. CHARGER FAILURE
A1-3	A2-3	A3-3	A4-3	A5-3	A6-3	A7-3	A8-3	A9-3
69KV TIE LINE LOCK-OUT RELAY TRIPPED	GENERATOR LOCK-OUT RELAY TRIPPED	RES. AUX. TRANS. TROUBLE	MAIN POWER TRANS. SUDDEN PRESSURE	480V FEED PARALLELED	2400V SWGR 1A OR 1B DC CONT VOLT LOW	HYDROGEN SYSTEM TROUBLE	EMER. HYDROGEN SEAL OIL PUMP RUNNING	REACTOR PLANT 125VDC GROUND
A1-4	A2-4	A3-4	A4-4	A5-4	A6-4	A7-4	A8-4	A9-4
69KV TIE LINE BACKUP RELAY TRIPPED	GEN. FIELD GROUND RELAY TRIPPED	RES. AUX. TRANS. SUDDEN PRESSURE	MAIN POWER TRANS. COOLING FEED AUTO TRIP	480V. ESS. BUS. 1A OR 1B D.C. VOLTAGE LOW	480V. SW. GR. 1A OR 1B D.C. CONT VOLT LOW	TURBINE EXPANSION VIBRATION REC.	120 V. AC NON-INTERR. BUS RES SUPPLY	GENERATOR PLANT 125VDC GROUND

ANNUNCIATOR "A" WINDOW ENGRAVING (A76, PNL. C LEFT)

B1-1	B2-1	B3-1	B4-1	B5-1	B6-1	B7-1	B8-1	B9-1	B10-1	B11-1	B12-1	B13-1	B14-1
TURBINE-COND AUX. DEVICE TRIPPED	TURBINE ROTOR STOPPED	TURBINE EXHAUST HOOD HIGH TEMP	HOTWELL CONDENSATE HIGH CONDUCT.	GLAND STEAM GENERATOR LEVEL (HI)	LOW SET REACTOR SAFETY VALVE OPEN	HEATER NO. 1 LEVEL (HI)	DIESEL HP SERVICE WATER PUMP RUNNING	HEATING BOILER WATER (HI-LO)	SCREEN DIFF. LEVEL (HI)	AIR COMPRESS. 1A OR 1B LUBE OIL PRESS (LO)	CHANGE ROOM DRAIN TANK HI (200 GAL)	B26. OIL HDR. PRESS (LO) RFP TRIP	MAIN STM. BY-PASS VALVE ACQU. TANK PRESS (LO)
B1-2	B2-2	B3-2	B4-2	B5-2	B6-2	B7-2	B8-2	B9-2	B10-2	B11-2	B12-2	B13-2	B14-2
TURBINE OIL PUMP RUNNING	TURBINE OIL TANK (HI-LO)	H.P. SERVICE WATER TANK LEVEL (HI-LO)	HOTWELL LEVEL (HI-LO)	GLAND STEAM PRESS. (LO)	HIGH SET REACTOR SAFETY VALVE OPEN	HEATER NO. 2 LEVEL (HI)	DIESEL HP SERVICE WATER TROUBLE	HEATING BOILER STEAM (HI-LO)	DEEPWELL WATER PRESS (HI-LO)	CONTROL AIR HEADER PRESS (LO)	CHANGE ROOM DRAIN TANK OVER-FLOW	FEEDWATER AUX. OIL PUMPS AUTO START	MAIN STEAM BY-PASS VALVE ACQU. TANK PRESS (LO)
B1-3	B2-3	B3-3	B4-3	B5-3	B6-3	B7-3	B8-3	B9-3	B10-3	B11-3	B12-3	B13-3	B14-3
EMER. BRG OIL PUMP RUNNING	TURBINE LOW VACUUM PRE-TRIP	TURBINE METAL OIL SEAL OIL PUMP RUNNING	HOTWELL COND. TO DEMIN. FLOW (LO)	FLASH TANK WATER LEVEL (HI)	DEMIN. VIRGIN WATER TANK LEVEL (HI-LO)	HEATER NO. 3 LEVEL (HI)	DIESEL HP SERV. WATER CONT. MAN-OFF	HEATING BOILER FLAME FAILURE	LP SERV. WATER PUMP 1A OR 1B AUTO-START	STATION AIR HEADER PRESS (LO)	FEEDWATER HYD. COUP TANKS TEMP (HI)	FEEDWATER BRO. OIL TANKS LEVEL (LO)	MAIN STEAM BY-PASS VALVE ACQU. TANK PRESS (LO)
B1-4	B2-4	B3-4	B4-4	B5-4	B6-4	B7-4	B8-4	B9-4	B10-4	B11-4	B12-4	B13-4	B14-4
INVERTER MOTOR OVERLOAD	CONDENSER VACUUM PUMP TROUBLE	TURBINE EXHAUST WATER SPRAY ON	COND. DEMIN. OUTLET CONDUCT. (HI)	DEMIN. WATER TRANS. HDR. PRESS (LO)	CONDENSER STORAGE TANK LEVEL (HI-LO)	MAKE-UP DEMIN. TROUBLE	FIRE HEADER WATER PRESS (LO)	STEAM SEPARATOR H ₂ O STOR. TANK LEVEL (HI)	STM. RE-HEATER WATER LEVEL (HI)	MOTOR BRG. TEMP HIGH RECORDER D-13	FEEDWATER HYD. COUP TANKS LEVEL (LO)	FEEDWATER HYD. COUP WATER PUMP 1A OR 1B TRIP	AIR COMPRESSOR 1A OR 1B AUTO. START

ANNUNCIATOR "B" WINDOW ENGRAVING (B49, PNL. C RIGHT)

C1-1	C2-1	C3-1	C4-1	C5-1	C6-1	C7-1	C8-1	C9-1	C10-1	C11-1	C12-1	C13-1	C14-1
REACTOR WATER LEVEL (HI)	CONTROL RODS HYD. RESERV. (LO)	SODIUM PENT. INJECTION	SHUTDOWN COND. WATER LEVEL (HI-LO)	STACK GASEOUS RADIATION (HI)	AIR EJECTOR OFF-GAS MONITOR (HI)	CONTAIN. BLDG. AIR EXH. GAS RADIATION (HI)	CONTAIN. BLDG. DAMPERS CLOSED	COMPONENT COOLING WATER RADIATION (HI)	CHANNEL 1 LOW COUNT RATE	CHANNEL 3 SHORT PERIOD	CHANNEL 5 FLUX LEVEL (LO)	CHANNEL 5 FLUX LEVEL (HI)	REACTOR BLDG. SUMP LEVEL (HI)
C1-2	C2-2	C3-2	C4-2	C5-2	C6-2	C7-2	C8-2	C9-2	C10-2	C11-2	C12-2	C13-2	C14-2
REACTOR WATER LEVEL (LO)	CONTROL RODS N ₂ BOTTLE PRESS (LO)	SODIUM PENT. LEVEL (LO)	OVERHEAD STORAGE TANK LEVEL (HI)	STACK PARTICULATE RADIATION (HI)	AIR EJECTOR OFF-GAS MON. TROUBLE	CONTAIN. BLDG. AIR EXH. PARTIC. RADIATION (HI)	REACTOR VENT EXH. FLOW (LO)	COMPONENT COOLING WATER MON. TROUBLE	CHANNEL 2 LOW COUNT RATE	CHANNEL 4 SHORT PERIOD	CHANNEL 6 FLUX LEVEL (LO)	CHANNEL 6 FLUX LEVEL (HI)	WASTE DISPOSAL BUILDING SUMP LEVEL (HI)
C1-3	C2-3	C3-3	C4-3	C5-3	C6-3	C7-3	C8-3	C9-3	C10-3	C11-3	C12-3	C13-3	C14-3
STEAM TO TURB. PRESS. (LO)	CONDENSER PUMP 1A OR 1B AUTO START	SODIUM PENT. TEMP. (LO)	OVERHEAD STORAGE TANK LEVEL (LO)	STACK MONITOR TROUBLE	LIQUID WASTE E SERV. WATER RADIATION (HI)	CONTAIN. BLDG. AIR EXH. MON. TROUBLE	TURB-COND PARTICULATE RADIATION (HI)	MOBILE PLANT PARTICULATE MON. RAD. (HI)	CHANNEL 1 SHORT PERIOD	NUCLEAR INSTRUMENTATION TROUBLE	CHANNEL 7 FLUX LEVEL (LO)	CHANNEL 7 FLUX LEVEL (HI)	WASTE GAS STORAGE VAULT SUMP LEVEL (HI)
C1-4	C2-4	C3-4	C4-4	C5-4	C6-4	C7-4	C8-4	C9-4	C10-4	C11-4	C12-4	C13-4	C14-4
REACTOR STEAM OUTLET PRESS. (HI)		REACTOR VESSEL TEMP. (LO)	REACTOR BLDG. PRESS. (HI)	SODIUM PENT. LEVEL LOW	LIQUID WASTE E SERV. WATER MON. TROUBLE	AREA MONITORS RADIATION (HI)	TURB-COND. COOLING WATER MON. TROUBLE	MOBILE PLANT PARTICULATE MON. TROUBLE	CHANNEL 2 SHORT PERIOD		CHANNEL 8 FLUX LEVEL (LO)	CHANNEL 8 FLUX LEVEL (HI)	TURB. ROOM SUMP LEVEL (HI)

ANNUNCIATOR "C" WINDOW ENGRAVING (C-20, PNL. D RIGHT)

D1-1	D2-1	D3-1	D4-1	D5-1	D6-1	D7-1
CONTROL ROD GAS PRESS (LO)	REACTOR WATER LEVEL (HI-LO)	MAIN STEAM ISOLATION VALVE NOT FULL OPEN	DFACTOR PARTIAL SCRAM	CHANNEL 3 SHORT PERIOD	CHANNEL 7 FLUX LEVEL (HI)	REACTOR MCC 1A VOLTAGE (LO)
D1-2	D2-2	D3-2	D4-2	D5-2	D6-2	D7-2
CONTROL ROD OIL LEVEL (LO)	REACTOR WATER LEVEL (LO-LO)	MAIN STEAM SHUT-OFF VALVE NOT FULL OPEN	REACTOR ALL ROD SCRAM	CHANNEL 4 SHORT PERIOD	CHANNEL 8 FLUX LEVEL (HI)	2400V BUS 1A VOLTAGE (LO)
D1-3	D2-3	D3-3	D4-3	D5-3	D6-3	D7-3
	REACTOR STEAM OUTLET PRESS (HI)	TURBINE STOP VALVE NOT FULL OPEN	REACTOR BLDG. MANUAL SCRAM	CHANNEL 5 FLUX LEVEL (HI)	POWER FLUX LEVEL CHANNELS IN 1 OF 2 LOGIC	2400V BUS 1B VOLTAGE (LO)
D1-4	D2-4	D3-4	D4-4	D5-4	D6-4	D7-4
	POWER RECIRC. FLOW ABNORMAL	TURBINE CONDENSER VACUUM (LO)	KEY SWITCH IN BY-PASS	CHANNEL 6 FLUX LEVEL (HI)		

ANNUNCIATOR "D" WINDOW ENGRAVING (D20A, PNL. D LEFT)
"FIRST OUT" ALARMS

E1-1	E2-1	E3-1	E4-1	E5-1	E6-1	E7-1	E8-1	E9-1	E10-1	E11-1	E12-1	E13-1	E14-1	E15-1	E16-1
REACTOR AUX. DEVICE TRIPPED	F.C. PUMP 1A TRIPPED	POWER RECIRC. FLOW ABNORMAL		SEAL INJECT RES. LEVEL (HI)	F.C. PUMP 1A SEAL INJECT FLOW (LO)	SEAL INJECT SUPPLY FLOW (LO)	F.C. PUMP 1A LEAKOFF FLOW (HI-LO)	CONTROL ROD HYDR. PUMP 1A OR 1B RUNNING	MIXED BED EXCH. CONDUCT. (HI)	RECOMB. COMP. PRESS (HI)	12,000 GAL. STOR. TANKS 1A OR 1B PRESS. (HI)	COMP. COOL SURGE TANK LEVEL (LO)	EMERG. CORE SPRAY FLOW (LO)	REACTOR AUX. TEMP. (HI)	
E1-2	E2-2	E3-2	E4-2	E5-2	E6-2	E7-2	E8-2	E9-2	E10-2	E11-2	E12-2	E13-2	E14-2	E15-2	E16-2
HYD. VALVE ACCUM. WATER LEVEL ABNORMAL	F.C. PUMP 1B TRIPPED	F.C. PUMP 1A OR 1B DEVIATION	F.C. PUMPS HYD. COUPL. OR OIL PUMP AUTO START	SEAL INJECT RES. LEVEL (LO)	F.C. PUMP 1B SEAL INJECT FLOW (LO)	SEAL INJECT TEMP. (HI)	F.C. PUMP 1B LEAKOFF FLOW (HI-LO)	SHIELD COOL FLOW (LO)	MIXED BED EXCH. TO REGEN. COOLER FLOW (LO)		12,000 GAL. STOR. TANKS EXP. MIXTURE	COMP. COOL WATER TEMP. (HI)	FUEL ELEM. STOR. WELL LEVEL (LO)		
E1-3	E2-3	E3-3	E4-3	E5-3	E6-3	E7-3	E8-3	E9-3	E10-3	E11-3	E12-3	E13-3	E14-3	E15-3	E16-3
HY. VALVE ACCUM. WATER PRESS. ABNOR.	F.C. PUMP 1A COUPL. WATER TROUBLE	F.C. PUMP 1A LUBE OIL TROUBLE		F.C. PUMP 1A LEAKOFF TEMP. (HI)	F.C. PUMP 1A BACKUP SEAL LEAKOFF (HI)	F.C. PUMPS SEAL WATER FILTER (DP) HI	CONTROL ROD SEAL WATER FLOW (LO)	SHIELD COOL SURGE TANK LEVEL (LO)	MIXED BED EXCH. INLET TEMP. (HI)	BLDG. STEAM TO RECOMB. FLOW (LO)	6,000 GAL. RETEN. TANK 1A OR 1B LEVEL (HI)	DECAY HEAT COOLER OUTLET TEMP. (LO)	WASTE WATER STORAGE TANKS LEVEL (HI)		
E1-4	E2-4	E3-4	E4-4	E5-4	E6-4	E7-4	E8-4	E9-4	E10-4	E11-4	E12-4	E13-4	E14-4	E15-4	E16-4
HYD. ACCUM. SUMP TANK LEVEL (LO)	F.C. PUMP 1B COUPL. WATER TROUBLE	F.C. PUMP 1B LUBE OIL TROUBLE		F.C. PUMP 1B LEAKOFF TEMP. (HI)	F.C. PUMP 1B BACKUP SEAL LEAKOFF (HI)	CONTROL ROD SEAL WATER FILTER (DP) HI	CONTROL ROD NOZ. EFFLUENT FLOW (LO)	SHIELD COOL PUMP DISCH. FLOW (LO)	EJECTOR AFTER COOLER HOLD-UP TEMPERATURE (HI)	RECOMB. OUTLET TEMPERATURE (HI)	SPENT RESIN TANK LEVEL (HI)	COMPONENT COOLING FLOW (LO)			

ANNUNCIATOR "E" WINDOW ENGRAVING (F48, PNL. E)

A = reset gain of control system controller
 C = rate gain of control system controller
 K = proportional gain of controller
 N = pump impeller speed
 P_2 = turbine inlet pressure

P_{set} = pressure set point
 S = laplace operator
 X = scoop tube position
 ϵ = error signal
 τ_p = time constant of pressure transducer

FORCED-CIRCULATION-PUMP SCOOP TUBE POSITION CONTROL

FIG. 8.2

P_2 = turbine inlet pressure

P_{set} = pressure set point

S = laplace operator

S'_{BP} = a function of bypass valve position

ϵ = error signal

σ = control signal to bypass valve actuator

τ_{BP} = time constant of bypass valve actuator

MAIN STEAM BYPASS VALVE CONTROL

FIG. 8.3

RAMP DECREASE IN TURBINE POWER
 DEMAND OF 1.5 MWE/MIN
 INITIAL POWER = 100%

FIG. 8.4

RAMP INCREASE IN TURBINE POWER
DEMAND OF 1.5 MWE/MIN
INITIAL POWER = 60%

FIG. 8.5

REACTOR VESSEL
PRESSURE, P_1 , psia

1300
1290
1280

TURBINE INLET
PRESSURE, P_2 , psia

1265
1260
1255
1250

RECIRCULATION
FLOW RATE,
 W_{in} GPM

35,000
30,000
25,000
20,000

REACTOR POWER
LEVEL, n %

120
110
100
90
80

MAXIMUM HEAT
FLUX, $Q_{B, max}$
Btu/sec - ft²

140
120
100
80

0 5 10 15 20 25 30 35 40 45 50 55 60 65 70
TIME, SEC

STEP INCREASE IN TURBINE POWER
DEMAND OF 5 MWE
INITIAL POWER = 90%

FIG. 8.6

REACTOR VESSEL
PRESSURE P_1 , psia

1290
1280
1270

TURBINE INLET
PRESSURE P_2 , psia

1265
1255
1245

RECIRCULATION
FLOW RATE,
 W_{in} GPM

30,000
20,000
10,000

REACTOR POWER
LEVEL, n %

100
80
60

MAXIMUM HEAT
FLUX Q_B^{max}
Btu/sec - ft²

120
100
80
60
40

0 5 10 15 20 25 30 35 40 45 50 55 60 65 70 75
TIME, SEC

STEP INCREASE IN TURBINE POWER
DEMAND OF 5 MWE
INITIAL POWER = 60%

FIG. 8.7

STEP DECREASE IN TURBINE POWER DEMAND
OF 25 MWE
INITIAL POWER = 100%

FIG. 8.8

POWER-FLOW RELATIONSHIPS FOR AUTOMATIC CONTROL

FIG. 8.9

NUCLEAR INSTRUMENTATION RANGES

FIG. 8.10

LACBWR INSTRUMENT SLEEVE LOCATION PLAN AND ELEVATION

FIG. 8.11

LACBWR NUCLEAR INSTRUMENTATION FUNCTIONAL BLOCK DIAGRAM

FIG. 8.12

LACBWR CONTROL ROD DRIVE MECHANISM SHOWING
SEAL INJECTION, FLUSHING, AND BLOWDOWN PORTS
(FOR DETAILS SEE AMENDMENT 11)

FIG. 8.13

BUFFER WATER DISPLACEMENT FLOW PATHS

FIG. 8.14

NOTE 1
S9 - THREE POSITION SWITCH
POSITION 1-NORMAL
POSITION 2-VALVE TEST FOR BORON INJECTION
POSITION 3-ALARM TEST

SYMBOLS

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

LACBWR
LOGIC BLOCK DIAGRAM
ROD CONTROL

DETAILED DESIGN-CONTRACT NO. AT (11-1) 8801
DWG. NO. 41-503-209
SHEET 1 OF 1

LACBWR LOGIC BLOCK DIAGRAM ROD CONTROL

LACBWR SAFETY SYSTEM FUNCTIONAL BLOCK DIAGRAM

FIG. 8.16

LACBWR SAFETY SYSTEM FUNCTIONAL BLOCK DIAGRAM

FIG. 8.17

FIG. 8.18

LACBWR IN CORE FLUX MONITORING BLOCK DIAGRAM

IN CORE FLUX MONITORING LAY OUT

FIG. 8.20

LACBWR HEALTH AND RADIATION MONITORING SYSTEM BLOCK DIAGRAM

9. ELECTRICAL DISTRIBUTION (Figs. 9.1 through 9.3)

This section supersedes and makes obsolete Sec. 8 of the Hazards Summary Report for Construction Authorization of LACBWR (ACNP-62574).

9.1 POWER REQUIREMENTS

The main electrical auxiliary supply is shown in Fig. 9.1. The generator plant electrical auxiliary supply is shown in Figs. 9.2 and 9.3. For normal operation, all power for the reactor and turbine auxiliaries is supplied at 2400 v through the main feed breakers from the unit auxiliary transformer connected to the generator. During startup of the reactor and turbine plant or if the normal auxiliary source fails during power operation, power is supplied through reserve feed breakers from the reserve auxiliary transformer connected to the 69-kv bus of the external grid. Any automatic trip of a main feed breaker automatically closes the corresponding reserve feed breaker. The unit auxiliary transformer is rated at 13.2 - 2.4 kv, 5000 kva, and the reserve auxiliary transformer is rated at 69 - 2.4 kv, 5000 kva. Therefore, both 2400-v busses (1A and 1B) have alternate supplies of full capacity.

The 2400 - 480-v auxiliary transformers (1A and 1B) are supplied by the 2400-v busses (1A and 1B). Bus 1A also supplies power directly to reactor forced-circulation Pump 1A, while Bus 1B supplies power directly to reactor forced-circulation Pump 1B. Thus, duplicate equipment is supplied by separate busses to increase reliability.

The 2400 - 480-v auxiliary transformers (1A and 1B) feed the 480-v busses (1A and 1B) through separate circuit breakers. The 480-v busses (1A and 1B) feed the off-gas stack blowers (1A and 1B), seal injection pumps (1A and 1B), and lighting feeds (1A and 1B). Again, duplicate equipment is supplied from separate busses for increased reliability. A 480-v bus tie circuit breaker provides a backup power supply. This circuit breaker can be closed manually to feed either 480-v bus from the other bus and will close automatically upon trip of the main feed circuit breaker connecting to either Bus 1A or 1B.

The 480-v bus (1A) also supplies the turbine building motor control center (1E) and the 480-v essential bus through an automatic throwover switch. The 480-v essential bus supplies power for the emergency core spray pumps (1A and 1B) and feeds the turbine building motor control center (1A) and the reactor building motor control center (1A) that supply all auxiliaries for emergency operation. Duplicate equipment is again split between turbine building motor control center (1A) and reactor building motor control center (1A). The 480-v bus (1B) also provides power for the reactor building crane, reactor building motor control center (1B), and the waste disposal building motor control center.

The turbine building 120-v bus is fed from TBMCC 1A through a 480-120-v transformer. The 120-v bus feeds the turbine building 120-v regulated bus through a 15 kva Solatron. This regulated bus is used for equipment that requires voltage regulation.

The reactor plant 125-v d-c bus is fed from a battery bank charged by a charger supplied from TBMCC 1A. The charger normally carries the load on the 125-v d-c bus and maintains the battery bank at full charge. Periodic hydrometer tests and visual inspections are made to check the condition of the batteries. The 125-v d-c bus feed an inverter set, a portion of the reactor building lighting system, and other essential d-c services. The 120-v a-c non-interruptible bus, supplied from the rotary inverter set, supplies essential reactor instrumentation and control circuits.

Figure 9.2 shows the entire 2400-v and 480-v switchgear, including that for the generator plant equipment. Figure 9.3 shows the motor control centers for the generator plant equipment.

9.2 EMERGENCY POWER SYSTEM

Emergency power to TBMCC 1A and RBMCC 1A and to the 480-v essential bus is supplied by a 250-kw engine-generator set through the automatic throwover switch. This set will be started automatically upon loss of power from the main and reserve sources of auxiliary power. Power to the 120-v a-c non-interruptible bus is not interrupted before the engine-generator assumes the emergency load, since the rotary inverter set that supplies power to this bus is continuously energized by the battery bank. When power is restored to TBMCC 1A by the engine generator, the battery charger again picks up the 125-v d-c load and returns the batteries to full charge.

If the inverter set fails, feed from the turbine building 120-v a-c regulated bus is automatically supplied to the 120-v non-interruptible bus. Normally, these two busses are not connected.

The engine generator can be tested by either of two methods. In the first method, the engine generator is started manually (by placing) the control switch in the "run" position) and is run with the test load resistor providing a 100 kw load. In this manner, the electrical distribution system is not disturbed. The engine generator can also be tested by placing the control switch in the "auto" position and pushing the auto transfer switch "test" pushbutton at the main panel. This operation starts the engine generator and operates the auto-transfer switch when the generator voltage is sufficient. The auto transfer switch breaks the normal feed to the 480-v essential bus and connects the engine generator to the 480-v essential bus. Since the normal feed breaks before the emergency feed closes, no synchronization is necessary. There is, however, a momentary interruption of power to all equipment on the essential bus. Because the under-voltage relay on MCC 1A is time delayed to override such momentary interruptions caused by switching, no scram occurs. All equipment on the essential bus which was operated before the transfer would continue to operate after the transfer except for the component cooling pumps, which might have to be manually restarted if the interruption is long enough to drop out the starter.

9.3 LIGHTING

The normal lighting load throughout the plant is supplied by 480-v busses (1A and 1B) which are fed by the reactor 2400 - 480-v auxiliary transformers (1A and 1B). Upon power outage, a portion of the reactor building lighting system is supplied by the 125-v d-c bus. In addition to the emergency d-c lights in the control room, emergency lighting is also supplied to the control room from TBMCC 1A.

APPROVED FOR
CONSTRUCTION

SYN.	DESCRIPTION	BY	CHK.	APP.	DATE
5	MISC. REVISIONS	JLR	JLR		12-23-64
4	MISC. REVISIONS	JLR	CK		10-30-64
3	REV. TURB. BLDG. 120V BUS & MISC.	JLR	JLR		10-3-64
2	MISC. REVISIONS	JLR	KSR		8-22-64
1	REVISED FOR FABRICATION	JLR	KSR		5-5-64
0	APPROVED FOR CONSTRUCTION	JLR	HSR		3-31-64
-	ISSUED FOR APPROVAL	JLR	JLR		2-7-64

NO.	NOTES
1.	FOR SYMBOLS & DEVICE NUMBERS SEE S&L STANDARDS EC-110, EC-111 & EC-115.
2.	SYMBOL INDICATES OVERLOAD TRIP UNIT
REFERENCE DRAWINGS	
DWG. NO.	TITLE
503621	SINGLE LINE DIAGRAM - UNIT I
503623	KEY DIAGRAM - 2400 & 480V. SWGR - UNIT I
DRAWN J. RIVERA	
DATE 2-7-64	
CHECKED J. L. ROGERS	
ENGINEER	
APPROVED	
JOB NO. 3055	
SARGENT & LUNDY ENGINEERS CHICAGO, ILLINOIS	
FOR ALLIS-CHALMERS MANUFACTURING COMPANY ATOMIC ENERGY DIVISION NUCLEAR POWER DEPARTMENT WASHINGTON, D. C.	
TITLE SINGLE LINE DIAGRAM ELECTRICAL SYSTEM LACBWR PROJECT U.S. ATOMIC ENERGY COMMISSION GENOA, WISCONSIN	
DWG. NO. 41-503626	REV. 5

FIG. 9.1

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CONSTRUCTION**

DRAWN J. RIVERA	SARGENT & LUNDY	
DATE 8-30-63	ENGINEERS	
CHECKED J. L. ROBERTS	CHICAGO, ILLINOIS	
ENGINEER <i>C. J. [unclear]</i>	[FOR]	
APPROVED <i>[Signature]</i>	ALLIS-CHALMERS	
JOB NO. 3201.3055	MANUFACTURING COMPANY	
APPROVALS	DATE	POWER EQUIPMENT DIVISION
A. C. APP.		MILWAUKEE, WISCONSIN
A. S. APP.	9-23-63	
A. C. APP.	4-7-63	
CUSTOMERS APP.		KEY DIAGRAM
REA APP.		2400 & 480V. SW. GR. UNIT 1
		LACBWR GENERATOR PLANT
		DAIRYLAND POWER COOPERATIVE
		GENOA, WISCONSIN
DO NOT SCALE DWG.	DWG. NO.	503623
SCALE	MAN	REV. 3

NOTE:
(1) INDICATES MCC SIDE 1
(2) INDICATES MCC SIDE 2

				DRAWN J. RIVERA DATE 12-4-69 CHECKED J. L. ROGERS ENGINEER <i>W. M. H. H. H.</i> APPROVED <i>W. M. H. H. H.</i> JOB NO. 3201		SARGENT & LUNDY ENGINEERS CHICAGO, ILLINOIS [FOR] ALLIS-CHALMERS MANUFACTURING COMPANY POWER EQUIPMENT DIVISION MILWAUKEE, WISCONSIN
				APPROVALS DATE A. C. APP. A. C. APP.		
				A. C. APP. A. C. APP.		
				A. C. APP. A. C. APP.		
				A. C. APP. A. C. APP.		TITLE 480 V. M.C.C. & 125 V. D.C. AUX. DIAG. LACBWR GENERATOR PLANT DAIRYLAND POWER COOPERATIVE GENOA, WISCONSIN
				A. C. APP. A. C. APP.		
				A. C. APP. A. C. APP.		
				A. C. APP. A. C. APP.		
				A. C. APP. A. C. APP.		503624 SHEET OF 3
				A. C. APP. A. C. APP.		
				A. C. APP. A. C. APP.		
				A. C. APP. A. C. APP.		

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FIG. 9.3

10. SHIELDING

10.1 SHIELDING DESCRIPTION

The LACBWR plant is contained in three buildings: the containment building, the turbine building, and the waste treatment building. Plans of the main, mezzanine, and grade floors are shown in Figs. 10.1, 10.2, and 10.3. A vertical section of the containment building is shown in Fig. 10.4.

The major radioactive components in the plant are the primary system (including the reactor, external recirculation system, main steam line, turbine, condenser, and the feedwater system), the reactor-water purification system, the waste-disposal system, and the spent fuel. The floor plans and general-arrangement drawings in Sec. 6 show the locations of these components.

10.1.1 Containment Building

In general, the shielding around the reactor consists of ordinary concrete ($\rho = 2.35 \text{ g/cm}^3$) around the sides of the reactor and heavy concrete ($\rho = 3.65 \text{ g/cm}^3$) above and below the reactor. Table 10-1 lists the thicknesses of the shielding materials around the core, both axially and radially, and Fig. 10.4 shows the location of these materials around the core.

The containment building shell is steel plate 1.16 in. thick on the cylindrical portion and 0.60 in. thick on the hemispherical top head. The interior surface of the containment building, except for the hemispherical top head, is lined with 9 in. of concrete. There is no concrete lining in the top head (see Fig. 10.7). The main steam line cut-out in the concrete biological shield is shielded by 5.4 in. of steel from el 656 ft to el 674 ft (see Fig. 10.5). The nuclear instrument sleeves in the biological concrete are shielded with plugs 45 in. long consisting of steel shell filled with heavy concrete. Concrete shielding is employed as follows: ordinary concrete walls 1 ft thick around the retention tanks and 2 ft thick around the demineralizers; heavy concrete walls around the recirculation pumps (the inner walls and the wall on the building periphery are 2 ft-10 in. thick); ordinary concrete 5 ft-8 in. thick for the fuel storage pool and for the wall between the containment shell and the stored fuel; ordinary concrete 4 ft-9 in. thick for the other two walls and 4 ft-8 in. thick for the floor (see Fig. 10.6); if required, there will be a 1-ft thick concrete block wall, 3 ft high, on each side of the pool at el 667 ft (the stored fuel is shielded on top by 12 ft-6 in. of water). The floor at el 642 ft-9 in. is ordinary concrete 1 ft-0 in. thick, and a 2 in.-thick lead shield is on the main steam line.

TABLE 10-1

CORE SHIELDING MATERIALS

<u>material</u>	<u>thickness, in.</u>
<u>radially</u>	
water reflector ($\rho = 0.71 \text{ g/cm}^3$)	17.0
steel thermal shield	1.25
water gap ($\rho = 0.71 \text{ g/cm}^3$)	0.5
steel vessel wall	3.625
external thermal shield:	
steel cladding.	1.5
lead	2.0
ordinary concrete ($\rho = 2.35 \text{ g/cm}^3$)	96.0
<u>axially above core</u>	
water ($\rho = 0.426 \text{ g/cm}^3$)	---
steam	---
steel vessel head	3.625
air space	138.0
heavy concrete ($\rho = 3.65 \text{ g/cm}^3$)	45.0
steel in concrete plug	1.5
<u>axially below core</u>	
water ($\rho = 0.71 \text{ g/cm}^3$)	103.0
steel vessel	3.625
steel shield plug	16
target plate	4

Most of the recirculation-loop piping is within the primary biological shield. The 16-in. forced-circulation inlet and outlet lines and the 16-in. inlet and outlet manifolds are all within the biological shield. Part of the 20-in. piping that connects the recirculation pumps to the manifolds is also within the shield. The rest of the 20-in. piping is inside the pump cubicles. The pump areas are restricted during normal operation.

10.1.2 Turbine Building

The shielding in the turbine building is ordinary concrete ($\rho = 2.35 \text{ g/cm}^3$), except for a few cases where heavy concrete ($\rho = 3.65 \text{ g/cm}^3$) is used. Ordinary concrete walls are around the periphery of the building on the grade and mezzanine floors. The walls are 2 ft-0 in. thick, except at the southeast corner of the building around the high-pressure heater, where they are 2 ft-6 in. thick. Except for the control room, concrete does not line the periphery of the main floor. The shielding around the full-flow demineralizers is ordinary concrete 2 ft-0 in. thick, and the shielding walls around the reheater and separator are ordinary concrete 3 ft-4 in. thick.

The shielding walls for the air ejector are heavy concrete. The southwest wall is 3 ft-0 in. thick, and the north wall, where the operating valves are located, is 2 ft-6 in. thick. The remaining section of the north wall between the turbine supports is 2 ft-0 in. thick, and the main floor (el 668 ft) above the air ejector is heavy concrete 3 ft-0 in. thick.

The 8-in.-dia air suction line from the condenser to the air ejector is clad with 1 in. of lead from the condenser to the point where the pipe goes under the 3-ft heavy-concrete main floor.

Ordinary concrete walls 2 ft thick are above and below the water boxes on the north end of the condenser, and running between these two walls are two steel plates 3 in. thick (on sides of the two cylindrical water boxes) and a steel plate 4 in. thick (between the two boxes). The remaining wall west of the condenser is ordinary concrete 2 ft-0 in. thick.

The shielding wall for the high-pressure heater is heavy concrete 2 ft-0 in. thick. The main floor (el 668 ft) over this area is ordinary concrete 2 ft-0 in. thick, and the mezzanine floor section under the main steam line and stop valves is ordinary concrete 2 ft-0 in. thick.

The location of the shielding walls for the turbine is shown in Fig. 10.1. The side walls are ordinary concrete 1 ft-8 in. thick and 7 ft-0 in. high, and the wall at the east end of the turbine is 1 ft-0 in. thick and 7 ft-0 in. high.

Ordinary concrete is used as shielding for the control room (see Fig. 10.9) and is designed to protect personnel should the maximum credible accident (MCA) (see Sec. 14) occur. The cross section in Fig. 10.7 shows only the east wall, but the locations and thicknesses of the shielding in the south wall, which also faces the containment building, are the same as in the east wall.

10.1.3 Waste Treatment Building

The waste treatment building is northeast of the containment building and contains facilities for the collection, processing, storage, and disposal of radioactive waste materials.

The grade floor of the waste treatment building contains a shielding compartment that encloses the evaporator, condenser, spent-resin tank, water-collecting tank, and the evaporator-feed tank. Figure 10.3 shows the locations of the shielding walls for this compartment. The portion of the south wall that encloses the evaporator, condenser, spent-resin tank, and the evaporator feed tank is 2 ft-6 in. thick, and the portion that encloses the water collecting tank is concrete 11-5/8 in. thick. The west wall of the decontamination area is concrete 11-5/8 in. thick. The portion of the concrete east wall that is nearest the spent-resin tank is 3 ft thick. The other portion of this wall in the spent-resin compartment is 2-ft-6 in. thick. The northern half of the east wall and the whole north wall are concrete 11-5/8 in. thick. The walls that are between the spent-resin and the evaporator compartments and between the evaporator and the evaporator feed tank compartments are concrete 2 ft thick. The concrete north walls of the spent-resin and evaporator compartments are 2-ft-6 in. thick and 2 ft thick, respectively. A separate below-grade enclosure houses the two 1800-ft³ (12,000 gal) gas-storage tanks, and this enclosure is connected to the waste treatment building by a pipe tunnel. There is 1 ft of concrete and a 3-ft fill of dirt as shielding above the gas-storage tanks.

Beneath the grade floor is the concentrated-waste tank cell. The shielding walls inside the building are 2 ft-6 in. thick, and (because this cell is below grade) the outer walls are only 1 ft-6 in. thick, since the dirt fill around these walls provides the necessary additional shielding.

10.2 PLANT DOSE LEVELS

The calculational methods and models used in the shielding analysis, along with the sources of activity, are given in Appendix C.

10.2.1 Shielding Design Criteria

The criteria used in designing the shielding for the La Crosse Boiling Water Reactor and in designating restricted areas conform to accepted practices and standards. Radiation levels in unrestricted areas of the plant meet the requirements of 10 CFR 20, including amendments, as published in the Federal Register for Sept. 7, 1960 (F. R. Doc 60-8211).

Figures 10.8 through 10.11 give the general dose levels for LACBWR. All dose levels given have been increased by a factor of 2.5 over those calculated in the analysis, so as to include any uncertainty in computing the following:

- (1) sources of radiation,
- (2) attenuation coefficients,
- (3) buildup factors,
- (4) source models, and
- (5) shielding and structural geometric representation.

All areas in which dose levels are 2.0 mr/hr or less are unrestricted areas. (Unless otherwise specified, the term "mr/hr" means "mrem/hr.") Areas in which dose levels are 2.0 mr/hr to 100 mr/hr are radiation areas, and all areas greater than 100 mr/hr are high radiation areas. Entrance into radiation areas and high radiation areas is restricted during full-power reactor operation.

10.2.2 Containment-Building Dose Rates

Dose rates at points on the shield face around and above the core are less than 2.0 mr/hr. Doses from the main steam line (MSL) are limited by the MSL shielding to below 2.0 mr/hr in normal working areas. Since the pumps are in separate compartments, limited maintenance may be performed on one pump while the plant operates at partial power with the second pump. During maintenance operations, the radiation level in the shutdown pump compartment will be monitored continuously.

The control-rod-drive room and all areas below el 642 ft are restricted areas during normal reactor operation. The dose rate in the control-rod-drive room below the bottom shield-plug is about 2000 mr/hr. Of this dose, about 13 mr/hr is the direct dose rate from the core, and the remainder is due to streaming through the control-rod drives.

The dose rates outside the shielded cubicles of the retention tanks and demineralizers are 4 mr/hr and 8 mr/hr, respectively. The dose rates for all other areas below el 642 ft, including those around the forced-circulation pump and the decay heat cooler, are between 2.0 and 100 mr/hr. The dose rate for all areas outside the containment building is less than 2.0 mr/hr at grade level.

Figures 10.12 and 10.13 give the gamma dose rate and the neutron fluxes axially from the core. Figures 10.14 and 10.15 give the neutron fluxes and gamma dose rate in the radial direction. Figure 10.16 gives the axial distribution of the gamma dose rate along the inner face of the concrete biological shield. All gamma dose rates and neutron fluxes given in Figs. 10.12 through 10.16 have been increased by a factor of 1.5 to include the maximum uncertainty in the analysis. This uncertainty factor covers any error in the gamma and neutron source spectra, the attenuation coefficients, and the buildup factors. However, when data from these curves are used in calculating dose levels outside the shielding, the values are increased by an additional factor to account for the increased error because of the greater penetration through shielding material. (This total correction factor of 2.5 corresponds to the factor of 2.5 used in calculating the general dose levels.)

10.2.2.1 Integrated Neutron Flux. The total fast neutron flux above one Mev at the inner face of the pressure vessel on the core centerline is 1.53×10^{10} n/cm²-sec. Based on an operating period of 20 years, the integrated fast flux (nvt) at the vessel becomes 9.66×10^{18} n/cm².

10.2.2.2 Dose Rates During Reloading. The dose rate at the surface of the shield pool during transfer of a spent fuel element is 0.8 mr/hr, and during transfer of a control rod, 0.5 mr/hr (assuming the element or control rod is at its maximum height in the pool). Table 10-2 gives the gamma dose rate in air, 1 ft from the vessel closure. The dose rate above the reactor, on the main floor, is 2.5 mr/hr, assuming the vessel is flooded, the top shield plug is removed, and the reactor has been shut down for 12 hr.

10.2.2.3 Instrument-tube Streaming. The dose rate at the end of the nuclear instrument tube is 2.0 mr/hr outside the 45-in. heavy-concrete plugs.

10.2.3 Electrical Cable Room Dose Rates

The dose rate in the electrical cable room between the containment building and the turbine building is 2.0 mr/hr. The dose rates in the room below the cable room at el 629 ft are 2.0 mr/hr outside the 4500-gal liquid-waste-tank cubicle and 12.5 mr/hr outside the off-gas holdup-tank cubicle.

The main steam line (MSL) passes through this area near the south wall. The dose rates in air 5 ft and 10 ft from the MSL are 270 mr/hr and 83 mr/hr, respectively.

The dose rates in the area next to the recombiner and its related equipment vary from 2.0 mr/hr to 8.3 mr/hr, assuming no buildup of active particulate material on the recombiner.

TABLE 10-2

DOSE RATES ONE FOOT FROM VESSEL CLOSURE

<u>source</u>	(mr/hr)			
	<u>top surface</u>		<u>bottom surface</u>	
	<u>no decay</u>	<u>one-day decay</u>	<u>no decay</u>	<u>one-day decay</u>
activated carbon steel	1.7	---	4.4	0.02
activated stainless steel	0.3	0.04	7.2	1.1
total	2.0	0.04	11.6	1.12

10.2.4 Turbine Building Dose Rates

Because of the short half-life of N-16 and O-19, and provided there are no appreciable amounts of gaseous fission products in the steam, entry into radiation and high radiation areas in the turbine building for routine maintenance is allowed shortly after shutdown.

10.2.4.1 Normal Dose Rates. During full-power operation, the dose rate in the area next to the air-ejector valves is 17-25 mr/hr, making this a radiation area. However, entrance to this area is required only during startup, when the dose rate is 2.0 mr/hr or less. The area inside the turbine shielding wall is a high radiation area during full-power operation, and access will not be allowed, except for routine walk-through inspections, and at low power levels.

The dose rates for the control room, machine shop, laboratories, meeting rooms, and offices are less than 0.25 mr/hr. The dose rates for areas outside the shielding of the condenser hotwell, air ejectors, full-flow demineralizers, off-gas blowers, and the off-gas compressor are below 2.0 mr/hr.

The dose rate for all areas outside the turbine building, at grade level, is 2.0 mr/hr or less. At the air ejector dose point directly outside the south turbine-building wall, the dose is 75 mr/hr, but since this point is 18 ft above grade level there is no radiation hazard to personnel at grade level.

10.2.4.2 Emergency Dose Rates. The maximum dose rate in the control room during the MCA is 4.2 r/hr, including direct and sky-shine contributions. The maximum integrated dose for the 24-hr period after the MCA is 15.5 r. The average dose rate in the control room during the MCA is 1.7 r/hr, and the average integrated 24-hr dose is 6.3 r.

10.2.5 Waste-Treatment-Building Dose Rates

The dose rates inside the waste treatment building, in the areas where operating personnel are normally located, are 2.0 mr/hr or less. The dose rates outside the waste treatment building are 0.25 mr/hr or less.

10.3 GAMMA HEATING

All heating rates given have been increased 50 percent unless noted otherwise, to cover calculational uncertainties.

10.3.1 Vessel and Thermal Shield Gamma Heating

Only gamma heating was considered, since the heat contribution from neutrons is negligible. The maximum gamma heating rates for the thermal shield and vessel wall are along the horizontal centerline of the core. The heating rates for the thermal shield and vessel wall are 8.3×10^4 Btu/(hr)(ft³) and 3.25×10^4 Btu/(hr)(ft³), respectively.

10.3.2 Biological Shield Gamma Heating

The gamma heating along the inner face of the concrete biological shield is shown in Fig. 10.17. The maximum heating rate occurs about 70 cm above the top of the core.

The maximum occurs above the top of the core because the lead thermal shield extends only 10 in. above the top of the core. The maximum gamma heating rate for the lead thermal shield on the face of the concrete biological shield is 1100 Btu/(hr)(ft³) and occurs along the horizontal centerline of the core.

10.3.3 Gamma Heating Below the Core

The gamma heating rates below the core (increased by a factor of two to cover calculational uncertainties) for the vessel and the steel shield plug are 80 and 3.5 Btu/(hr)(ft³) respectively.

10.3.4 Gamma and Neutron Heating in the Grid Plate

The maximum gamma and neutron heating in the upper and lower sections of the grid plate, along the vertical centerline, are (including a 20 percent uncertainty factor) 200,000 Btu/(hr)(ft³) and 80,000 Btu/(hr)(ft³), respectively. The 200,000 value was used in a stress analysis reported in Sec. 4 of Appendix A.

10.4 FIGURES

SPENT FUEL
STORAGE WELL

CONTAINMENT BUILDING

GENERATOR PLANT
& REACTOR PLANT
CONTROL ROOM

TURBINE BUILDING

MAIN FLOOR OF TURBINE BLDG, EL. 668' -0"

FIG. 10.1

SPENT FUEL
STORAGE WELL

CONTAINMENT BUILDING

TURBINE BUILDING

MEZZANINE FLOOR OF TURBINE BLDG, EL. 654' -0"

FIG. 10.2

GRADE FLOOR OF TURBINE BLDG, EL. 640' -0"

FIG. 10.3

CONTAINMENT BUILDING ELEVATION

FIG. 10.4

MAIN STEAM LINE SHIELDING
(at el. 660' -0")

FIG. 10.5

FUEL-STORAGE POOL

FIG. 10.6

CONTROL ROOM SHIELDING

FIG. 10.7

FIG. 10.8

SPENT FUEL
STORAGE WELL

STEAM JET
AIR EJECTOR

TURBINE BUILDING

CONTAINMENT BUILDING

TRUE NORTH

DOSE LEVEL - MEZZANINE FLOOR OF TURBINE BLDG, EL. 654' -0"

FIG. 10.9

DOSE LEVEL - MAIN FLOOR OF TURBINE BLDG, EL. 668' -0"

FIG. 10.10

DOSE LEVEL—CONTAINMENT BUILDING ELEVATION

FIG. 10.11

AXIAL NEUTRON FLUX AND GAMMA
DOSE RATE ABOVE CORE

FIG. 10.12

AXIAL NEUTRON FLUX AND GAMMA DOSE
RATE BELOW THE CORE

FIG. 10.13

FAST AND THERMAL NEUTRON FLUX RADIALLY FROM CORE, at axial maximum

Fig. 10.14

RADIAL GAMMA DOSE RATE

FIG. 10.15

GAMMA DOSE RATE ALONG INNER
FACE OF CONCRETE BIOLOGICAL SHIELD

FIG. 10.16

GAMMA HEATING ALONG INNER
FACE OF CONCRETE BIOLOGICAL SHIELD

Fig. 10.17

11. OPERATING PRINCIPLES AND SAFEGUARDS

This section summarizes the operating principles and associated safeguards for the various individual systems of the reactor plant. The philosophy and administrative controls for overall plant operation are discussed in Sec. 13.

The detail in which the operating principles are presented is dependent upon the relation of each system to plant safety. The plant operators will be guided by the "LACBWR Operating Manual," copies of which will be available at the plant. Section 11.1 is a summarized sample of the individual system operating instructions for the operating manual. A summary of instructions for overall plant operation is given in Sec. 13.4.

11.1 FORCED-CIRCULATION SYSTEM

The forced-circulation system is described in Sec. 5.1.2 and the system flow diagram is shown in Fig. 5.3. A summary of the operating instructions for the system follows.

11.1.1 Normal System Startup

The conditions prior to normal startup of the forced-circulation system are as follows:

- (1) The reactor is shut down.
- (2) The primary system is not pressurized.
- (3) The hydraulic valve accumulator is in service and set for automatic operation.
- (4) The lube oil system for each of the forced-circulation pumps is in service.
- (5) The fluid coupling water supply system for each of the forced-circulation pumps is ready for service.
- (6) The seal injection system is in service, with water entering the upper injection point of each forced-circulation pump.
- (7) Alarms pertaining to the above four systems are cleared.

Startup proceeds by the following steps:

- (1) The pump speed control switches are placed in the Pull-Out position.
- (2) The pump suction valves are opened and the keys that lock the switches in position are removed.

(3) The discharge bypass valve auto-manual key switches are turned to the Auto position, and the keys are removed.

(4) Both forced-circulation pump discharge valves are closed. (bypass valves should open automatically).

(5) One of the pump motors is started with the pump motor control switch; the pump ammeter is watched for excessive current.

(6) The coupling water pump is started for the operating pump.

(7) The pump speed control switch is pushed in. The pump goes to 40 percent speed. Pump current is observed for normality at this speed.

(8) The pump discharge valve control switch for the operating pump is turned to Auto, and the key in the base is removed, locking the valve in Auto. The discharge valve opens and the associated bypass valve closes.

(9) The second pump is started in the same manner, and the discharge valve switch is locked in Auto.

(10) The individual pump speed control switches are adjusted until the two pumps are operating at the same speed.

(11) The speed of the two pumps is simultaneously increased with the gang speed control switch to ~ 625 rpm; the pump ammeters are observed.

THE MOTORS ARE NOT TO BE OVERLOADED, SINCE THEY ARE NOT SIZED FOR FULL SPEED OPERATION WITH COLD WATER.

(12) When reactor pressure is increased above ~ 500 psig, the seal injection is transferred to the lower injection points with the transfer switches on the console.

11.1.2. Normal Operation

During reactor power operation, the forced-circulation system is operated in one of three modes:

(1) Constant Pump Speed

With the pumps at constant speed, reactivity compensations are made with the control rods. The pump speed must be high enough to avoid the power-flow scram and, must be compatible with the pump suction subcooling, to avoid cavitation.

(2) Manual Pump Speed Control

The control rods and the pump speed are adjusted until the total forced-circulation flow and the reactor power correspond to a point on the normal Flow-Power curve (Fig. 4.68). The reactor power can now be controlled with the gang pump speed control switch. The switch is turned to "Lower" to decrease the pump speed and reactor power, and to "Raise" to increase the pump speed and reactor power.

(3) Completely Automatic

In this mode automatic variations of pump speed maintain a constant turbine inlet pressure. The system is placed in automatic operation as follows:

(a) The control rods and pump speeds are adjusted until the total forced-circulation flow and reactor power correspond to a point on the normal Flow-Power curve (Fig. 4.68).

(b) The pump speed controllers are transferred to "Manual."

(c) The Manual output adjusting knobs on the two speed controllers are turned until the two null indicating lights are out, indicating that the Jordan amplifiers are balanced.

(d) The reactor pressure control transfer switch is turned to Auto and held until the amber light comes on. The operator now has manual control of the two pump speed controllers.

(e) The reactor pressure controller is transferred to Manual, and the speed controllers to Manual-Balance.

(f) The manual output knob on forced-circulation pump 1A speed controller is turned until it shows zero deviation.

(g) Forced-circulation pump 1A speed controller is transferred to Auto.

(h) The bias knob on forced-circulation pump 1B speed controller is turned until the controller shows zero deviation.

(i) This controller is transferred to Auto. The operator now has manual control of both pumps with the reactor pressure controller.

(j) The reactor pressure controller is transferred to Manual-Balance and the set point knob is adjusted until the deviation is zero.

(k) The reactor pressure controller is transferred to Auto. The system is now on complete automatic control.

The pumps can be returned to manual control at any time by momentarily turning the reactor pressure control transfer switch to Manual.

11.1.3 Normal Shutdown

During a normal plant shutdown, the forced-circulation pumps are operated at a constant speed of 625 rpm or less (after the corresponding reduction in reactor power level), which precludes cavitation as feedwater flow is reduced, and prevents overloading of the pump motors when the water cools.

When reactor pressure decreases below 600 psig, the seal water is transferred to the upper injection point of each seal. The forced-circulation pumps may be shut down in the following manner when the primary water has cooled to within approximately 100 F of the seal water temperature:

- (1) The speed of both pumps is simultaneously reduced at the maximum rate with the ganged speed control (to ~40 percent speed).
- (2) The individual pump speed control switches are simultaneously put on the Pull-Out position. The pump speeds then drop to zero. The discharge valves remain open (one valve may begin to close and then immediately re-open).
- (3) The hydraulic coupling water pumps are turned off.
- (4) The forced-circulation pump motors are tripped.
- (5) The pump seal injection and lube oil systems may be shut down after both pump shafts have stopped rotating.

11.1.4 Special Operations

During initial testing, the procedures outlined below will be checked with the reactor subcritical to determine the optimum initial flowrates and to assure that the transfer can be made without sudden changes in core flow.

11.1.4.1 Transfer From Two-Pump to One-Pump Operation. Transfer from two-pump to one-pump operation must be done with the pump speeds in manual control, with a flow low enough to be delivered by one pump operating at a higher speed. The reactor power must be reduced to a value compatible with this flow:

- (1) The speed of the pump to be shut down is decreased, while the speed of the other pump is increased to hold a constant total flow.
- (2) The speed of the pump being taken out of service is decreased in this manner to the limit with the manual speed control switch (to ~40 percent of full speed).

While the slow pump is moving toward shutoff on its head-capacity curve, the flow from the fast pump will increase for a given speed, because of lower pressure drop through the system.

(3) The pump discharge valve control switch of the pump being stopped is unlocked and turned to "Close".

(4) As the discharge valves close, the speed of the other pump is increased to keep a constant amount of core flow (while the discharge valve is closing the associated bypass valve is opening).

(5) When the discharge valve is closed, the pump speed control switch is put on the Pull-Out position to decrease the speed to zero.

(6) The pump discharge valve switch is returned to Auto, locked, and the key removed. The valve can then open automatically if the operating pump should fail. This provision is desirable for natural circulation.

(7) The hydraulic coupling water system for the down pump is stopped.

(8) The pump motor is turned off.

(9) The suction valve and the discharge bypass valve on the stopped pump are left open to allow some water to circulate through the loop to keep it hot.

11.1.4.2 Transfer From One-Pump to Two-Pump Operation. Transfer from one-pump to two-pump operation must be done with the pump speeds in manual control. Normally, when one pump is not in service its suction valve and discharge bypass valve are kept open to keep the loop hot. But if this procedure is not followed and the loop is cold, the suction valve must be opened first, and the discharge bypass valve placed in manual control and opened to regulate the warm-up rate of the cold loop.

If seal water has not been entering the shutdown pump, injection should be initiated at least 30 min before the pump is started.

When the forced-circulation loop temperature difference indicator shows negligible difference between the two loops, the shutdown pump is brought on line in the following manner:

(1) The lube oil system for the shutdown pump is started.

(2) The hydraulic coupling water system for the shutdown pump is prepared for operation.

(3) The speed of the operating pump is maintained or increased so that the flow rate through that loop is greater than 14,000 gpm (i.e., greater than the total flow with both pumps at 40 percent speed). Reactivity compensations are made by moving the control rods.

- (4) The pump discharge bypass valve key switch is turned to Auto, and the key removed.
- (5) The pump discharge valve control switch for the down pump is unlocked and turned to Close.
- (6) The speed control switch for the down pump is to be in the Pull-Out position.
- (7) The pump motor is started.
- (8) The coupling water pump is started for the pump being brought on line.
- (9) The speed control switch for the pump being started is pushed to Neutral which brings the pumps to ~ 40 percent speed. Flow begins to pass through the open discharge bypass valve slightly increasing the core flow. The slight reactivity change is countered by either reducing the speed of the other pump or slightly inserting the control rods.
- (10) The pump discharge valve is turned to Auto. The speed of the pump that has been on line is reduced to keep the total flowrate and the reactor power constant. The change is not large, since the pump being brought on line is at 40 percent speed and operating near shutoff on the head-capacity curve.
- (11) The speed of the slow pump is gradually increased while the speed of the fast pump is decreased, to keep the total core flow constant.
- (12) When the pumps are at the same speed, operations can proceed as explained in Sec. 11.1.2.

11.1.5 Abnormal Operation

11.1.5.1 Reactor Scram. If the reactor is operating in the automatic pressure control mode and a scram occurs, the pumps automatically transfer to manual control. Depressing the all-rod-insert button also transfers the pumps to manual operation.

If the pumps are operating above 80 percent speed when a scram occurs or when the all-rod-insert button is pressed, the pumps automatically slow down to 80 percent to prevent cavitation. This action occurs whether initially the pumps were in automatic or manual. If the pumps were initially operating at 80 percent speed or below they will remain at the same speed. In either case, the forced-circulation system requires no operator action.

11.1.5.2 Loss of Electrical Power. Loss of power to both 2400-v busses initiates an all-rod scram, trips both forced-circulation pumps, and trips all the forced-circulation pump 480-v auxiliaries. The seal water accumulators supply the pump seals during pump coast-down. The suction and discharge valves remain open. The forced-circulation system is safe under this condition.

Loss of power to one 2400-v bus initiates a partial scram and trips the forced-circulation pump supplied by that bus. When the pump speed falls below 40 percent, its discharge valve closes. The other forced-circulation pump remains in operation. If the initial speed was above 80 percent, the pump slows to 80 percent. All of the forced-circulation pump 480-v auxiliaries remain in operation. The forced-circulation system is safe under this condition, and no immediate operator action is required.

Undervoltage on 480-v bus 1A initiates an all-rod scram and trips seal injection pump 1A. A loss of power to this bus is probably accompanied by a loss of power to the 2400-v busses and 480-v bus 1B. However, if there is still power to these other busses, seal injection pump 1B starts automatically and the forced-circulation pumps remain in operation. The forced-circulation pump 480-v auxiliaries (other than seal injection pump 1A) also remain in operation. The forced-circulation system is safe under this condition.

Loss of power to 480-v bus 1B without a loss of power to the 2400-v busses and 480-v bus 1A trips all forced-circulation pump 480-v auxiliaries, except seal injection pump 1A.

The resulting loss of lube oil pressure trips both forced-circulation pumps. The valves remain open to allow natural circulation, but the reduction of primary coolant flow initiates a power-flow scram.

A complete loss of 125-v d-c control power is extremely unlikely since this circuit is backed up by the battery bank. However, there could be a loss of control power to individual components. Upon loss of 125-v d-c control power, the forced-circulation system components fail as follows:

- (1) Forced-circulation pump motor breakers fail in position.
- (2) Forced-circulation pump suction and discharge rot valves fail open.
- (3) Forced-circulation pump discharge bypass valves fail closed in automatic or manual control.
- (4) Forced-circulation pump seal leakoff valves fail open.
- (5) Variable speed fluid coupling scoop tube fails in position.
- (6) Seal injection pump motor breakers fail in position.

The other forced-circulation pump 480-v auxiliaries are not affected.

Loss of control power to a forced-circulation pump seal leakoff valve while the seal is isolated opens the valve and permits the outleakages of reactor water if the reactor is pressurized. The seal leakoff manual stop valve must be closed to isolate the seal.

11.1.5.3 Forced-Circulation Pump Trip. A forced-circulation pump is automatically tripped by the following:

- (1) motor overload
- (2) 2400-v bus undervoltage
- (3) high seal leakoff temperature
- (4) low seal water flow
- (5) low lube oil pressure
- (6) pump suction valve partially closed
- (7) low pressure differential between pump suction pressure and seal injection pressure at the seal injection pump discharge
- (8) short circuit
- (9) electrical ground

Tripping one forced-circulation pump with the reactor at power causes a reduction in the primary coolant flow with a corresponding reduction in the power level. The discharge valve for the tripped pump closes automatically to restrict reverse flow through the downloop, and the discharge bypass valve is interlocked to open preventing the formation of a cold water leg. If both pumps are tripped, the pump discharge valves are interlocked to remain open so that the natural circulation flow will be established at a corresponding low power level. The established reactor thermal-hydraulic criteria are not violated during the transient, and the transient is terminated with the reactor in a safe condition (Sec. 14.3.9).

11.1.5.4 High Temperature of Pump Lube Oil. High lube oil temperature is alarmed in the control room but does not trip the pumps. The most likely cause of high temperature is insufficient cooling water flow through the lube oil cooler. If the malfunction cannot be corrected in accordance with specified instructions, the reactor must be shut down and the forced-circulation pumps tripped.

11.1.5.5 Low Pressure of Pump Coupling Water. Low coupling water pressure at the outlet of the coupling water cooler causes automatic start of the standby pump since the probable cause of the low pressure is loss of the operating pump. The standby pump startup is annunciated in the control room. If normal pressure is not restored, the coupling water will gradually boil off, resulting in a gradual reduction in forced-circulation pump speed. If the pump speed continues to decrease, it is tripped and corrective maintenance performed. Loss of coupling water, however, does not damage the fluid coupling.

11.1.5.6 High Temperature and Abnormal Water Level of Pump Coupling Water Reservoir. High coupling water temperature and high or low water level is alarmed in the control room. Possible causes of high temperature include insufficient cooling water flow through the coupling water cooler, coupling water pump failure, and low water reservoir level. Possible causes of abnormal water level include loss of the coupling water pumps and loss of return coupling water flow to the reservoir. Temperature indication is given in the control room and each water reservoir is equipped with a gauge glass for local level indication.

Operation with high temperature or low water level may continue for short periods of time without damage to the couplings. However, if the temperature or level cannot be returned to normal before pump speed drops to 40 percent or excessive bearing temperature is indicated, the pump must be tripped.

11.1.5.7 Low Pressure or Low Water Level of Hydraulic Valves Accumulator. Low pressure or low level in the hydraulic valve accumulator is alarmed in the control room. The alarm setpoints are such that all rotovalve actuators remain operable after the setpoints are reached (Sec. 5.2.12.1). Unless the alarm condition can be corrected promptly, the reactor will be shut down to allow maintenance work on the hydraulic valve accumulator system.

11.1.6 Precautions to Avoid Pump Damage

11.1.6.1 Pump Buffer Seal. The buffer seal would be damaged if the pump is operated while the seal is hot. If the seal is isolated for more than 1-1/2 hr while the reactor is hot, seal water is supplied to the seal for 30 min before pump startup.

Whenever a seal is isolated, the seal leakoff valve is checked to ensure automatic closure. Hot reactor water leakage could cause seal damage.

The seal would be severely damaged if seal water flow is interrupted while the pump is operating. Extreme caution is taken whenever valves in the seal water system are being operated to ensure against inadvertent shutoff of the seal water supply.

11.1.6.2 Fluid Piston Bearing. The fluid piston bearings may rub during pump startup and shutdown when the pump speed is < 40 percent. Interlocks prohibit continuous operation in this speed range to prevent bearing damage. However, when pump speed is changed between 0 to 40 percent, the operator must ensure that the pump speed controller does not stop at some intermediate speed.

The fluid piston bearing can also be damaged by reverse rotation. The discharge valves are interlocked for automatic closure to avoid reverse rotation; however, the operator must ensure that the discharge valves function as required.

11.1.6.3 Pump Motor. The forced-circulation pump motors are sized for 100 percent pump speed at reactor design temperatures. At lower reactor temperatures or during one pump operation, it may be necessary to operate the pump(s) at reduced speeds to prevent motor overload that could result in a motor overload trip. (Pre-operational testing will determine if this precaution is necessary).

11.1.6.4 Pump Speed. The subcooling effect of the feedwater at normal operating conditions provides sufficient net positive suction head to allow pump operation at 100 percent speed. A low feedwater flow results in pump cavitation when saturated reactor water is recirculated in the high pump speed range. Cavitation is avoided by attention to proper pump speed during reactor warmup and cooldown.

11.1.7 Alarms

Table 11-1 lists the forced circulation system alarms.

TABLE 11-1

FORCED-CIRCULATION SYSTEM ALARMS

<u>window</u>	<u>name plate</u>	<u>actuating device</u>
D2-4	power-recirc. flow abnormal	reactor safety system full and partial scram Channels 1 and 2
E1-1	reactor auxiliary power device tripped	control switch and auxiliary pumps motor trip
E1-2	hyd valve accum water level abnormal	level switches
E1-3	hyd valve accum water press. abnormal	pressure switches
E1-4	hyd accum sump tank level low	level switch
E2-1	F.C. pump 1A tripped	circuit breaker "a" contacts
E2-2	F.C. pump 1B tripped	circuit breaker "a" contacts
E2-3	F.C. pump 1A coupling water trouble	temp switch and level switches
E2-4	F.C. pump 1B coupling water trouble	temp switch and level switches
E3-1	power-recirculation flow abnormal	reactor safety system anticipatory alarm Channel 2
E3-3	F.C. pump 1A lube oil trouble	level switches

<u>window</u>	<u>name plate</u>	<u>actuating device</u>
E3-4	F.C. pump 1B lube oil trouble	level switches
E4-2	F.C. pumps hyd coup. or oil pump auto start	
E5-1	seal inject. res. level high	level indicator switch
E5-2	seal inject. res. level low	level indicator switch
E5-3	F.C. pump 1A leakoff temp high	temp. indicator switch
E5-4	F.C. pump 1B leakoff temp high	temp. indicator switch
E6-1	F.C. pump 1A seal inject flow low	flow switch
E6-2	F.C. pump 1B seal inject flow low	flow switch
E6-3	F.C. pump 1A backup seal leakoff high	level switch
E6-4	F.C. pump 1B backup seal leakoff high	level switch
E7-1	seal inject. supply flow low	flow indicator switch
E7-2	seal inject. water temp high	temp. switch
E7-3	F.C. pumps seal water filter high Δp	press. differential indicator
E7-4	control rod seal water filter high Δp	press. differential indicator
E8-1	F.C. pump 1A leakoff flow (hi-lo)	flow indicator switch
E8-2	F.C. pump 1B leakoff flow (hi-lo)	flow indicator switch

11.2 SEAL INJECTION SYSTEM

The seal injection system is described in Sec. 5.2.4, and Fig. 5.9 is the system flow diagram.

11.2.1 General Operation

Seal water supply is maintained in the seal injection reservoir by an automatic level control valve in the makeup line from the full flow condensate demineralizers. Seal injection flow and system pressure are maintained automatically as described in Sec. 5.2.4.8. The seal injection piston accumulators, and one seal injection pump are on automatic standby control to ensure a continuous seal water supply.

Plant operational tests will determine if the forced-circulation pumps provide sufficient driving head to maintain an operating blowdown flow of 29 gpm through the decay heat cooling loop. If such operation is confirmed, the control-rod drive housing blowdown

pump will be shut down when forced-circulation pumps are operating. The valving will be adjusted, if necessary, to maintain the blowdown flowrate.

When the plant is shut down, the seal injection system will remain in operation until the primary system temperature has dropped to about 150 F and the forced-circulation pumps have been shut down. When the forced-circulation pumps are shut down, the seal injection pump can be shut down, and the seal injection system secured by isolation with the available manual valves. The piston accumulators will then be removed from automatic operation by turning the hand control switches of the seal injection pumps to the off position.

11.2.2 Engineered Safeguards

The following subsections describe the design philosophy to protect against possible system malfunctions.

11.2.2.1 Seal Injection Pump Failure: Failure of a seal injection pump will automatically trigger the following sequence of events:

(1) A drop of differential pressure between the pump discharge supply header and the reactor to the set value of 180 psi opens the solenoid valve in the nitrogen supply line to the piston accumulators. The nitrogen supply, automatically regulated to 250 psi above reactor pressure, discharges water from the accumulators at a flow rate governed by the forced-circulation pump seal flow control instrumentation.

(2) The accumulators exhaust their water supply in approximately 1-1/2 min; the pistons then seal the discharge to prevent nitrogen from entering the seal injection system.

(3) Before the accumulator water supply is exhausted, pressure is restored in the seal injection pump discharge supply header by a low flow automatic start signal to the standby pump.

(4) Restoration of pressure in the seal injection system closes the nitrogen supply valve and opens a nitrogen vent valve; system pressure then refills the accumulators with water. The down seal injection pump is repaired as the system operates.

(5) If the standby seal injection pump fails to start and differential pressure from the seal injection pump discharge header to the reactor continues to drop to a set value of 160 psi, the forced-circulation pumps are automatically tripped, leading to an automatic reactor scram.

11.2.2.2 Low Pump Seal Leakoff Flow. The leakoff flowrate from the forced-circulation pump seals is maintained at a minimum of 0.2 gpm to protect the seal

rings from damage. An alarm is actuated when this flow drops below 0.5 gpm, and the forced-circulation pump is automatically tripped at a flow below 0.2 gpm.

11.2.2.3 High Temperature of Pump Leakoff Water. Failure to maintain seal water pressure above reactor pressure would result in leakage of primary water into the normal seal leakoff line that leads back to the seal injection reservoir, but a high temperature signal in the seal leakoff line trips the forced-circulation pump if leakage occurs.

11.2.2.4 Forced-Circulation Pump Backup Seal Failure. A normally dry leakoff line routes malfunction leakage to retention tanks if the backup seal fails on either of the forced-circulation pumps. A loop in the leakoff line collects the leakage; a liquid level switch monitoring the loop actuates an alarm to signal the malfunction leakage. The forced-circulation pump is then shut down, and the failed backup seal replaced.

11.2.2.5 Seal Water Makeup Supply Failure. Lack of normal makeup water supply from the condensate demineralizers is signaled by a low level alarm monitoring the seal injection reservoir. A valve in the emergency supply line from the overhead storage tank is automatically opened by the level controller; the backup water supply feeds by gravity.

11.2.2.6 Loss of Air to Control Valves. The air-controlled valves in the normal leakoff lines from the forced-circulation pump seals fail open upon loss of air, to allow leakage flow during pump coastdown. Flow control valves in the inlet lines to pump seals and control-rod drive seals also fail open, to allow seal water flow to continue.

Loss of air to the pilot valve for the three-way switching valve between the upper and lower injection points of a forced-circulation pump seal fails in the vented position, allowing the three-way valve to be normally deenergized. The three-way valve fails in position, and the pump is shut down.

The control valve in the makeup line to the seal injection reservoir fails open to avoid starvation of the seals. The control valve in the line from the overhead storage tank to the seal injection reservoir fails closed to avoid draining of the overhead storage tank.

11.2.2.7 Loss of Power to Solenoid Valves. The nitrogen supply valve to the back-up accumulators for seal water supply fails open and the nitrogen vent valve fails closed to ensure operation of the accumulators in an emergency.

The pilot valve for the forced-circulation pump normal leakoff control valve fails open and vented to ensure an open control valve during pump coastdown.

11.2.3 Alarms

Seal injection system alarms signaled in the control room are given in Table 11-2.

TABLE 11-2

SEAL INJECTION SYSTEM ALARMS

<u>window</u>	<u>name plate</u>	<u>actuating device</u>
E5-1	seal injection reservoir level high	level indicator switch
E5-2	seal injection reservoir level low	level indicator switch
E5-3	F.C. pump 1A leakoff temperature high	temp. indicator switch
E5-4	F.C. pump 1B leakoff temperature high	temp. indicator switch
E6-1	F.C. pump 1A seal injection flow low	flow switch
E6-2	F.C. pump 1B seal injection flow low	flow switch
E6-3	F.C. pump 1A backup seal leakoff high	level switch
E6-4	F.C. pump 1B backup seal leakoff high	level switch
E7-1	seal injection supply flow low	flow indicator switch
E7-2	seal injection water temperature high	temperature switch
E7-3	F.C. pumps seal water filter high Δp	pressure differential indicator
E7-4	control-rod seal water filter high Δp	pressure differential indicator
E8-1	F.C. pump 1A leakoff flow (hi-lo)	flow indicator switch
E8-2	F.C. pump 1B leakoff flow (hi-lo)	flow indicator switch
E8-3	control-rod seal water flow low	flow switch
E8-4	control-rod seal water flow low	flow switch

11.3 ROD CONTROL AND SAFETY SYSTEM

The rod control and safety system and its operating principles are described in Sec. 8.5. Table 11-3 gives the system alarms.

TABLE 11-3
SAFETY SYSTEM ALARMS

<u>window</u>	<u>name plate</u>	<u>actuating device</u>	<u>remarks</u>
C1-1	reactor water level high	safety system reactor water level Channel 1	anticipatory alarm
C1-2	reactor water level low	safety system reactor water level Channel 1	anticipatory alarm
C1-3	steam to turbine pressure low	alarm unit	trips reactor bldg. steam isolation valve
C1-4	reactor steam outlet pressure high	safety system reactor pressure Channel 2	anticipatory alarm
C10-1	Channel 1 low countrate	N.I. Channel 1 level trip	reactor start
C10-2	Channel 2 low countrate	N.I. Channel 2 level trip	reactor start
C10-3	Channel 1 short period	N.I. Channel 1 period trip	withdraw permit
C10-4	Channel 2 short period	N.I. Channel 2 period trip	withdraw permit
C11-1	Channel 3 short period	N.I. Channel 3 period trip	withdraw permit
C11-2	Channel 4 short period	N.I. Channel 4 period trip	withdraw permit
C11-3	nuclear instrument trouble	N.I. Channels 3 & 4 downscale level trip and Channels 7 & 8 trouble monitor	trouble alarm
C11-4	flux level channels in 1 of 2 logic	safety system coincidence relay	normal alarm condition indicates 2 out of 4 coincidence
C12-1	Channel 5 flux level low	N.I. Channel 5 downscale level trip	anticipatory alarm

TABLE 11-3 - Safety System Alarms (continued)

<u>window</u>	<u>name plate</u>	<u>actuating device</u>	<u>remarks</u>
C12-2	Channel 6 flux level low	N.I. Channel 6 down-scale level trip	anticipatory alarm
C12-3	Channel 7 flux level low	N.I. Channel 7 down-scale level trip	anticipatory alarm
C12-4	Channel 8 flux level low	N.I. Channel 8 down-scale level trip	anticipatory alarm
C13-1	Channel 5 flux level high	N.I. Channel 5 level trip	anticipatory alarm
C13-2	Channel 6 flux level high	N.I. Channel 6 level trip	anticipatory alarm
C13-3	Channel 7 flux level high	N.I. Channel 7 level trip	anticipatory alarm
C13-4	Channel 8 flux level high	N.I. Channel 8 level trip	anticipatory alarm
D1-1	control rod accumulator gas pressure low	(29) pressure switches	partial scram
D1-2	control rod accumulator oil level low	(29) pressure switches	partial scram
D2-1	reactor water level high-high	safety system level Channels 1 and 2	full scram
D2-2	reactor water level low-low	safety system level Channels 1 and 2	full scram
D2-3	reactor steam outlet pressure high	safety system pressure Channels 1 and 2	full and partial scram
D2-4	power recirc. flow abnormal	safety system power-flow Channels 1 & 2	full and partial scram
D3-1	turbine bldg. steam isolation valve not full open	auxiliary relay-limit switch	full scram
D3-2	reactor bldg. steam isolation valve not full open	auxiliary relay-limit switch	full scram

Table 11-3 - Safety System Alarms (continued)

<u>window</u>	<u>name plate</u>	<u>actuating device</u>	<u>remarks</u>
D3-3	turbine stop valve not full open	auxiliary relay-limit switch	partial scram
D3-4	turbine condenser vacuum low	auxiliary relay-vacuum switch	full scram
D4-1	reactor partial scram	partial scram prohibit relay	partial scram
D4-2	reactor all-rod scram	full scram prohibit relay	full scram
D4-3	reactor building manual scram	auxiliary relay	full scram
D5-1	Channel 3 short period	N.I. Channel 3 period trip	full scram
D5-2	Channel 4 short period	N.I. Channel 4 period trip	full scram
D5-3	Channel 5 flux level high	N.I. Channel 5 level trip	full scram
D5-4	Channel 6 flux level high	N.I. Channel 6 level trip	full scram
D6-1	Channel 7 flux level high	N.I. Channel 7 level trip	full scram
D6-2	Channel 8 flux level high	N.I. Channel 8 level trip	full scram
D7-1	reactor MCC 1A voltage low	voltage relay	full scram
D7-2	2400-v Bus 1A voltage low	voltage relay	partial scram
D7-3	2400-v Bus 1B voltage low	voltage relay	partial scram
E3-1	power recirc. flow abnormal	safety system power-flow Channel 2	anticipatory alarm

11.4 MAIN STEAM AND FEEDWATER SYSTEM

The steam and feedwater system is described in Sec. 5.1.1 and Figs. 5.1 and 5.2 are the system flow diagrams. System operation is fundamental to the plant operating procedures described in Sec. 13.4. Table 11-4 gives the system alarms.

TABLE 11-4

MAIN STEAM AND FEEDWATER SYSTEM ALARMS

<u>window</u>	<u>name plate</u>	<u>actuating device</u>	<u>remarks</u>
B6-1	1390 psig safety valve open	temperature switch	
B6-2	either 1426 psig safety valve open	temperature switch	
B12-3	feedwater hyd. coupling tanks temp. high	temperature switches	
B12-4	feedwater hyd. coupling tanks level low	level switches	
B13-1	bearing oil header press. low R.F.P. trip	pressure switches	
B13-2	feedwater aux. oil pumps auto-start	pressure switches	
B13-3	feedwater bearing oil tanks level low	flow switches	
B13-4	feedwater hyd. coupling water pump trip	auxiliary switches	
B14-2	main steam by-pass accumulator pump 1A or 1B auto start	auxiliary contact	
B14-3	main steam by-pass valve not closed	limit switch	
B14-4	air compressor 1A or 1B auto start	auxiliary contact	
C1-3	steam to turbine pressure low	alarm unit	

Table 11-4 - Main Steam and Feedwater System Alarms (continued)

<u>window</u>	<u>name plate</u>	<u>actuating device</u>	<u>remarks</u>
C1-4	reactor steam outlet pressure high	safety system reactor pressure Channel 2	
D2-3	reactor steam outlet pressure high	safety system pressure Channels 1 and 2	full and partial scram
D3-1	turbine bldg. steam isolation valve not full open	auxiliary relay-limit switch	full scram
D3-2	reactor bldg. steam isolation valve not full open	auxiliary relay-limit switch	full scram
D3-3	turbine stop valve not full open	auxiliary relay-limit switch	partial scram
D3-4	turbine condenser vacuum low	auxiliary relay-vacuum switch	full scram
E1-1	reactor auxiliary device tripped	auxiliary switch	

11.5 DECAY HEAT COOLING SYSTEM

The decay heat cooling system is described in Sec. 5.2.1, and Fig. 5.5 is the system flow diagram.

11.5.1 General Operation

The decay heat system outlet valve is locked closed while the reactor is in operation (see Sec. 5.2.1.1). Prior to using the decay heat loop for continuing the controlled cooling after reactor shutdown, the system will be heated at a controlled rate (not to exceed 50 F per hour) until it approaches reactor water temperature. To heat the system, the by-pass valve around the discharge check valve is regulated to control reverse system flow by the forced-circulation pump head.

After a reactor shutdown, the reactor cooldown rate is initially controlled by regulation of the control rod drive seal water injection flowrate and by flashing steam to the main condenser, thus preventing excessive thermal stresses on the reactor vessel. After the reactor water temperature is reduced to 470 F or below, the reactor building

steam isolation valve is closed. At this time the reactor might be flooded beyond the upper flange to accelerate the cooldown. This is done by allowing seal water to accumulate, and by adding feedwater. The reactor is then cooled at 50 F per hour by circulation through the decay heat cooler and back to the reactor via the 2-in. line to the reactor head. The cooling rate will be controlled by regulating the flow of reactor water and cooling water through the decay heat cooler.

Operating procedures incorporate the following precautions:

- (1) During cooldown, temperatures sensed by thermocouples on the vessel wall are watched to ensure that the allowable vessel cooldown rate and the allowable temperature difference between two points on the vessel are not exceeded.
- (2) The maximum temperature drop on the tube side of the decay heat cooler is < 150 F, to prevent overstress in the tube sheet for the 1000 design cooldown cycles.
- (3) The maximum allowable cooling water outlet temperature for the decay heat cooler is 180 F, to minimize the temperature drop on the tube side and thus minimize stresses in the tube sheet.
- (4) While the decay heat pump is operating, the minimum allowable flow is 100 gpm and the maximum allowable temperature of water entering the seal is 160 F.
- (5) When the water temperature in the decay heat pump is above a given value, cooling water flow is maintained through the pump seal and bearing jackets, and the pump pedestal.
- (6) To prevent inadvertent draining of the reactor, the system blowdown valve and all component vents and drains are closed before system valves are opened.

11.5.2 Engineered Safeguards

Inadvertent draining of the primary water through the blowdown line to the main condenser is prevented by the following engineered safeguards:

- (1) Loss of Air to Blowdown Control Valve - The blowdown control valve fails closed upon loss of air.
- (2) Power to Solenoid Valve - The three-way solenoid valve in the air supply line to the blowdown control valve fails in the vented position upon loss of power, thus shutting off air to the blowdown valve and keeping the blowdown valve closed. The solenoid valve also receives a low level signal from the reactor safety system that maintains the valve in a vented position to prevent an open blowdown valve while reactor water level is low.

To prevent an accidental cold water reactivity insertion from the decay heat cooling loop a valve in the discharge line from the decay heat pump is locked closed at all times

while the reactor is in operation. Keys are maintained under control of the shift supervisor.

11.5.3 Alarms

Table 11-5 gives the decay heat cooling system alarms.

TABLE 11-5
DECAY HEAT COOLING SYSTEM ALARMS

<u>window</u>	<u>name plate</u>	<u>actuating device</u>	<u>remarks</u>
E13-3	decay heat cooler outlet temp. low	temperature switch	decay heat pump should not be started until the alarm is clear to avoid high thermal stress at the decay heat loop discharge nozzle
E1-1	reactor auxiliary device tripped	decay heat pump motor trip	if pump operation is not desired switch is placed in pullout position. After trip of operating pump, reason for trip is determined, and pump restarted.

11.6. SHUTDOWN CONDENSER SYSTEM

The shutdown condenser system is described in Sec. 5.2.2, and Fig. 5.6 is the system flow diagram.

11.6.1 General Operation

During reactor operation, the shutdown condenser system is normally isolated from the reactor plant by control valves and is started automatically when the reactor building steam isolation valve or the turbine building steam isolation valve is not fully open, or when the reactor pressure exceeds 1325 psig. These are emergency conditions that also provide a scram signal to the reactor safety system.

The startup sequence is as follows:

- (1) The steam inlet and tube-side vent control valves open instantaneously.
- (2) The condensate outlet control valve opens after a 10-sec delay.

(3) The shell-side level controller positions the demineralized water control valve to maintain a constant level.

(4) If the demineralized water supply does not maintain the normal level with the demineralized water valve wide open, a signal from the level controller opens the service water valve.

When the shutdown condenser starts automatically, the valves remain open until the initiating condition is corrected.

If system operation is initiated by high reactor pressure, the shutdown condenser valves close when the reactor pressure is sufficiently reduced. The main steam by-pass valve then dumps steam to the main condenser to maintain reactor pressure between 1265 and 1280 psig. Thus when system operation is initiated by high reactor pressure, no operator action is necessary during the startup, operation, or shutdown phases.

If system operation is initiated by a closure or partial closure of the reactor building steam isolation valve or the turbine building steam isolation valve, the system operates until the valve reopens or the interlock is bypassed. The steam inlet valve is fully open and its hand controller is bypassed, so reactor pressure reduction and cooldown rates are uncontrolled. The operator then regains control of the steam flow by bypassing the interlock and controlling the steam inlet control valve manually.

If reactor cooldown must be done with the reactor plant isolated from the main condenser, the shutdown condenser operates with a controlled cooling rate until the reactor water is cooled to 470 F. The reactor is further cooled down in accordance with the operating procedure for the decay heat cooling system.

After each operation, the shell side of the shutdown condenser is refilled with demineralized water and chemicals are added to reduce corrosion.

Operating procedures include the following precautions:

(1) The manual shutoff valves in series with the system control valves are kept open while the reactor is operating so that the system starts automatically when required.

(2) Cooldown procedures for the reactor vessel are followed carefully to avoid damage from thermal shock or high thermal stresses.

(3) Drain lines from the equipment and piping of the shutdown condenser tube side must be checked periodically to prevent cold water accumulation from valve leakage, residual water after system operation, etc.

(4) The shell side of the shutdown condenser will be sampled periodically for corrosion protection. Samples taken during and subsequent to operation are checked for activity to detect any leaks that may develop.

11.6.2 Engineered Safeguards

The following subsections describe the design philosophy to protect against system malfunctions.

11.6.2.1 Loss of Valve Control Air. The steam inlet, condensate outlet, and demineralized water supply control valves fail open upon loss of control air. The tube side vent and service water control valves fail closed. The manual valves in each line are positioned to operate the system as required by reactor and generator plant conditions.

11.6.2.2 Loss of Power to Solenoid Valves. The steam inlet, condensate outlet, and condenser vent valves fail open upon loss of control power. Dual solenoids ensure valve operation if one solenoid should fail. Since d-c control power to these valves is supplied by multiple sources, power loss is very unlikely.

11.6.2.3 Loss of Demineralized Water Supply. Loss of demineralized cooling water is automatically remedied by the shell-side level controller, which opens the service water supply control valve.

11.6.2.4 Power Failure. Upon a complete loss of electrical power, the diesel engine driven auxiliary service water pump supplies 200 gpm to the reactor building for the shutdown condenser, emergency core spray cooling system, and the shield coolers.

11.6.2.5 Failure of High Pressure Channel. If the pressure transmitter fails to signal the shutdown condenser upon high reactor pressure (1325 psig), a second pressure transmitter (set for 1350 psig) supplies a signal to another reactor pressure channel.

11.6.2.6 Alarms. Shutdown condenser system alarms are given in Table 11-6.

TABLE 11-6

SHUTDOWN CONDENSER SYSTEM ALARMS

<u>window</u>	<u>name plate</u>	<u>actuating device</u>	<u>remarks</u>
C1-4	reactor steam outlet pressure high	reactor pressure Channel No. 2 trip unit	reactor safety system anticipatory alarm
C4-1	shutdown condenser water level high-low	alarm units	
D2-3	reactor steam outlet pressure high	reactor pressure	reactor safety system scram alarm
D3-1	reactor bldg. steam isolation valve not full open	limit switch	reactor safety system scram alarm
D3-2	turbine bldg. steam isolation valve not full open	limit switch	reactor safety system scram alarm
E15-1	reactor auxiliary temperature high	temperature recorder	

11.7 PRIMARY PURIFICATION SYSTEM

The primary purification system is described in Sec. 5.2.3.2, and Fig. 5.8 is the system flow diagram.

11.7.1 General Operation

The purification system operates continuously while water is being circulated through the reactor, except under the following conditions:

- (1) during the initial cooldown period after reactor shutdown and prior to reactor flooding (operation here might cool the reactor water much faster than the vessel head and cause excessive thermal stresses),
- (2) after boron injection for emergency shutdown or control rod calibration (system operation would remove the boron from the reactor water), and
- (3) while the purification system is used to blow down reactor water to the over-head storage tank.

During system operation, the radiation levels at the ion exchangers and the filters are periodically checked. The filters are cleaned and the resin replaced before the radiation level creates problems in handling the resin or filter. The filters are also cleaned upon indication of an excessive pressure drop across the filter; the resin is replaced when water conductivity measurement indicates that the resin is depleted or when pressure drop across the resin bed is excessive.

Operating procedures for the purification system incorporate the following precautions:

- (1) The system is not operated at a flowrate of less than 25 gpm, so that the possibility of channeling in the ion exchangers is reduced and good filtering of suspended solids is provided.
- (2) If the ion exchangers are bypassed, a minimum allowable flow (10 gpm at 250 F) is maintained to prevent cavitation of the purification pump.
- (3) The temperature of seal water leaving the purification pump seal is kept below 160 F.
- (4) Cooling water flow is maintained through the pump bearing and seal chamber jackets to keep the water temperature in the purification pump within allowable limits.
- (5) The system is operated with a purification cooler outlet temperature less than 120 F to prevent resin damage.

(6) The vents and drains on the system components and the blowdown valves are potential means of draining the reactor. These valves are kept closed when not in use.

11.7.2 Engineered Safeguards

11.7.2.1 Loss of Control Air. The purification system control valve at the inlet to the regenerative cooler fails closed upon loss of air; the purification pump is interlocked to trip when the control valve is closed to avoid pump cavitation. The pump can also be tripped from the control room.

11.7.2.2 Purification Cooler Tube Failure. If a purification cooler tube fails, reactor water leaks into the reactor building component cooling water system. The subsequent high radiation in the component cooling water system is indicated locally and sounds an alarm in the control room. A leak in the regenerative cooler would be indicated by abnormal temperature and flow behavior on purification system indicators provided in the control room. Both of these coolers can be isolated between the purification system control valve and the purification pump discharge check valve by closure of the control valve, using the hand control switch provided in the control room.

11.7.3 Alarms

Primary purification system alarms are given in Table 11-7.

TABLE 11-7

PRIMARY PURIFICATION SYSTEM ALARMS

<u>window</u>	<u>name plate</u>	<u>actuating device</u>	<u>remarks</u>
E10-1	mixed bed exchanger conductivity high	conductivity relay	depletion of mixed bed resins; channeling in mixed bed resins
E10-2	mixed bed exchanger to regen. cooler flow low	alarm unit	
E10-3	mixed bed and cation exchanger inlet temp- erature high	temperature switch	

11.8 BORON INJECTION SYSTEM

This system is described in Sec. 5.2.5, and Fig. 5.14 is the system flow diagram.

11.8.1 Standby Conditions

The standby condition of the boron injection system before system operation is as follows:

- (1) The following valves are open:
 - (a) boron tank outlet valve (locked open)
 - (b) boron tank to pumps control-valve-inlet stop-valve (locked open)
 - (c) boron tank to pumps control-valve-outlet stop-valve (locked open)
 - (d) core spray pumps suction valves
 - (e) core spray pumps discharge valves
- (2) The following valves are closed:
 - (a) boron tank drain to retention tanks stop-valve (locked closed)
 - (b) boron tank to pumps control-valve
 - (c) pumps to reactor forced-circulation header control-valve
- (3) The core spray pumps are not operating.

11.8.2 Operation

The boron injection system can be placed in operation by pushing the boron injection button in the control room following a reactor full scram condition. An alarm indicates if any rods fail to insert fully upon a full scram signal. If needed to complete the shut-down, the boron injection system can be actuated. Pushing the button accomplishes the following:

- (1) Closes the control valve in the line from the overhead storage tank to the core spray pumps suction. This valve is part of the emergency core spray system (Sec. 5.2.6).
- (2) Closes the control valve in the core spray line to the vessel. This valve is part of the emergency core spray system.
- (3) Opens the boron tank to pumps control-valve.
- (4) Opens the pumps to reactor control-valve.
- (5) Closes the primary purification system control valve, which is interlocked to trip the purification pump.
- (6) Starts both core spray pumps.

When the boron tank level drops to the low-low level alarm point, the button is released to stop boron system operation. Boron removal from the reactor is discussed in Sec. 11.13.2.

11.8.3 Special Instructions

The boron injection system is tested during reactor shutdown periods to ensure that the pumps are operable and the boron injection control valve is not clogged. Manual gate valves are positioned so that water supply to the pumps is taken from the overhead storage tank during these tests. A key switch can be placed in "Valve Test" position to permit control-valve testing when the reactor is not scrammed. Placement of the switch in the "Alarm Test" position allows testing of the alarm that signals all rods are not fully inserted within 3 sec after a full scram.

While the reactor is operating, samples are periodically drawn from the boron supply tank to ensure that the line and valves to the discharge control-valve are not clogged. If there is plugging, this line can be flushed through the sample connections.

The boron system operating precautions are as follows:

(1) The primary purification system pump must be manually tripped if it is running when the boron injection system is started, since the purification system discharge control valve closes automatically during boron injection.

(2) Boron injection is to be stopped when the boron tank level drops to the low-low level alarm point to prevent core spray pump operation without water supply. The water will then be supplied from the overhead storage tank, flushing the injection line until the pumps are tripped manually or by high reactor water level.

(3) The boron tank and lines from the tank to the first normally closed valve must be maintained at 100-110 F to assure that the sodium pentaborate does not precipitate from solution.

11.8.4 Engineered Safeguards

Two solenoid valves installed in parallel admit nitrogen to open the boron injection control-valves upon a signal from the control room. This double solenoid arrangement provides reliability for positioning the control-valves for boron injection.

Two nitrogen bottles provide the gas pressure to open the boron injection control-valves, making them independent of the control air supply.

Electric heating elements are provided in the boron tank, and electric heating cables are provided for the boron injection piping and valves. Temperature switches control the heaters to maintain the standby boron solution, piping and valves at 100-110 F, protecting against boron injection line clogging by precipitated sodium pentaborate.

The boron injection control valves cannot be opened if power fails to the dual solenoids. A bypass line around the initial injection control valve contains a manual gate valve which can be opened to allow boron injection through the core spray header. Prior to opening the bypass valve, a manual gate valve in the line from the overhead storage tank must be closed.

A single boron injection pushbutton in the control room provides the signal to start both core spray pumps and position the control valves for boron injection; the start circuit cannot be closed unless there is a full scram signal. Both core spray pumps are connected to the engine-generator set emergency power supply to provide additional reliability.

Normally open manual valves in the line from the boron supply tank to the emergency core spray pumps are locked in the open position to provide administrative control protection against inadvertent closing of those valves.

11.8.5 Alarms

Boron injection system alarms are listed in Table 11-8.

TABLE 11-8
BORON SYSTEM ALARMS

<u>window</u>	<u>name plate</u>	<u>actuating device</u>
C3-1	sodium pentaborate injection	auxiliary relay
C3-2	sodium pentaborate level (low)	level switch
C3-3	sodium pentaborate temperature low	temperature recorder
C5-4	sodium pentaborate level (lo-lo)	level switch
E15-1	reactor aux. temperature (hi)	temperature recorder

11.9 EMERGENCY CORE SPRAY SYSTEM

Section 5.2.6 describes this system, and Fig. 5.14 is the system flow diagram.

11.9.1 Standby Conditions

During reactor operation the emergency core spray system is valved open from the overhead storage tank to the core spray header via the core spray pumps. Both pump control switches are on AUTO, and the selector switch is turned to the pump that will start automatically. The lines from the boron injection tank and from the pumps discharge to the forced-circulation system header are valved closed as indicated in Sec. 11.8.1.

11.9.2 General Operation

The reactor scrams and the emergency core spray cooling system is started automatically when the reactor water level drops to the low-low level switch set point. A supply of water from the overhead storage tank is always available to the emergency core spray cooling pumps suction header. One of the pumps starts automatically to supply water to the reactor vessel spray header at a rate of ~ 50 gpm. Manual control switches are provided in the control room so that the second pump can be started to obtain a total flow-rate of ~ 100 gpm.

If signals are received from both the reactor water low level alarm and the reactor low pressure transmitter, the low pressure core spray control valve opens automatically so that water flows directly from the overhead storage tank or the service water system (when opened) to the core spray header.

The overhead storage tank low level alarm warns the operator that the tank water level is low, and the level switch causes the control valve in the demineralized water supply line to open. If additional core spray water is needed, the control valve in the high pressure service water line is opened from the control room.

If the operating pump trips automatically because of motor overload, the standby pump starts automatically.

The emergency core spray system is shut down by tripping the pumps from the control room.

Operating precautions are as follows:

- (1) The core spray pumps must be tripped or supplied with service water when the overhead storage tank is empty, to prevent damage to the pumps.

(2) The system valves must be in their normal position at all times while the reactor is operating, so that the system can be operated if it is required.

(3) The stop valve in the water supply line to the pumps must be closed while the reactor is not pressurized, to prevent flow through the idle pumps to the reactor. The valve must be reopened on reactor startup.

11.9.3 Engineered Safeguards

All control valves in the emergency core spray system fail open on loss of nitrogen supply or loss of power to the solenoid valves.

The control valve in the high pressure service water supply line to the emergency core spray system fails closed on loss of air or control power, preventing service water flow to the reactor if the reactor is not pressurized.

The core spray pumps are connected to the engine-generator set emergency power supply. If one of the pumps trips because of overcurrent, the standby pump starts automatically.

The reactor can also be supplied with core spray by gravity feed, for a MCA type situation. A supply (15,000 gal) of the water in the overhead tank is reserved for the core spray by appropriate location of outlet connections on the tank.

The manual gate valve in the core spray supply line is normally locked open to provide administrative control against inadvertent closing.

11.9.4 Alarms

Emergency core spray system alarms are listed in Table 11-9.

TABLE 11-9
EMERGENCY CORE SPRAY ALARMS

<u>window</u>	<u>name plate</u>	<u>actuating device</u>	<u>remarks</u>
C1-2	reactor water level low	reactor level	reactor safety system anticipatory alarm
C4-2	overhead storage tank level (high)	level switch	demineralized water supply control valve is closed
C4-3	overhead storage tank level low	level switch	demineralized water supply control valve opens
D2-2	reactor water level low-low	reactor level	reactor safety system, reactor scram, and one emergency core spray pump starts automatically
E14-1	emergency core spray flow (low)	flow indicator switch, plus low reactor water level	second emergency core spray pump should be manually started

11.10 COMPONENT COOLING SYSTEM

This system is described in Sec. 5.2.8, and Fig. 5.16 is the system flow diagram.

11.10.1 General Operation

Prior to system operation, the inlet component cooling water valves to all components are opened. The component cooling water outlet valves on equipment such as the shield cooler, purification cooler, and forced-circulation pump auxiliary equipment coolers are about half opened to allow a recirculation path for the component cooling water.

The hand control switch for one of the component cooling pumps is used to start that pump. Two pumps can be used if needed.

The valves in the discharge lines of the various coolers are adjusted until the estimated required flows (Sec. 5.2.8.1) are obtained.

As the rest of the plant is brought on stream, the temperature indicators are monitored and the cooling water flows adjusted as required.

Flows through the system are monitored at the outlet of all heat exchangers. The inlet and exit temperatures of the component coolers are noted, in addition to the temperatures of the various other exchangers being supplied with cooling water. The level control maintains the correct surge tank water level. For an excessively low level, the level switch initiates the alarm in the control room.

Routine checks for increased activity level in the component cooling water are made by use of the radiation monitor.

The system is shut down by using the hand control switch to trip and shut down the component cooling water circulating pump(s).

11.10.2 Loss of Power

Upon loss of power, the component cooling water pumps are automatically supplied with emergency power from the engine generator set.

11.10.3 Alarms

Component cooling system alarms are listed in Table 11-10.

TABLE 11-10

COMPONENT COOLING SYSTEM ALARMS

<u>window</u>	<u>name plate</u>	<u>actuating device</u>
C9-1	component cooling water radiation high	radiation monitor
C9-2	component cooling water monitor trouble	flow switch
E12-4	component cooling flow low	flow indicator switch
E13-1	component cooling surge tank level low	level switch
E13-2	component cooling water temperature (high)	temperature switch

11.11. SHIELD COOLING SYSTEM

This system is described in Sec. 5.2.9, and Fig. 5.17 is the system flow diagram.

11.11.1 General Operation

The hand control switch for one of the shield cooling pumps is used to start the pump and the coolant flow through the system.

Initially the globe valves in the exit side of each cooling circuit are fully open to allow full flow through the various cooling coils.

After the pump is started, the globe valve in the exit pipe from each cooling circuit is adjusted until the indicated flow approximates the value given in Sec. 5.2.9.1.

After the plant reaches and maintains full power, the temperature indicators for the various thermal shield cooling circuits and for the shield surface are monitored. The circuit outlet valves are adjusted as required to control the shield temperatures. The flowrates resulting from the adjustments are logged and used for balancing the system for future startups.

The inlet and exit temperatures of the shield cooler are maintained. Service water is adjusted so that proper temperatures are maintained. The level control maintains the correct surge tank water level. For an excessively low level the level switch operates the alarm in the control room.

The shield cooling system water is sampled periodically and tested for pH, chloride, total solids, and long-lived gross gamma activity. If the water conductivity is above maximum limits the portable demineralizer is inserted into the system.

The system is shut down by using the hand control switch to trip and shut down the shield cooling pumps.

The operator is alert for the following:

- (1) Cooling water must be provided to the shield cooling circuits whenever the reactor is at or approaching power.
- (2) Cooling water supply to the shield water cooler must be assured at all times to provide a heat sink for the shield water.
- (3) The quality and pH of the shield water must be closely monitored and adjusted if necessary, to assure minimum corrosion and/or scaling on the embedded shield cooling coils.

11.11.2 Loss of Power

On loss of power, the shield cooling pumps are automatically supplied with emergency power by the engine generator set.

11.11.3 Alarms

The shield cooling system alarms are listed in Table 11-11:

TABLE 11-11

SHIELD COOLING SYSTEM ALARMS

<u>window</u>	<u>name plate</u>	<u>actuating device</u>
E9-2	shield cooling flow (low)	any of six flow switches
E9-3	shield cooling surge tank level (low)	level switch
E9-4	shield cooling pump discharge flow (low)	flow switch
E15-1	reactor aux. temperature high	temperature recorder

11.12 STORAGE WELL COOLING SYSTEM

The system is discussed in Sec. 5.2.10 and Fig. 5.18 is the system flow diagram.

11.12.1 General Operation

Initially the pool is filled with water to the normal level of operation, and the valves are open as follows:

- (1) filter inlet and outlet valve
- (2) suction and discharge valves of one pump
- (3) cooler inlet and outlet tube side valves
- (4) cooler inlet and outlet shell side valves

All other system valves are initially closed; all instrumentation and controls have been checked and calibrated. Normally only one pump is used.

If a system failure that results in leakage of storage well water occurs, makeup water can be supplied from the overhead storage tank at a rate of 830 gpm.

The system is placed in operation by manual opening of valves so that pool water flows from the upper inlet and skimmer in the well, through the filter, pump, and cooler, and back into the bottom of the well. The pump is then started with the manual switch in the control room, or with the local control switch.

In this mode of operation the well water is recirculated through the filter and the cooler. The well water temperature is kept below 120 F. The storage well inlet temperature is controlled with the bypass around the cooler. Filter pressure drop in the clean condition should be about 3 psi.

If the water becomes excessively contaminated with dissolved impurities, the portable demineralizer can be connected into the cooling system.

If it becomes necessary to remove impurities from the lower portion of the pool, the system suction can be shifted from the skimmer to the bottom drain by suitable valve manipulation. Fluid flow through the components is otherwise the same.

11.12.2 Alarms

The alarms in the system are as follows:

<u>window</u>	<u>name plate</u>	<u>actuating device</u>
E14-2	fuel element storage well level low	level switch
E15-1	reactor aux. temperature high	temperature recorder

11.13 DRAIN AND LIQUID WASTE DISPOSAL SYSTEM

Section 5.3.1 describes the system, and Fig. 5.37 is the system flow diagram.

11.13.1 General Operation

When sufficient liquid waste has been collected to justify dumping, each retention and waste water tank is examined to determine the quantity and condition of the water.

Before disposing of the water from the waste water tanks, one tank is isolated and all normal drainage is transferred to the other tank by valving at the tank headers.

Since retention tank 1A cannot be isolated, the contents of tank 1A are pumped into tank 1B. Tank 1B is then isolated and all normal drainage is transferred to tank 1A.

The tank contents are recirculated to produce a homogeneous mixture in the tank. The contents are pumped out at the bottom and discharged back into the top. A sample is withdrawn from the tank and checked for radioactivity level.

If the radioactivity level is low enough for discharge to the river without processing, further analysis is not necessary. If a high activity indication precludes direct disposal, the sample is analyzed for conductivity. If high solids are indicated, the tank contents are transferred to the evaporator feed tank. If the solids content is not excessive, the water is processed through the Radwaste ion exchanger.

11.13.1.1 Discharge to the River. After analysis of the collected waste batch has determined a safe activity level for discharge to the river, the discharge rate for satisfactory dilution with main condenser circulating water is determined. The permissible discharge rate is a function of the radioactivity level -- the higher the level, the more dilution required. The total-flow indicator is read and recorded. The globe valve in the discharge line to the river is unlocked by a key, provided proper operation of the radiation monitor system in the liquid waste and service water discharge line is confirmed. The retention or waste water storage tank pump is started. The normally locked globe valve in the discharge line is gradually opened until the flowrate for proper dilution is obtained. When the tank has been emptied, the pump stops automatically.

11.13.1.2 Evaporation Cycle. The contents of the retention or waste water storage tanks are transferred to the evaporator feed tank. Care is taken that the evaporator feed tank does not overflow if more than 1000 gal is in the retention or waste water storage tank at the time of transfer (as with condensate regeneration waste solutions).

The liquid waste demineralizer pump then recirculates the evaporator feed tank contents through the neutralizer eductor and back to the feed tank. When a homogeneous mixture is assured, the contents of the feed tank are sampled and the pH determined. The proper amount of additive (pH control and/or defoaming agents) is added to the chemical mixing tank. The contents of the mixing tank are drawn into the waste solution through the eductor. The evaporator feed tank contents are periodically checked until the pH is adjusted. The liquid waste demineralizer pump is then stopped.

The evaporator level controls are activated, and the evaporator feed tank outlet is opened to the evaporator inlet line. The pump is started and operated until the evaporator feed flow control valve is automatically positioned to "not full open." The heat steam stop valve is opened and steam flows at 100 psig through the shell side of the evaporator, to maintain an evaporator pressure of 15 psig. The evaporator feed is automatically regulated to maintain the correct level in the evaporator; if the feed control valve closes completely because of high level or for other reasons, the evaporator feed pump stops automatically. As soon as the feed control valve begins to open, the evaporator feed pump starts automatically if the control has not been repositioned.

The vapor generated in the tube side of the evaporator exits to the water collection tank through the steel wool packed demister and evaporator condenser. The service water flow through the tube side of the evaporator condenser is adjusted until the condensed vapor is subcooled to at least 120 F. The evaporator demister is backflushed from the water collection tank after completion of each batch. The demister drains to the evaporator bottoms section.

The contents of the water collection tank are sampled for radioactivity, conductivity, pH, and chlorides. If there is room in the main condensate water surge tank, and if the analyses are satisfactory, the water in the water collection tank is pumped to the main condenser hot well with the process water transfer pump. The excess water introduced to the hot well is transferred by the condensate pump to the main condensate water surge tank.

If storage space is not available in the main condensate water surge main tank, and if the activity level will meet 10 CFR 20 standards after mixing with condenser circulating water, the water is pumped to the river with the same precautions and in the same manner as previously described. If the activity level of reclaimed water in the water collection tank is too high for discharge to the river or for storage, the water is further processed through the Radwaste ion exchanger.

After each batch of liquid waste (~ 1000 gal) is processed in the liquid waste evaporator, the solids concentration in the evaporator bottom is ~ 35 w/o, and the temperature ~ 215 F. The evaporator residue is preferentially packaged directly from the evaporator. However, if the evaporator load requirements make necessary immediate further evaporator operation, the residue is drained to the concentrated waste storage tank in the basement of the waste treatment building.

The evaporator bottom discharge nozzle is manifolded to two lines, one leading to the drumming station and the other to the concentrated waste storage tank. Each line has an isolation plug valve with a stem extension through the shielded wall. If the drumming station is not available, and if continued use of the evaporator is required, the heating

coils on the line to the concentrated waste tank are energized to preheat the pipe to ~ 150 F. The extended stem plug valve in the line to the storage tank is then opened to drain the residue into the tank. After draining is completed, the tank steam sparger is operated as necessary to maintain the fluidity of the stored concentrated waste.

11.13.1.3 Drumming of Concentrated Waste. The drumming operation precludes direct contact between the operator and the concentrated waste piping. The basic container is a 55-gal drum. Normally, the liquid waste is mixed with cement and absorbent within the entire 55-gal drum. However, if the radiation level of the liquid waste is above what the cement self shielding can keep within 10 CFR 20 limits, shielding cement is poured into an annulus formed by a 35-gal drum within the 55-gal drum. Empty drums are stored in an outside storage facility. The ground floor of the waste treatment building is kept free of cement and sawdust.

The drum, containing dry cement and absorbent, is lowered from the ground floor through a hatch, into the basement and onto a dolly within a concrete block box immediately below the hatch. The hatch is ~ 13 ft inside the double doors of the east corner. The concrete block box is high enough to contain any splashing during filling and mixing of waste. A monorail chain hoist overhead of the ground floor is used for handling the drums, which weigh 650 lb filled.

The drum is clamped in position on the dolly by a screw jack that has a handle extension through the concrete block box. The dolly is positioned beneath the mixer with the jack handle, and the dry cement and absorbent are mixed with the drilling machine.

After the dry cement mixture is homogeneous, the evaporator residue is either drained into the drum or pumped into the drum after going to the concentrated waste storage tank. If pumped from the concentrated waste storage tank, the waste volume is controlled by setting the revolution counter on the shaft of the positive displacement pump; waste volume from the evaporator is judged by the level within the drum. A mounted mirror provides a remote view of the drum level. The pipe line that carries the evaporator residue to the drum is heated and shielded where exposed to operating personnel. A manually operated quick closing valve with an extended handle is at the end of the tubing that opens into the drum.

After cement is mixed with radioactive liquid waste, the barrel is lifted from the box to the grade level floor, where it stands until the cement sets.

After final checks of activity and labeling, the top of the barrel is welded in place. The capped barrel is then removed for outside storage in a controlled area.

After a batch of evaporator residue is drummed, the pipelines handling the concentrated waste are backflushed into the evaporator feed tank to prevent plugging with caked concentrated waste.

11.13.1.4 Radwaste Ion Exchanger Process. Liquid waste in the retention or waste water storage tank to be processed through the Radwaste ion exchanger is pumped through the Radwaste ion exchanger into the water collection tank. No more than 1000 gal of liquid waste is processed at a time, to avoid overflowing of the collection tank.

11.13.1.5 Collection of Spent Resin. Resin whose ion exchange capacity has been depleted is sluiced to the spent resin tank from the Radwaste ion exchanger, the primary purification system cation exchanger, and the primary purification system mixed bed ion exchanger. Condensate demineralizer resin that is fouled or otherwise not regenerable is also sluiced to the spent resin tank.

The spent resin storage tank is sized to accept one complete condensate demineralizer charge of resin and its accompanying sluice water. The resins can be stored in the tank for short-lived decay.

The spent resin tank has an overflow which drains to the waste treatment building sump. The spent resin tank outlet to this line is protected with a fine screen, to retain the resin in the tank if excessive sluice water is added and overflows. The overflow line has a loop seal in the normally dry pipe run. The loop has a level switch and alarm. If sluice water overflows the spent resin tank, the level rises in the loop seal, triggers the level switch, and activates alarms locally and on the condensate demineralization system control board.

The spent resin tank has sufficient sluice water to carry out the contained resin. The tank is pressurized with plant air and the resin is discharged to drums in the same manner as the evaporator bottoms. Drumming is controlled in accordance with spent resin activity level.

11.13.2 Removal of Boron from Reactor Water Systems

After boron solution has been injected into the reactor and the reactor has been cooled, boron is initially removed by filling the reactor with demineralized water and draining to the retention tanks (without uncovering the core) for several cycles, until the reactor water boron concentration is reduced to the desired value. The primary purification system is then put on-stream and boron removal is completed by ion exchange.

11.13.3 Alarms

The alarms for the drain and liquid waste disposal system are listed in Table 11-12.

TABLE 11-12

DRAIN AND LIQUID WASTE DISPOSAL ALARMS

<u>window</u>	<u>name plate</u>	<u>actuating device</u>
C6-3	liquid waste and service water radiation high	radiation monitor
C6-4	liquid waste and service water monitor (trouble)	flow switch
C8-3	turbine-cond. cooling radiation (high)	radiation monitor
C8-4	turbine-cond. cooling water monitor (trouble)	flow switch
C14-1	reactor building sump level high	level switch
C14-2	waste disposal building sump level high	level switch
C14-3	waste gas storage vault sump level high	level switch
E12-3	6000 gal retention tank 1 or 2 level high	level indicator switch
E14-3	waste water storage tanks level high	level switch

11.14 VENT AND WASTE GAS DISPOSAL SYSTEM

Section 5.3.3 describes the system, and Fig. 5.38 is the system flow diagram.

11.14.1 Normal Operation

The system is ready for normal operation under the following conditions:

- (1) All gaseous waste vent lines are open to the stack.
- (2) The stack blowers are operating and the stack dampers are open.
- (3) The reactor building ventilation fans are operating and ventilation discharge duct valves are open to the tunnel.
- (4) The waste treatment building ventilation fan is operating.
- (5) The air-operated diaphragm valve in the off-gas line to the recombiner steam ejector is closed.
- (6) The steam supply to the recombiner steam ejector is shut off.
- (7) The remotely operated diaphragm valve in the gas storage tank discharge line to the stack is closed.
- (8) All drain valves on the gaseous waste system equipment are closed.
- (9) Instrumentation and controls have been calibrated and are operative.
- (10) Steam is available for the recombiner steam ejector.
- (11) Cooling water is available for the waste gas compressor and the waste gas compressor aftercooler.
- (12) The gas inlet valve to one of the gas storage tanks is open and the valve to the other storage tank is closed.
- (13) Control valve bypass valves are closed.
- (14) Instrument air is available for the operation of the air-operated diaphragm valves and for instruments and controls.
- (15) Cooling water is flowing through the recombiner condenser.

As soon as the main condenser air ejectors are started, gas flows from the main condenser to the stack via the holdup tank. In the stack the gases are mixed with outside air. The waste gas leaving the holdup tank is monitored continuously.

The vented equipment in the reactor containment building, the waste treatment building, and the turbine building are "breathing" on the vent headers connected to the stack inlet plenum. The negative pressure maintained in this plenum assures flow to the stack from these locations.

11.14.2 Abnormal Operation

11.14.2.1 High Radiation Alarm from Holdup Tank Effluent. When the holdup tank effluent exceeds the high radiation alarm setpoint (Sec. 8.8.1.2) the following actions take place automatically:

- (1) The control valve in the gaseous waste line from the holdup tank to the stack closes.
- (2) The control valve in the gaseous waste line to the recombiner steam ejector opens.
- (3) The control valve in the steam supply line to the recombiner steam ejector opens.
- (4) The control valve in the bypass line from the suction side of the waste gas compressor to the discharge side of the waste gas compressor aftercooler opens.
- (5) The heating coils on the inlet line to the recombiner are energized.
- (6) A radiation alarm is given in the control room.

This alignment of valves allows gas to flow to the gas storage tanks by the driving force of the recombiner steam ejector. The waste gas compressor cannot be started while the combustible gas analyzer indicates an explosive mixture. The recombiner vessel and inlet piping are warmed up before compressor startup to prevent steam condensation, which would impair recombiner efficiency. If equipment warmup can be done satisfactorily before the storage tank pressure exceeds 3 psig (a 15 to 30 min period if the tank is initially at atmospheric pressure), and the gas analyzer in the recombiner condenser effluent line indicates a non-explosive mixture (less than 4 percent hydrogen), the waste gas compressor can be manually started, and no further operating limitations apply. However, if the storage tank pressure is allowed to exceed 3 psig before the recombiner is operating at full efficiency (as determined by recorded readings from the combustible gas-analyzer), the hydrogen bearing gas mixture must be vented from the storage tank prior to compression beyond 15 psig. Short-term releases of radioactivity will not appreciably affect the annual average release. Since the system piping and vessels are designed to withstand the explosive force of an explosive mixture initially at 15 psig, a mixture that might be explosive will never be compressed beyond that value.

When the compressor is started, the control valve in the bypass line from the compressor suction to the compressor aftercooler discharge closes automatically to prevent recirculation of gases to the compressor suction; gas flow to the storage tanks continues. The waste gas compressor is manually shut down when the high radiation signal ceases, or for a high pressure alarm from the compressor discharge or gas storage tanks. In the latter case, it may be necessary to shut down the plant to solve the radioactivity problem.

The gas storage tank(s) can be vented to the stack by regulating the control valve in the gas outlet header and observing that the activity level indicated by the stack gaseous and particulate monitors remains within permissible discharge levels. The vent control valve can be manipulated in the control room by a hand indicating-controller. One of the tanks is vented while the other is available to collect system off-gases without compressor operation.

11.14.2.2 High Radiation Alarm from Reactor Containment Ventilation Effluent. In the event of excessive radioactivity (alarm level reached) in the discharge of the reactor containment ventilation fan, the dampers in the 24-in. ventilation inlet duct and in the 20-in. ventilation outlet duct close automatically, as described in Sec. 5.3.3.11, and the ventilation system recirculation valve opens to allow containment building air recirculation to cool the forced-circulation pump cubicles. The failure mode of these valves is discussed in Sec. 6.6.4. The 3-way control valve in the 4-in. vent header leaving the vapor container is controlled to route the waste gas from the reactor cavity vents directly to the stack, instead of via the reactor building ventilation system. This valve operation isolates the reactor containment ventilation system but does not interfere with the venting of any reactor plant equipment.

After resolution of the activity problem, the valve positions are reversed to restore the original mode of operation.

11.14.2.3 High Radiation Alarm from Stack Effluent. If excessive particulate or gaseous activity is indicated by the stack monitors (at a level between one and ten times the alarm level) (see Sec. 8.8.1.3), the control valve in the 4-in. reactor plant vent header to the stack is closed and the control valve off the vent header to the suction line of the waste gas compressor is opened by manual (control room) operation of a common pilot solenoid valve. The waste gas compressor is started manually compressing the vent gases into the gas storage tank(s). This operation does not interfere with the normal routing of condenser gases to the stack, since the two venting routes are independent. After resolution of the radioactivity problem the control valve positions are reversed manually back to their normal position, and the waste gas compressor is shut down manually.

11.14.2.4 Reactor Containment Building Isolation. If the reactor containment is isolated because of high reactor pressure or high containment pressure, the isolation control valve in the 4-in. vent header leaving the reactor containment building closes automatically, as do the dampers in the inlet and outlet duct of the reactor ventilation system.

A second control valve outside the reactor containment building, in the 4-in. vent header to the stack, can be manually closed from the control room if the other valve malfunctions.

11.14.3 Infrequent Operation

11.14.3.1 Main Condenser Vacuum Pump Operation. During main condenser startup the vacuum pump may be operated to speed main condenser evacuation. The large quantities of air discharged by the vacuum pump are routed directly to the stack through a separate 6-in. vent header.

11.14.3.2 Shutdown Condenser Operation (See Sec. 11.6 and Fig. 5.6). Upon operation of the shutdown condenser, the shutdown condenser vent control valve is opened automatically to vent the noncondensables from the condenser. The gases are routed to the holdup tank, and from there follow the same routes as the off-gas from the main condenser (which normally go directly to the stack). For excessive activity levels, the gases automatically are routed through the recombiner, and from there to the gas storage tanks (Sec. 11.14.2.1).

The shutdown condenser vent control valve is interlocked with the gas analyzer to prevent compression of a combustible gas mixture above 3 psig. This interlock automatically closes the control valve 2 min. after the gas storage tank pressure exceeds 3 psig while an explosive mixture exists at the recombiner condenser outlet. The 2-min delay provides sufficient time to purge the shutdown condenser of noncondensables. After control valve closure, the shutdown condenser operates "bottled up." Venting of accumulated noncondensables during bottled-up operation can be accomplished by manual (control-room) operation of the shutdown condenser vent valve.

11.14.4 Precautions

11.14.4.1 Radioactive Gas Leakage. All vents of the gaseous waste system are maintained at a slightly negative pressure by the stack blowers. Leaks in the vent piping system would cause air inflow into the system. Air in-leakage cannot occur, however, in the waste gas compressor and in the piping system from the compressor through the gas storage tanks to the stack. This system contains highly radioactive gas at pressures up to 300 psig. The integrity of this high pressure system is important. Potential sources of trouble are the valve stem packing boxes and leakage across the valve seats of check valves, control valves, and shutoff valves.

11.14.4.2 Combustible Gas Mixture. The manual startup controls of the compressor are interlocked with the gas analyzer so the compressor cannot operate while there is above 4 v/o of hydrogen in the gas sample. Frequent gas analyzer calibration checks will be made to ensure that the instrument senses and indicates correctly.

11.14.4.3 Control Valve Failure. Upon loss of air or power the control valves fail as follows:

<u>service</u>	<u>failure position</u>
containment vent header isolation valve (HC)	closed
containment vent header isolation valve (Auto)	closed
holdup tank discharge to stack	closed
holdup tank discharge to recombiner steam ejector	closed
steam supply to steam ejector	open
bypass waste gas compressor	open
reactor plant waste gas vent header to stack	closed
waste gas of reactor plant to waste gas compressor	open
stored waste gas to stack	closed
shutdown condenser vent	closed

The isolation valves in the 4-in. vent header from the containment building fail closed, isolating the vents of the reactor plant equipment. There may be a resultant gradual pressure buildup in the vent system with some gas leakage into the vapor container through valve packings, which should be corrected as soon as possible.

The control valve in the gas discharge line from the holdup tank to the stack fails closed, which interrupts venting of noncondensables from the main condenser and dissipates the condenser vacuum. The situation can be corrected by opening the manual bypass valve in the control valve bypass line. If diversion of waste gas is then required because of high activity alarm from the holdup tank effluent monitor, the bypass valve of the control valve is manually closed and the bypass valve allowing flow to the recombiner is manually opened, since the control valve in the 4-in. waste gas pipe line to the recombiner steam ejector also fails closed (on loss of air). Waste gas is then routed through the recombiner to the gas storage tanks, since the control valve for steam supply to the recombiner steam ejector and the control valve allowing bypass of the waste gas compressor both fail open. Operation of the recombiner and compressor is described in Sec. 11.14.2.1.

The control valve in the 4-in. vent header for waste gases venting from the containment building to the stack fails closed, and the control valve in the vent line connecting that vent header to the suction side of the waste gas compressor fails open. The reactor plant vent gases can then be routed to the gas storage tanks by manually starting the waste gas compressor.

The control valve which permits venting of stored waste gas to the stack fails closed, and remote control from the control room is lost. However, the bypass can be manually controlled to release gas to the stack if necessary.

The control valve in the vent line from the shutdown condenser fails closed to effect isolation. If it fails, noncondensables would not be vented from the shutdown condenser during the initial operation. This situation can be corrected immediately by opening the bypass on the control valve.

11.14.5 Alarms

Waste gas disposal system alarms are listed in Table 11-13.

TABLE 11-13

WASTE GAS DISPOSAL SYSTEM ALARMS

<u>window</u>	<u>name plate</u>	<u>actuating device</u>
C4-4	reactor building pressure high	pressure switch
C5-1	stack gaseous radiation high	radiation monitor
C5-2	stack particulate radiation high	radiation monitors
C5-3	stack monitor trouble	radiation monitors
C6-1	air ejector off-gas monitor high	radiation monitor
C6-2	air ejector off-gas monitor trouble	radiation monitor
C7-1	contain. bldg. air exh. gas radiation high	radiation monitor
C7-2	contain. bldg. air exh. partic. radiation high	radiation monitors
C7-3	contain. bldg. air exhaust monitor trouble	radiation monitors
C7-4	area monitors radiation high	area monitors
C8-1	containment building dampers closed	limit switches

TABLE 11-13 - Waste Gas Disposal System Alarms (continued)

<u>window</u>	<u>name plate</u>	<u>actuating device</u>
C8-2	reactor ventilation exhaust flow low	flow indicator
E10-4	ejector aftercooler holdup tank flow (lo)	flow indicator
E11-1	recombiner compressor pressure high	pressure indicator
E11-3	building steam to recombiner flow low	flow indicator
E11-4	recombiner disch. outlet temperature high	temp. recorder
E12-1	12,000 gal storage tank 1A or 1B press. high	pressure indicator
E12-2	12,000 gal storage tank 1A or 1B exp. mixture	alarm units

12. ENVIRONMENTAL SURVEY

A continuing regional environmental monitoring program will be conducted. Sampling has been initiated in order to provide approximately one year's data before the commencement of reactor power operations.

The purposes of this environmental monitoring program are to:

- (1) collect sufficient data to allow measuring of the base radiation levels in the vicinity of the reactor site,
- (2) measure the radiation levels in the plant environs throughout the operation of the reactor, and
- (3) compare operational levels with preoperational levels to determine if any significant differences exist and, if so, if those differences are related to LACBWR operations.

The scope of the LACBWR environmental monitoring program will be compatible with the environmental features and usage of the surrounding terrain. The program will also be compatible with other environmental monitoring programs to be conducted in that area by the U. S. Public Health Service, the U. S. Atomic Energy Commission, and the States of Wisconsin, Iowa, and Minnesota.

Present plans call for analysis of the following types of samples:

- (1) air,
- (2) Mississippi River water and silt,
- (3) well and tap water,
- (4) rain and snow,
- (5) fish and small animals,
- (6) milk, and
- (7) vegetation and soil.

The analyses will determine gross beta, gamma, and alpha levels and identify various specific isotopes as appropriate.

The program will be coordinated with the State of Wisconsin Department of Health, and results will be routinely furnished to that Department, the U. S. Public Health Service, and the U. S. Atomic Energy Commission.

13. ADMINISTRATIVE AND OPERATIONAL PROCEDURES

This section presents information pertaining to the organization of the operating staff and to operating procedures for LACBWR. This information was not included in ACNP-62574.

13.1 ORGANIZATION

During startup and initial testing, approximately 55 people will be present at the plant. This number will vary somewhat, depending on the tests underway, the number of shifts operated, and the amount of maintenance required.

The organization is structured to accomplish two primary objectives:

- (1) to provide a safe, efficient startup and testing program
- (2) to provide the maximum participation and training of the utility personnel, within the limits of Allis-Chalmers' responsibility for the plant

13.1.1 Reactor Preoperational and Low-Power Testing

The organization chart for preoperational and low-power testing is shown in Fig. 13.1.

13.1.1.1 Generator Plant Staff. Dairyland personnel are administratively organized as shown on the left of Fig. 13.1. Within this organization they will perform standby maintenance for the generator plant and maintain reactor plant services (demineralized water, heating steam, instrument and control air, etc.).

Personnel from the generator plant staff will also be assigned to the reactor staff to assist with the testing program and thus become familiar with the reactor systems. While working in the reactor plant they are under the supervision of the reactor startup staff.

13.1.1.2 Reactor Plant Staff. The reactor plant staff for preoperational and low-power testing is shown on the right of Fig. 13.1. The basic staff consists of approximately nine Allis-Chalmers personnel. These will be supplemented by Allis-Chalmers technical personnel as required for special testing; by craft labor to be supplied by the construction contractor if required; and by Dairyland operators, instrument and electrical technicians, and mechanical maintenance personnel.

The reactor staff is divided into four functional groups, all reporting to the Project Operations Manager.

The Project Operations Manager directs the startup organization and has complete responsibility for the operation and safety of the reactor plant. He works with the

Dairyland Plant Superintendent to assure coordination of plant planning and testing. He also works with the AEC Site Representative to assure that the Commission's interests and requirements are observed.

13.1.1.3 Operations Group. The Operations Group is charged with the actual performance of all tests in accordance with approved written procedures. This group collects the data prescribed in the procedures and reports and logs any abnormal or unexpected occurrences during the various tests. Each member of the group must be able to operate the reactor and all auxiliary equipment safely and efficiently.

Since many of the preoperational tests are instrument checks, and since close cooperation between the operators and the instrument technicians is essential at all stages, the instrument technicians are assigned to this group. Dairyland operators will be assigned to the Operations Group to work under the supervision of the A-C Operations Supervisor and the A-C shift supervisors.

13.1.1.4 Test Group. The Test Group coordinates the test schedules, consolidates the data, and prepares test reports for the reactor systems and components. This group consists of a Test Coordinator, a reactor physicist, and additional technical personnel from Allis-Chalmers as required for special tests.

The Dairyland Assistant Plant Superintendent or the Process Engineer and the Engineering Aide work closely with this group to familiarize themselves thoroughly with the reactor plant.

13.1.1.5 Health Physics Group. The Health Physics Group is responsible for conventional safety and fire protection, and industrial hygiene, as well as radiation safety. The group is to be organized near the end of preoperational testing and consists of an A-C health physicist and the Dairyland health and safety engineer and health physics technicians.

13.1.1.6 Plant Engineering Group. The Plant Engineering Group is responsible for heavy maintenance and for the correction of design deficiencies or faulty equipment in the reactor plant. The group consists of a Plant Engineering Supervisor, additional engineers from A-C Bethesda as required, and craft labor supplied by the construction contractor.

13.1.2 Power Testing

The interactions between the reactor and generator plants require complete coordination of operations for power testing. The organization for power testing is shown in Fig. 13.2. Operations are supervised by the A-C staff. Job descriptions for key personnel are given in App. D. The responsibilities and staffing of the functional groups are described in the following subsections.

13.1.2.1 Operations Group. The Operations Group consists of approximately 32 operators, supervisors, and instrument and electrical maintenance personnel. They are responsible for the safe and efficient operation of the entire plant and for the maintenance of all instrumentation, controls, and switch gear.

13.1.2.2 Test Group. The Test Group is responsible for test coordination, data consolidation, and preparation of test reports for all plant systems. The group consists of the Test Coordinator, the Reactor Physicist, and the Dairyland Assistant Plant Superintendent or the Process Engineer and the Engineering Air, supplemented by additional personnel as required.

13.1.2.3 Health Physics Group. The basic responsibilities of the Health Physics Group are the same for power testing as for low-power testing, but their activities are expanded because of the increased radiation levels throughout much of the plant and the increased amounts of radioactive waste.

13.1.2.4 Plant Engineering. The Plant Engineering staff consists of 11 people, supplemented by outside craft labor as necessary.

Allis-Chalmers personnel are principally responsible for reactor plant equipment, and Dairyland personnel for generator plant equipment. The two groups cooperate in the performance of all mechanical maintenance work.

13.1.3 Routine Plant Operation

The Dairyland Organization Chart for routine plant operation is shown in Fig. 13.3. Job descriptions for the key personnel are given in App. D.

13.1.4 Administrative Procedures

The reactor and its associated systems have been designed to reduce the possibility of component failures and to minimize the consequences of such failures or of operating errors. However, the operating staff is ultimately responsible for operating the plant safely and efficiently.

Administrative controls to further reduce the possibility of human error are detailed as rules and procedures in three basic control documents, discussed below in order of priority.

(1) LACBWR Technical Specifications

The Technical Specifications list significant design and operating procedures which must be adhered to in the absence of specific authorization from the Atomic Energy Commission. They represent the parameters that define the boundaries of licensed activity which the Commission has evaluated and approved from a safety standpoint.

(2) LACBWR Health Physics Manual

The Health Physics Manual is concerned principally with the radiological safety of all personnel on the site or in its vicinity. This manual describes in detail the following:

- (a) the responsibilities and duties of all personnel as related to radiological safety
- (b) the allowable exposure limits for both normal and emergency conditions
- (c) the procedures and forms to be used in maintaining a radiation exposure history for all personnel entering potentially radioactive areas
- (d) the criteria to be used in determining the radiation areas and the posting requirements for these areas
- (e) the radiation monitoring instrumentation
- (f) the procedures for radiation and contamination control
- (g) decontamination procedures
- (h) the procedures and limitations governing disposal, storage, and shipment of radioactive material.

The procedures and rules described in this manual must conform to the Technical Specifications and to all applicable federal and state regulations.

Changes to the Health Physics Manual must be reviewed and approved by the Health Physicist, the Operations Supervisor, the Project Operations Manager and the La Crosse Reactor Safety Committee.

(3) LACBWR Operating Manual

The LACBWR Operating Manual contains detailed operating procedures for all reactor components and systems, and procedures for integrated plant operation. These procedures cover both routine operation and foreseeable plant emergencies, including detailed emergency procedures to be followed by Allis-Chalmers and Dairyland personnel in the event that specific evacuation situations occur which are attributable to nuclear startup, testing programs, or commercial operations. (Also see Sec. 15.)

The procedures must conform to the Technical Specifications, to all applicable federal and state regulations, and to the rules established in the Health Physics Manual.

Revisions to the Operating Manual must be reviewed and approved by the Operations Supervisor, the Health Physicist, the Project Operations Manager and the Safety Committee.

In addition to these basic documents, special procedures will be written for initial startup and testing and for any future tests. These test procedures must conform to the Technical Specifications and to the rules established in the Health Physics Manual. Further, the test procedures shall not disregard any warnings or precautions in the LACBWR Operating Manual.

All test procedures shall be reviewed and approved by the Operations Supervisor, the Health Physicist, the Project Operations Manager, and the Safety Committee. In every case each member of the operating staff shall be familiar with the written procedures for which he has partial or complete responsibility.

At least one copy of the latest revisions of the LACBWR Technical Specifications, the LACBWR Health Physics Manual, and the LACBWR Operating Manual will be kept in the main control room, the Shift Supervisor's office, and in the plant technical library at all times.

The day before any special test is to be performed, a copy of the approved test procedure will be placed in the control room and remain there until the test has been completed. Copies of the test procedures will also be supplied to the supervisors responsible for performing the test, and placed on file in the plant general office prior to the test.

13.2 TRAINING

13.2.1 Initial Startup and Routine Operation

Each member of the A-C group for initial startup and operation has had operating experience on another reactor. Each is thoroughly familiar with LACBWR by knowledge of its design, by helping to prepare the Operating Manual, or by an informal training program. Each operations group member is required to pass comprehensive in-house examinations in reactor fundamentals, health physics, and the LACBWR systems. The crew is to acquire further experience by preparing and conducting the preoperational tests.

Dairyland Power personnel who assist in the initial startup and operation of LACBWR will eventually assume responsibility for routine operation of the plant. All of these personnel have prior reactor operating experience and, in addition, will complete a three-phase training program prior to assisting in the operation of LACBWR.

The first phase of the training program is conducted by Dairyland Power and covers basic mathematics, thermodynamics, and the various turbine plant systems. The second phase is a 12-week academic program conducted by Allis-Chalmers personnel. This program consists of courses in reactor fundamentals, reactor instrumentation and control, mechanical design, process systems, reactor operations, health physics, and reactor hazards. The third phase of the formal training program is a temporary assignment to an operating reactor.

Dairyland operators will obtain additional experience and familiarity with LACBWR by observing and assisting with the preoperational and nuclear tests.

13.2.2 Continuing and Replacement Training

One of the responsibilities of the Plant Superintendent and the Assistant Plant Superintendent is the continued competence of the operating staff.

Each operator is required to pass an annual written examination, prepared under the direction of the Assistant Plant Superintendent and approved by the Safety Committee. These examinations are concerned principally with reactor operations and administrative and emergency procedures.

New or replacement personnel receive on-the-job training, combined with an assigned study program. Scheduled consultation periods with technical and operations supervisors determine the progress of the trainee and resolve any questions.

13.2.3 Coordination with Local Officials

Under the direction of the Plant Superintendent, the Health and Safety Engineer arranges with local and state authorities to present informational and educational lectures to public assistance groups which might be called upon if there were an emergency at the site requiring their assistance.

13.3 PROCEDURES FOR INITIAL PLANT STARTUP

13.3.1 Preoperational Testing

Prior to fuel loading, the various systems and components will be thoroughly tested, in accordance with detailed written procedures, to assure that they operate as intended.

Except for certain special tests, the tests will be performed by utility personnel working under the direct supervision of the Allis-Chalmers staff. The planned order of test performance may vary, depending on test priority and availability of key personnel and specific equipment, etc. Where possible, no equipment will be tested until all auxiliary components or systems required for the operation of that component have been completely checked out and tested.

The primary breakdown of the preoperational tests is by systems. Within a system, the tests are further broken down into various component and integrated system tests. The number of subtests depends on the complexity of the system, the components to be checked, and the interrelationship among the various systems.

The systems to be tested and the scope of the various subtests are as follows:

(1) Power Distribution

The power distribution tests consist of specific tests to assure that the various transformers, switchgear, motor control centers, and emergency power sources operate as required. All regulation and protection devices will be checked to assure that the control settings have been adjusted as required.

(2) Instrument and Control Air Distribution

Control air is supplied to reactor auxiliaries from the generator plant. Tests will be conducted to assure that properly regulated air is supplied to all air operated valves, liquid level bubblers, etc.

(3) Demineralized Water Distribution

Demineralized water to the containment building and the waste treatment building is supplied from the generator plant demineralized water system. Tests will be conducted to assure that an adequate supply is available at all required locations.

(4) Liquid Waste

Tests will be conducted to assure that the components and portions of the liquid waste system in the containment building operate as required for the collection and disposal of spills, overflows, and drains. Further tests will be conducted to assure that the means for transferring liquid waste and depleted resins to the liquid waste treatment buildings are adequate. The operation of the liquid waste demineralizer, the liquid waste evaporator, and the equipment and procedures for packaging and handling solid and concentrated wastes will also be tested.

(5) Overhead Storage Tank

The overhead storage tank will be filled and the makeup control and level instrumentation will be tested. The building spray nozzles will be tested with compressed air.

(6) Component-Cooling Water System

The component-cooling system instrumentation will be checked and calibrated. The system will be filled and operated to assure that the required flow can be supplied to each of the components cooled by the component cooling system. The alarms and controls will also be checked for proper operation.

(7) Heating, Air Conditioning, and Ventilation System

The containment building heating, air conditioning, and ventilation systems will be tested, and the waste treatment building heating and ventilation will be checked. The ventilation tests will include operation of the stack blowers and balancing of flows in the turbine building and in the containment and waste treatment buildings.

Actual performance of heating and air conditioning, in terms of capacity, will be proven during the course of the first year of operation, under different weather conditions and under various heat loads from the reactor primary systems.

(8) Storage-Well Cooling System

The storage-well cooling instrumentation will be checked and calibrated. The system will be filled and operated in each of the modes for which it was designed, including those of flooding and of return to the overhead storage tank. This testing will also prove the adequacy of the pool liner and canal gate. The system alarms and controls will be checked for proper operation.

(9) Shield-Cooling System

The shield-cooling system instrumentation will be checked and calibrated. The system will be filled and operated. The various cooling loops will be balanced to assure that proper flow can be established in each. The system alarms and controls will be checked for proper operation.

(10) Condensate-Demineralizer System

The condensate-demineralizer system instrumentation will be checked and calibrated. System operation will be checked by circulating water from the condenser hotwell through the demineralizer and back to the hotwell, using the condensate pumps. The condensate-demineralizer regeneration system will be tested to determine performance and to determine the amount of water used.

(11) Emergency Core Spray

The emergency core spray system instrumentation will be calibrated and checked. System operation will be tested to assure that the controls function as required and that the required flow is delivered to the emergency core spray nozzles.

(12) Hydraulic Accumulator System (for rotovalves)

The hydraulic accumulator system for the rotovalves will be tested and the valve opening times adjusted.

(13) Feedwater System

The feedwater system test will be accomplished as the primary system is being filled. Each of the components of the three-element feedwater control system will be tested and calibrated. An overall system check will be made, using test input signals, while the feedwater pumps are being operated to fill the primary system.

(14) Decay-Heat System

The decay-heat system instrumentation will be checked and calibrated. The system will be operated to assure that the design requirements have been met.

(15) Primary Purification System

The primary purification system instrumentation will be checked and calibrated. The system will be operated to assure that all components operate properly. When the resin has been sufficiently exhausted to warrant replacement, it will be sluiced to the waste treatment building, and the ion exchangers will be recharged by the normal operating procedures.

(16) Boron-Injection System

The system instrumentation will be checked and calibrated, the sodium pentaborate tank filled with demineralized water, and the system operated to assure that all components perform as required.

(17) Shutdown Condenser

The shutdown condenser system instrumentation will be checked and calibrated. The system valves and controls will be operated to assure that they function as required.

(18) Seal-Injection System

The system instrumentation and controls will be tested and calibrated. The system will be filled and operated to make final adjustment to the controls. All interlocks and trips will be checked.

(19) Forced-Circulation-Pump Auxiliaries

Instruments will be checked and calibrated and the bearing oil and the hydraulic coupling water systems of each of the forced-circulation pumps will be operated to assure that they function properly.

(20) Forced-Circulation-Pump Loops

The system instrumentation will be checked and calibrated. The loops will then be operated individually and data taken to permit calibration of the elbow tap flow meters and the proper settings for the limit switches on the hydraulic coupling scoop tube actuators. These adjustments will be made with the pumps shut down. Additional tests will then be made to determine both steady-state and transient characteristics with dual pump operation.

(21) Hot-Primary System Tests

The reactor vessel head will be installed. The system will be heated to between 250 and 300 F using the building steam connection to the decay heat cooler. The forced-circulation pumps will be operated while the temperature is in this range and sufficient data will be taken to verify the predicted effects of primary water density decrease. The pumps will also be operated at the maximum speed possible without exceeding the motor rating, to determine at what point, if any, cavitation occurs while saturated water is pumped.

(22) Vent and Gaseous Waste

The system instrumentation will be checked and calibrated. The operation of the various valves, the compressor, and the recombiner heater will be tested using manual controls and simulated alarm signals. The compressor will be operated until the off-gas storage tanks are at 300 psig. The compressed gas will then be discharged to the stack, using the remote-controlled storage tank discharge valve.

(23) Nuclear Instrumentation System

Each component of each of the eight nuclear channels will be tested for proper operation. All interlocks, alarms, and control switches will be checked and all necessary calibrations carried out.

(24) Reactor Safety System

Each component of each safety system channel will be calibrated and tested. Simulated input signals or actual tripping of the sensing device will be used to test all alarm, scram, and process output actions.

(25) Rod-Control System

Static tests will be made, without actually moving the control rods, to assure that the various interlocks and bypasses operate as required. Operational testing of the rod-control system will be combined with the control-rod drive testing.

(26) Control-Rod Drives

The control-rod drive oil and gas charging systems will be thoroughly checked and tested. Each drive will be tested in the rod test mode to assure proper operation and to determine scram time.

The entire bank will be withdrawn and partial or full scrams initiated by real or simulated signals from each scram-causing device. When this has been completed, the scram time for each drive will again be checked.

The rods will also be withdrawn and then inserted with the all-rod-insert push-button.

(27) Fuel Handling System

With the canal plug and canal gate removed, the reactor vessel will be flooded to the vessel flange. The failed-fuel element location bundle will then be removed to the storage well and the upper vessel cavity and storage well will be flooded.

The remote handling tools will be used to remove and reinstall a control rod, a shroud can, a simulated fuel element, and a simulated source. All operations will be done as if the components were highly radioactive. When testing of the handling tools has been completed, the vessel and storage well will be drained to the level of the upper flange, and the failed-fuel location bundle will be reinstalled.

(28) Radiation-Monitoring System

Each of the fixed and portable radiation monitors and survey meters will be tested and calibrated. All sampling equipment, such as pumps, blowers, and flowmeters will be checked for proper operation.

(29) Reactor Pressure-Control System

Each component of the pressure control system will be tested and calibrated. Test signals will be used to verify that the speed of the forced-circulation pumps varies as required.

(30) Main-Steam Bypass Valve

Each component of the main steam bypass valve controls will be tested and calibrated. The valve operation will be checked and the opening time measured. Pressure signals will be simulated to assure that the overall system, including the valve, operates as intended.

(31) In-Core Flux-Monitoring System

All monitoring components, including electronic components and the drives, will be tested and calibrated. The drive motor limit switches will be adjusted as required.

(32) Failed-Fuel-Element Location System

All electronic components, including the detectors, will be tested and calibrated as required. The wire drives on the electrostatic precipitators will be checked for proper operation.

13.3.2 Initial Reactor Criticality Tests

The preoperational testing program for LACBWR will be completed and the shroud cans will be installed prior to fuel loading and initial reactor critical testing. Detailed procedures for each individual startup program test will be prepared and submitted to the LACBWR Safety Committee for review. After Safety Committee approval, the reactor will be loaded, tested, and operated under the supervision of Allis-Chalmers AEC licensed personnel responsible for the testing and operation of the nuclear plant.

A 5-curie Pu-Be source is used until just prior to power operation; then an Sb-Be source is used. All temporary in-core instrumentation and necessary fuel loading equipment is checked for safe and operable condition. The in-core instrumentation consists of two BF₃ neutron detectors and one compensated ionization chamber. All three channels are read in the control room and are connected into the regular scram circuits to provide period protection and flux level protection. Additional indicating equipment for one BF₃ channel is in the containment building as close as possible to the loading bridge. The in-core instrumentation and the additional indicating equipment are used until the Sb-Be source is loaded, and then they are removed.

During initial loading to critical mass, all required plant systems are operated according to the procedures in the LACBWR Operations Manual, except when modified by the individual test procedures. Fuel is added in increments to the flooded reactor (water level is approximately 1 ft below the vessel flange) until a critical mass is obtained. During the loading of any fuel element, the control rod in the center of the previously loaded fuel configuration is in "cocked safety" position and ready to scram, while other control rods are fully inserted. Background neutron counts are taken prior to the installation of the Pu-Be source; neutron counts are then taken after the source installation with control rods inserted and withdrawn, and during, and after, each fuel addition. Reciprocal source multiplication curves are plotted and extrapolated to predict the size of the critical core. The initial loading increment consists of four fuel elements. Subsequent additions will be limited to half the number predicted for criticality by the 1/M curves (counting fractional increments as a whole element), but will not exceed three elements. If the three in-core channels do not coincide in determination of the required number of

additional elements for criticality, the lowest (most conservative) predicted number governs the size of the next increment. The following rod configurations are used to obtain neutron countrates after each fuel increment has been added to the core during the approach to critical mass:

- (1) most reactive rod out, all others in,
- (2) all rods out, and
- (3) most reactive rod in, all other rods fully withdrawn.

The reactor shutdown safety margin with the most reactive rod out and all others in, shall never be less than 0.5 percent $\Delta k/k$. The values of $1/M$ obtained with all rods out are extrapolated to determine the amount of fuel to be added in the next increment. Rod-worth information is thus provided before criticality is achieved. After each loading increment, the outermost set of four equidistant rods is withdrawn, in clockwise sequence, until all rods except the center rod are fully withdrawn. The excess reactivity of the first critical core is only a fraction of the worth of a fuel element. Since the worth of the center rod is greater than the excess reactivity of the initial critical core, criticality is achieved during the withdrawal of the center rod, after all other rods are fully out.

The reliability of the control rods and control rod system is checked prior to fuel loading. Rod scram time is checked to meet the design requirement of 2.5 sec.

Nuclear instrumentation reliability is checked at the start of each fuel loading shift. Sufficient data are taken to ensure no changes in channel operating characteristics.

The loading crew is familiar with fuel loading equipment and procedures. The procedures show the order of element loading. The fuel is lowered into the core slowly and the fuel handling tool is not disengaged from the element until the Operations Supervisor in the control room so orders. This precaution allows immediate element withdrawal if the countrate is not satisfactory. A Health Physicist in the fuel loading area ensures radiological safety.

The boron injection system is ready as a backup if there is control-rod failure. The system can inject enough boron solution to reduce reactor power from hot full power to hot subcritical in one minute (Sec. 5.2.5.1).

Containment building access is limited to the personnel required to load fuel or take data. The shift supervisor keeps check on the number and authority of persons in the containment building. No personnel are allowed in the containment building when a control rod is being withdrawn during initial testing with the reactor vessel head removed.

Core Loading Procedures. After criticality has been established and the initial core excess reactivity determined, a few more fuel elements may be added for a better core

geometry and to provide sufficient reactivity for preliminary rod calibration. The criteria for fuel increments are:

(1) The number of fuel elements added in any increment is not more than half the number of additional elements that would reduce the shutdown safety margin of the reactor to 0.5 percent $\Delta k/k$ with the most reactive rod fully withdrawn.

(2) The number of fuel elements added in any increment will not add more than 3 percent reactivity to the core. The "cocked safety" concept is used.

The worth of the central control rod is measured against the bank position, and the excess reactivity of the core is determined (bank refers to control rods surrounded by fuel). Outer rods not surrounded by fuel are ineffective and will be fully withdrawn during rod calibration.

As fuel loading proceeds, rods are calibrated as required to measure the excess reactivity. The fuel elements are loaded incrementally, with the cocked safety concept, as during previous loading operations. Countrates are obtained, with the center rod fully withdrawn, as a function of the number of elements in the core. The subcriticality of the core with the most reactive rod fully withdrawn and the amount of reactivity added to the core by each increment are determined to assure the safety of an additional increment.

All applicable safety measures for initial loading to critical mass remain in effect. As fuel is added, the in-core detectors and/or the Pu-Be source must be relocated. This operation is done with all control rods fully inserted. The shutdown countrate and the ion chamber current are measured before and after relocation of a detector or the Pu-Be source. No more than one detector is relocated after a loading increment.

The shroud cans will consist of a mixture of zirconium and stainless steel. Fuel loading will proceed until the core configuration, which is composed of 72 elements, is reached. The shutdown safety margin (>0.5 percent $\Delta k/k$) will be checked at each fuel-loading step. After the core is fully loaded, shroud cans will be changed as required to obtain a startup core configuration.

After the core is fully loaded, the critical position of the 28-rod bank, as well as the subcritical counts with the 28-rod bank fully inserted and the central rod fully withdrawn, is determined. The subcritical counts are also determined for withdrawn rods representative of all other positions in the symmetrical core.

The temperature coefficient of the final core is measured from ambient temperature to approximately 170 F. A negative temperature coefficient is expected for this core between ambient and design temperature, but if a positive coefficient is found, the shutdown safety margin criteria shall be satisfied at all times.

13.3.3 Zero and Low Power Physics Tests

13.3.3.1 Core Excess Reactivity. The final determination of core excess reactivity is made by adding boron to the reactor water. The boron concentration is increased in steps, and the control rods are calibrated between steps. The excess reactivity is the total worth of the 29-rod bank from the critical position (with no boron) to the full-out position. The calculated excess reactivity of the probable startup configuration of 28 stainless-steel and 44 zirconium shroud cans is 15.8 percent $\Delta k/k$. The most important criterion for the final core configuration is the shutdown safety margin.

13.3.3.2 Neutron Flux Distribution. During excess reactivity measurements, the core neutron flux distribution is measured, using different boron concentrations, for control-rod configurations typical of those encountered during initial reactor power operation. An initial calibration of the detectors in the instrument tubes is also obtained from these measurements.

13.3.3.3 Installation of Sb-Be Source and Vessel Closure. Prior to any further nuclear testing, the reactor Sb-Be source is loaded into one of the eight empty fuel locations, and the Pu-Be source removed. A zirconium shroud can is used for the Sb-Be source; the other seven are of stainless steel. The temporary in-core detectors are relocated in their normal operating positions. The reactor vessel head is installed, and the required tension is placed on the studs.

13.3.3.4 Temperature Coefficient. The temperature coefficient is measured between ambient temperature and ~ 577 F. The system is heated to ~ 300 F by the decay heat system. After the maximum temperature attainable with the decay heat system has been reached, reactor power is increased until a satisfactory heating rate is established. During heatup, the center rod is gradually moved and its critical position is determined as a function of temperature. At approximately 577 F, the reactor power is controlled to maintain a constant system temperature, in order to obtain an approximate value of the system heat losses.

13.3.4 Tests at Increasing Power

13.3.4.1 Power Calibration. The reactor power is raised and held in increments of ~ 20 percent of full power, and the power level is determined from thermal measurements taken at steady-state conditions. Correction factors for the nuclear channel readings are determined and applied up to an estimated power level of 80 to 90 percent, after which the nuclear channels will be calibrated to read the calculated reactor thermal power. During the initial increase to an unexplored power level, the startup program test procedures require at least the minimum specified flow circulation and reactor pressure for that power level. The control rods are not operated in a bank position but in a

control-rod program finalized prior to initial power operation. Access to the containment building is limited to personnel required to record data or to service equipment.

13.3.4.2 Shielding Survey. At approximately 10 percent and 100 percent power, the biological shield and the reactor auxiliary systems in service are monitored for gamma and neutron radiation levels. The radiation levels are compared to the predicted design values, and additional shielding will be added if required.

13.3.4.3 Power Reactivity. The amount of reactivity in voids and Doppler is determined at each power increment by comparing the critical rod configuration with that for the previous power level. There is a reactivity difference of approximately 2 percent between the hot, clean zero-void condition and the 100 percent power, 15 percent average voids condition. (See footnote to Table 4-2.)

13.3.4.4 Xenon Reactivity. The reactivity associated with equilibrium xenon for 20 and 100 percent power is determined by operating until near equilibrium conditions exist and by comparing the critical rod configuration with that for no xenon in the core. As xenon builds to a peak and then decays following a shutdown, the resultant reactivity changes are also measured.

13.3.4.5 Water Level Calibration. At each power increment, the reactor water gauge glass, the level recorder in the control room, and the two-phase interface level are intercalibrated with in-vessel probes. True-level vs. indicated level curves are established for several power levels for normal power operation. During this test, the reactor steam exit quality is measured as a function of water level.

13.3.4.6 Stability Margin. The stability of the reactor at 24, 60, and 100 percent power is determined during the initial power escalation program. Fluctuations in reactor power are analyzed to monitor any oscillation tendencies. Stability at higher power levels is predicted before reactor power is increased.

13.3.4.7 Response to Fluid Dynamics Effects. The transient effects at various power levels are investigated for the following:

- (1) changes in forced-circulation flow
- (2) changes in feedwater flow
- (3) changes in feedwater temperature
- (4) changes in steam pressure
- (5) sudden opening of the emergency shutdown condenser valves
- (6) sudden opening of the main steam bypass valve
- (7) sudden closure of the reactor building steam isolation valve.

No reactor limiting conditions (see Sec. 4.6) or any other plant design limitations are exceeded during these tests. Observations during these tests will determine if any instabilities exist.

13.3.4.8 Response to Partial Scram. Partial-scram reactor shutdown and the amount of time required to recover the plant without exceeding design limitations is tested at various power levels. The negative reactivity inserted by the partial scram should be sufficient to at least cancel out the positive reactivity introduced by loss of core voids and reduction of fuel temperature.

13.3.5 Full Power Tests

13.3.5.1 Initial Full Power Run. At 100 percent power, sufficient data are obtained for a final heat balance of the entire plant. Final results are compared with design expectations to ensure that the plant meets contractual obligations.

13.3.5.2 Plant Response to Load Changes. Plant response to various electrical load rate changes are determined for both manual and automatic operation. The load change rates include normal and emergency rates for the turbo-generator between 60 and 100 percent power. Previous tests involving changes in forced-circulation flow, feedwater flow and temperature, and reactor pressure help in predicting plant behavior during load changes. Transient effects that exceed any plant design limitation are not allowed to occur.

13.4 OPERATING PROCEDURES

A summary of the normal operating procedures for the plant follows.

13.4.1 Cold Startup

Prior to withdrawing control rods for the approach to power, the various system and subsystem prestartup check sheets are completed and signed by the shift supervisor. The rod drop test is performed if this is the first approach to criticality for that month.

The initial conditions prior to startup are as follows:

- (1) Forced-circulation pumps are operating at ~625 rpm (18,000 gpm).
- (2) Reactor building steam isolation valve and its bypass are closed.
- (3) Turbine building steam isolation valve is open.
- (4) Turbine stop valve is closed.
- (5) Control valve in steam line to reheater is closed.
- (6) Turbine condenser cooling water pumps are operating.

- (7) One condensate pump is operating while the other is on automatic standby.
- (8) Feedwater pumps are in standby.
- (9) Turbine is rolling on the turning gear.
- (10) All scram accumulators are fully charged and their respective bypass switches are in the normal position.
- (11) The following scrams are bypassed:
 - (a) main condenser vacuum low. (This bypass also permits the reactor building steam isolation valve to be opened with a low vacuum.)
 - (b) reactor building steam isolation valve not fully open. (This bypass also prevents automatic opening of the shutdown condenser when the isolation valve is not open.)
 - (c) turbine stop valve not fully open.

Conditions for reactor start and rod withdraw permits (Sec. 8.5.2.3) are satisfied and reactor startup proceeds as follows:

- (1) "Reactor Start" pushbutton is pressed.
- (2) A base countrate for source range channels Nos. 1 and 2 is obtained with the scaler, and recorded.
- (3) Rod withdrawal is begun as specified on the rod programming sheet while observing the period and countrate indicators for source range channels Nos. 1 and 2. In accordance with the rod programming plan described in Sec. 4.5, the reactor is normally operated with certain rods fully inserted. Before a cold approach to criticality after significant operation at power, a rod programming sheet will be prepared, and approved by the assistant superintendent. The sheet indicates which rods are to remain in the core and which rods are to be withdrawn, and gives an estimate of the position of withdrawn rods at the cold critical condition.
- (4) Starting with the rod of lowest number (nearest the center), each specified rod is withdrawn ~4 in. This procedure is repeated to keep the withdrawn rods at approximately the same level (within a 4 in. interval) throughout the approach to criticality.
- (5) When the countrate indicated on source range channels Nos. 1 and 2 has increased by a factor of ~10, the rods are checked individually to assure that they are operating properly. All rods but one are left in position, and that rod is withdrawn until a definite increase in countrate is noted. The rod is then returned to its former position, and the procedure repeated with each rod until it is confirmed that all rods and drives are operable. The approach to criticality then continues.

(6) When criticality is reached, the withdrawn rods are adjusted until they are within a 1.0-in. vertical interval of each other with the reactor still critical. These positions are recorded.

(7) One of the rods is withdrawn enough to place the reactor on a period of ~ 30 sec, and the power is allowed to increase. At approximately 10^5 cps in the source range channels, but not before other nuclear channels are indicating neutron level, source range high voltage is turned off. At the time, the source range period interlocks are bypassed. At a higher power level (to be specified in the LACBWR Operating Manual) the two Log-N period scram interlocks will be bypassed.

(8) After reaching a power level of ~ 6 Mwt, the primary system will heat up at ~ 50 F/hr. The actual limits on heatup rate are based on the thermal stresses in the reactor vessel. The reactor operator observes the reactor vessel wall temperature recorder in the control room and controls the heatup rate so not to exceed the following limits.

(a) temperature increase rate of 100 F/hr for any of the 24 thermocouples on the vessel wall,

(b) temperature difference between adjacent thermocouples on the vessel wall as specified in Fig. 13.4.

(9) As the Water temperature increases, the control rods are withdrawn to overcome the negative temperature coefficient and to maintain the desired heating rate. The rods withdrawn are always kept within a 1.0-in. vertical interval of each other.

(10) During heatup, excess water in the reactor vessel is dumped into the main condenser via the decay heat system.

(11) When the reactor pressure reaches ~ 550 psig, turbine plant warmup is begun. First the bypass around the reactor building steam isolation valve is opened, then the isolation valve itself.

(12) When steam is available in the turbine building the gland steam generator is started.

(13) The vacuum pump is started. Controls are set to obtain 5 to 8 in. of vacuum.

(14) Rod withdrawal is continued as required to maintain reactor vessel heating.

(15) The setpoint on the wide-range main steam bypass valve controller is adjusted to a pressure above reactor pressure, and turned to automatic.

(16) One feedwater pump is started at low speed, but the feedwater control valve is left closed.

(17) The setpoint on the wide range main steam bypass valve controller is lowered until the valve begins to open. The control rods are adjusted as necessary to hold the reactor pressure, and the feedwater control valve is opened as required to keep a constant water level in the vessel. The control rods and the feedwater control valve are adjusted as required for a steam flow of $\sim 50,000$ lb/hr. The limits shown on the allowable steam flow vs. reactor pressure curve (Fig. 13.5) are not to be exceeded. During subsequent warmup of the turbine, reactor heatup is continued. The setpoint on the wide range main steam bypass controller is changed in small increments to higher pressures. Control rods are adjusted as necessary to maintain constant steam flow. The rate of increase in pressure is such that the reactor vessel temperatures stay within design limitations.

(18) The turbine IPR system is put in operation, and its control setpoint is adjusted to a value below the existing pressure so that the output actuator will move to its maximum steam flow (top position).

(19) With load limit in "Valves Closed" position, the stop valve is cycled open and closed with the stop valve control button to warm the steam chest. The main steam bypass valve moves automatically to reduce the amplitude of reactor pressure changes.

(20) Steam chest warmup is performed on a "lapsed time with steam contact" method. When warmup has been completed (20-30 min.) the stop valve is left open.

(21) Steam seal in-leakage produces some turbine cylinder heating. If necessary, additional steam is admitted through governing valves. Flow is controlled so that the unit stays on turning gear, and the load limit is used to crack the valves.

(22) After preliminary warm-up, governing valves are closed with the load limit control. Reactor pressure should be ~ 900 psig when turbine cylinder warmup is completed.

(23) Vacuum is increased to 20-22 in.

(24) The turbine speed changer is adjusted to the 3600 rpm zero load setting.

(25) Holding the reactor power constant, the governing valves are opened gradually by means of the load limit, to take the turbine off turning gear. As the steam flow to the turbine is increased, the turbine bypass valve automatically closes slightly to keep pressure constant.

(26) Open steam to reheater valve slowly, immediately after the turbine is off the turning gear.

(27) The steam jet air ejector is put in operation and adjusted for maximum vacuum. The vacuum pump is shut off.

(28) Turbine speed is increased slowly to 600 rpm and held there while it is determined that the turbine is operating properly. The operator will listen for any rubbing sounds and check supervisory instruments for indications of any abnormalities.

(29) The turbine is accelerated at a uniform rate, first by increasing vacuum to maximum value, then by increasing governing valve opening. Turbine metal temperature rise and vibrations or noise related to rubbing determine the time of acceleration to 3600 rpm.

(30) When the vacuum reaches the maximum value, the bypass on main condenser vacuum low-scam is removed.

(31) The bypass on reactor building steam isolation valve position scam is removed.

(32) The pressure setpoint on the main steam bypass valve is continuously raised, and the control rods are adjusted as required to continue vessel heatup.

(33) When the turbine reaches ~3450 rpm, operation of the load limit does not increase speed and can be set at 50 percent. The speed changer is then used to control acceleration.

(34) As the turbine is accelerated, the control rods are withdrawn to increase the steam flow to approximately 100×10^3 lb/hr. The reactor vessel three-element feedwater controller may be put in operation.

(35) When the turbine speed reaches 3600 rpm, the excitation on the generator field is brought to the normal no load value and system voltage is matched. The synchroscope, electrical controls, and meters are used to match line voltage, frequency, and phase.

(36) The 69-kv oil circuit breaker is closed to connect the generator to the line, and the load is immediately raised to 3 - 6 Mwe ($45 - 90 \times 10^3$ lb/hr). As

the load is increased, the reactor power remains constant. The main steam bypass valve automatically closes enough to keep the pressure constant.

(37) The bypass key on the turbine stop valve position scram is removed.

(38) Turbine load is increased by manual control of the speed changer, as limited by the rate of turbine metal temperature increase and by any vibration. The setpoint on the main steam bypass valve control is raised until all the steam is flowing to the turbine. The control rods are adjusted as required for reactor vessel heatup limits. The electrical load will then be ~ 10 Mwe and the vessel pressure ~ 1200 psig.

(39) The wide-range main steam bypass valve controller is put on manual. The narrow-range controller on the valve is already set to open upon overpressure. (If reactor vessel heatup has been slow and the pressure is ≤ 1100 psig, the turbine is placed in IPR control to increase the pressure.)

(40) When the reactor is at or above 1100 psig, load increase continues by manual increase of speed changer control, within the allowable loading rates. At the same time, the control rods are adjusted as required to increase reactor pressure until the turbine inlet pressure reaches 1250 psig.

(41) Turbine loading continues with the speed changer control (the load limit is raised as required). The reactor control rods are adjusted as required to maintain the turbine inlet pressure at 1250 psig.

(42) As indicated in Fig. 8.9, reactor power should not exceed 125 Mwt without an increase in forced-circulation flow.

(43) Normal procedure is to place the reactor in automatic pressure control at 125 Mwt.

13.4.2 Normal Power Operation

During normal operation, the Dairyland dispatcher will specify the power level at which the unit is to be operated. He will also specify the changes in load required by the system and, normally, the rate at which these changes are to be made.

The turbine operator meets the load requirements by using the speed changer to adjust the turbine power. If the reactor is in the automatic pressure-control mode, the forced-circulation pumps automatically speed up or slow down to change reactor power as required to maintain constant steam pressure at the turbine inlet.

If the reactor is in the manual mode, the reactor operator may adjust either the speed of the forced-circulation pumps or the position of the control rods to control the power level. The normal practice will be to vary the pump speed, since this method avoids the possibility of an accidental power-flow scram, which would occur if the operator increased the reactor power without maintaining proper flow.

Occasionally some control-rod movement will be necessary to compensate for burnup and to keep the reactor operating on or near the normal operating curve, shown in Fig. 8.9. As described in Sec. 4.5.1, this interim shimming will normally be accomplished by moving the center control rod. At intervals of three to five months, depending on the power levels at which the reactor is operated, certain rods will be interchanged in accordance with the control-rod program.

These interchanges will be made in accordance with detailed written procedures which assure that:

- (1) the control rods are attached to the drives
- (2) no rod configuration occurs that results in any rod being worth more than 2.6 percent $\Delta k/k$ (see Sec. 14.2)
- (3) no power distributions occur that would cause the allowable burnout margin to be exceeded. (If necessary, rod interchange will be scheduled to coincide with periods of light load operation.)

The principal additional duties of the operators, during normal power operation, consist of routine monitoring and checking of plant equipment.

13.4.3 Normal Shutdown

Normally the reactor is brought from power operation to cold subcritical as follows:

- (1) The dispatcher is advised of a planned shutdown.
- (2) The turbine operator reduces load at the scheduled rate, using the turbine speed changer control. The reactor follows the change in either automatic or manual control. Power is reduced by reducing pump speed or by a combination of control rod insertion and reduction of pump speed to below 80 percent of full speed.
- (3) When electrical power reaches ~ 30 Mw, the automatic pressure control system no longer functions and reactor power is reduced by control-rod insertion.

(4) The turbine operator continues to reduce load. The reactor operator then inserts the control rods at a rate that causes reactor pressure to decrease. The pressure is decreased at a rate that enables the turbine operator to control the rate of electrical power reduction. The reactor operator also watches that reactor vessel temperatures are within cooldown limitations. The turbine operator watches turbine temperatures and differential expansion, and alters the rate of load change or if necessary requests the reactor operator to alter the rate of steam temperature change. He also observes exhaust hood temperatures to confirm automatic operation of exhaust sprays if the temperature rises.

(5) As pressure is decreased, the setpoint on the main steam bypass controller is reduced to remain ~50 psi above the pressure at the turbine stop valve.

(6) When steam pressure at the turbine stop valve is ~1100 psig, the interlock that closes the reactor building steam isolation valve upon low pressure at the turbine stop valve is bypassed.

(7) Power decrease is continued while turbine and reactor pressure is lowered. The reactor power for a given pressure must be kept within the limits shown in Fig. 13.5.

(8) Depending on the rate at which turbine load can be decreased, reactor pressure is ~700 - 800 psig after load reduction to 5 Mwe.

(9) The setpoint on the main steam bypass controller is slowly reduced until the bypass valve begins to open, then is slightly increased so the bypass valve is just closed.

(10) Since the turbine should not operate for extended periods at low power, electrical load is dropped from the 5 Mw in 3 to 5 minutes.

(11) As the turbine governor valves close, the reactor operator inserts the control rods, using the ALL ROD INSERT pushbutton if necessary. The main steam bypass valve opens to relieve any pressure rise and then begins to close as the rods are driven fully in.

As soon as the neutron level is within the range of the source range channels, the high voltage is turned on and the two source range period scram interlocks are placed "in circuit"; keys are removed.

(12) When the turbine governor valves are closed, the turbine operator trips the turbine and sees that the generator oil circuit breaker trips.

(13) The operator ensures that the motor auxiliary oil pump cuts in when the relay oil pressure from the main shaft driven pump drops to ~160 psig because of turbine speed decrease. At 1800 rpm the field is taken off.

(14) With the turbine stop valve closed and all control rods full in, the operator takes the appropriate actions to continue cooldown, depending on the core decay heat. The decay heat depends on the time and power levels of previous operation, and the rate of shutdown.

(15) If there is considerable decay heat, the operator gradually decreases the setpoint on the wide range main steam bypass valve controller. The valve opens, allowing steam to pass to the main condenser. If the makeup rate is low, it may be necessary to control the water level in the reactor vessel with the feedwater control valve. The cooling rate and temperature differentials in the reactor vessel must be within the specified limits at all times.

(16) When the turbine spindles reach full stop, as indicated by the zero speed indicator light, the turning gear is engaged and put in mesh. The gear motor is started. The turning gear ammeter is observed for a possible rise in turning gear power.

(17) When the reactor pressure falls below 500 psig, the main steam bypass valve is placed in manual control.

(18) If the reactor vessel water cools too rapidly, the desired cooling rate is obtained by the following:

(a) Main steam bypass valve is closed.

(b) Reactor building steam isolation valve is closed, stopping steam flow to the air ejector and the gland steam generator.

(c) Seal-water flow to the control-rod drives is reduced.

(19) If the upper part of the vessel cools too slowly, the water level in the vessel may be raised to provide more uniform cooldown rates. Before the level is raised above the steam nozzles, the reactor plant steam isolation valve must be closed. Vessel heat losses and any cold water added to raise the level contribute to the cooldown.

(20) When the cooling rate begins to slow, the decay heat system is started. The cooling rate is adjusted by control of flow through the decay heat cooler and the decay heat cooler bypass. One forced-circulation pump can also be stopped.

(21) When primary water temperature reaches ~ 150 F, the second forced-circulation pump is stopped.

(22) Cooldown is continued until the reactor water is < 120 F. The decay heat system valves are adjusted as required to maintain the desired reactor temperature.

13.4.4 Scram Shutdown

Conditions that cause a partial or full scram are given in Table 8-2.

In the event of a scram, the reactor operator confirms that 13 rods go to the fully inserted position for a partial scram, or that all rods go to the fully inserted position for a full scram. The operator then ensures that:

- (1) nuclear power is decreasing
- (2) reactor pressure is not increasing or rapidly decreasing. (The main steam bypass valve or the shutdown condenser may have opened automatically to prevent over-pressure.)
- (3) water level in the reactor vessel is within normal operating limits
- (4) forced-circulation pumps are at or below 80 percent speed

The operator then determines the cause of the scram. In cases where transient conditions or a sequence of safety actions cause more than one scram signal, the original cause of scram registers as "first out" on the scram annunciator panel. The first scram signal can be distinguished from following signals by the appearance of a red light on the annunciator panel.

When the reactor scrams, the turbine operator checks to see that the house load automatically switches to the auxiliary breaker. If the source of scram is the reactor or reactor auxiliaries, the turbine does not trip. However, a turbine trip will always scram the reactor.

After a scram that does not trip the turbine, the pressure at the turbine throttle valve begins to decrease. The turbine initial pressure regulator begins closing the turbine governor valves to prevent sudden reactor pressure decrease. The rate of generator load drop depends on the amount of decay heat in the core. When electrical power drops to ~ 5 Mwe, the turbine is manually tripped, which opens the 69-kv oil circuit breaker.

If a turbine trip occurs simultaneously with reactor scram, the pressure at the turbine stop valve rises and the main steam bypass valve opens automatically.

After the cause of scram has been established, the reactor is either shut down completely and cooled, or restarted. If the reactor is cooled, the procedure is essentially the same as for normal shutdown (Sec. 13.4.3), except that the main steam bypass valve is used from the start of pressure reduction.

If feasible, the scram condition is corrected with the reactor hot and pressurized, and the reactor is prepared for restart as follows:

- (1) The bypass keys are removed from the intermediate channel period scrams.
- (2) The control-rod drive accumulator gas-pressure-low scram is bypassed.
- (3) The control-rod drive accumulator oil-level-low scram is bypassed.
- (4) The turbine stop valve-not-fully-open scram is bypassed.
- (5) The following scrams, if signaled after the original scram is initiated, are also bypassed.
 - (a) reactor building steam isolation valve not fully open.
 - (b) main condenser vacuum low.

The scram relays can then be reset. As soon as the partial scram is reset, the scram solenoid valves will reset, allowing oil to flow from the charging pumps to the individual control-rod drive accumulators. Approximately 5 min. is required to recharge the accumulators. As the last accumulator returns to normal, the annunciator bell sounds, and the following three windows on the warning annunciator of Panel D begin flashing:

- (1) "Gas Pressure Warning Control Rods"
- (2) "Control Rods Oil Failure Warning"
- (3) "Control Rod Oil Pump Charging"

These signals indicate that the control-rod-drive accumulators are completely recharged.

(6) The control-rod-drive accumulator oil-low scram bypass and the control-rod-drive accumulator gas-low scram bypass are removed, and the keys are withdrawn. This clears the rod withdraw-permit circuit.

(7) The startup-channels high voltage is turned on when the neutron level is within range of the startup channels. Reactor startup then proceeds.

- (8) The "Reactor Start" pushbutton is pressed.
- (9) Rod withdrawal is begun, in accordance with the "Rod Programming" sheet. (Sec. 13.4.1).
- (10) When criticality is approached, a check is performed to determine that each control rod being withdrawn is attached to its drive. One at a time, each rod is withdrawn a sufficient amount for a definite increase in count rate and is then returned to its previous position. After this check, the programmed rods are withdrawn until sufficient power is generated to increase the reactor pressure. The high voltage to the startup channels is turned off at $\sim 10^5$ counts/sec, but not before other nuclear channels begin to indicate neutron level.
- (11) The setpoint on the main steam bypass valve is reduced and the control rods are adjusted until approximately 100,000 lb/hr of steam flows to the condenser.
- (12) The turbine load control is set to the "Valves Closed" position.
- (13) The turbine stop valve is opened. If the steam chest has cooled appreciably, the stop valve is cycled between the open and closed positions to warm the steam chest to within 100 F of the main steam temperature.
- (14) The scram bypass for turbine stop valve not fully open is removed, and any other scram previously bypassed is also removed.
- (15) If the turbine has been placed in turning gear, governing valves are gradually opened by means of the load limit control, to take the turbine off the turning gear. As the steam flow to the turbine increases, the bypass valve automatically closes.
- (16) If the unit has been on the turning gear for some time, there may be some temperature distribution change in the cylinder and support structure. If there is such a change, the turbine is rolled at 600 rpm for 10 to 15 min to equalize cylinder temperatures before bringing the turbine to speed.
- (17) The turbine is then returned to full speed in approximately 20 min. During this interval, the reactor pressure is increased to normal operating pressure by control-rod withdrawal and by raising the setpoint on the main steam bypass controller. The rate of pressure increase is such that vessel temperature limitations are not exceeded.
- (18) The generator field excitation is brought to the normal no-load value, and system voltage is matched. The generator is synchronized with the system.
- (19) Load increase is continued by manual increase of the speed changer, until the bypass valve is closed. Loading rate is governed by the thermal and differential expansion limits.

(20) The turbine is then loaded fairly rapidly and stabilized when the electrical load corresponding to the cylinder expansion measurements is reached. During this procedure, the reactor operator withdraws control rods and, if necessary, increases pump speed.

(21) When design pressures have been reached, and when the load has been stabilized, the reactor operator makes a final adjustment of control rods and pump speed to return to the normal power-flow curve. He then bypasses the period scrams on the intermediate channels and returns to normal operating procedures.

13.5 MAINTENANCE PROCEDURES

13.5.1 Major Maintenance Tasks

The major maintenance operations for the plant are turbine-generator overhaul, reactor core refueling, and leak testing of the containment building. All three are scheduled as far in advance as possible, to minimize down-time. Turbine overhaul, normally an annual operation, is scheduled in conjunction with either core refueling or containment building leak testing. (Core refueling and leak testing must be done separately.) Of the major maintenance tasks, only core refueling and turbine overhaul could result in increased radiation exposure of the personnel involved. No work is allowed to proceed until detailed procedures have been prepared, and approved by the Operations Supervisor (Assistant Plant Superintendent) and the Radiological Physicist.

A preventive maintenance schedule for plant equipment is to be prepared in accordance with manufacturers' instructions, and it will be modified as indicated by operating experience. This program includes a schedule for exercising normally idle components. Records of each component's history include the extent and type of repair work, any non-routine maintenance, and all preventive maintenance. If possible, maintenance of containment building equipment is performed without relocating the equipment. Whenever equipment must be removed from the containment building for repairs, permission is required from the Health Physics Department. If decontamination of the removed component is necessary before repair, the component is taken to the decontamination room inside the Waste Treatment Building. Decontamination is done under the supervision of Health Physics personnel.

During normal plant operation, every shift is required to take readings of plant parameters and conduct inspection tours of plant equipment. If peculiarities in component operation are noted, the Shift Supervisor evaluates the situation and takes corrective measures to protect the equipment. Where duplicate equipment is provided or where the component is not essential to safe plant operation, the Shift Supervisor's decision does not require complete plant shutdown if the equipment must be removed from service. The Shift Supervisor, however, retains the authority to shut down the plant if necessary. His decision always gives priority to protection of the public, plant personnel, and property,

and at no time are these considerations subordinated to the testing program or the generation of electric power. The Shift Supervisor informs the Operations Supervisor (Assistant Plant Superintendent) of his decision as soon as possible whenever a major plant component or safety system is involved.

13.5.2 Maintenance of Safeguards Systems and Instruments

All safety systems are maintained in operating condition once initial core loading starts. Before each startup after a 24-hr shutdown, the nuclear instrumentation system and the safety system are tested for proper operation. In addition, both systems can be tested periodically, and certain components can be replaced with spare units without causing a reactor scram during plant operation. The monitors of the radiation monitoring system are recalibrated at least annually, a one-point check being performed more frequently.

Preventive maintenance on all components is performed according to manufacturers' recommendations. Radiation monitors may be repaired during plant operation as long as there is backup monitoring (either from a similar monitor or from intermittent sampling).

The Operations Group keeps instrument calibration and maintenance records for the nuclear and safety systems. The Health Physics Department is responsible for the radiation monitoring system.

13.5.3 Protection of Maintenance Personnel

The Project Operations Manager (Plant Superintendent) has primary responsibility for the protection of all site personnel. The Operations Group, in cooperation with the Health Physics Department, ensures that all radiation exposures are at a minimum within the acceptable limits.

The guides and procedures established by the LACBWR Health Physics Manual and 10 CFR 20 are observed during all maintenance operations. Additional procedures covering specific maintenance operations are issued as required. Self-reading dosimeters and film badges are worn by all personnel in radiation areas. Work in high radiation areas, airborne contamination areas, and other designated areas require a radiation work permit (RWP). The RWP assures that the persons involved are adequately informed of the hazards of the particular job, and that all precautionary measures for the job have been taken.

Supervisors see that all personnel under their supervision know, understand, and comply with the regulations in the Health Physics Manual. In addition, shift supervisors keep informed of all current maintenance work and see that no radiation safety rule is violated.

13.6 MATERIAL SURVEILLANCE PROGRAM

13.6.1 Reactor Vessel Material Irradiation

The program that has been proposed will indicate the radiation damage to the reactor vessel during its useful life.

The bombardment of ferritic steels by fast neutrons (energy >1 Mev) increases the ultimate strength, yield strength and hardness, while decreasing the ductility. This irradiation also affects the ductile behavior of such materials by raising the ductile-brittle transition temperature so that less energy is required to fracture notched impact specimens. Such specimens will be used to analyze the ductile-brittle transition temperature of the LACBWR reactor vessel material (ASTM A-302, Gr. B) as a function of vessel life.

Prior to reactor startup, capsules containing ASTM A-302, Gr. B reactor vessel material will be installed in the reactor. The capsules will be mounted about the inner periphery of the internal thermal shield on a 93.5-in. dia centered about the midpoint of the core vertical axis. Each capsule contains all the necessary vessel material test specimens for the planned evaluation. Capsules will be removed from the reactor at appropriate intervals.

Since the capsules are mounted closer to the core than to any portion of the vessel, the specimens will receive a larger radiation dose than will the vessel for a corresponding time. The specimens will therefore furnish data that may be used to predict vessel radiation damage and, hence vessel integrity and useful life. Provision will be made for determining the integrated fast neutron dose received by the test specimens, so that radiation damage can be correlated with exposure.

The specimens contained in each capsule consist of material of the same heat as the vessel material and have been subjected to the same heat treatment. Specimens will be removed periodically to determine the increase in nil-ductility transition temperature (NDT) resulting from neutron irradiation.

13.6.2 Primary System Corrosion

Provisions have been made to allow insertion of corrosion test coupons into the forced-circulation loops. Two decontamination flange fittings in each loop are designed to allow insertion of test samples. The blind flange caps on the decontamination fittings have fixtures for holding test coupons.

13.7 REFUELING PROCEDURES

13.7.1 New Fuel

All new fuel is delivered to the Allis-Chalmers accountability representative (or his authorized alternate) at the LACBWR site, who supervises its transfer to the new fuel storage area on the reactor building main floor.

The packing cases are then opened and the contents checked against shipping form AEC-101. Any fuel assemblies that do not meet specifications or that are not as described on the shipping forms, or that appear to have been tampered with or to have any other deficiency, are repacked and held for disposition instructions from the Chicago Operations Office and the fuel manufacturer. When a fuel assembly inspection is complete the element is placed in one of the three new fuel storage racks in the biological shield, at the main floor level. There is space for 84 new fuel assemblies (28 per rack). As each assembly is placed in one of the new fuel storage racks, the accountability representative keeps a record of the assembly number, the new fuel storage position number, the net weight of source and special material, and the date. After the fuel assemblies are placed in the storage racks, the accountability representative takes inventory and compares the actual total with that shown in his records. If there are any discrepancies, the storage racks and shipping containers are rechecked until all records agree. The storage racks are then locked, and only opened by the accountability representative or his authorized alternate.

All shipping forms, vendor's packing lists, or other incidental shipping documents and records are filed together for final accounting.

13.7.2 Refueling

Refueling is conducted under the direction of the Assistant Plant Superintendent. Detailed refueling procedures are prepared in advance by the Operations Group with the assistance of the Assistant Plant Superintendent or the Process Engineer. No refueling operations are done until the procedures have been approved by the Assistant Plant Superintendent and the Health and Safety Engineer.

A member of the Health Physics Department is present during the entire refueling operation. The guides and procedures established by the LACBWR Health Physics Manual and 10 CFR 20 are observed at all times.

During the transfer of the spent assemblies from the core to the fuel element storage well racks, the reactor upper cavity, fuel transfer canal, and storage well are flooded. At no time shall less than 10 ft of water be above the fuel assemblies. Prior to flooding, valve check lists are used to ensure that all valves are as required for refueling. Any

valve is to be tagged if its accidental opening could lower the water level in the storage well or the reactor upper cavity.

Either the accountability representative or his authorized delegate is at the fuel loading area during fuel assembly transfer to maintain the required records. The records include the core position of each element prior to removal, and its new position in the storage well racks. All records are filed for final accounting. Any discrepancies in the records must be resolved before the vessel head is bolted on.

13.7.3 Spent Fuel

The spent fuel storage rack capacity is 84 fuel assemblies. The shortest distance between any two racks is 13 in. It is physically impossible to place a fuel assembly in the 13 in. space between the racks. After refueling operations are completed, the storage well water level is lowered, and maintained at ~el 679 ft-11 in. during the spent fuel cooling period prior to shipment. Before the irradiated assemblies are loaded into the shipping cask, the water level is raised as required to ensure a minimum of 10 ft of water above the assemblies during the transfer.

All operations are conducted according to detailed written procedures approved by the Assistant Plant Superintendent and the Health and Safety Engineer. Compliance with the LACBWR Health Physics Manual and 10 CFR 20 is enforced at all times. In addition, the shipping cask carrying irradiated fuel assemblies conforms to all ICC regulations before shipment from the site.

ORGANIZATION CHART, PREOPERATIONAL AND LOW POWER TESTING

FIG. 13.1

ORGANIZATION CHART, POWER TESTING

FIG. 13.2

ORGANIZATION CHART, DAIRYLAND POWER COOPERATIVE LACBWR PLANT

FIG. 13.3

LACBWR PRESSURE VESSEL THERMOCOUPLE LOCATIONS
AND MAXIMUM ALLOWABLE TEMPERATURE DIFFERENCES

FIG. 13.4

ALLOWABLE STEAM FLOW AND REACTOR POWER AT REDUCED PRESSURE

FIG. 13.5

14. SAFEGUARDS EVALUATION

This section summarizes the design features that help ensure safe operation of the reactor, analyzes the inherent safety of the reactor, and discusses and analyzes the effects of possible accidents, including the maximum credible accident. This section revises and expands the preliminary evaluation presented in ACNP-62574.

14.1 DESIGN SAFEGUARDS

14.1.1 General

Safe power production is the primary consideration in the design and operation of LACBWR. The plant conforms to all applicable standard codes, and has been designed to minimize the likelihood of accidents and to limit the consequences of accidents that could result from credible abnormal conditions. Although accidents requiring a secondary pressure containment are of very low probability, the reactor plant is housed in a pressure vessel that will withstand the most serious accident credible, as a final safety consideration to ensure protection of the general public.

Previous sections of this report describe in detail the design features incorporated in the plant and the plant equipment to assure a high degree of operational reliability and overall safety; the administrative controls to assure responsible and efficient operation of the plant and to protect against human error that might be adverse to plant safety have also been discussed.

A review of the final plant design and the proposed operation with regard to generally accepted safety criteria has indicated that the plant can be operated as proposed without undue hazard to the health and safety of the public.

14.1.2 Safety Systems and Actions

The plant safety system protects against conditions that might damage the reactor or endanger plant personnel or the general public. This function is carried out in part by the scram and alarm circuitry, which ensures that:

(1) conditions which could become hazardous if not corrected actuate an alarm signal, and

(2) conditions which are an immediate hazard cause a partial or full scram of the reactor.

A complete list of LACBWR scrams is given in Table 8-2, along with the safety actions that are initiated at the scram setpoints.

Alarms that are essential to plant safety give visual and audible annunciation in the control room. These are shown in Fig. 8.1 and, except for the generator plant alarms, are listed in Sec. 11. Other alarms are strategically located in the plant to facilitate plant operation. After an alarm signal, the operator may correct annunciated conditions during operation or, if necessary, shut down the reactor manually.

The shutdown condenser (Secs. 5.2.2 and 11.6) acts as an alternate heat sink for the reactor and is capable of condensing all steam generated immediately after shutdown. The shutdown condenser system is automatically started when the reactor is isolated from the main condenser (i.e., when either the reactor building or the turbine building steam isolation valve is not fully open) or when the primary system pressure is abnormally high (1325 psig, with a backup signal at 1350 psig).

The liquid-poison injection system (Secs. 5.2.5 and 11.8) backs up the rod control system and can admit a solution of sodium pentaborate into the reactor vessel in the event of control-rod malfunction. This solution contains enough boron to make the reactor subcritical in the cold clean condition with all control rods fully out. An alarm indicates if any rods fail to insert fully upon a full scram signal. If needed to complete the shutdown, the boron injection system can be actuated.

When the core is in danger of losing coolant (low level), at least 50 gpm of water are automatically pumped to the core through spray nozzles located above each fuel assembly (see Emergency Core Spray System, Secs. 5.2.6 and 11.9). The emergency cooling water is supplied from the overhead storage tank, with high pressure service water serving as a backup. A containment building spray system (Sec. 6.7) is also provided to condense vapors accidentally released from the primary system and to wash out, at least partially, released halogens and solid fission products from the containment atmosphere. The system is manually operated from either the control room or inside the containment building. It delivers water by gravity flow from the overhead storage tank to reduce containment building pressure caused by release of primary-system energy.

In the event containment isolation is necessary (e.g., following a major accident), the reactor building steam isolation valve is closed automatically to isolate the primary system within the reactor building (Sec. 5.1.1). Low-steam pressure at the turbine stop valve or low reactor water level (both indicative of a major system rupture) close the isolation valve, which in turn scrams the reactor and initiates shutdown condenser operation. (This same sequence of actions occurs upon low main condenser vacuum, which would indicate loss of the main heat sink.) The containment building ventilation system inlet and exhaust ducts are also closed (Sec. 6.6.4) by signals that would occur in the event of a major accident, and an automatic bypass permits recirculation of building air through the filters and exhaust blower. The ventilation system valves close on:

(1) high activity measured by any of the containment building air exhaust gaseous and particulate monitors (Sec. 8.8.1.4),

(2) high containment building pressure, and

(3) high reactor pressure (Sec. 8.5.2.5.2).

Signals from (2) and (3) above also cause closure of the control valve in the 4-in. vent header from the containment building. The signal from (3) ensures that the containment building is isolated before the main steam safety valves open.

Operational procedures and automatic actions are designed to prevent the uncontrolled release of radioactivity from the liquid, gas, or solid waste disposal systems. Solids are packaged in drums for shipment to an authorized disposal site (Sec. 11.13.1.5), and all liquid and gaseous waste is monitored before being released from the plant (Secs. 8.8.1.1 through 8.8.1.3).

Three safety valves on the 10-in. steam line within the containment building prevent excessive pressurization of the primary system and thus prevent component damage. The valves provide a final backup to other automatic pressure limiting actions initiated by high system pressure (Sec. 5.1.1.1), including opening of the main steam turbine bypass valve, partial and full scrams, and shutdown condenser operation. Each safety valve releases steam through separate vertical vents which discharge at an elevation of approximately 698 ft toward a steel plate located on the bottom of the main floor. The locations of the vents are such that it is unlikely that anyone will be in contact with very hot steam. The platform that is provided for the failed-fuel-element location system valving station (elevation 684 ft) is in a potentially dangerous area, but even if steam release occurred while a person were on the platform, he would not be hit directly with the hot steam. The temperature in the containment building after steam release has been calculated to be 135 F, and as such, is not hazardous. Blowing of the safety valves would cause radiation doses to persons in the containment building. Section 14.2.5 of ACNP-62574 discusses the gamma dose received during a 15-min period from Nitrogen-16 (70 mrem) and the dose from inhalation of radioiodines (100 mrem). Radioiodines released to the containment building were based on the presence of 44 defective fuel rods (producing 1 Mw of power) in the core at the time of safety valve opening. The calculated doses are conservative, since in ACNP-62574 it was assumed that the safety valves would reseal at a pressure not greater than 7 percent below the opening pressure. They will actually reseal at 4 percent below the opening pressure, and thus less steam would be released to the containment building than was previously assumed. The doses are well below the allowable quarterly exposures given in 10 CFR 20.

In addition to the basic safety actions summarized above, numerous safeguards are incorporated in the design of individual systems to ensure safe and proper integrated operation of

the entire plant and to protect individual equipment components from damage. These safeguards include various interlocks to assure safe operation of valves, pumps, and control rods; fail-safe position of valves on loss of power or control air; backup pumps; emergency power supply; pressure relief valves for individual equipment components; and redundancy and reliability incorporated in the design of the instrumentation and control system. In all cases, safety considerations have received priority over operating or other considerations. Detailed descriptions of design features for individual systems are discussed in Secs. 5 and 8, and operating principles and associated safeguards are discussed in Sec. 11.

14.1.3 Limitations on Reactivity Insertion

The rod-control system design (Sec. 8.5.2.3), in conjunction with the fuel management and control-rod program (Sec. 4.5), precludes the insertion of reactivity in amounts that would damage the core. Only one control rod may be withdrawn from the core at a time. The limits on the rate of rod withdrawal are such that, even if a rod is accidentally withdrawn, there is little probability of core melting (Sec. 14.3.1). Interlocks are provided to prevent an increase in the forced-circulation pump speed or the opening of a pump discharge valve while a control rod is being withdrawn, and to prevent control-rod withdrawal while a signal for pump speed increase exists. In addition, the maximum worth of a control rod in planned operating patterns is not sufficient to cause fuel damage in the event of a rod-drop accident (Sec. 14.3.3).

The automatic plant control system is designed to maintain a constant system pressure, so that pressure-related reactivity increases will not normally occur. The reactor is scrammed in the event of abnormally high primary system pressure. In addition, conditions which could lead to high pressure (including closure of the reactor building or turbine building main steam isolation valves, closure of the turbine stop valve, or loss of main condenser vacuum) cause a scram.

The feedwater control system regulates the unvoided water level in the reactor vessel and tends to maintain a feedwater flow rate equal to steam flow rate (Sec. 5.1.1.2.2). This provision stabilizes the nonboiling height in the core, thus minimizing reactivity insertion caused by loss of steam voids.

Reactivity increase caused by a rapid increase in coolant flow rate is prevented both by design features and by administrative control. The rate of increase in recirculation pump speed is limited by the forced-circulation pump speed control system and is such that no serious consequences can result from uncontrolled pump speed increases.

Operational procedures (Sec. 11.1.1) and interlocks (Sec. 5.1.2.5) enforce proper recirculation pump startup. A pump cannot be started unless its discharge valve is closed, and the discharge valve cannot be subsequently opened while there is a high water temperature difference between the two forced-circulation loops. This provision prevents insertion of

cold water (and hence reactivity) from a shutdown loop into the reactor (Sec. 14.3.2.1). Operational procedures also minimize the chance of cold water being inserted into the reactor from the shutdown condenser (Sec. 11.6.1) or from the decay-heat cooling line (Sec. 11.5.1). The tube side of the shutdown condenser is drained periodically to prevent the buildup of cold water, and the decay heat cooling line is kept at reactor temperature when not in use by a backflow of reactor water through the system.

14.2 INHERENT SAFETY

14.2.1 Reactor Stability

A general discussion of stability in boiling water reactors, including oscillations in coolant flow rate, in power level, and in neutron flux distributions, is given in Sec. 14.1.2.3 of ACNP-62574. The same section presents a table (Table 14-2*) comparing the stability factors of LACBWR and other boiling water reactors and indicates that the LACBWR design is well within general limits of stable operation. On the basis of power density and operating pressure, the stability limit of LACBWR is likely to be at least 175 percent of the design power level. However, since fuel-center melting is expected to be near 120 percent of the design power level, the power level instability factor is not the operating limitation.

The analog simulator has been used to verify LACBWR stability for various operating conditions. The frequency response characteristics of the open-loop transfer function were obtained by applying a sinusoidal reactivity input to the kinetics model and by measuring the feedback Doppler and void reactivities (refer to LACBWR simulator diagram, Fig. 1 of ACNP-64572, Amendment 6 to the Construction Application). A frequency generator was used to obtain the sinusoidal input reactivity, which for all cases had a peak-to-peak magnitude of 10¢ and which varied over a frequency range of 0.01 cps to 10 cps.

The gain-and-phase-shift vs. frequency plots for four separate operating conditions are presented and discussed in ACNP-65511 (Appendix C of Amendment 9 to the Construction Application). The four cases considered and the phase margins (180 deg plus phase shift at a gain of 0 decibels) and gain margins (negative of the gain for -180 deg phase shift) are given in Table 14-1.

Case I represents normal operation and Case II is the lower load limit for automatic load-following operation. Cases III and IV are for power levels corresponding to the burnout safety limit curve at the recirculation flow rates of Cases I and II (see Fig. 8.9).

It is generally considered that a phase margin of approximately 50 deg and a gain margin of 10 to 15 db provides a reasonable degree of stability.** Since LACBWR is well above

* Later calculations indicate the reactivity in steam voids should be 2.1 percent instead of the value of 4.2 percent given in this table.

** J. G. Gille, M. J. Pelegrin, P. Decauline, Feedback Control System, McGraw-Hill Publishing Co., p. 281

these limits for the conditions considered, stable operation is indicated and overpower limitations should be based on the burnout safety margin rather than on stability considerations.

TABLE 14-1

PHASE AND GAIN MARGINS FOR SELECTED
LACBWR OPERATING CONDITIONS

<u>case</u>	<u>reactor power level (% of full power)</u>	<u>recirculation flow rate (gpm)</u>	<u>phase margin (degrees)</u>	<u>gain margin (decibels)</u>
I	100	30,000	110	>25
II	60	13,200	110	26
III	120	30,000	108	26
IV	80	13,200	96	28

14.2.2 Response to Reactivity Additions

An analysis of the inherent shutdown tendency of LACBWR is presented in Sec. 14.1.2.2 of ACNP-62574. The analysis of reactor response to step and ramp insertions of reactivity has been modified to account for refined values of the effective delayed neutron fraction, the prompt neutron lifetime, the Doppler coefficient of reactivity, and the maximum full-power fuel temperature. The results of the final analysis are described in the following subsections.

14.2.2.1 Experimental Results and Applications to LACBWR. Two series of experiments have been performed to determine the effects of reactivity increases in heterogeneous, water-moderated cores. The initial series was in the Boiling Reactor Experimental Facility (BORAX) at the National Reactor Testing Station in Idaho. The second series was conducted as part of the AEC reactor safety program in the Special Power Excursion Reactor (SPERT).

Results from the BORAX-IV and SPERT-I cores, which utilized oxide fuel rods with a long thermal time constant, are especially significant for LACBWR. Other cores tested in the experiments contained fuel elements of the MTR type with short thermal time-constants and are therefore of limited applicability to LACBWR. However, test results on the MTR type of core which relate the general characteristics of reactor response to reactivity increases without specific dependence on any shutdown mechanism may be applied to LACBWR. Section 14.1.2.2.3 of ACNP-62574 discusses in some detail the portions of the BORAX and SPERT series of experiments which can be applied to an analysis of step and ramp reactivity insertions in LACBWR. It also discusses relevant analytical methods. In assessing the safety of LACBWR,

the response to increases in reactivity would be more similar to the response of the SPERT constrained core than to that of the unconstrained core. The unconstrained fuel rods had complete freedom to move radially, whereas the fuel rods of a LACBWR fuel element are constrained by spacers at three positions along the length. This constraint limits rod bowing and the possible increase in reactivity associated with it.

LACBWR, like other boiling water reactors, has a high degree of inherent stability with respect to increases in reactivity. Increases in reactor power are accompanied by increases in the amount of voids within the core and the temperature of the fuel. Since both these effects reduce reactivity, the reactor tends to shut itself down as power increases. Because of the fairly long time-constant of the uranium dioxide fuel, void formation is not an effective shutdown mechanism for very short transients. However, experience with BORAX-IV has shown that the inherent shutdown capability of boiling water reactors is independent of the fuel time-constant for reactivity increases approximately equal to or less than the prompt critical value. Results from the SPERT experiments on a low-enrichment oxide-fueled core show that void formation is a significant shutdown mechanism for reactor periods on the order of the fuel thermal time-constant, but becomes less important for reactor periods which are small compared to the thermal time-constant. Thus, for all but the largest ramp increases of reactivity and for step increases below that associated with prompt criticality (for which the period equals 180 msec), void formation is an important shutdown mechanism in LACBWR.

For step increases of reactivity above the prompt critical value, and for large ramp increases of reactivity, the fuel-temperature increase is an important shutdown mechanism. This has been demonstrated in the SPERT experiments on an oxide core, where it was shown that, for a range of initial periods down to 3.2 msec, excursions were terminated by the Doppler effect without core damage.

A study has been made of the inherent safety afforded by these two shutdown mechanisms in LACBWR. This study was for a range of initial reactor conditions and both step and ramp increases of reactivity. For the purpose of the study, inherent safety was defined as the ability to terminate the initial power burst of a transient when the external safety circuits were inoperative, without melting of the fuel in the region of highest neutron flux. External safety circuits were neglected during the initial transient because there are lags between the onset of a scram condition and the decrease in reactivity caused by rod insertion. However, before additional power bursts occurred there would be sufficient time for the scram circuits to trip and the rods to be inserted.

Three initial conditions were assumed: (1) a full-power reactor, (2) a zero-power reactor at operating pressure and saturation temperature, and (3) a zero-power reactor at room temperature and pressure. Damage to the fuel was considered the limiting criterion for reactivity insertion, even though limited fuel melting in itself does not necessarily represent fuel failure or a hazardous condition.

Damage to the fuel-element cladding was not considered, since for any short period transient it would not be a more limiting condition than center melting. The amount of heat transferred from fuel to cladding would be small because of the long thermal time-constant of the fuel. For short period transients, the fuel would melt under less severe conditions than would cause the cladding temperature to approach the melting point. Thus, the criterion of center melting used in the study of inherent safety is conservative.

Results obtained for maximum allowable step or ramp reactivity insertions have in general been derived using simplified methods of analysis and conservative assumptions. Effects such as the increase in the effective Doppler coefficient during adiabatic transients that are due to non-uniform power shape have been neglected. The reported values are useful in that they define the lower limits of permissible reactivity insertions, and these limits may be used to show that many of the reactivity-insertion type accidents discussed in Sec. 14.3 are not hazardous. However, in analyzing certain specific accidents, such as the control-rod drop accident (Sec. 14.3.3) or accidents where use has been made of the LACBWR analog simulator, refined methods of analysis and assumptions have been used (refer to Amendments 6 and 11 for these refinements). Therefore, the results in Secs. 14.2.2.2-14.2.2.4 represent only conservative bounds which are not applied to all the accidents analyzed.

14.2.2.2 Step Increases in LACBWR. The effects of step increases of reactivity are summarized in Table 14-2. The table shows, for each operating condition, the maximum reactivity increase which may be handled by either shutdown mechanism without damage. Reactivity compensation was assumed directly proportional to the energy produced during the excursion, and the power level was assumed symmetrical about the peak point. The equations used for reactivity compensation are given on pages 14-15 and 14-16 of ACNP-62574.

For excursions terminated by fuel-temperature increase, the permissible reactivity increase consists of the delayed neutron fraction plus the amount of prompt excess reactivity (i.e., reactivity above the prompt critical value) compensated at peak power. The prompt excess reactivity compensated by void formation is assumed to be negligible, because of the long thermal time-constant of the fuel. Permissible increases for shutdown by void formation are thus given as the delayed neutron fraction. The combined effects of shutdown by void formation and fuel-temperature increase have not been determined. However, the permissible reactivity would certainly be greater than for either mechanism alone.

It may be seen from Table 14-2 that a step increase of reactivity of less than 0.91 percent will not damage the core. If the reactivity of the operating reactor (at full power) were increased by this amount, the energy input to the fuel in the region of highest flux would be just enough to melt the fuel at the fuel-rod centerline. The total reactivity compensated during the transient would be equal to twice the increase in prompt excess reactivity. After the initial power transient, the reactor would still be supercritical by an amount equal to the delayed neutron fraction minus the initial prompt excess reactivity, or 0.59 percent Δk .

TABLE 14-2

EFFECTS OF STEP INCREASES OF REACTIVITY*

<u>initial conditions</u>		<u>shutdown mechanism</u>	<u>operating limits**</u>	
<u>power</u>	<u>moderator temp., F</u>		<u>reactivity, % $\Delta k/k$</u>	<u>period, sec</u>
full	577.5	fuel-temp rise	0.91	0.028
		void formation	0.75	0.180
zero	577.5	fuel-temp rise	1.42	0.007
		void formation	0.75	0.180
zero	68	fuel-temp rise	1.49	0.006
		void formation	0.75	0.180

* based on the following initial values:

delayed neutron fraction, $\beta = 0.75\%$

prompt neutron lifetime, $\ell = 5.0 \times 10^{-5}$ sec

power coefficient of reactivity (Sec. 4.4): at full power = $-2.0 \times 10^{-5} \Delta k/k/C$
 average value between zero and
 full power = $-2.2 \times 10^{-5} \Delta k/k/C$

** limit set by fuel melting

The reactor power would go through an initial peak and be reduced to somewhat below its original value. It would then start to rise again on a slow period because of the remaining 0.59 percent Δk of delayed excess reactivity. This rise would be compensated by void formation, and the reactor could be scrammed before there would be further core melting.

For the less severe cases of zero-power operation at 577.5 F and 68 F, respective step reactivity increases of at least 1.42 and 1.49 percent may be tolerated without fuel melting. After each of these excursions, the reactor would still be supercritical by an amount equal to the delayed neutron fraction minus the initial prompt excess reactivity. However, the period would be so long that sufficient time would be available for scram.

14.2.2.3 Ramp Increases in LACBWR at Operating Conditions. Ramp increases of reactivity in a core initially at saturation conditions are compensated by void formation. Unless the ramp insertions are very rapid, reactor power rises in equilibrium with void content. The reactor power under steady-state conditions is limited to slightly more than 120 percent of full power, beyond which fuel would begin to melt in the center of the fuel pin in the region of highest flux. For a rise in power of 20 percent from the full-power condition, the increase in void content would be approximately 4.0 percent and the increase in reactivity held in voids would be approximately 0.96 percent.* An additional 0.09 percent reactivity would be compensated by the increase in average fuel temperature of approximately 80 F caused by a 20 percent rise in power. Thus, a total ramp activity increase of 1.05 percent would not have serious consequences in the operating reactor.

14.2.2.4 Ramp Increases in LACBWR at Zero Power. The analysis of ramp increases of reactivity in a zero-power core was based on the SPERT results, which showed that the consequences of a ramp increase of reactivity with a minimum period of τ_m are the same as the consequences of a step increase of reactivity with a stable period τ_m .

Reactivity increase rates of up to $2.1 \times 10^{-2} \Delta k/k/\text{sec}$ are possible at zero power, operating pressure, and saturation temperature without damage to the fuel elements during the initial power transient. The minimum period reached during this transient would be 0.007 sec. In a zero-power core at room temperature, rates of reactivity increase up to $2.9 \times 10^{-2} \Delta k/k/\text{sec}$ --which result in a minimum period of 0.006 sec--would be possible. For both these cases, scram is necessary to prevent additional power bursts that would damage the core.

If it is assumed that the first power burst does not scram the reactor, the maximum rate of reactivity increase that will still avoid fuel-element damage is much lower. A conservative value for this maximum rate of increase can be based on the criterion that the power

* based on a void coefficient of $-2.4 \times 10^{-3} \Delta k/k/\% \text{ void}$.

level in the primary transient shall not exceed the steady-state power level that will avoid centerline melting. Melting would not occur during the initial transient. A reactivity increase rate of $5.0 \times 10^{-4} \Delta k/k/sec$ to the room-temperature core gives a maximum power of 196 Mwt (approximately 120 percent of full power) and a minimum period of 125 msec. An increase rate less than $5.0 \times 10^{-4} \Delta k/k/sec$ could therefore be tolerated even if scram did not occur because of the first power burst.

14.3 EVALUATION OF POSSIBLE ACCIDENTS

The accidents that could occur through equipment malfunction or operational error are analyzed qualitatively and evaluated in ACNP-62574. Several accidents have since been analyzed by use of the analog simulator described in ACNP-64572 (Amendment No. 6 to Application for Construction Authorization Docket No. 115-5). Results of these dynamic analyses are reported in detail in Amendments 7 and 9 to the construction application. All accident analyses are summarized in the following subsections, which supersede the preliminary analyses given in Sec. 14.2 of ACNP-62574.

14.3.1 Inadvertent Control-Rod Withdrawal

The rod control system is designed so that only one control rod may be withdrawn from the core at a time. The rod withdrawal rate is 20 in./min, and the maximum rate of reactivity increase corresponds to movement of the rod near its point of highest incremental worth. The consequences of inadvertent rod withdrawal during startup and at full power (with the reactor plant on manual or automatic control) are discussed below.

14.3.1.1 Startup Condition. The shortest period transient would result from withdrawal of a control rod with the reactor initially subcritical and at room temperature. Under these conditions, short periods are reached at low power levels before inherent limiting mechanisms become effective. The calculations were for the worst case--that of continuous rod withdrawal from shutdown with the period and high-flux scram circuits inoperative--even though rod withdrawal from the shutdown core is not possible unless these scram circuits are operative.

The shutdown power level is at least 1 w, because of source neutrons. The strong antimony-beryllium source installed in the reactor for startup remains in the reactor during operation.

In determining the minimum period and maximum power reached during the initial transient, the only reactivity compensation was assumed to be that due to the negative fuel temperature coefficient of reactivity. The maximum insertion rate of a control rod in a cold-clean (and just critical) core is $1.2 \times 10^{-3} \Delta k/k/sec$ (16¢/sec) if an abnormal control rod pattern were used, even though such a pattern would not be permitted for actual operation. As a control rod is withdrawn from the shutdown core, the reactor period decreases to a minimum of 0.07 sec, which will be reached 7 sec after criticality. Power will peak at 320 Mw

shortly thereafter due to the Doppler effect and will then decrease further as voids are created. The maximum peak fuel temperature would be about 1514 F. No fuel damage would occur.

14.3.1.2 Full Power Condition. The effects of a continuous reactivity insertion rate caused by accidental rod withdrawal have been analyzed using the analog simulator. The results are presented in Sec. 3.1 of Appendix C to Amendment 9 (ACNP-65517), and are based on an assumed reactivity insertion rate of 5¢/sec, with scram protection included. The assumed insertion rate is conservative relative to those associated with the maximum individual rod worths predicted for power operation (≤ 2 percent $\Delta k/k$). The rod configurations of high individual worths considered in Sec. 14.3.3.2 are not applicable to the full-power case because the peaking factors associated with such configurations would be intolerable in the case of full-power operation.

14.3.1.2.1 Reactor on Automatic Control. At full power operation, reactivity increase caused by rod withdrawal is compensated by void formation. If a control rod is withdrawn while the reactor is on automatic control, the speed of the forced-circulation pumps, and hence the coolant flow rate, decreases, which increases the void content in the core. The reactor power remains essentially constant during the initial part of the transient. Since, however, the recirculation flow rate continues to decrease, a reactor scram results when the power-flow relationship exceeds its permissible limit. No core damage occurs and the maximum primary system pressure rise is small. If the power-flow relationship failed to cause a scram, the reactor power would eventually begin to rise and would cause a scram at 115 percent of full power. The reactor pressure at this time would have risen to approximately 1315 psig. No core damage would result.

14.3.1.2.2 Reactor on Manual Control. If a rod is withdrawn while the reactor is on manual control, reactivity is compensated by the void formation and fuel temperature rise which result from the increase in reactor power level. As the rod is withdrawn the power level continues to rise, causing reactor scram at 115 percent of full power. The maximum power reached is below 120 percent of full power and, hence, no fuel melting occurs. The rise in primary system pressure is small. If scram failed to occur on high power level, backup would be provided by the power-flow relationship scram; however, in this case the centers of a few of the fuel rods might begin to melt.

14.3.2 Cold-Water Insertions

The introduction of cold water into the reactor vessel would increase reactivity because of

- (1) the moderator negative temperature coefficient of reactivity and, more importantly,
- (2) the decrease in average void content.

The five possible sources of cold-water insertion into the reactor that could cause a significant reactivity increase are:

- (1) the forced-circulation loops,
- (2) the return line from the shutdown condenser,
- (3) the feedwater line,
- (4) the liquid poison injection line, and
- (5) the emergency core cooling system.

The effects of cold water insertions from sources (1) through (3) above have been analyzed using the LACBWR analog simulator, and are discussed in detail in Secs. 3.4 and 3.5 of Appendix C to Amendment 9 (ACNP-65517).

14.3.2.1 Forced-Circulation Loop. Cold water could be introduced into the reactor vessel by starting a recirculation pump after a long shutdown for maintenance or repair. Normally, a second forced-circulation pumping loop is not put into operation when the reactor is operating at a significant power level with one forced-circulation loop. However, the possible adverse effects of starting a second loop have been investigated using the analog simulator.

Interlocks prevent a pump from starting unless its associated discharge valve is closed and prevent the discharge valve from opening unless the water temperatures of the two forced-circulation loops have equalized. Thus, it is unlikely that large amounts of cold water from a shutdown loop will be added to the reactor by operating the forced-circulation pump with its discharge valve open.

A bypass line around the discharge valve of each loop permits a small backflow through the loop and thus allows temperature equalization between loops. When a second pump is brought into operation, the pump is started and brought from zero to 40 percent of its full speed against a closed discharge valve, with the bypass line open. The pump discharge valve is then automatically opened, and the valve in the bypass line automatically closes, provided that loop temperatures have equalized. Thus, the maximum cold water addition from a cold recirculation loop protected by interlocks is from the cold water flow through the discharge valve bypass line when the associated pump is brought to speed.

The accident was analyzed assuming that:

- (1) room-temperature water is present in the shutdown loop,
- (2) full flow through the bypass line is established instantaneously.

(3) the associated pump is brought from zero to full speed before the temperature starts to equalize between loops,

(4) the initial reactor power level is equal to the highest possible value on the normal power-flow curve (Fig. 8.9),

(5) the reactor is under manual control, and

(6) protection afforded by the scram circuitry is neglected. These assumptions give a cold water injection rate of 25 gpm into the reactor under initial conditions of 77 percent of full power and 19,000 gpm recirculation flow.

The analog results indicate that an instantaneous reactivity insertion of $\sim 60\text{c}$ would result, with the reactor power level peaking sharply to 200 percent of full power and returning to a steady-state power level of 100 percent. Fuel centerline temperature and surface heat flux do not exceed the steady-state values corresponding to 100 percent power level, so core damage is unlikely. However, for the actual case with scram circuits operative, high flux-level and power-flow scrams will assure that core damage does not occur.

14.3.2.2 Condensate From the Shutdown Condenser. Normally closed valves are located in the inlet and discharge lines for the shutdown condenser. Any main steam leaking through these valves would condense within the tube side of the shutdown condenser and accumulate as cold water. On initial operation of the shutdown condenser, this water could be inserted into the reactor at a maximum rate of 62,000 lb/hr. A standard plant procedure is to drain the tube side of the shutdown condenser periodically to prevent the buildup of large amounts of water. Furthermore, reactor scram is always coincident with shutdown condenser operation, so the reactor would be subcritical before the cold water reached the core. Thus, the addition of cold condensate from the shutdown condenser would not normally represent any reactivity problem.

However, the addition of ambient temperature condensate at a rate of 62,000 lb/hr (~ 140 gpm) to the operating reactor was analyzed using the analog simulator. This addition would correspond to manual initiation of the shutdown condenser operation at a time when a substantial amount of cold water was present in the tube side of the condenser.

Results were obtained for reactor operation initially at both 60 and 100 percent of full power under manual control. For both cases, there are moderate initial transients, with return to steady-state power levels 10 percent higher than the initial power levels. Although scram signals would be generated by the initial power pulses (power-flow scram for the 60 percent case, and both power-flow and high-power-level scram for the 100 percent case), scram would not be necessary to prevent core damage.

14.3.2.3 Feedwater Line. A sudden drop in the temperature of feedwater to the recirculation system would cause core inlet subcooling to increase and thus result in a positive reactivity effect. Normally, feedwater temperature varies by approximately 20 F over the 60 to 100 percent power operating range, and this temperature change is gradual. If one of the three feedwater heaters were lost, a drop of approximately 70 F could result in feedwater temperature.

An investigation of the effect of loss of a feedwater heating stage was made using the analog simulator. Three operating conditions were analyzed: 100 percent power under both manual and automatic control, and 60 percent power under manual control. For each case a 70 F drop in feedwater temperature was assumed to occur within 5 sec.

Manual operation at 100 percent power was found to be the most severe case. Under this condition core excess reactivity increases to a maximum of 7¢ before being returned to zero by void formation. The reactor power level rises initially to approximately 114 percent of full power, and then gradually rises to a steady-state value of 116 percent. Even in the absence of a scram (scram would occur at 115 percent of full power) the maximum pressure reached is only 1322 psia. These results show that a 70 F drop in feedwater temperature may be safely accommodated without either excessive pressure or core damage.

14.3.2.4 Liquid-Poison Injection and Emergency Core Cooling. The maximum flow of cold water to the reactor vessel from either the liquid-poison system line or the emergency core spray system is approximately 100 gpm (~45,000 lb/hr) and is less than the flow caused by accidental opening of the shutdown condenser valves. Cold-water injection from either of these sources would not be hazardous.

14.3.3 Control-Rod Drop Accident

The control-rod drop accident is discussed in detail in Amendment No. 11 to Application for Construction Authorization Docket No. 115-5.

A control-rod-drop accident is postulated as follows: it is assumed that a control rod becomes separated from its drive mechanism. The drive mechanism is then moved to its fully withdrawn position, and, for some reason, the control rod remains fully inserted in the core. Subsequent jarring of the control rod causes it to fall freely from the core, thus inserting reactivity.

Following a control-rod dropout, the reactivity insertion will result in a transient power excursion leading to a transient heat generation in the fuel rods. The most pertinent data on the effects of large transient heat inputs to stainless-clad oxide fuel rods come

from a series of tests in the TREAT facility.* In these tests, an energy input of 240 cal/g of oxide was found to be the threshold input for fragmentation of the oxide fuel and dispersal into the water coolant, even though a portion of oxide fuel and cladding melted at lower energy inputs. On the basis of the TREAT tests, it is concluded that a reasonable threshold energy input for fuel fragmentation and dispersal in LACBWR is 230 cal/g, which corresponds to the TREAT energy threshold of 240 cal/g corrected for a higher initial fuel temperature. In order to maintain the burst energy to the maximum-power fuel rod at less than 230 cal/g, the increase in reactivity caused by a dropped control rod from a critical LACBWR core must not exceed 2.6 percent $\Delta k/k$. The effects of a control rod dropping from the critical core for both normal and abnormal rod patterns is discussed below, although the abnormal pattern could occur only if the planned rod program and administrative controls are totally disregarded.

14.3.3.1 Actual Control-Rod Patterns. For the planned control-rod patterns to be used in operation, the maximum worth of a control rod in a critical core is less than 2 percent $\Delta k/k$. This rod worth is well below the estimated minimum of 2.6 percent $\Delta k/k$ that might cause fuel damage. Thus, with the planned control-rod patterns, a rod-drop accident would not result in fuel damage or other hazards.

14.3.3.2 Abnormal Control Rod Patterns. A high rod worth for an individual control rod can occur when the reactor is critical with a very localized region unrodded. This hypothetical condition is more severe than any that will occur, because the reactor, by administrative control, will only be brought to criticality with a fairly uniform control-rod removal; not by withdrawing all the control rods of a local region.

To evaluate this abnormal condition, it was assumed that the reactor is brought to criticality by completely withdrawing a cluster of adjacent control rods and that the next rod in the cluster drops from the reactor. For example, in the cold, clean condition the reactor can be made critical by withdrawing two adjacent rods near the center of the core; a rod adjacent to these two was then removed and considered as the potential dropped rod. The reactivity insertion due to a dropped rod was determined by assuming the rod worth was equal to the average worth of the rods in the removed cluster.

* (1) Reactor Development Program Progress Report, ANL-6944, pp 82-84 (Sept. 1964).

(2) Liimatainen, R. C., Ivins, R. O., Deerwester, M. F., and Testa, F. J., Studies of Metal-Water Reactions at High Temperatures II. Treat Experiments Status Report on Results with Aluminum, Stainless Steel-304, Uranium, and Zircaloy 2, ANL-6250 (Jan. 1962).

The worth of the dropped rod was determined for three separate cases (Table 14-3): a cold-clean core, a hot zero-power core, and a hot core containing the average full power void percent. The hot zero-power case is the most severe, since it provided the highest rod worth (4.9 percent $\Delta k/k$) and the least effective Doppler compensation. It is also more severe than the hot-voided case since the burst energy for a given ramp-insertion rate is greater when starting from a lower initial-power level.

TABLE 14-3

REACTIVITY INSERTION BY A DROPPED CONTROL ROD

<u>core condition</u>	<u>reactivity insertion</u>
cold clean	3.3 percent
hot: zero power	4.9 percent
hot: average full-power void percent	4.2 percent

The transient during the rod drop was analyzed as a prompt ramp insertion of reactivity (effect of delayed neutrons was neglected), with the Doppler effect providing the only shutdown mechanism. Rod position vs. time was determined assuming that the rod falls freely from the core, with only a buoyancy retarding force. In analyzing the power burst behavior, the maximum ramp insertion rate (22.75 percent $\Delta k/k/\text{sec}$) that occurs during rod dropout was assumed to exist throughout the transient. The effects that a non-uniform power shape has in increasing the effective Doppler compensation for a given rise in average core temperature were accounted for.

Following control-rod dropout, a peak power level of 305,000 Mw is reached, and the total energy generation during the transient is 3115 Mw-sec. On the basis of an adiabatic heat addition (no heat loss) to the fuel elements and an assumed low surface-heat transfer coefficient, the cladding melting point is reached in more than half the fuel rods in the central region (which does not contain control rods) before the transient is terminated. The average temperature of the oxide fuel in the central core region reaches 4230 F, while the maximum temperature reaches 6900 F. Approximately 11 percent of the oxide in the core would be heated to the melting point, with only a little more than 2 percent being completely melted. The amount of fuel experiencing energy inputs in excess of 230 cal/g would be 1480 lb, or approximately 7 percent of the total fuel in the core. Thus, up to 7 percent of the total core fuel might then be released to the coolant. The TREAT stainless-clad oxide pins and the oxide core destructive testing in SPERT 1 indicate substantially different pressure effects on dispersal of hot or molten oxide in water than have been observed from the rapid meltdown of plate type U-Al alloy fuel elements. Explosive pressure increases have been observed for plate-type core

meltdown but were not observed in the TREAT or SPERT 1 destruct tests. Thus, while it cannot at this point be proven, it appears unlikely that this accident would cause a destructive pressure increase of sufficient magnitude to rupture the reactor vessel.

Even if vessel rupture did occur, however, the consequences of a rod-drop accident would be no more severe than those of the maximum credible accident (Sec. 14.3.18). Actually, less fission products would be released to the containment building than for the MCA, since initial meltdown would occur in a water-filled core and the water would retain a fraction of the halogen activity (for the MCA, meltdown is assumed to occur in a dry environment). If the entire energy of the transient, plus the energy content of the primary water and steam, is added to the containment building instantaneously and if no heat losses from the building or heat absorption by internal structures are accounted for, a peak containment pressure of 53.9 psig would result. This is only 2 psi above the containment building design pressure of 52 psig and is 6 psi below the containment building test pressure. Thus, containment integrity would be maintained.

14.3.4 Fuel Loading Accident

During fuel-assembly loading, the reactor period and flux scram circuitry are in operation. Of the 29 control rods, 28 will be fully inserted in the core, with the central rod withdrawn, or "cocked," so that it can be scrambled if necessary. The fully loaded reactor is at least 0.5 percent $\Delta k/k$ subcritical under these conditions, so that during the loading of even the last assembly, the amount of subcriticality is never less than 0.5 percent. If, despite this, it is assumed that the reactor is just critical when a fuel assembly is added, the reactivity worth of the cocked rod more than offsets that of the added fuel. Although a fuel-loading accident is therefore not considered credible, it is briefly analyzed.

If in the calculation of core reactivity vs. the number of added fuel assemblies it is assumed that the calculated reactivity values are lower than the actual values by 3 percent $\Delta k/k$, the reactor could be just critical after the addition of 40 fuel assemblies (central rod withdrawn). The reactivity worth of the next fuel assembly would then be approximately 0.4 percent $\Delta k/k$. Assuming the rapid insertion of 0.4 percent reactivity to a just critical core, it was shown in Sec. 14.2.2.2 that no fuel damage would occur. The transient would be limited by fuel temperature increase alone, even if a scram did not occur during the initial power burst. Thus, even under the above assumption, a fuel-loading accident would have no serious consequences.

14.3.5 Reactor Building Steam Isolation Valve Closure

If the reactor building steam isolation valve were accidentally closed, the reactor would be completely cut off from its major heat sink. However, as soon as the isolation valve moves from its fully open position, a full scram occurs and operation of the shutdown condenser system is automatically initiated. If the reactor fails to scram on signal from

isolation valve closure, a second scram signal is given when main steam pressure increases to 1325 psig. The analog simulator has been used to analyze system behavior for the following conditions:

- (1) isolation valve closure, with scram on start of closure,
- (2) isolation valve closure, with scram on high system pressure.

For each case initial operation at 100 percent of full power with manual plant control and constant recirculation flow was assumed. A detailed discussion of these calculations and the results is presented in Sec. 3.6 of Appendix C to Amendment 9 (ACNP-65517).

14.3.5.1 Scram on Start of Closure. The closure time of the isolation valve is approximately 10 sec, but because of the valve characteristics steam flow through the valve will be reduced to zero within 6 sec. Since reactor scram is essentially complete within 2 sec, the reactor will be generating only decay heat by the time the flow through the isolation valve is completely stopped. Because the rate of core heat generation drops off rapidly and the shutdown condenser has sufficient capacity to remove all decay heat, the primary system pressure rise will be small. The analog results show that the pressure oscillates approximately ± 1 psi for about 7 sec after start of isolation, and then gradually decreases. Thus, initiation of scram and shutdown condenser operation on start of isolation valve closure provides adequate protection against primary system pressure rise.

14.3.5.2 Scram on High Pressure. If the shutdown condenser were placed in operation but scram action failed on closure of the isolation valve, the reactor pressure would not reach the reactor vessel design pressure (1400 psig). A high pressure scram signal would be given when system pressure reached 1325 psig, and backup scram signals would be provided by the 1350-psig scram and high-power-level scram during the resulting transient. The analog results indicate that a peak pressure of 1355 psig is reached, after which the pressure returns to an equilibrium value of approximately 1348 psig. The reactor power level rises to approximately 140 percent of full power as a result of the positive reactivity effect of the pressure increase, but the rapid power reduction caused by scram prevents fuel meltdown.

14.3.5.3 Failure of Shutdown Condenser Inlet or Outlet Valve to Open. In the event the shutdown condenser inlet and/or outlet valves fail to open, the three safety valves in the 10-in. steam line protect the integrity of the reactor vessel and primary piping by venting steam directly to the containment building atmosphere (Sec. 14.1.2), which is isolated on a high reactor pressure signal. The feedwater supply, backed up by the emergency core spray, will provide core cooling water to prevent fuel damage. Prior to opening of the relief valves, the isolation valve closure will provide a scram signal that is backed up by scram signals due to high reactor pressure, high reactor power level, and an unsafe power-flow relation (Sec. 8.5.2).

14.3.6 Turbine Trip

If the turbine trips, closure of the turbine main stop valve initiates a scram signal and causes a transient increase of main steam pressure by preventing steam flow to the turbine. This pressure rise causes the main steam bypass valve to open (time constant of 1 sec*), allowing steam to be dumped directly to the turbine condenser. This condition and the case in which the bypass valve fails to open have been analyzed using the analog simulator. The results and further discussion are presented in Sec. 3.3 of Appendix C of Amendment 9 (ACNP-65517). Neither case results in a hazardous condition.

14.3.6.1 Scram and Opening of Bypass Valve. If the turbine trip is followed by reactor scram and opening of the bypass valve the reactor power level goes through a peak of 112 percent of full power (because of the positive reactivity effect of a pressure increase), and the reactor pressure goes to a peak of 1306 psig. Both power level and pressure subsequently decrease.

14.3.6.2 Scram, with Bypass Valve Stuck Closed. If the turbine trip is followed by scram only, the reactor power level behavior is almost identical to the case when the bypass valve opens. However, the reactor pressure rises to beyond 1325 psig, and hence the shutdown condenser system would automatically be placed in operation and system pressure would be reduced. The analog simulation did not include operation of the shutdown condenser and thus showed a gradual pressure rise after an initial peak at 1328 psig.

14.3.7 Opening of Main Steam Bypass Valve

If the main steam bypass valve accidentally became fully opened during 100 percent power operation, an additional steam flow up to 100 percent of the full-power flow could be withdrawn from the primary system, and thus primary system pressure would decrease. Analog simulation studies reported in Sec. 4.2.6 of Amendment 7 (ACNP-64604) show that reactor vessel and turbine inlet pressure start to decrease as soon as bypass valve opening is initiated. If the valve is opened at its maximum rate and the operation of the initial regulator system is neglected, the rate of pressure decrease is essentially constant after 2 sec, at a rate of 20 psi/sec. This rate of depressurization has been used in the reactor vessel design analyses (Sec. 4.2.1). A decreasing primary system pressure has a negative reactivity effect, so reactor power level also begins to decrease as valve opening is initiated. In addition, the reactor water level would fall at a rate of approximately 1 in./sec.

*Because of the valve actuator characteristics, the time constant is equal to the time required for full opening of the valve when the valve stem is moved continuously at its maximum rate.

When the water level drops to a preset level above the top of the core, the following automatic actions occur: (1) reactor scram, (2) reactor building steam isolation valve closure (which initiates a separate scram signal and shutdown condenser system operation), (3) emergency core spray system operation, and (4) closure of the primary water blow-down valve (if open). Thus, the reactor would be shut down and power would decrease to decay-heat levels. Further loss of steam would be prevented by the closed isolation valve, and the shutdown condenser would remove all heat generated within the core. The automatic safety actions are sufficiently rapid to avoid danger of fuel meltdown caused by uncovering of the core. The feedwater pump would be operated at full speed by the automatic feedwater control system prior to restoration of reactor water level by the feedwater and emergency core spray.

For actual plant operation, the initial pressure regulator system will assume control of turbine inlet pressure when the turbine inlet pressure decreases below a set value. This system regulates the turbine inlet valves to close with further reduction in pressure. Thus, the rates of decrease of primary system pressure and reactor water level would be less than the values given above, and the effects would be less serious than those projected.

14.3.8 Feedwater System Failure

Normally, one feedwater pump is used to return turbine condensate to the reactor, with the second pump on ready standby. Under special conditions the two pumps may be used in parallel, each pump supplying one-half the required feedwater flow. Since the scoop tube actuator on the pump drive provides for maximum flow output on loss of control air, such failure could cause an increase in feedwater flow rate from a full power rate of 1310 gpm to a maximum of 1580 gpm with one pump operation and 2800 gpm with two pump operation. Accidental closing of the feedwater control valve, on the other hand, would cause feedwater flow to decrease to zero. Both increase and loss of feedwater flow have been analyzed with the aid of the analog simulator. Further discussion and results are presented in Sec. 3.4 of Appendix C (ACNP-65511).

14.3.8.1 Increase in Feedwater Flow. The increase in feedwater flow rate is reflected by an increase in core inlet subcooling, and, hence, reactivity. Analog investigations of system response following feedwater flow increases were performed for the cases of both single and dual feedwater pump operation. In both cases initial reactor operation at 100 percent of full power was assumed. It was also conservatively assumed that the reactor was on manual control so that reductions in reactor power level by reductions in the recirculation flow rate were not possible. Further, it was assumed in both cases that feedwater flow rate is increased from its initial value to the maximum value at a constant rate equal to the maximum rate.

The increase in feedwater flow rate from 1310 to 1580 gpm, corresponding to single-pump operation, results in a rise of reactor power to a steady-state value of 117 percent (if high power scram is neglected), and of primary system pressure to approximately 1320 psig. This power level and pressure rise is not excessive. Thus, even in the absence of the scram at 115 percent of full power, no serious effects result from an increase in the feedwater flow rate from its normal value to the maximum flow obtainable with a single feedwater pump.

The increase in feedwater flow rate from 1310 to 2800 gpm, corresponding to dual-pump operation, requires scram to prevent excessive power levels. Reactor scram takes place when the power level reaches 115 percent of full power and causes a rapid drop in power level. The maximum surface heat flux from a fuel rod and the primary system pressure are also kept low enough by the scram to prevent any serious consequences.

14.3.8.2 Loss of Feedwater Flow. If loss of feedwater flow were to occur because of accidental closure of the feedwater control valve or for any other reason, the decrease in core subcooling would cause a drop in reactor pressure and power level. Analog results indicate that evaporation would rapidly reduce the reactor water level to the scram setpoint, and neither the reduced power level nor the turbine initial pressure regulator system would delay the time for low-low-level scram, (~14 sec). Reduction of water level to the scram setpoint will also cause closure of the reactor building steam isolation valve, operation of the shutdown condenser, and introduction of core coolant through the emergency core spray system. Thus, the reactor water level is automatically restored and serious consequences do not result from the accident.

14.3.9 Loss of Forced-Circulation Flow

Abnormal operating conditions of the forced circulation system, including the loss of the forced-circulation pumps, are discussed in Sec. 11.1.5. A rapid reduction in recirculation flow rate will result in a corresponding drop in reactor power level, but the drop in core heat flux will lag the power generated because of the thermal time-constant of the fuel. Therefore, the loss of recirculation flow accident was analyzed with the analog simulator to assure that the burnout safety margin is maintained above 1.5 during the transient. The transient values of maximum core heat flux, core exit quality, recirculation flow rate, and reactor pressure were determined; and these parameters were used to determine the burnout safety margin as a function of time following the loss of pumping power. It was conservatively assumed that each pump was completely decoupled from its motor, and the natural-circulation driving head was neglected. Results indicate that the burnout safety margin decreases very slightly from its initial value of 1.86 to approximately 1.84 in the first 0.1 sec. following loss of pumping power, and increases thereafter. Thus, the minimum burnout safety margin is well above 1.5.

Digital computer analyses for operation at 10 percent, 20 percent, and 30 percent of full power indicate that natural circulation flow can remove up to 30 percent of full-power heat generation at 1285 psig (with or without subcooling by feedwater), without violating thermal-hydraulic design criteria (Table 14-3A). Thus, it is expected that flow coastdown after loss of pumping power will be at a slower rate than indicated by the analog model; and the reactor will stabilize at some low-power level with natural circulation cooling.

In the event that feedwater flow is slowed or stopped during natural circulation cooling of the shutdown reactor, steam flow to the main condenser would lower the reactor water level to the scram setpoint, thus initiating emergency core spray, closure of the reactor building steam isolation valve, and shutdown condenser operation. An interlock insures that the forced-circulation pump suction and discharge valves are open when the shutdown condenser is placed in operation, and shutdown cooling by natural circulation through the shutdown condenser would proceed in a normal manner. The fuel would not be damaged.

TABLE 14-3A

NATURAL CONVECTION RECIRCULATION FLOW*
(pressure = 1300 psia)

power	6%**	10%	20%	30%
flow, gpm	2700	3200	3800	4000
average steam voids, fraction	0.17	0.23	0.31	0.37
maximum exit voids, fraction	0.36	0.44	0.57	0.64
minimum burnout ratio	25.0	14.0	6.9	4.7
forced-circulation system				
pressure drop, ft				
pumps	0.30	0.35	0.45	0.50
rest of loop	0.55	0.70	0.85	1.00

*assuming that feedwater is at saturation temperature

**extrapolated from 10 percent power

14.3.10 Power Failure

The design features of the electrical distribution system (described in Sec. 9) that minimize the frequency and effect of power outages include:

- (1) automatic reserve power to all equipment,

(2) bus splitting to provide isolation of faults and to supply separate power feeds to duplicate equipment where desirable,

(3) bus ties to provide alternate power feeds,

(4) automatic emergency diesel generator power to all essential equipment within 15 sec of loss of normal power,

(5) non-interruptible power (both a-c and d-c) to all essential instrumentation,

(6) instrumentation for monitoring and alarming. Corrective or preventive action is taken as required.

Loss of voltage to either 2400-v bus is monitored by time-delay relays. A partial scram is effected if voltage to either bus is lost for longer than the time required for automatic switching to reserve power. An all-rod scram is effected if voltage is lost from both busses for longer than the time required for automatic switching to reserve power. The voltage to reactor building motor control center 1A is similarly monitored to effect an all-rod scram if voltage is lost to the control-rod drive electric motors for longer than the time for automatic switching to reserve power.

Although loss of main and reserve sources of auxiliary power would trip the forced-circulation pumps and the circulating water pumps, serious consequences would not result. The reduced recirculation flow would reduce reactor power, and loss of condenser cooling water would eventually lead to loss of condenser vacuum with reactor scram. Adequate core cooling is then maintained by operation of the shutdown condenser. Cooling water to the shutdown condenser is supplied by the demineralized water transfer pumps, which are fed by the emergency generator after loss of normal power. Final backup cooling water is supplied by the diesel-engine-driven auxiliary service water pump.

14.3.11 Instrument Failure

Fail-safe design features have been incorporated in safety system equipment. Channel component failure that prevents proper scram channel operation actuates channel trip stages. Equipment failure in any channel does not affect the operation of any other channel. Each channel has a separate power supply incorporating independent overload protection. Redundancy and/or backup are provided throughout the system by duplicate channels which monitor the same variable, by alternate inputs to actuate the safety system, and by a full scram backup for some partial scram signals to act if the scram condition worsens. Safety features of all reactor plant instrumentation are discussed in detail in Sec. 8.

14.3.12 Failure of a Control-Rod-Drive Mechanism

There would be no serious consequences if mechanical failure of a control-rod drive mechanism prevented a completely withdrawn rod from entering the core under scram conditions. The core would be subcritical by at least 0.5 percent Δk , even if the failed rod were a rod of maximum worth.

The scrambling force for each of the 29 control rods is supplied by a continuously monitored hydraulic accumulator. Low gas pressure or oil level in any accumulator initiates a partial scram, so that the worst consequences of an accumulator failure would be a nuisance scram.

A separate feature of the rod drive mechanism (Sec. 8.5.1) assures rod insertion into the core if hydraulic pressure is lost while the reactor is in a scram condition. If the hydraulic motor drives slower than the electric motor during rod insertion, the electric motor exerts torque to drive the control rod completely into the core. Upon the completion of scrambling motion, or completion of shimming motion at any position along its stroke, the control rod is positively prevented from further vertical motion by the action of the mechanical brake which is always engaged. The effects of a control rod dropping out of the core after a postulated separation from its drive mechanism is discussed in Sec. 14.3.3.

14.3.13 Fuel Element Failure

This section supersedes Sec. 14.2.14 of ACNP-62574, and uses the data collected by the anemometer at the reactor site (see Sec. 3.4.4).

The consequences of failures related to small defects in the fuel cladding are slight. Fission product release from defective fuel elements is expressed by an escape-rate coefficient which is defined as the fraction of contained nuclides that escape from a fuel rod each second. Escape-rate coefficients are on the order of $1.3 \times 10^{-8}/\text{sec}$ for the gaseous fission-products xenon, krypton, iodine, and bromine*. If defects occur in the fuel, nearly all of the halogen activity released from the fuel, plus any released solid fission-product activity, are expected to be retained in the reactor water. Most of the small amount of halogens and solid fission products that escape with steam from the reactor vessel are condensed or dissolved in the condensate in the turbine condenser. Thus, the only significant activity release from the stack after fuel-element failure is from gaseous xenon and krypton nuclides which are removed by the main condenser air ejectors.

The concentrations of the xenon and krypton nuclides downwind of the stack have been computed, based on release from defective fuel containing equilibrium concentrations of the various nuclides. A 10-min holdup time exists between release from the fuel and release from the stack. Release severity is estimated by summing the ratios

*WAPD-TM-29, "Radiochemistry of Third PWR Fuel Material Test X-1 Loop NRX Reactor," p. 38 (Feb. 1, 1957).

of the actual concentration vs. maximum permissible concentration of each nuclide for unrestricted areas. The sum of these ratios should not exceed unity. The activity released from the stack for 1 Mwt of failed fuel-rods (44 average-power fuel rods) and the nuclide distribution are given in Table 14-4; the magnitude of release from a given number of defective fuel rods is obtained by multiplying the given release rates by the number of defective rods and dividing by 44.

TABLE 14-4
GASEOUS FISSION-PRODUCT RELEASE FROM THE
STACK FROM 1 Mwt OF DEFECTIVE FUEL RODS

<u>isotope</u>	<u>release rate, curies/sec</u>
Xe-131m	3.3×10^{-6}
Xe-133m	1.7×10^{-5}
Xe-133	7.2×10^{-4}
Xe-135m	1.2×10^{-4}
Xe-135	6.4×10^{-4}
Kr-83m	5.7×10^{-5}
Kr-85m	1.1×10^{-4}
Kr-85	1.3×10^{-6}
Kr-87	2.7×10^{-4}
Kr-88	4.0×10^{-4}

A summary of the number of fuel rods that can be defective without exceeding the maximum permissible concentration (MPC) at the receptor, for various meteorological conditions, is given in Table 14-5. Each case listed in the table is discussed in more detail in the following paragraphs of this section.

The highest instantaneous concentrations at ground level occur during fumigation conditions. The concentrations during fumigation at the exclusion area boundary were determined from Gifford.* The Type-B meteorological conditions shown in Fig. V-1 (p. 17) of Gifford were used, representing the moderately unstable condition expected to exist below the inversion lid. An inversion lid height equal to the effective stack height of 618 ft (determined from the Bryant-Davidson correlation on page 72 of Meteorology and Atomic Energy) and a wind speed of 1 m/sec were also used. With these parameters, the sum of the ratios of actual concentration to MPC for all xenon and krypton nuclides at the site boundary would not exceed a value of 1, provided that there were defects in no more than 48 fuel rods.

*The Problem of Forecasting Dispersion in the Lower Atmosphere, USAEC, Oak Ridge, Tennessee, July 1961, p. 24, Eq. 5.

TABLE 14-5

ALLOWABLE DEFECTIVE FUEL RODS WITHIN MPC FOR STACK RELEASE DURING PLANT OPERATION

activity concentration considered	meteorological category	wind speed, m/sec	effective stack height, ft	directional frequency or reduction factor	distance from stack, ft	no. of allowable defective fuel rods without exceeding MPC
instantaneous fumigation to plain	Type B, below inversion lid	1.0	618	--	1109 (excl. radius)	48
average fumigation to plain	Type B below inversion lid	1.0	618	30% (reduc. factor)	1109 (excl. radius)	3800
long-period average to plain	Type A	3.0	408	39%	1480	1710
long-period average to plain	Type B	3.0	408	39%	2600	3200
long-period average to bluff	Types D, E, F	1.4	515	7.9%	2300-3000	925-975
long-period average to bluff	Type B	1.4	515	7.9%	1640	1540
long-period average to bluff with 50% inversion frequency	Type F	1.0	618	8.9%	8500	all
average fumigation to bluff	Type B below inversion lid	1.0	618	10% (reduc. factor)	2460	6240
instantaneous fumigation to bluff	Type B below inversion lid	1.0	618	--	2460	26
long-period average to bluff based on 1 hr/day of fumigation and 23 hr/day of Type-B condition		--	--	--	~1640	1300*

*MPC for setpoint of alarms are based on this condition since Type-B meteorological conditions are predominant at the site.

In addition to this calculation for an instantaneous condition, the average concentrations from fumigation were calculated for a 1-yr period. Fumigation can be expected for no more than 1 hr in any given day. Also, a reduction factor of 30 percent was used because of wind direction variations. Defects could be present in 3800 fuel rods without exceeding permissible average concentrations at the exclusion area boundary.

When averaged over a year, ground concentrations are actually higher for meteorological conditions other than fumigation. A method of determining long-term average concentrations is given by Eq. 7 on p. 24 of the Gifford reference. The equation contains an f-factor defined as the wind frequency toward a particular direction, in percent per radian. It also contains a vertical dispersion coefficient that depends on the distance from the source and on meteorological conditions. The equation was graphically maximized with respect to distance for various meteorological conditions, using the f-factor and the corresponding directional average wind velocity that give the largest concentration. From the wind roses in Fig. 3.8, an f-value of 39 percent and an average wind speed of 3.0 m/sec for winds from the south-southeast direction were obtained. This wind speed results in an effective stack height of 408 ft. The Type-A meteorological condition (extremely unstable) gives the highest ground concentration (at 1480 ft), the Type-B condition the next highest value (at 2600 ft). There could be defects in up to 1710 fuel rods for the Type-A condition and 3200 fuel rods for the Type-B condition without exceeding MPC. It is felt that the Type-B condition best represents the condition expected at the site.

The foregoing results assume there is no difference in elevation between the base of the stack and the surrounding area. There is, however, a steep bluff to the east of the site. The bluff rises sharply to a height of 360 ft above the base of the stack, then more gradually to a plateau 500 ft higher than the stack base. Bluff height vs. horizontal distance from the stack is given in Table 14-6. These distances were obtained from U. S. Coast and Geodetic quadrangle maps (see Fig. 1 of Appendix A in ACNP-62574).

TABLE 14-6
BLUFF HEIGHT VS. DISTANCE FROM STACK

<u>horizontal distance from stack, ft</u>	<u>height of bluff above base of stack, ft</u>
0 - ~ 600	0
1140	160
1380	260
1630	360
2460	460
6250	490
8400	500
> 8400	≤ 500

The concentrations of activity at various points on the bluff were determined from meteorological condition Types A through F, using Fig. V-2 of the Gifford reference. The equation for long-term average concentrations was used. From the wind roses in Fig. 2.8, an f-factor of 7.9 was used for wind blowing from the west toward the bluff. The corresponding average wind speed in this direction is 1.4 m/sec, giving an effective stack height of 515 ft. The stack height at each distance from the stack was assumed to be equal to the effective stack height measured from the base of the stack, minus the height of the bluff above the stack base. This assumption is conservative since there would be a tendency for the effective stack height to be increased by the surface boundary layer of winds blowing toward and over the bluff.

Meteorological condition Types D, E, and F give the highest maximum activity concentrations, all about a factor of 3.5 greater than the maximum average yearly concentration calculated for the Type-B condition with a flat plain. These maximums occur at distances of 2300 to 3000 ft from the stack. For the Type-B condition, which represents the most probable average condition at the site, the maximum concentration is at a distance of 1640 ft from the stack, and is a factor of about 2 greater than the Type-B flat plain condition.

The Type-F inversion condition on the bluff was also considered. For this case, a wind speed of 1 m/sec (giving an effective base stack height of 618 ft) and an f-factor of 8.9 percent were used to represent the conditions at night, when inversions are most likely.

An inversion frequency of 50 percent was used based on Table 3-9 corrected to account for wind conditions at the site (see Sec. 3.4.4.5). The maximum yearly average concentration occurs on the top of the bluff, ~ 8500 ft from the stack, and is a factor of 6 less than the maximum Type-B concentration on the bluff.

Maximum instantaneous and yearly average concentrations on the bluff caused by fumigation with Type-B meteorological conditions were also calculated. For the instantaneous case, it was found that 26 fuel rods would have to be defective for MPC to be exceeded (at a distance of 2460 ft from the stack). The average yearly concentration caused by fumigation for 1 hr/day is a factor of 240 less than the instantaneous concentration, based on a reduction factor of 10 percent to account for the wind directional frequency toward the bluff.

The calculations indicate that both the maximum instantaneous and yearly average activity concentrations caused by defective fuel rods would occur on the bluff to the east of the reactor site. The presence of defects in more than 26 rods is unlikely. When yearly average bluff concentrations for both fumigation and the Type-B condition are added to obtain a total yearly exposure, there could be defects in ~ 1300 fuel rods without exceeding average yearly permissible exposures.

A gross fuel-element rupture could result in much higher total fission-product release rates than those for fuel-rod defects. If excessive activity were detected by the fission-product monitor downstream of the holdup tank of the vent and waste gas disposal system, or by a monitor in the stack, a high radioactivity alarm would be given. In addition, a valve in the line to the stack would be automatically closed, and the primary-system off-gases would be automatically diverted to two 12,000-gal off-gas storage tanks (Sec. 5.3.3.11). Thus, excessive activity would not be released. The storage tanks can hold all off-gases produced by three days of full-power operation. If release from a gross fuel-element failure continued for more than three days, the plant would be shut down and failed elements removed. Failed elements would be identified by the failed fuel element location system (Sec. 5.2.7). After the initial burst of fission products following a gross failure, however, activity released from a fuel element might eventually decrease to acceptable levels.

The fission-product monitor is in the off-gas line upstream from the point where gas is diverted to the storage tanks. Thus any fission-product activity released from the primary system is indicated continuously, even if no gas flows to the stack. Release to the stack could restart whenever the fission-product monitor indicated low activity. Radioactive gas collected in the storage tanks could be bled to the stack to relieve storage-tank pressure. Since all gas from the storage tanks is also monitored by the stack gaseous and particulate monitor, this gas could be bled to the stack without release of excessive activity.

14.3.14 Primary-System Ruptures

Safety features of the plant design minimize the probability of primary-system ruptures caused by over-pressurization. Nevertheless, the effects of primary-system ruptures both within and without the containment building have been considered, and design measures have been taken to minimize their radiological hazards.

14.3.14.1 Ruptures Outside the Containment Building. A rupture in the main steam line outside the containment building would release primary-system steam to the atmosphere until closure of the reactor building steam isolation valve (closing time of ~ 10 sec). The valve is closed by either of two signals that would be initiated by a system rupture:

- (1) low reactor water level (which also causes scram), and
- (2) low pressure at the turbine stop valve.

Even if the core became completely uncovered prior to the signal for valve closure, fuel rods would not begin to melt in the 10 sec period required for isolation. Thus, no fission products from melted fuel would escape to the surrounding atmosphere after a main steam line rupture. Actually, it is doubtful that any fuel melting would occur since feedwater

flow to the primary system would continue after a rupture and would tend to restore the water level, which would prevent core damage.

The only doses received following a steam-line rupture would be those caused by release of activity contained in steam and primary-system water. ACNP-62574 discusses the calculational assumptions and methods used to determine the doses outside the containment building. The total integrated doses received at the exclusion area radius of 1109 ft (0.21 m) were found to be 1 rem to the thyroid and 4 mrem to the bone. For any distance beyond approximately 30 ft from the point of release the thyroid dose was found to be under 300 rem. It is therefore concluded that the radiological hazard from a steam-line rupture outside the containment building is not excessive.

The effects of a rupture in the feedwater line at a point outside the containment building would be less serious than the effects of a steam-line rupture. A check valve in the feedwater line at a point within the containment building prevents significant escape of primary-system water to the atmosphere. A shutoff valve may also be closed if required. Thus, the only significant effect of a rupture in the feedwater line is a loss of feedwater flow to the primary system, which is discussed in Sec. 14.3.8.2.

14.3.14.2 Ruptures Within the Containment Building. In the event of primary-system rupture within the containment building, the building would be automatically sealed off. The dampers in the ventilation inlet and exhaust ducts would close upon high reactor pressure, high gaseous or particulate activity, or high building pressure; the control valve in the 4-in. vent header leaving the containment building would close upon high reactor pressure or high building pressure; the reactor building steam isolation valve would close on low reactor water level or low steam pressure at the turbine stop valve; and the check valve in the feedwater line would prevent any backflow out of the containment building through this line. Low reactor water level would also cause a scram as would isolation valve closure. Additional valves that are located in piping penetrations of the containment vessel to assure containment isolation are described in Sec. 6.5.

The effects of a primary-system rupture within the containment building depend upon the size and location of the rupture. Three types of ruptures are discussed below: a major rupture above the top of the core, and minor and major ruptures below the bottom of the core.

14.3.14.2.1 Major Rupture Above Core. A major rupture in a line above the top of the core would partially uncover the core. However, flashing of water within the core and an emergency core spray initiated by low-low water level would cool the core sufficiently to prevent meltdown until the core was again covered. After the water level is restored by the feedwater supply system, continued flow of at least 50 gpm of emergency coolant would adequately cool the core.

14.3.14.2.2 Minor Rupture Below Core. For minor ruptures below the core that cause single blowdown openings of less than 1.5 in. equivalent diameter, reactor water level would be maintained by the feedwater supply and core meltdown would not result. For a 1.5 in.-dia rupture the flowrate from the reactor vessel at full system pressure is approximately equivalent to the feedwater flowrate.

14.3.14.2.3 Major Rupture Below Core. The severity of below-core ruptures increases with the rupture size. A major rupture in one of the 20-in. forced-recirculation lines is the worst case. The core would then quickly become completely uncovered, and feedwater flow would not restore the vessel water level. However, the emergency core-cooling system would be put into operation very shortly after the rupture. At least 50 gpm of cooling water would be directed at the top of the core from the spray nozzles. It is expected that spray water flow to the fuel elements would provide sufficient cooling to prevent melting. However, more pessimistic assumptions as to operation of the emergency cooling system following rupture in a 20-in. recirculation line form the basis for the maximum credible accident analysis in Sec. 14.3.18.

14.3.15 Inadvertent Release of Activity

14.3.15.1 Release to the Mississippi River.

14.3.15.1.1 Liquid Release. Gross spillage of radioactive water into the plant vicinity, which might cause contamination of the Mississippi River, is considered practically impossible. All radioactive water leaking from the primary system, and other liquid wastes produced within the containment building, are collected in two 6000-gal retention tanks. If there were a major primary-system rupture, all water in the primary system, together with the water in the overhead storage tank, would be collected in the base of the containment building. No water can leak from the containment building. Waste liquids produced in the turbine building are also collected in tanks. Waste liquids from both the turbine building and the containment building may be pumped to the waste-treatment facility. The piping tunnels leading to the waste-treatment facility are lined with concrete. Also, tanks of the waste-treatment facility are placed in concrete compartments so that, even if leakage did occur from ruptures in either the line or the tanks, the radioactive liquids would be contained.

The largest tank in the waste-treatment facility has a capacity of 1000 gal. If this tank is assumed to be filled with primary-system water, the total contained activity, based on the figures of Tables 10-1 and 10-2 of ACNP-62574, would be approximately 1.7 curies. Based on the dilution factors given in Sec. 3.4.5.5 for slug release to the river, and assuming (very conservatively) the release of all this activity, the maximum river water contamination 40 miles downstream would be $1.4 \times 10^{-6} \mu\text{c/ml}$. No river water is used for municipal

water supply for a distance of at least 40 miles downstream. Assuming no Ra-226 or Ra-228, this activity level is a factor of 14 higher than the permissible average yearly level in unrestricted areas. Since it is an instantaneous and not an average value, it does not represent a hazard.

14.3.15.1.2 Release Following MCA. The worst possible contamination of the river is associated with the maximum credible accident. This has been determined under the assumption that one-half the iodine and solid fission-product activity released from the containment building following the MCA finds its way to the river and is mixed uniformly with a river flow of 27,970 cfs (see Table 3-12). Average concentrations over a 1-year period were determined, assuming that iodine release continues for 1 year after the accident and that solid fission-product activity continues for 30 days. Other assumptions include a 100 percent fuel meltdown and a constant leakage rate of 0.1 percent/day of the contained activity, and a reduction of 50 percent of the iodines in the containment building. The results are presented in Table 14-7 along with the maximum permissible concentrations (MPC).

The isotopic concentration in the river, when averaged over the year following MCA, would be less than the maximum permissible concentration. The calculations are conservative. Complete fuel meltdown has been used even though less than 35 percent of the fuel rods melt during the MCA (see Sec. 14.3.18.2). A 0.1 percent/day constant leakage rate is used, although pressure reduction in the containment building would lead to much lower release rates soon after the MCA. Since the 0.1 percent leakage rate is used for one year for the iodines, their calculated multiples of MPC are especially conservative. Also, instead of the 50 percent reduction of the iodines, TID-14844 states that absorption, adherence, and settling are estimated to give a 3 - 10 reduction factor. This factor does not even include the washdown effect from the building spray system.

TABLE 14-7
MISSISSIPPI RIVER CONTAMINATION FOLLOWING MCA,
AVERAGED OVER ONE YEAR

<u>nuclide</u>	<u>concentration, μ c/ml</u>	<u>concentration/MPC</u>
I-131	2.5×10^{-7}	0.83
I-132	4.5×10^{-9}	negl.
I-133	5.8×10^{-8}	0.06
I-134	2.2×10^{-9}	negl.

Table 14-7 - Mississippi River Contamination Following MCA, Averaged Over One Year
(cont'd)

<u>nuclide</u>	<u>concentration, μ c/ml</u>	<u>concentration/MPC</u>
I-135	1.6×10^{-8}	negl.
Sr-89	2.9×10^{-8}	0.01
Sr-90, Y-90	2.8×10^{-9}	0.01
Ce-141	3.3×10^{-10}	negl.
Ce-144	3.8×10^{-8}	negl.
Pr-144	1.3×10^{-11}	negl.
Total	4.0×10^{-7}	0.91

14.3.15.2 Airborne Stack Effluent. Section 14.3.13 discusses isotopes released to the stack via the gas holdup tank if defects were present in fuel rods. The activity concentrations downwind of the stack were computed for various meteorological conditions.

Gaseous activity could also be released to the stack from the 12,000-gal gas storage tanks of the vent and waste gas disposal system. As discussed in Sec. 5.3.3.7, each tank can store system gases for 36 hr of full power operation. These gases do not include halogen nuclides retained in the reactor water. The doses were calculated assuming that one tank is filled to capacity at 300 psig and then releases its entire contents because of an inadvertent opening of the valves in the line to the stack.

A gaseous activity release rate from the holdup tank to the stack that, if continued, would cause a yearly average ground concentration of 10 or more times the MPC results in automatic closure of the valve in the line to the stack (Sec. 8.8.1.2). The gases are then diverted to the gas storage tanks. For the present calculation it was assumed that gases causing 10 MPC downwind were added continuously to a storage tank for 36 hr. A constant maximum leakage rate from the storage tank to the stack was then used. Fumigation was also assumed during the period of activity release. Only the xenon and krypton nuclides were considered since short-lived nuclides such as N-16, N-17, and O-19 would have substantially decayed before release.

The maximum activity concentration occurs at a point 2460 ft from the stack at a height of 460 ft on the bluff east of the reactor site. The total dose was found to be less than 1/4 rem and does not represent a significant hazard.

14.3.16 Fire

The 500 gpm high-pressure service water pump delivers water from the low-pressure service water system to hose connections and hydrants throughout the plant. A 700 gpm diesel-engine-driven auxiliary service water pump in the crib house can deliver river water and serves as a backup source for fire protection.

Portable fire extinguishers are placed at convenient locations throughout the containment building. In addition there are two hose connections on each floor. Combustible materials are kept out of the building, and none of the construction materials are combustible. The possibility of fire within the containment building is therefore slight, and a local fire could produce little damage before being extinguished.

Combustible material is present in the turbine building, but there is adequate fire protection. There are five hose cabinets and additional hose connections in the building. Areas of above normal fire potential, such as near the turbine boil system, are protected by manual water spray systems.

Other fire protection equipment includes a hose cabinet and various hose connections in the waste treatment building, and five fire hydrants located at various points around the containment, turbine, and waste treatment buildings.

14.3.17 Accidents of Nature

14.3.17.1 Earthquakes. The site is in a tectonically stable area where no earthquakes causing major damage have been experienced during the history of earthquake observation. The Uniform Building Code (1958 edition) includes the site in zone O, which indicates no significant damage would occur in the event of a seismic disturbance at the site.

14.3.17.2 Floods. The site is filled to a grade elevation of 639 ft. As discussed in Sec. 3.4.5.4 of ACNP-62574, it is estimated that there is less than one chance in two hundred that a flood of sufficient magnitude to raise the river water to the fill level would occur in any given year.

14.3.17.3 Tornadoes. As discussed in Sec. 3.4.4.6 of ACNP-62574, there is approximately one chance in 2000 that in any given year a particular square mile in the La Crosse area would be struck by a tornado. The chance of a tornado crossing directly over the reactor site is even less.

14.3.18 Maximum Credible Accident (MCA)

The maximum credible accident is postulated and analyzed in Sec. 14.3 of ACNP-62574. The analysis has since been modified to account for the design changes in shadow shielding discussed in ACNP-63584 (Amendment 3 to ACNP-62574): a concrete lining thickness of 9 in. instead of 2 ft in the cylindrical portion of the containment building, and the elimination of concrete lining the upper hemispherical head. Revised doses reported in Amendment 3 are based on calculations that include sky-shine dosage and on more conservative assumptions:

(1) 100 percent meltdown of fuel rods is assumed instead of the 18.2 percent meltdown calculated and used in the initial analysis, and

(2) a constant leakage rate of 0.1 percent/day of the contained activity is assumed instead of a pressure-dependent leakage rate decreasing with time following the MCA.

In addition to the revisions reported in ACNP-63584, energy release from the MCA has been recalculated to reflect refined values for the containment-free volume and the primary water inventory, and other calculations relating to the MCA have been reviewed to reflect the final design.

14.3.18.1 Definition. The maximum credible accident is the accident postulated to cause more serious consequences than any other accident deemed credible. This accident is sometimes referred to as the maximum design accident, since the reactor plant is designed to protect the general public from the conditions following the postulated accident. The credibility of such an accident is generally argumentative and the design protection is therefore based on very conservative assumptions. For LACBWR the maximum credible accident is postulated to be a sudden, gross loss of coolant. The hypothetical sequence of events is:

(1) A rupture completely opens one of the 20-in.-dia forced-recirculation lines.

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

(3) Since the reactor will be scrammed on low water level and since loss of moderator would make the reactor subcritical by a large amount, the reactor is assumed to generate only decay heat at the end of the blowdown period.

(4) There is a 30-sec delay after blowdown until emergency cooling water from the emergency core-spray pumps or the overhead storage tank reaches the core through the core spray nozzles.

(5) Since no heat is removed during this 30-sec interval, the fuel rod temperatures rise from decay heat.

(6) All water from the core spray ring flows through the core spray nozzles and then down the shrouds, with no direct cooling of the core. Each shroud section surrounds a fuel assembly. Heat is removed from these assemblies only by radiation.

14.3.18.2 Fuel Meltdown and Fission Product Release to Contamination. The maximum credible accident is evaluated for an assumed 100 percent meltdown of fuel rods when considering doses to the areas outside the reactor containment building (Sec. 14.3.18.6). However, the detailed analysis of fuel-rod meltdown indicates that this assumption is extremely conservative.

A core meltdown fraction of 18.2 percent was previously calculated and presented in Sec. 14.3.3. of ACNP-62574 together with a detailed discussion of the assumptions and methods used in the calculation. The final calculation, based on these methods but somewhat different assumptions regarding power peaking and fuel exposure, indicates instead that a maximum 34.9 percent of the fuel rods might experience melting somewhere along their length. Both results represent upper estimates because of the conservative assumptions employed in the analysis. The major factors of conservatism in the analysis are as follows:

(1) The fuel temperature is assumed to remain constant during the nearly instantaneous loss of reactor coolant. Actually, void formation would reduce nuclear heat generation as the core becomes uncovered, and flashing of coolant would lower the coolant temperature and tend to lower the fuel temperature.

(2) It is assumed that there is a 30-sec interval from loss of core coolant to cooling by water from the core spray nozzles. It is more realistically estimated that full flow from the core spray will be developed within 10 sec of the low reactor vessel water level signal for automatic start of the core spray pumps. Also, it is estimated that 5 to 10 sec would be required to uncover the core instead of the assumed nearly instantaneous blowdown. Thus, the actual period that the core would be without coolant is probably 0 to 5 sec, rather than the assumed 30 sec.

(3) It is assumed that emergency cooling water flows down the shrouds without direct cooling of the fuel, and that heat is transferred by radiation only. Further, it is assumed that only the outside row of fuel rods radiate heat directly to the shroud, while other rods radiate heat to the adjacent outer row.

(4) It is assumed that the temperature across each fuel rod is uniform at all times. This is conservative since the fuel center will be at a higher temperature than the cladding; the cladding will not actually melt until the average fuel temperature is well above the melting point of the cladding.

(5) It is assumed that the fraction of fuel rods which experience melting somewhere in their length is equivalent to the meltdown fraction. The actual fraction of meltdown is, of course, much less.

(6) It is assumed that the MCA occurs when the fission-product inventory of the core is at a maximum.

Initial calculations indicated that portions of the cladding would melt on 18.2 percent of the fuel rods in the core; these rods would contain a maximum 24.4 percent of the total fission-product inventory if power distribution is assumed to be that for a startup core. Dose calculations for the MCA (reported in ACNP-62574) were therefore based on release to the containment atmosphere of 24.4 percent of the noble gases, 12.2 percent of the halogens, and 0.244 percent of the solid fission products contained in the core after an average fuel exposure of 15,000 Mwd/ST.

The final design calculations and fuel management program for LACBWR permit revision of the calculated values for actual fission-product release to account for the following:

(1) the axial maximum-to-average heat flux peaking factor is 1.87 instead of the 1.45 used in the original calculation, and

(2) the maximum percentage of total fission-product inventory contained in the melted rods can be more accurately determined when based on the actual fuel management program (described in Sec. 4.5) instead of on an average fuel exposure of 15,000 Mwd/ST for the entire core.

Fission-product inventory in melted fuel rods depends on both fuel exposure time and power density. The longest fuel exposure time occurs for the equilibrium core, which has an approximate exposure of 15,000 Mwd/ST for the central third of fuel, 10,000 Mwd/ST for the intermediate third, and 5000 Mwd/ST for the outer third. The maximum power densities occur for the startup core before flux peaking has been reduced. The revised analysis assumes exposure times based on the equilibrium core, and power densities based on the startup core. The reference data and method for calculation of isotopic activity are described in Sec. 14.3.3.4 of ACNP-62574. The fuel rod meltdown percentage is increased from 18.2 to 34.9 because of the higher peaking factor. The effective meltdown fraction applicable to fission-product release is somewhat greater because the molten rods will have experienced greater exposure. An effective meltdown fraction of 37.8 percent has been calculated, based on appropriate weighting of the fission product distribution. The portion of the core fission-product inventory released to the containment building is therefore revised to 37.8 percent of the noble gases, 18.9 percent of the halogens, and approximately 0.4 percent* of the solid fission products. The values for core and containment building fission-product activity are given in Table 14-8.

*not 0.378 percent, since the activity of the solid fission products reaches different fractions of saturation in the three fuel exposure regions

TABLE 14-8

CORE AND CONTAINMENT BUILDING FISSION-PRODUCT ACTIVITY

<u>isotope</u>	<u>core activity, c</u>	<u>released activity* to cont. bldg., c</u>	<u>released activity** to cont. bldg., c</u>
I-131	4.16×10^6	7.86×10^5	2.08×10^6
I-132	6.17×10^6	1.17×10^6	3.09×10^6
I-133	9.18×10^6	1.74×10^6	4.59×10^6
I-134	7.96×10^6	1.50×10^6	3.98×10^6
I-135	7.81×10^6	1.48×10^6	3.91×10^6
Xe-131m	4.18×10^4	1.58×10^4	4.18×10^4
Xe-133	9.22×10^6	3.49×10^6	9.22×10^6
Xe-135m	2.33×10^6	8.81×10^5	2.33×10^6
Xe-135	8.28×10^6	3.13×10^6	8.28×10^6
Kr-83m	7.74×10^5	2.93×10^5	7.74×10^5
Kr-85m	1.40×10^6	5.29×10^5	1.40×10^6
Kr-87	3.77×10^6	1.43×10^6	3.77×10^6
Kr-88	5.33×10^6	2.02×10^6	5.33×10^6
Sr-89	6.38×10^6	2.41×10^4	6.38×10^4
Sr-90	3.84×10^5	1.79×10^3	3.84×10^3
Y-90	3.91×10^5	1.82×10^3	3.91×10^3
Ce-141	7.46×10^6	2.82×10^4	7.46×10^4
Ce-144	4.96×10^6	2.28×10^4	4.96×10^4
Pr-144	3.27×10^6	1.39×10^4	3.27×10^4
Total solids (above isotopes)	2.39×10^8	9.10×10^5	2.39×10^6

*based on a 37.8 percent effective core meltdown

**based on a 100 percent core meltdown. This is conservatively used to calculate doses outside the containment building after a further reduction of 50 percent of the iodines is accounted for in the inhalation dose calculations, because of absorption onto internal surfaces (Sec. 14.3.18.6.5).

14.3.18.3 Pressure Variation Within Containment Building. The pressure within the containment building following the maximum credible accident was recalculated to include final values (see Sec. 6.2.8) of primary system energy and of containment free volume. The increase in building volume caused by reduction in shadow shielding compensates for the higher values for mass and energy released from the primary system, so that there is no significant change from the results reported in ACNP-62574.

Following a primary-system rupture (MCA) within the containment building, an initial peak pressure and peak temperature of 48.5 psig and 273 F are rapidly attained. If a rupture occurred while the reactor were flooded during normal cooldown procedure, the maximum pressure and temperature would be 50.6 psig and 275 F; but the flooding operation is not until several hours after reactor shutdown, at which time a core uncovering would result in a much lower rod meltdown percentage. The calculation assumes that no heat either is absorbed by internal structures or lost from the building, and that the air within the building at the time of rupture is 80 F. Energy from a zirconium-water reaction is not applicable, since the only zirconium in the core is in the shrouds and vertical posts, and neither would be heated to reaction temperatures.

After peak pressure buildup, activation of the building spray system will rapidly reduce the building pressure and will also wash fission products from the building atmosphere. The 42,000-gal storage tank in the top of the reactor building can deliver ~ 27,000 gal of water at 1000 gpm to the building spray system. In addition, competing mechanisms will tend to change the containment building pressure; the addition of decay heat from the core to the building tending to increase the pressure, and heat losses from the building and heat absorption by internal structures tending to decrease the pressure.

The decrease of containment-building pressure with time, caused by building spray system operation, has been calculated with a number of conservative assumptions:

- (1) Although water in the overhead storage tank is normally close to 80 F, the ambient temperature of the containment building, it is assumed to be 100 F.
- (2) Whereas decay heat from the core is assumed to increase building pressure during the 27-min spray period, heat losses to internal structures of the building and heat transfer from the building to the outside are neglected.

The results are plotted in Fig. 14.1. Fifteen minutes after a rupture, the building pressure has been reduced to about 24 psig and the temperature to about 230 F. At the end of the spray period, or after 27 min, building pressure has been reduced to about 16 psig and the temperature to about 205 F.

After the 27-min spray period, the flow from the overhead storage tank to the building spray system is stopped but water from the overhead storage tank continues to the core spray ring at 50 gpm. Continued decay-heat generation would act to increase building pressure. However, if heat absorption by internal structures and heat losses from the building are accounted for, only a small increase in building pressure is expected after the 27-min spray period. Eventually, pressure begins to decrease with time.

14.3.18.4 Damage to the Containment Building. The maximum credible accident is not expected to damage the containment building. Thus, building leakage should not be greater than the 0.1 percent/day of the contained volume, which is the specified leak rate at the design pressure of 52 psig. Preliminary leak tests (Sec. 6.3) indicate that the leak rate will be much less than allowable.

The peak pressure in the containment building immediately following a major rupture is only 48.5 psig, which is under the design pressure. Since the building is designed according to Code standards, a building pressure > 200 psig could probably be contained without rupture.

The maximum rate of pressure rise within the containment building following a major rupture is ~ 5 psi/sec. The building is designed to withstand easily this loading rate.

High pressure portions of the primary system, including the reactor vessel, steam line piping, recirculation loop piping, and pumps are enclosed by structural and biological shielding concrete that will serve as an effective barrier to any missiles that might be generated.

The only auxiliary components not surrounded by substantial amounts of structural concrete are the emergency core spray system and portions of the primary purification system. The nine inches of concrete shadow shielding should provide adequate protection for the containment shell from damage by any missile generated, in the unlikely event of a non-ductile rupture in these systems.

14.3.18.5 Fission-Product Release from Containment Building. The leakage rate from the containment vessel is specified to be less than 0.1 percent of the contained volume per day (equivalent to release of 1.16×10^{-8} /sec of total activity) at the design pressure of 52.0 psig. The building spray system decreases building pressure as described in Sec. 14.3.18.3, which would, in turn, reduce the leakage flow. However, leakage flow is conservatively assumed to be constant at 0.1 percent. The leakage rate, $Q(c/sec)$, for a given nuclide or mixture of nuclides is obtained by multiplying the appropriate value in Table 14-8 by 1.16×10^{-8} after first reducing the iodine inventory by a factor of 2 (see below).

Washout and settling of fission products within the containment building are ignored, with the following exceptions:

(1) Inhalation doses for the iodines were calculated on the basis of a 50 percent reduction in the iodines present within the containment building immediately after the rupture. Because of the chemical nature of iodine isotopes, at least 50 percent of the iodines would be removed from the building atmosphere by absorption onto internal surfaces (see TID-14844, Calculation of Distance Factors for Power and Test Reactor Sites, p. 14).

(2) Although no settling or washout of solid fission products is assumed during the first 2 hr, a large fraction of these nuclides would be removed by the building spray system, and, over long periods of time, a significant amount of the solid fission products settle out of the building atmosphere. Therefore, for calculations of the doses received for an indefinite period of time, after the accident, it was assumed that after 30 days all solid activity is removed from the building atmosphere.

For all release calculations, the fission products are assumed to emanate at ground level from a single point of the building.

14.3.18.6 Effects of MCA Outside the Containment Building.

14.3.18.6.1 Meteorological Considerations. The inhalation dosage predicted downwind from the plant as a result of the maximum credible accident is greatly dependent upon the choice of diffusion equation and the meteorological parameters used in performing the calculations. Sutton's diffusion equation has long been used to express the activity concentration downwind from the point of release. However, the difficulty in obtaining accurate experimental values for the parameters needed in Sutton's equation has led to more widespread use of the Gaussian interpolation formula. As stated by Gifford:*

$$X = \frac{Q}{\pi \sigma_y \sigma_z \bar{u}} \exp \left[-\frac{1}{2} \left(\frac{y^2}{\sigma_y^2} + \frac{h^2}{\sigma_z^2} \right) \right]$$

where,

X = downwind concentration in curies per cubic meter,

Q = source strength in curies per second

$\bar{u}$ = mean wind speed in meters per second

*Gifford, F. A., Jr., The Problem of Forecasting Dispersion in the Lower Atmosphere, p.22, Eq. 1.

y = cross-wind distance in meters from the plume axis, which is assumed to coincide with the mean wind direction

h = source height in meters, and

σ_y, σ_z = dispersion coefficients in meters.

For a Gaussian distribution of concentration, the dispersion coefficients σ_y and σ_z may be interpreted as the standard concentration deviations in the y and z direction. Thus, the equation in this form permits direct interpretation of experimental data obtained from field measurements.

The dispersion coefficients of the Gaussian interpolation may be expressed in terms of the Sutton parameters by:

$$\sigma_y = \frac{1}{\sqrt{2}} C_y d^{1-n/2} \quad \text{and}$$
$$\sigma_z = \frac{1}{\sqrt{2}} C_z d^{1-n/2}$$

where,

C_y, C_z = Sutton's diffusion coefficients

n = stability factor, and

d = distance downwind from the source

Families of curves of σ_y and σ_z as a function of distance from the source and for various meteorological categories are presented on p. 17 of the Gifford reference.* These curves are based on the work of Pasquill and Meade. The meteorological categories vary from a Type-A regime (extremely unstable) to a Type-F regime (moderately stable). According to TID-14844, the Pasquill Type-F regime represents a moderately stable regime that could occur between 15 and 25 percent of the time in most areas of the United States (frequency data is actually for England). The frequency of unfavorable regimes at the site should not be significantly greater than for the average United States values. Therefore, the meteorological parameters used are based on the Type-F regime, which represents the worst conditions that might be expected at the time of the accident. Calculations were based on the following values of Sutton's parameters:

$$\bar{u} = 1 \text{ meter/sec}$$

$$C_y = 0.40 \text{ meters}^{n/2}$$

*Ibid., p. 17

$$C_z = 0.07 \text{ meters}^{n/2}$$

$$n = 0.5$$

$$\sigma_y = \frac{0.40}{\sqrt{2}} d^{0.75}$$

$$\sigma_z = \frac{0.07}{\sqrt{2}} d^{0.75}$$

When these values are used to solve for σ_y and σ_z as a function of distance, the curves for the Type F regime presented by Gifford are approximated.

To determine the amount of material deposited from a plume on ground surfaces, the suggested method of Chamberlain* has most often been used. Chamberlain defined a deposition velocity (V_g) as:

$$V_g = \frac{\text{amount deposited per cm}^2 \text{ of horizontal surface per sec}}{\text{volumetric concentration above this surface}}$$

Thus, the rate of deposition upon a horizontal surface in curies per cm^2 per sec equals V_g times the plume concentration in curies per cm^3 . This method is used to determine the deposition rate of iodine and solid fission-product activity following the maximum credible accident.

Values of V_g have been experimentally determined. It has been found that deposition velocities for chemically active materials such as iodine are in general an order of magnitude higher than deposition velocities for inert material of similar size.** The deposition velocities assumed are:

$$V_g = 0.02 \text{ m/sec (for iodines)}$$

$$V_g = 0.001 \text{ m/sec (for solid fission products)}$$

The deposition velocity for the solid fission products corresponds to a particle size in the submicron range.

14.3.18.6.2. Direct Radiation Doses. Observers outside the containment would be subjected to direct gamma radiation from the fission products within the containment building, which will be attenuated in part by the concrete lining of the vertical walls.

*Chamberlain, A. C., "Aspects of Travel and Deposition of Aerosol and Vapor Clouds," British Report AERE-HP/R-1261 (September 17, 1953)

**Gifford, F. A., Jr., and D. H. Pack, Surface-deposition of Airborne Material - Some Recent Measurements, p. 8, unpublished (January 29, 1962)

This direct gamma dose is calculated by regarding the total fission products within the containment as two point sources, one corresponding to the 20 percent of free volume contained in the unshielded top hemispherical dome and the other to the 80 percent of free volume contained in the shielded cylindrical section.

For a specific gamma-emitting nuclide, the dose rate from a point source is given by:

$$D = C \frac{P_o B e^{-\mu t}}{4 \pi d^2}$$

where,

- D = dose rate, rem/sec
- P_o = source energy release rate, gammas/sec
- B = buildup factor, dimensionless
- μ = linear absorption coefficient, cm⁻¹
- d = distance from source, cm
- t = thickness of attenuating material, cm, and
- C = conversion factor, rem/gamma/cm²-sec

The initial source strengths for each of the iodine, krypton and xenon nuclides, and for the mixture of solid fission products, was obtained from Table IV of TID-14844, which also gives an average gamma energy for each of the nuclides. These source strengths were modified to correspond to 165 Mw and an equivalent core meltdown of 100 percent.

Initial dose rates from the equivalent point sources were determined for each gamma energy using air attenuation and appropriate buildup factors, and including attenuation through 0.60 in. of steel for the source in the top dome and attenuation through 1.16 in. of steel plus 9 in. of concrete for the source in the cylindrical portion.

The time-integrated doses from the various nuclides were determined allowing for the effect of decay. Doses were determined (1) for 2-hr exposure and (2) for infinite time exposure with the solids contributing only for the first 30 days.

The total doses resulting from direct radiation from the building are tabulated in Table 14-9.

TABLE 14-9
DIRECT RADIATION DOSES

<u>distance</u>	<u>dose (rem/2 hr)</u>	<u>dose (rem/∞ or rem/30 days)</u>
400 ft	230	735
900 ft	29.3	93.9
1109 ft*	10.9	34.9
3000 ft	0.14	0.45
5280 ft	0.04	0.13
3 mi**	---	---
19 mi***	---	---

* minimum radius of the exclusion area

** radius of low-population zone

*** distance to La Crosse, the nearest large population center

14.3.18.6.3 Sky-Shine Doses. The point source calculations with buildup, described in the preceding section, include the total direct and scattered contribution from the source volume within the upper dome. The source volume in the shielded cylindrical section, however, will contribute a sky-shine dose through the unshielded dome which was not accounted for in the calculation of direct doses.

This sky-shine dose was estimated by treating 80 percent of the total source as an equivalent point source located at the midpoint of the cylindrical volume, and calculating the intensity of the single air-scattered gamma radiation at ground-level points outside the building. Equation (13) on page 85 of MIT-5007, Final Hazards Summary Report for the MIT Research Reactor, was used to calculate the intensity of scattered radiation:

$$I = \frac{NS}{4x} \left(\frac{d\sigma}{d\Omega} \right) (\psi_2 - \psi_1) \left[\pi - \theta_1 - \left(\frac{\psi_1 + \psi_2}{2} \right) \right]$$

where,

I = intensity of radiation, gammas/cm²-sec

N = electrons/cm³ of air

S = source strength, photons/sec

$\frac{d\sigma}{d\Omega}$ = Klein-Nishina differential cross section for scattering

x = distance from source, cm

ψ_1 = angle formed by intersection of lines from source to receptor and from source to top edge of cylinder, radians

ψ_2 = $\pi - \psi_1$, radians

ϕ_1 = angle formed by intersection of lines from receptor to source and from receptor to top edge of cylinder, radians

A scattering angle of 90 deg was assumed. This is conservative because most of the scattering is through a larger angle, and the scattering cross section in general and the final photon energy decrease with increasing scattering angle.

The effects of attenuation in air and through the building wall and the effects of multiple air scattering are approximated by attenuating the scattered photons exponentially over the straight-line distance between source and receptor and by assuming a buildup factor. The scattered flux is converted to a dose rate at the degraded photon energy and the effect of fission product decay is included, as described by the following equation for the time-integrated dose:

$$D = I \text{ Be}^{-\Sigma \mu d} C \int_0^t \Gamma(t) dt$$

where,

D = dose in rem

B = buildup factor

μ = linear absorption coefficient, cm⁻¹

d = thickness of attenuating material, cm

$\Gamma(t)$ = decay law

C = conversion factor, rem/gamma/cm²-sec

The total dose is the sum of doses from all source nuclides. The results are given in Table 14-10 for various distances.

TABLE 14-10

SKY-SHINE DOSES

<u>distance</u>	<u>rem/2 hr</u>	<u>rem/∞ or 30 days</u>
400 ft	152	486
900 ft	27.6	88.4
1109 ft	9.65	30.9
3000 ft	0.39	1.25
1 mi	0.05	0.16
3 mi	--	--
19 mi	--	--

14.3.18.6.4. External Doses from Leakage Plume. When the receptor and cloud centerline coincide with ground level, the centerline concentration factor downwind of a continuous point source (see Sec. 14.3.18.6.1) is:

$$\frac{X}{Q} = \frac{1}{\pi \mu \sigma_y \sigma_z}$$

where,

the dispersion coefficients σ_y and σ_z are the standard vertical and horizontal deviations of the cloud concentration.

The centerline concentrations at various distances from the source are given in Table 14-11 for $C_y = 0.4$ and $C_z = 0.07$.

TABLE 14-11

CENTERLINE CONCENTRATION FACTOR
(sec/m³)

<u>distance</u>	<u>X/Q</u>
400 ft	1.70×10^{-2}
900 ft	5.05×10^{-3}
1109 ft	3.66×10^{-3}
3000 ft	8.20×10^{-4}
1 mi	3.51×10^{-4}
3 mi	6.73×10^{-5}
19 mi	4.25×10^{-6}

The gamma envelopment dose is conservatively calculated by assuming an observer to be enveloped by a half-space of uniform source concentration equal to the calculated centerline concentration. The dose rate at any time is

$$D(t) = 1.5 \frac{S(t) C}{2 \mu}$$

where the volumetric source $S(t)$ is determined by the leakage rate Q for each nuclide and the downwind centerline concentration factor X/Q , and where μ and C are the air attenuation coefficient and the dose-to-flux conversion factor for the gamma energy of interest. The factor of 1.5 is included to approximate the effect of a linear buildup from sources beyond one mean free path from the receptor and to account for the small dose contribution from beta decay.

Source terms were obtained for each nuclide and for the mixture of solid fission products from the data presented on p. 27 of TID-14844.

The time-integrated gamma exposure from all nuclides in the plume was determined considering decay within the containment building previous to release, but not after release from the building. The resulting external doses are presented in Table 14-12.

TABLE 14-12
EXTERNAL DOSE FROM LEAKAGE PLUME

<u>distance</u>	<u>rem/2 hr</u>	<u>rem/∞ or 30 days</u>
400 ft	14	238
900 ft	4.0	68.0
1109 ft	2.90	49.3
3000 ft	0.66	11.2
1 mi	0.29	5.3
3 mi	0.06	1.16
19 mi	--	0.09

14.3.18.6.5 Inhalation Doses. Inhalation doses were computed for the nuclides I-131, I-132, I-133, I-134, I-135, Sr-89, Sr-90, Y-90, Ce-141, Ce-144 for varying distances from the containment vessel. Doses were computed from the following formula:

$$TID = \frac{X}{Q} \cdot BZ \int Q_i dt$$

where,

TID = total integrated dose, rem

X/Q = concentration factor, sec/m^3

B = breathing rate, m^3/hr

Z = TID per curie inhaled, rem/c

Q_i = activity leakage rate, c/sec, and

t = time, hr

The concentration factors at different distances from the source are given in Sec. 14.3.18.6.4.

The activity leakage rate was calculated by first multiplying the activity values within the containment building (Table 14-8) by 1.16×10^{-8} (Sec. 14.3.18.5) after reducing these by a factor of 50 percent for iodines only*. This reduction factor has been used only in the inhalation calculations. Decay within the containment building was accounted for, but not decay after release.

*TID-14844 (p.14) assumes that only 50 percent of the iodines in the containment building are available for release to the atmosphere.

For the iodines, inhalation times of 2 hr and infinity were used and, for solids, 2 hr and 30 days. Breathing rates of $1.25 \text{ m}^3/\text{hr}$ for the 2-hr inhalation period and $0.833 \text{ m}^3/\text{hr}$ for the infinite period were assumed (see p. 23 of TID-14844). Calculated values of Z^* for the various nuclides are listed in Table 14-13.

TABLE 14-13

INTEGRATED DOSES PER MICROCURIE INHALED OF SELECTED NUCLIDES

isotope	Z (rem/c inhaled)	organ affected
I-131	1.484×10^6	thyroid
I-132	0.054×10^6	thyroid
I-133	0.399×10^6	thyroid
I-134	0.025×10^6	thyroid
I-135	0.124×10^6	thyroid
Sr-89	0.528×10^6	bone
Sr-90 - Y-90	24.3×10^6	bone
Ce-141	0.020×10^6	bone
Ce-144 - Pr-144	1.23×10^6	bone

The total integrated doses for each of the nuclides are given in tables 14-14 and 14-15; the sums of the thyroid doses from the iodines are plotted in Figs. 14-4 and 14-5.

TABLE 14-14

TOTAL INTEGRATED DOSES FOR MAXIMUM EXPOSURE TIME OF 2 HR

nuclide	dose, rem						
	400 ft	900 ft	1109 ft	3000 ft	1 mi	3 mi	19 mi
I-131	760	225	164	36.6	15.7	3.01	0.19
I-132	31	9	7	1.5	0.6	0.12	0.01
I-133	436	129	94	21.0	9.0	1.73	0.11
I-134	12	4	3	0.6	0.3	0.05	--
I-135	108	32	23	5.2	2.2	0.43	0.03
total iodine	1,350	399	291	64.9	27.8	5.34	0.34

*Morgan, Hart A., ed., Health Physics, Off. J. Health Phys. Soc., Vol. 3 (June 1960)

Table 14-14 - Total Integrated Doses for Maximum Exposure Time of 2 Hr (cont'd)

nuclide	400 ft	900 ft	1109 ft	dose, rem	1 mi	3 mi	19 mi
				3000 ft			
Sr-89	15.1	4.5	3.2	0.73	0.31	0.06	----
Sr-90	47.8	14.2	10.3	2.30	0.99	0.19	0.01
Y-90	61.9	18.4	13.3	2.98	1.28	0.25	0.02
Ce-141	0.7	0.2	0.2	0.04	0.02	----	----
Ce-144	38.7	11.5	8.3	1.86	0.80	0.15	0.01
Pr-144	5.1	1.5	1.1	0.25	0.11	0.02	----
total solids	169	50.3	36.4	8.16	3.51	0.67	0.04

TABLE 14-15

TOTAL INTEGRATED DOSES FOR MAXIMUM EXPOSURE TIME OF INFINITY
(GASES) AND 30 DAYS (SOLIDS)

nuclide	400 ft	900 ft	1109 ft	dose, rem	1 mi	3 mi	19 mi
				3000 ft			
I-131	70,600	20,950	15,200	3,400	1,460	279	17.6
I-132	45	13	10	2	1	----	----
I-133	4,510	1,340	970	217	93	18	1.1
I-134	12	4	3	1	----	----	----
I-135	383	114	83	18	8	2	0.1
total iodine	75,600	22,400	16,300	3,640	1,560	299	18.8
Sr-89	2,980	884	643	144	62	11.8	0.74
Sr-90	11,500	3,400	2,470	553	237	45.4	2.86
Y-90	1,930	572	416	93	40	7.6	0.48
Ce-141	129	38	28	6	3	0.5	0.03
Ce-144	8,960	2,660	1,930	431	185	35.5	2.23
Pr-144	5	2	1	----	----	----	----
total solids	25,500	7,560	5,490	1,230	527	101	6.34

14.3.18.6.6 Ground Contamination and External Doses from Ground Contamination.

The ground deposition rates for iodine and the solid fission-product nuclides have been calculated at distances varying from 50 ft to 19 miles downwind from the containment vessel. The rate of ground deposition is given by the following equation (see p. 93 of Meteorology and Atomic Energy):

$$\text{rate of deposition (c/m}^2\text{-sec)} = \frac{2 Q V_g}{\bar{u} \pi C_y C_x d^{2-n}} \exp - \left(\frac{4 V_g d^{n/2}}{n \bar{u} \pi^{1/2} C_z} \right)$$

The release rates are $1.16 \times 10^{-8}/\text{sec}^{-1}$ and the meteorological parameters are discussed more fully in Sec. 14.3.18.6.1. In determining deposition rates the iodine isotopes are treated as vapors, and the so-called solid nuclides as particles.

In determining the ground contamination in curies/ m^2 at a time (t), account was taken of source decay within the containment vessel and the decay of material deposited before time (t).

Time-integrated doses were determined at points one meter above the ground, based on these calculated ground concentrations. The resulting doses at all points downwind were small relative to the other sources of whole body radiation and the results are therefore not presented. This would be true even if the deposition velocities were optimized to give the highest possible dose.

14.3.18.7. Total Radiation Exposure from MCA and Site Acceptability. Tables 14-16, 14-17, and 14-18 and Figs. 14.2 through 14.5 summarize the results of the calculations of whole body exposure and thyroid dosages as described in Sec. 14.3.18.6. These results permit direct evaluation of the suitability of the site with regard to the criteria defined for an exclusion area and for distances to a low population zone and to the nearest population center.

TABLE 14-16

WHOLE BODY DOSE FOR MAXIMUM EXPOSURE TIME OF 2 HR

<u>distance</u>	<u>direct radiation</u>	<u>leakage plume</u>	<u>sky shine</u>	<u>total</u>
400 ft	230	14	152	396
900 ft	29.3	4.0	27.6	60.9
1109 ft	10.9	2.90	9.65	23.5
3000 ft	0.14	0.66	0.39	1.19
1 mi	0.04	0.29	0.05	0.38
3 mi	0.01	0.06	--	0.07
19 mi	--	--	--	--

TABLE 14-17

WHOLE BODY DOSE FOR INFINITE EXPOSURE TIME (30 DAYS FOR SOLIDS)

<u>distance</u>	<u>dose, rem</u>			
	<u>direct radiation</u>	<u>leakage plume</u>	<u>sky shine</u>	<u>total</u>
400 ft	735	238	486	1460
900 ft	93.9	68.0	88.4	250
1109 ft	34.9	49.3	30.9	115
3000 ft	0.45	11.2	1.25	12.9
1 mi	0.13	5.3	0.16	5.59
3 mi	---	1.16	---	1.16
19 mi	---	0.09	---	0.09

TABLE 14-18

THYROID DOSE FOR 2-HR AND INFINITE EXPOSURE TIME

<u>distance</u>	<u>dose, rem</u>	
	<u>2 hr</u>	<u>infinite</u>
400 ft	1350	7.56×10^4
900 ft	399	2.24×10^4
1109 ft	291	1.63×10^4
3000 ft	64.9	3640
1 mi	27.8	1560
3 mi	5.34	299
19 mi	0.34	18.8

The AEC requirements in 10 CFR 100 are for:

(1) An exclusion area of such size that an individual at any point on its boundary for 2 hr immediately following the postulated fission-product release would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

(2) A low-population zone of such size that an individual at any point on its outer boundary who is exposed to the radioactive cloud resulting from the postulated fission-product release (during the entire period of its passage) would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

(3) A population center distance of at least 1-1/3 times the distance from the reactor to the outer boundary of the low-population zone.

It can be seen from Figs. 14.2 through 14.5 that the thyroid dose determines the calculated exclusion area, low population zone, and population center distance. The actual distances characterizing the site are an 1109 ft exclusion radius, a distance of three miles to the outer boundary of the low population zone, and 19 miles to La Crosse, Wisconsin, the nearest population center. These actual distances compare with the calculated values as follows:

	<u>exclusion radius</u>	<u>low population radius</u>	<u>population radius</u>
calculated:	1090 ft	<3 mi	4 mi
actual:	1109 ft (0.21 mi)	3 mi	19 mi

Thus, the calculated values of the exclusion and population radii are within the actual distances. Doses at the actual exclusion area radius of 1109 ft and at the inner radius of the low-population zone are below the limiting doses of 25 rem to the whole body and 300 rem to the thyroid. At 1109 ft the whole-body dose would be 23.1 rem and the inhalation dose to the thyroid would be 291 rem for a 2-hr exposure period. Persons at the outer radius of the low-population zone for an infinite period after the accident would receive a whole-body dose of 1.16 rem and 299 rem from inhalation of iodine.

The maximum doses received at the nearest population center (the city of La Crosse) during an infinite time after the maximum credible accident would be 0.09 rem from the whole body exposure and 18.8 rem from iodine inhalation.

The populated area nearest the site is the town of Genoa (pop. ~340). It is about one mile north of the site and lies within the low population radius. Total doses received

at this distance for the 2-hr period from the accident would be 27.8 rem from iodine inhalation and 0.37 rem from whole body irradiation. For an infinite time period after the accident the corresponding doses would be 1560 rem and 5.55 rem.

In summary, the calculated dosage associated with the maximum credible accident is indicated to meet the requirements of 10 CFR 100. These calculations are consistently conservative. Particularly conservative assumptions are:

- (1) that 100 percent rod meltdown occurs relative to a maximum predicted value of 35 percent, and
- (2) that the containment building pressure and hence the leak rate, remain at design values, despite the pressure reduction that will occur through activation of the building spray system.

VARIATION OF CONTAINMENT-BUILDING PRESSURE AND
TEMPERATURE FOLLOWING MCA

FIG. 14.1

TOTAL WHOLE-BODY DOSE FOLLOWING MCA:
2-HR EXPOSURE

FIG. 14.2

TOTAL WHOLE-BODY DOSE FOLLOWING MCA: INFINITE EXPOSURE

FIG. 14.3

DOSAGE TO THYROID FOLLOWING MCA:
2-HR EXPOSURE

FIG. 14.4

DOSAGE TO THYROID FOLLOWING MCA: INFINITE EXPOSURE

FIG. 14.5

15. LA CROSSE BOILING WATER REACTOR EVACUATION PROCEDURES

The LACBWR Operating Manual (Vol. I Sec. 3) contains detailed emergency procedures to be followed by Allis-Chalmers and Dairyland personnel in the event specific evacuation situations occur which are attributable to nuclear startup and testing programs conducted by Allis-Chalmers and commercial operations conducted by Dairyland Power Cooperative. These procedures are:

- (1) to classify specific evacuation situations
- (2) to define areas affected
- (3) to denote alarms
- (4) to designate personnel duties, responsibilities and authorities
- (5) to prescribe a course of action which is only summarized here.

15.1 EVACUATION CRITERIA

Criteria for the various situations that necessitate evacuation are divided into the following: Area Evacuation, Building Evacuation, and Site Evacuation.

If radiation levels in an occupied area within a building are unknown or other conditions make occupancy hazardous, an area evacuation is deemed necessary.

If it appears that the radiation levels within a building exceed acceptable limits or other conditions within the building make occupancy hazardous, a building evacuation is deemed necessary.

If it appears that personnel within the exclusion area will be exposed to excessive radiation doses, a site evacuation is deemed necessary.

15.2 EVACUATION ALARM SYSTEM

The LACBWR evacuation alarm system is arranged in the following manner.

15.2.1 Area Evacuation Alarm System

The area evacuation alarm system consists of the following components:

- (1) fifteen strategically located area radiation monitors and the plant mobile particulate monitor (Secs. 8.8.1.5 and 8.8.1.6) which provide local visual indications and alarms as well as remote alarming and indication in the control room, and

(2) the plant communication system (Sec. 7.8) available for use by the person sighting an unsafe condition or by the control room operator on receiving a high radiation alarm or upon being notified of a hazardous condition.

15.2.2 Building Evacuation Alarm System

The building evacuation alarm system consists of the same components as the area evacuation alarm system. In addition the containment building has a siren.

15.2.3 Site Evacuation Alarm System

The plant communication system (Sec. 7.8) is used for the evacuation of the LACBWR.

15.3 EVACUATION PLANS

LACBWR evacuation plans are divided as follows.

15.3.1 Area Evacuation Plan

Personnel will evacuate an area within the plant upon:

- (1) sighting an actuated area radiation monitor alarm,
- (2) receiving orders by way of the plant paging system to evacuate an area, and
- (3) recognizing that conditions in the area make occupancy hazardous.

Action of personnel within an area that must be evacuated is independent of the type of notification, in that the evacuee will evacuate the affected area and proceed to an area where radiation levels are known to be within exposure limits and where occupancy is deemed safe. When the evacuee has ascertained his own safety he will report conditions to the shift supervisor on duty.

15.3.2 Building Evacuation Plan

Personnel will evacuate a LACBWR building upon:

- (1) recognizing that radiation levels within the building are above normal and possibly hazardous,
 - (2) receiving orders by way of the plant paging system to evacuate the building,
- or

(3) recognizing that conditions in the building make occupancy hazardous. This includes containment building evacuation upon a sustained reactor high pressure (1350 psig) alarm in the control room since the safety valves may subsequently vent steam to the building atmosphere (Sec. 14.1.2).

A containment building evacuation can also be initiated by manual actuation of the siren in the building. There are switches for this purpose located in the control room and in the containment building adjacent to the interior door of the personnel airlock.

Action of personnel within a LACBWR building that must be evacuated is independent of the type of notification, in that personnel will evacuate the affected building and proceed to a building where radiation levels are known to be within exposure limits and where occupancy is deemed safe. When the evacuee has ascertained his own safety he will report conditions to the shift supervisor on duty.

15.3.3 Site Evacuation Plan

Personnel will evacuate the LACBWR site upon notification via the plant communication system when:

- (1) radiation levels at the site are excessive, and
- (2) conditions are such that radiation levels at the site may become excessive.

15.3.3.1 Assembly Points. Two personnel evacuation assembly points are designated as North Point and South Point. North Point is located in the Dairyland Power Cooperative Genoa Steam Power Plant. South Point is located south of the exclusion area adjacent to Highway 35.

15.3.3.2 Occurrence-Phase. When the emergency is detected, the following actions will be taken:

- (1) Site evacuation will be announced over the plant communication system.
- (2) If the reactor is operating and has not scrammed automatically, control room personnel will scram the reactor.
- (3) Control room personnel will announce the evacuation assembly point over the plant paging system.
- (4) All personnel within the LACBWR exclusion area but outside the control room, will evacuate via the nearest building exit or roadway, employing the most direct route to the designated evacuation assembly point. Shift Supervisors or control room assigned operators outside the control room will return to the control room exercising judgment to assure personal safety.

(5) Control room personnel will remain at assigned stations, exercising judgment to assure personal safety.

15.3.3.3 Immediate Action Phase - Allis-Chalmers Operating Authorization.

(1) If a Shift Supervisor is present in the control room, he will:

- (a) evaluate the emergency utilizing all available information,
- (b) account for all personnel on duty at the time emergency occurred, take appropriate emergency control, and rescue action and issue instructions to available personnel,
- (c) notify the following persons that an emergency exists, explaining his evaluation of conditions and actions taken:
 - 1. Allis-Chalmers Management Representative on call,
 - 2. Dairyland Power Cooperative Management Representative on call, and
- (d) ensure that outside assistance agencies (fire-fighting, ambulance, law enforcement, medical, etc.) are contacted if immediate help is required, provided that their presence in or near the exclusion area will not result in significant hazards to their health and safety.

(2) If a Shift Supervisor is not present in the control room and cannot be contacted, the Plant Operator assigned to the reactor control station will carry out steps (a) through (d) in Sec. 15.3.3.3.

(3) Allis-Chalmers Management Representative on call, when notified that an emergency exists, will assume the role of Emergency Control Team Director and will begin to organize an Emergency Control Team. The Emergency Control Team will include, but not be limited, to the following:

- (a) alternate Dairyland Power Cooperative Management Representative on call,
- (b) LACBWR Health and Safety Engineer and Health Physics Technicians,
- (c) Allis-Chalmers Shift Supervisor on duty,
- (d) Dairyland Power Cooperative Shift Supervisor on duty,
- (e) other Allis-Chalmers and Dairyland Power Cooperative plant personnel on duty who have been trained in emergency procedures, and
- (f) Atomic Energy Commission Site Representative.

The Emergency Control Team will be informed of the nature of the emergency and instructed to report to the designated assembly point as soon as possible. Shift Supervisors on duty at the time emergency occurred will remain at assigned stations, conditions permitting, exercising judgment to assure personal safety or until instructed to evacuate by the Control Team Director.

(4) The Dairyland Power Cooperative Management Representative on call, when notified that an emergency exists, will report the emergency to the following persons who are not members of the Emergency Control Team:

- (a) LACBWR Superintendent,
- (b) LACBWR Assistant Superintendent,
- (c) LACBWR Operations Supervisor,
- (d) LACBWR Process Engineer,
- (e) Dairyland Power Cooperative General Manager who will notify the Dairyland Publicity Representative,
- (f) Dairyland Power Cooperative Chief Engineer, and
- (g) Director, Health and Safety Division, USAEC, Chicago Operations Office.

15.3.3.4 Immediate Action Phase-Dairyland Power Cooperative Operating Authorization. When Dairyland Power Cooperative assumes responsibility for operating the Nuclear Power Plant, Allis-Chalmers duties specified in Sec. 15.3.3.3 will be performed by Dairyland personnel and the procedures set forth will remain in effect.

15.3.3.5 Recovery Phase. Upon arrival at the designated evacuation assembly point, the Emergency Control Team Director will further evaluate and analyze the emergency and will execute emergency control and recovery procedures which include:

- (1) performance of radiological surveys and establishment of the Emergency Control Area,
- (2) controlling access and occupancy of Emergency Control Area,
- (3) accounting for all personnel on duty when emergency occurred,
- (4) rescue of personnel,
- (5) medical treatment for injured personnel,
- (6) personnel decontamination,

(7) issuing calls for pre-arranged assistance from additional local, county, state and federal agencies as required,

(8) issuing calls for additional emergency assistance as required, and

(9) development and execution of recovery plans.

15.3.3.6 Emergency Equipment. Arrangements will be made to utilize locally available emergency equipment at the designated evacuation assembly points for use by the Emergency Control Team. Additional emergency equipment will be available in the control room and at the Dairyland Power Cooperative La Crosse Headquarters.

15.4 EVACUATION TRAINING

Evacuation drills will be conducted for each LACBWR operating shift semi-annually to ensure familiarity with established procedures. Personnel of local agencies likely to be called for emergency assistance will be given appropriate radiological safety training by LACBWR Health and Safety personnel.

APPENDIX A

SUMMARY DESIGN REPORT FOR
LACBWR REACTOR VESSEL AND INTERNALS

A.1 SUMMARY OF REACTOR VESSEL AND INTERNALS DESIGN REPORTS

A.1.1 STEADY-STATE PRESSURE STRESSES OF REACTOR VESSEL

Critical areas of the LACBWR reactor vessel have been subjected to a detailed stress analysis. The vessel areas which have been studied are shown in Fig. A.1. The maximum steady-state pressure stresses at these regions of discontinuity are given in Table A-1. The combined stresses of external reaction and pressure stresses are given in Table A-2. A plot of the calculated steady-state pressure stresses in the region considered is shown in Figs. A.2 through A.11.

TABLE A-1
STEADY-STATE PRESSURE INTENSITY STRESSES

vessel region (Fig. A.1)	Fig. ref.	adjacent components	stresses, lb/in. ²					max. primary stress intensity	max. primary plus secondary stress intensity
			membrane hoop	membrane axial	max. radial	max. hoop	max. axial		
A-2	1	A.2 upper closure head shell upper closure head flange	9340 --	9340 --	-1400 --	18734 --	34025 --	9985 --	34025 --
	2	A.3 vessel flange vessel cylinder shell	-- 19300	-- 8950	-- -1400	-- 24720	-- 25194	-- 19275	-- 25194
	3	A.9 lower closure head shell short cylinder	9000 19300	9000 8950	-1400 -1400	13060 16500	11465 10760	9700 19275	14460 17900
	4	A.9 short cylinder ring	19300 --	8950 --	-1400 -1400	16500 14615	10760 --	19275 --	17900 16015
	5	A.9 vessel cylinder shell ring	19300 --	8950 --	-1400 -1400	19400 --	10250 --	19275 --	19400 16015
	6	A.6 upper closure head shell & A.7 ring	9340 --	9340 --	-1400 -1400	13292 17370	17511 --	9985 --	17511 18770
	7	A.6 ring & A.7 short cylinder	-- 4440	-- 1518	-1400 -1400	17370 6934	-- 7430	-- 4325	18770 8334
	8	A.8 cover	--	--	17300	17300	--	--	17300
		3-1/2 Bolt P=0	--	--	--	--	--	--	60100
		P=1400	--	--	--	--	--	27600	52000
		1-3/8 Bolt P=0	--	--	--	--	--	--	35000
		P=1400	--	--	--	--	--	--	--

allowable total stress intensity for A-302 Gr. B A 336 (Code Case 1236) at 650 F is 39,000 lb/in.².

allowable total stress intensity for A-437 B4B at 650 F is 75,000 lb/in.².

allowable primary stress intensity for A-302 Gr. B A 336 (Code Case 1236) at 650 F is 20,000 lb/in.².

allowable primary stress intensity for A-437 B4B at 650 F is 33,300 lb/in.².

TABLE A-2

COMBINED STEADY-STATE PRESSURE AND EXTERNAL REACTIONS INTENSITY STRESSES(Allowable primary stress intensity is 20,000 lb/in.² at 650 F)(Allowable primary and secondary stress intensity is 39,000 lb/in.² at 650 F)

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location (see Figs. A. 10 and A. 11)	stress, psi							max. stress intensity at location
	from reaction loads				primary plus secondary			
	maximum loop		maximum axial		pressure stresses			
	inside	outside	inside	outside	hoop	axial	radial	
inlet circulation	188	-280	523	-821	12,188	11,523	-1400	13,588 (inside)
outlet circulation	963	-1281	642	-1012	20,163	9,442	-1400	21,563 (inside)
steam in and out	977	-1243	711	-977	22,777	9,311	-1400	24,177 (inside)
control-rod nozzles	101	-149	367	-533	9,101	9,367	-1400	10,767 (inside)
support lugs								
primary	768	-1092	2419	-3541	12,259	14,707	-1400	16,107 (inside top)
secondary	291	-535	1088	-1900				

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A.1.2 THERMAL STRESSES AND FATIGUE

The following vessel regions have been analyzed for thermal stresses and fatigue:

- (1) closure head bolting
- (2) closure head and flanges region
- (3) control rod nozzle region
- (4) recirculation outlet nozzle region
- (5) steam outlet nozzle bimetallic weld transition

The temperature distributions in the vessel were evaluated for the various transient conditions in Table A-3.

TABLE A-3
TRANSIENT CONDITIONS EVALUATED

<u>transient</u>	<u>no. of cycles</u>	<u>total change</u>	<u>rate of change</u>
1. normal startup and shutdown	150	120 F - 577.5 F	150 F/hr*
2. load change			
increase	7300	1300 - 1275 psia	2 psi/min
decrease	7300	1300 - 1325 psia	2 psi/min
3. reactor scram	250	1300 - 1100 psia	20 psi/sec
4. maximum credible accident	1	1300 - 65 psia 577.5 F - 270 F	200 psi/sec 20 F/sec
5. steam valve closure	60	1300 - 1340 psia	21 psi/sec
6. accidental bypass valve opening	20	1300 - 1100 psia	20 psi/sec

* startup analysis is based on 100 F/hr as explained below.

Transient 6 is identical to Transient 3. The temperature difference associated with Transient 2 is less than 2.6 F and therefore negligible. Only the pressure changes associated with Transient 2 need be considered. Similarly, the maximum temperature difference in the vessel for Transient 5 is less than 4 F, and only pressure changes need be considered for this transient.

The thermal stresses calculated for the vessel region under steady-state and various transient conditions are nowhere excessive. (See Figs. A.12 to A.16.) The primary and secondary pressure and steady-state thermal stress limit (i.e., three times the ASME primary Vessel Code allowable stress) is nowhere exceeded. The fatigue effect is determined by the cumulative usage factor (U), where the allowable value of $U = 0.80$. The manner in which the stress cycles are defined assumes vessel head removal with each shutdown cycle, which results in a conservative calculation of the usage factor. Any reasonable method of cyclic fatigue analysis, including superposition of individual transient cycles to determine the stress range, does not result in an increased value of the usage factor for the bolting, which is the critical component.

The startup and cooldown rate of 150 F/hr used in the analysis would result in an excessive fatigue usage factor for the closure head boltings. However, the actual startup rate will not exceed 100 F/hr. In addition, it is necessary to impose a limit of 50 F on the difference in mean temperatures between the vessel shell flange and the closure head flange. This limit may be expected to prevail during shutdown operations. For a startup rate of 100 F/hr, a cooldown rate not exceeding 150 F/hr, and operating within the mean temperature difference limitation, the cumulative usage factor for the bolting is satisfactory (<0.80). Hence, the peak stress intensity and the fatigue usage factor for the bolting (Fig. A.12) are based on a startup rate of 100 F/hr, a cooldown rate not exceeding 150 F/hr, and the mean temperature difference between the two flanges not exceeding 50 F. The peak stress intensity and the fatigue usage factor for the other regions considered (Figs. A.13 to A.16) are based on a startup and shutdown rate of 150 F/hr. Tables A-4 and A-5 list the maximum and allowable stresses and the usage factors for the various stress regions.

TABLE A-4

MAXIMUM VS. ALLOWABLE STRESSES

<u>Fig. ref.</u>	<u>region</u>	<u>total of max. primary, secondary, and thermal stress (at design temp. and press. during steady- state conditions), psi</u>	<u>Code Case 1273N-7 allowable stress, psi</u>
	steam nozzle bimetallic joints: ⁽¹⁾		
---	low-alloy steel nozzle	27,986	60,000
---	s. s. pipe	15,543	33,600
	flange:		
A.12	bolting ⁽²⁾	71,065 ⁽⁴⁾	100,000
A.13	junction of head and closure flange ⁽²⁾	44,905 ⁽⁴⁾	60,000
	control-rod nozzle:		
A.14	junction of head ⁽³⁾	22,733	53,250
A.15	head adjacent to nozzle ⁽³⁾	18,092	60,000
A.16	recirculation outlet nozzle ⁽³⁾	17,954	60,000

- (1) Fatigue analysis of this region is not necessary because of the low stress levels.
 (2) Maximum stress occurs during startup and shutdown cycle.
 (3) Maximum stress occurs during Maximum Credible Accident cycle.
 (4) No stress concentration factor.

TABLE A-5

USAGE FACTORS

Fig. ref.	region	fatigue usage factor transient						cumulative usage factor (allowable U is 0.80)
		1	2	3	4	5	6	
	flange:							
A.12	bolting	0.7500	0	0.0329	0.0026	0	0.0026	$U = \Sigma U_i = 0.7881$
A.13	junction of head and closure flange	0.0125	0	0	0.000083	0	0	$U = \Sigma U_i = 0.012500$
	control-rod nozzle:							
A.14	junction of head	0	0	0	0.00235	0	0	$U = \Sigma U_i = 0.00235$
A.15	head adjacent to nozzle	0	0	0	0.000024	0	0	$U = \Sigma U_i = 0.000024$
A.16	recirculation outlet nozzle	0	0	0	0.000053	0	0	$U = \Sigma U_i = 0.000053$

A.1.3 REACTOR INTERNAL COMPONENT STRESSES

The reactor internal components are described in detail in Sec. 4. Only the components (discussed below) that are subjected to significant loading or that require dimensional stability under operating conditions have been stress analyzed. The thickness and other characteristics of components for which stress analyses were not warranted were determined from physics and hydraulic considerations and from manufacturing requirements.

A.1.3.1 Bottom Grid Assembly

In the theoretical analysis of the grid plate assembly it is assumed that the vertical forces acting on the grid plate are resisted only by the lower grid assembly.

A.1.3.1.1 Lower Grid Assembly. The deflection and stress analysis of the grid plate is based on the following assumptions:

- (1) The lower grid plate is a grillage of intersecting beams.
- (2) The beams are supported at their ends.
- (3) Beam loading consists of:
 - (a) uniform load of 52,000 lb,
 - (b) point load from the intersection of beams,
 - (c) intermediate couples from the relative twisting of intersecting beams,
 - (d) torsional loads from the relative twisting action of intersecting beams.

The total effect of all loadings upon the deflection and slope at any nodal point is determined by superposition. This analysis includes the effect of shear upon the deflection and slope at a given point. The deflection and slope equations for each of the above loading cases were incorporated into an IBM-1620 digital computer program. The maximum calculated stress (2370 lb/in.^2) and deflection (0.006 in.) occur at the center of the grid plate.

A.1.3.1.2 Upper Grid Plate Analysis. The deflection of the upper grid plate is limited by the restraining action of the lower grid plate to a maximum of 0.006 in. The maximum stress in the upper grid plate was determined by considering the plate to be loaded uniformly so that the load produces the maximum deflection of the assembly. The maximum stress was calculated from this maximum deflection to be $\pm 282 \text{ lb/in.}^2$. The maximum thermal stress from gamma heating on the upper plate was calculated to be 8350 lb/in.^2 . The total combined stress in the upper grid plate is thus 8632 lb/in.^2 .

A.1.3.1.3 Control Rod Guide. The control rod guides that transmit the core load from the upper plate to the bottom plate of the grid plate assembly were analyzed for the following loading conditions:

- (1) Buckling as a column,
- (2) Water impulse load acting on the side of the guide,
- (3) External pressure acting on the side of the guide.

Stresses for conditions (1) and (2) above are negligible. The maximum stress occurs for condition (3) assuming a flat plate that acts as a cantilever of a length equal to half of the cruciform span and a 1-in. width, uniformly loaded with a pressure equal to the pressure drop across the core.

For normal operating conditions ($p = 2$ psi max) the maximum stress is 3750 lb/in.^2 .

A.1.3.1.4 Tie Rods. The eight tie rods on the outer periphery of the grid plate assembly transmit the load from the core, control rod guides, and bottom plate to the upper plate near its point of support on the core support skirt. The maximum tensile stress occurring on the pipes was calculated to be 1550 lb/in.^2 .

A.1.3.2 Core Support Skirt

Various regions of the core support skirt were stress analyzed. The maximum stress is at the junction of the stainless-steel cylinder and the Inconel bottom flanges and results from differential thermal expansion between the two materials.

The discontinuity effects from thermal expansion are minimized by cutouts in the bottom of the skirt. The junction of the cylinder and the ring is then formed by 12 cantilever beams. Each beam is assumed to be built in at the ring, and there is an applied moment and shear force at the opposite end where the beam joins the full cylinder skirt.

The maximum calculated stress from all loads is -7250 lb/in. outside the skirt and $+6000 \text{ lb/in.}$ inside the skirt.

Beam compression stress from the downward applied load is negligible: 640 lb/in.^2 , as compared to the critical buckling value of $44,400 \text{ lb/in.}^2$.

A.1.3.3 Plenum Separator Plate

Axial differential expansion between the stainless-steel core support skirt and the low-alloy carbon steel vessel applies a load on the annular plate at the inner radius. The magnitude of this load at operating condition depends on plate behavior under differential pressure loading. The plate is designed not to yield or become deformed under any loading condition. The loading itself limits leakage losses from the differential pressure.

A plate thickness of 0.5 in. is the optimum for this application. Sufficient bolting is provided at the vessel support ring to realize a fixed-end condition of the plate at the bolt circle. The core support skirt ring is set between 0.020 in. and 0.035 in. above the vessel ring, to provide adequate seating load. The seating load at operating conditions increases because of axial differential expansion between the vessel and the core support skirt. The maximum stress in the plate occurring at the bolt circle, calculated for the maximum deflection (0.082 in.), is 22,900 lb/in.².

A.1.3.4 Steam Dryer Closure Plate

The steam dryer closure plate analysis is based on the following assumptions:

- (1) There is essentially no pressure differential across the seal.
- (2) Since the plate is clamped at the inner radius and welded to a comparatively rigid ring at the outer radius, the edges are considered fixed at both the outer and inner radii.
- (3) The most severe loading condition is at operating temperature and zero pressure, since vessel pressurization results in an axial extension of the shell; hence the effective displacement between the inner and outer radii of the closure plate is lessened.

The maximum stress occurs in the outer radius at the ring junction, based on a maximum deflection of 0.435 in. This stress is 26,000 lb/in.².

A.1.3.5 Fuel Shroud

The maximum stress and deflection were determined by analysis, based on a minimum wall thickness for shrouds fabricated of Zircaloy and Type-304 stainless steel. The analysis is for a uniform pressure differential of 2 psi along the entire length of the shroud.

Because of the thin wall it is further assumed that the restraining effects of the bottom flange are negligible. The maximum stress occurs at the corner, and the maximum deflection is at the middle point between corners. The maximum stress for the Zircaloy is 6300 lb/in.² at a deflection of 0.015 in. That for the stainless steel is 8000 lb/in.² at a deflection of 0.006 in.

A.1.4 FUEL ASSEMBLIES

The LACBWR fuel element assembly consists of a 10 x 10 square array of fuel tubes arranged by machined grid castings at the tube ends and supported along their lengths by intermediate spacer grids. Each fuel assembly is within a stainless-steel or Zircaloy-2 shroud that rests on the core support grid. A nozzle end-fitting supports the fuel assembly in the bottom support grid, and the upper end adapter contains a handle for lifting the fuel assembly.

Except for four rods that each consist of four sections, the fuel rods are unsegmented. End plugs in the four segmented rods contain flanges that prevent any vertical grid movement. The rods are spaced on a 0.565-in. pitch and have an active length of 83 in. The fuel pins are 88.08 in. long from grid to grid, and the assembly is 102.34 in. long. The fuel pellets are 0.350 in. in diameter and have a minimum length to diameter (L/D) ratio of 1.0. At ambient temperature, the loaded pellets have a 6 mil nominal diametral clearance in the fuel tube, leaving a 3.67-in. gap at the end of the tube for the compression spring. This 3.67-in. gap also provides a containing volume for released fission gas. The fuel tube is sealed by an AISI Type-348 stainless-steel end plug. Eight of these plugs tie the 100-tube bundle to the AISI Type-304 s. s. end grid assembly. The fuel rods are supported at three intermediate axial positions between the end grid plates by spacer grids positioned vertically by flanges on the segmented fuel pin.

The fuel rods are composed of free standing cladding, filled with UO_2 pellets of 95 percent of theoretical density. The pellets and cladding are sized for a nominal diametral clearance of 6 mils, which gives a diametral clearance of 0.9 mil at maximum operating temperature. The pellets are dished a minimum of 7 mils on both ends to reduce the relative longitudinal expansion between the pellet column and the cladding. The cladding material is AISI Type-348 stainless steel with a nominal 0.020-in. wall thickness. The calculated minimum thickness to withstand collapse at the design pressure of 1450 psi at 640 F is 13 mils.

The free volume for the fission-gas space was sized to limit the hoop stress to 90 percent of the 2 percent yield strength of Type-348 stainless steel. The maximum allowable pressure for this hoop stress is 3200 psi. The design burnup of 15,000 Mwd/ST was raised to 33,000 Mwd/ST for the hot fuel pin, with an integrated radial peaking factor of 2.2 for calculating the fission gas generation. The fission gas release was conservatively assumed to be 30 percent. Fission-gas release, helium backfill at 14.7 psi, and the 35 ppm moisture content of the pellets require a free fuel-tube volume that corresponds to a tube length of 3.46 in. The free volume from the dished pellet or the diametral clearance was not considered. The volume of the spring was subtracted from the required free volume.

The thermal stresses from a radial temperature gradient of 100 F is 16,000 psi inside and 15,200 psi outside the cladding. The thermal stress from the axial temperature gradient of 100 F is 13,000 psi. These secondary stresses have no immediate detrimental effect but can become a fatigue problem. More than 1×10^6 thermal cycles are allowable, according to the alternating stresses from a modified Goodman diagram. Therefore, allowing the hoop stress to become 90 percent of the yield strength does not present a fatigue problem.

The temperature of the interface of the end pellet and end plug closure was considered. The Pathfinder Reactor fuel interface temperature was calculated to be 1100 F. Since the LACBWR power density is lower than the Pathfinder and since stainless steel has a higher conductivity than Zircaloy-2, the LACBWR interface temperature will be well below that for which there is any stainless steel- UO_2 reaction.

A vibration analysis of the fuel rods showed that, at a maximum coolant flow velocity of 6.6 ft/sec through a fuel assembly, the vibrational frequency was 8.8 cycles/sec. The amplitude of the vibration is 0.0026 in. This low value is consistent with tests on the Elk River Reactor fuel bundles that showed no measurable vibration at the low flow rate of 5 ft/sec. The stress from vibration is 120 psi, which allows an infinite number of cycles before failure.

The intermediate spacer grids are of Inconel-600 strip stock. Stainless steel was not used because of stress corrosion considerations and because the yield strength could be exceeded with small deflections. The spacer bars are 10 mils thick and can receive a deflection force of 3 mils without yielding. Thinner stock would allow greater deflection without yielding, but the grid would be difficult to fabricate, fragile, difficult to assembly, and susceptible to damage during shipping.

The support grid is made of Type-304 stainless steel casting alloy of a yield strength of 18,000 psi at 600 F. The grid was analyzed by selecting a web that contains two fuel rods that support the upper grid and two fuel rods that support the intermediate grids. Assuming that a 5-gravity load acts on the fuel assembly, the maximum bending stress is 14,010 psi on the web, and the shear stress is 6,715 psi on the circular ring of the web.

The fuel assembly bottom end adapter consists of a cast conical nozzle, shroud, and support grid. The nozzle seats in a conical surface in the fuel element shrouds that contain a ring for locking the shroud to the reactor core grid. A lug projecting from the nozzle surface prevents accidental rotation of the locking ring. The lug has been designed to accept the load of the fuel element assembly without shearing or bending. The end adapter shroud is sufficiently long to divert flow through the nozzle to the corner fuel pins.

The fuel assembly upper end adapter contains a handle, baffle, and support grid. Flow tests conducted during the failed-fuel-location-system research and development program show that the baffle is needed to obtain water samples from the outside fuel pins.

The calculated weight of the fuel element assembly is 373 lb, including 295 lb of fuel.

A.2 DESIGN AND MANUFACTURING SPECIFICATIONS

This section includes Allis-Chalmers Specification No. 41-473, which describes the fabrication and welding, inspection, and test procedures for the LACBWR pressure vessel. Figures 4.1 and 4.2 are the reference drawings (41-503-090 and 41-503-093) for the specification.

The location and details of major vessel components and welding joints are shown in Figs. A.17 through A.21.

Specification No. 41-473
Rev. 3

SPECIFICATION
FOR
LACROSSE BOILING-WATER REACTOR (LACBWR)
PRESSURE VESSEL

ALLIS-CHALMERS MANUFACTURING COMPANY
Atomic Energy Division
Bethesda, Maryland

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1. SCOPE

The vendor shall design, fabricate, inspect, test, and furnish to the Purchaser in accordance with this specification, one (1) Reactor Vessel complete and including the following:

Vessel Shell
Closure Head
Closure
Nozzles
Vessel Support

Core Support Skirt
Thermal & Shock Shields
Internal Piping, Supports & Lugs
Control Rod Nozzles
Plenum Separator Plate

2. APPLICABLE PUBLICATIONS, SPECIFICATIONS, CODES AND STANDARDS

The following codes, standards, and other publications shall form a part of this specification to the extent stated in subsequent sections.

2.1 AMERICAN SOCIETY OF MECHANICAL ENGINEERS (ASME)

(Latest edition unless otherwise noted)

2.1.1 Boiler and Pressure Vessel Code (hereafter referred to as the ASME Code) 1962 Edition

Section II Materials Specification
Section VIII Unfired Pressure Vessels, including latest revisions to the applicable Nuclear Code Cases
Section IX Welding Qualifications

2.1.2 ASA B1.4, Screw threads for high strength bolting

2.1.3 ASA B16.20, Ring Joint Flanges and Grooves for Steel Pipe Flanges

2.1.4 ASA B16.5, Steel Pipe Flanges and Flanged Fittings

2.1.5 ASA B31.1, Code for Pressure Piping

2.2 AMERICAN SOCIETY FOR TESTING AND MATERIALS (ASTM)

(Latest edition unless otherwise noted)

2.2.1 Pipe and Tubing

ASTM A376	ASTM A106
ASTM A213	ASTM A312
ASTM A335	ASTM B167

2.2.2 Forgings

ASTM A105
ASTM A336 (Modified in accordance with ASME Code Case 1236)
ASTM A182

2.2.3 Plate

ASTM A212
ASTM A302
ASTM A240

2.2.4 Bars and Billets

ASTM A276

2.2.5 Bolting

ASTM A193

ASTM A194

ASTM A437

2.2.6 Welding Rod

ASTM A316

ASTM A298

ASTM A371

2.2.7 Cladding

ASTM A264 (with ASTM A240 Cladding)

2.2.8 Test and Inspection

ASTM A370

ASTM E142

ASTM A275

ASTM E109

ASTM E165

ASTM E94

2.3 TENTATIVE STRUCTURAL DESIGN BASIS FOR REACTOR PRESSURE VESSELS AND DIRECTLY ASSOCIATED COMPONENTS, DEPT. OF COMMERCE NO. PB-151987

2.4 ALLIS-CHALMERS SPECIFICATIONS

ACP 6-31

ACP 6-61

ACP 6-32

43-101-132

ACP 6-60

43-101-147

2.5 DRAWINGS

The drawings listed below are by this reference made a part of this specification. The optional dimensions shown on these drawings do not necessarily meet all the requirements of this specification. The development and selection of the final design details shall be the Vendor's responsibility and is subject to Purchaser's approval. The reference drawings are:

41-503-090

41-503-093

2.6 PAINTING

Allis-Chalmers Specification ACP 5-46

2.7 ASSOCIATION OF AMERICAN RAILROADS, GENERAL RULES GOVERNING
THE LOADING OF COMMODITIES ON OPEN-TOP CARS

2.8 UNIFORM BUILDING CODE - 1958 EDITION

3. VESSEL DESCRIPTION AND DESIGN

3.1 GENERAL

The Reactor Pressure Vessel is to be used to contain the nuclear core of the LACBWR Nuclear Power Station. This station will generate a total of 50 net electric megawatts from the steam generated by 165 Mw of heat in the nuclear core. A direct steam cycle will be used with condensate returned at 289.3 F through two stages of feedwater heating. The steam flow at the turbine throttle will be 525,000 lb/hr at a pressure of 1265 psia and a temperature of 574 F.

The arrangement of the Reactor Vessel components and internals shall be as shown on Dwg. 41-503-090, which is part of this specification. The reactor has a single zone, single pass core. To cool the core, water enters the reactor vessel through four 16-in. inlet nozzles into an annular inlet plenum between the vessel and core support skirt. The coolant flows downward along the vessel bottom head into the core inlet plenum and then upward through the bottom grid assembly and into the active core. As the coolant passes up through the fuel element assemblies, it is heated to saturation conditions and leaves the core as a steam-water mixture. After leaving the core, the coolant is separated into steam and bulk water. A free water surface is maintained between the water and the steam space in the dome of the Reactor Vessel where most of the steam is disengaged. Located in the water annulus separating the active core from the inside surface of the thermal shield, are 16 centrifugal steam separators, 10 in. in diameter. The bulk water flows radially outward and passes through the centrifugal steam separators to completely degas the recirculating water before leaving the vessel through four 16-in. outlet nozzles located just below the bottom grid assembly. The mechanically separated steam joins the naturally separated steam above the two-phase interface and the steam passes through a chevron-type steam dryer to remove any entrained moisture before leaving the vessel through two 8-in. steam-outlet nozzles.

3.2 VESSEL SHELL

The vessel shell shall consist of a 99-in. ID cylindrical plate section, a shell flange forging, and a formed lower head. The inside height of the vessel assembly from head to head shall be 37 ft. The base metal shall be of ferritic steel (6.1) and the interior shall be fully clad with austenitic stainless steel (6.2). The form of the lower head shall be ellipsoidal or some other shape subject to approval by the purchaser. The shell flange forging shall have a minimum ID of 99 in. and shall be machined to provide the seal closure and to receive the upper head bolting. The cylindrical section shall be made so that girth and longitudinal welds are minimized in the region surrounding the active core. The vendor in his bid document shall specify the number and location of welds in this region. No attachments, internal or external, are to be made in this region. Limits of out-of-roundness of the vessel shell interior surface shall conform to the requirements of ASME Code (Section VIII) Paragraph UG-80 of (2.1.1) and meet the requirements of Section (3.8) of this specification. A total of 60 nozzles (3.5) shall be provided by the vendor and welded into the vessel shell as indicated on Dwg. No. 41-503-093.

The vendor shall design and supply the vessel external attachments required to ship, erect, and lift the vessel shell with the closure head removed. Welding nuts shall be attached to the exterior of the vessel shell and lower head to support 2-in. thick non-combustible insulation to be furnished and installed by others. The spacing of the nuts shall be a maximum of 18 in.

Twenty (20) thermocouple pads shall be attached to the vessel shell at locations to be specified later by the purchaser.

3.3 CLOSURE HEAD

The closure head shall be an ellipsoidal head or some other shape subject to approval by the Purchaser, consisting of a rolled or forged center section and a head flange forging. The closure head base metal shall be ferritic steel (6.1), and the interior shall be fully clad with austenitic stainless steel (6.2). Two 2-in. instrument lead ports and one 20-in. access nozzle as shown on Dwg. No. 41-503-093, are to be furnished by the vendor and welded into the closure head. The head flange forging shall be machined as necessary to receive double gaskets and head bolting. Suitable lifting lugs shall be provided for lifting the head. The instrument lead ports shall be provided with bolted flanges. Three (3) sets of gaskets and one (1) blind flange shall be provided for each instrument lead port. Four (4) thermocouple pads shall be attached to the closure head at locations to be specified later by the Purchaser.

3.4 CLOSURE

The vessel closure shall be of the bolted and double gasket type. Two concentric gaskets suitable for the purpose intended shall be used with the inner gasket of austenitic stainless steel or austenitic stainless steel jacketed. The gaskets shall be capable of containing full reactor pressure and leakage rates shall fall within the limits specified in (8.9). The space between the gaskets shall be connected to the gasket leakoff line. Provision shall be made for measuring gasket leakage rate. The gasket sealing surfaces in both the closure head and the shell flange forgings shall be of suitable hardened material to insure that the gaskets do not score the sealing surface. The design details of the gasket seating surfaces and the selection of gaskets shall be subject to the approval of the purchaser.

The stud bolting is to be designed in accordance with ASA B1.4 (2.1.2). Material shall be specified in Section (6). The vendor shall supply all special tools required for the removal and replacement of the bolt nuts. It shall be an objective of the design to achieve minimum removal and replacement time of the closure head. The bolting design shall include provisions for determining bolt elongation. Remote removal and replacement of the head is not a requirement of this specification. The bolts shall be designed so that they can be tensioned by bolt tensioners. The bolt tensioners will be supplied by others. Spherical washers are to be supplied to minimize stud bending or the vendor shall demonstrate that they are not required.

3.5 NOZZLES

A total of 60 vessel nozzles are required, 3 in the closure head and 57 in the vessel shell, as listed on Dwg. No. 41-503-093. Nozzle extensions are to have ends suitable for butt welding to the purchaser's pipe of the size, schedule, and material listed; the details of the weld joint shall be supplied by the purchaser. The details of the nozzle weld preparation area shall be in accordance with Section (4.7) and shall require approval by the Purchaser. All nozzle extensions shall be seamless pipe or forgings. Nozzles shall be designed to accommodate thrust loads (4.6) and thermal stresses (4.1 and 4.4) as specified in Section (5). Nozzles shall be provided with thermal sleeves as necessary. Where thermal sleeves are required, the sleeve flow diameter shall not be less than the inside diameter of the specified Purchaser's pipe. Nozzles shall be of ferritic steel (6.1) clad with austenitic stainless steel (6.2) except nozzles with nominal size 4 in. or less which may be austenitic stainless steel or some suitable material (6.4) subject to approval by purchaser. All nozzle extensions shall be of austenitic stainless steel or clad with austenitic stainless steel.

3.6 VESSEL SUPPORT

The vessel support shall be designed and supplied by the vendor. The material shall be ferritic steel (6.1) of the same type as the vessel shell. The vessel shall be supported by pads or a skirt located at the lower end of the vessel shell near the forced circulation nozzles (as shown on Dwg. No. 41-503-093). The vessel support pads or skirt is to be shimmed in the field and secured to the support structure furnished and installed by others. Wear pads or stabilizers shall be provided on the upper portion of the vessel shell as shown on Dwg. No. 41-503-093. The wear pads or stabilizers shall function to keep the vessel vertical, prevent rotation, and allow for thermal expansion in the radial and axial directions. The attachment of the vessel support to the reactor vessel shall be such that it is not the limiting factor in the establishment of the heating and cooling rates. The resultant of forces and moments acting upon the vessel through the nozzles shall be resisted by the pressure vessel support assembly. The support must also be designed to restrain the vessel against lateral movement due to seismic shock. Pertinent data for design are given in the design data Section (4).

3.7 CORE SUPPORT SKIRT

The core support skirt shall be designed and supplied by the vendor. The core support skirt shall be accurately centered and securely mounted to the inside of the lower head of the vessel so as to permit removal without cutting or distortion. The core support skirt shall be designed to support the vertical and lateral loads imposed on it by the core and the core support structure. The flange on the upper edge of the core support skirt shall be flat and normal to the vessel centerline within 0.010 in. total indicated runout, as measured after final vessel heat treatment and machining. The flange shall be machined to receive the bottom grid assembly bolting. The core support skirt shall be fabricated of austenitic stainless steel (6.2).

3.8 THERMAL AND SHOCK SHIELDS

The vendor shall provide a thermal shield which is located around the core and a thermal shock shield which is located at the top of the thermal shield, both shields are shown on Dwg. No. 41-503-090. The thermal shield shall be 1-1/4 in. $\pm$ 1/16 in. thick, right circular cylinder extending 20 ft above the vessel working point. The thermal shield shall be located to provide a 1/2 in. minimum flow annulus with the ID of the reactor vessel. The thermal shield shall be supported by a ledge attached to the vessel wall between the inlet and outlet forced circulation nozzles, as shown on Dwg. No. 41-503-090. The thermal shock shield shall be 1/4 in. thick cylinder as shown on Dwg. No. 41-503-090. The thermal shock shield shall be designed to allow cooling water and steam to circulate. The method of support and attachment of both shields to the reactor vessel shell shall be such that the flow between the shields and the vessel is not restricted. Both shields shall be designed for removal from the vessel without cutting or distorting them. The shields shall be fabricated of austenitic stainless steel (6.2).

3.9 INTERNAL PIPING, SUPPORTS AND LUGS

All internal piping, supports and lugs shall be furnished and installed by the vendor and shall be fabricated of or fully clad with austenitic stainless steel (6.2). All members which transmit loads to or from the shell of the pressure vessel shall be welded to the carbon steel base metal by removing the stainless clad and subsequently replacing the clad by overlay welding as specified in Section (6.2.5). Internal piping shall include the emergency cooling water piping to the boiler section of the core. Pipe size and location are as shown on Dwg. No. 41-503-090. The internal piping shall be adequately anchored to withstand the imposed loading due to flow, thermal expansion, earthquake, and shipping according to Section (2.3). The emergency cooling water distribution ring design shall be such that this assembly can be removed from the vessel without cutting or distorting it.

3.10 CONTROL ROD NOZZLES

The control rod nozzles, supplied and installed by the vendor in the bottom head of the pressure vessel, shall be fabricated either of clad carbon steel, Inconel, or austenitic stainless steel as specified in Section (2.2). They shall be fabricated according to the details supplied by the purchaser at a later date. A target plate shall be provided at the flange of the core support skirt. Location and alignment of nozzles shall satisfy the following requirements:

(a) Centerlines of nozzles at the far end and target points on the target plate located on the core support skirt flange shall be located on 9.009 in. $\pm$.015 in. control rod centers.

(b) Axes of nozzles inside the vessel shall intercept the target plate to within .060 in. of the corresponding target points. This requirement shall apply after final heat treatment.

Due consideration shall be given to the use of bimetallic weld joints as specified in Section (4.7).

3.11 DESIGN OF THREADED FASTENINGS AND OTHER SLIDING SURFACES

All threaded fastenings and sliding surfaces shall be designed and manufactured to prevent sticking or galling of the members after a design lifetime exposure to the operating conditions.

Stud bolts and nuts shall be threaded in accordance with the American Standard for Screw Threads for High Strength Bolting, ASA B1.4 Section (2.1.2), sizes 1 in. in diameter and smaller with the Coarse Thread Series, and sizes 1-1/8 in. in diameter and larger with the 8 pitch-thread series. Threads shall be finished for a Class 7 Fit as specified in ASA B1.4.

Bolting smaller than 5/8 in. diameter shall not be used. All threaded fastenings supplied by the vendor shall be considered to be exposed to water environments.

3.12 PLENUM SEPARATOR PLATE

The vendor shall supply a plate to separate the forced circulation inlet and outlet plenums. The plate shall be a flow barrier extending from the inner vessel wall to the core support skirt as shown on Dwg. No. 41-503-090. The plenum separator plate shall prevent coolant from bypassing the core. The plate shall be removable without cutting or distortion of any internal components.

4. DESIGN DATA

4.1 REACTOR PRESSURE VESSEL OPERATING CONDITIONS

The reactor pressure vessel shall be designed for the following conditions:

operating pressure, psia	1300
design pressure, psia	1415
operating temperature, °F	577.5
design temperature, °F	650
incident neutron flux on shell at core centerline, neutrons/cm ² -sec 1 Mev	1.9×10^{10}
heat generation for inside surface at core centerline, watts/cm ³	1*
absorption coefficient, cm ⁻¹	0.3*
maximum feedwater temperature decrease, °F/min	2.5
water temperature entering the emergency cooling water nozzle, °F	70
pressure vessel design lifetime, yr	20

4.2 WEIGHT LOADS

core, tons	22
core support structure, tons	6
water during refueling, tons	143
control rods and drives, tons	9
thermal shields and attached components, tons	35

4.3 DYNAMIC LOADS

vertical load at each control rod nozzle during reactor scram, lb (10 g deceleration)	5000
time of application of scram load, sec	0.150
seismic loading conditions for U.S. Coast and Geodetic Survey Major Damage Classification, Zone 0	2.8

4.4 PRIMARY SYSTEM TRANSIENTS

A complete description of the applicable reactor transients shall be supplied by the purchaser. This description shall include the initial conditions of the reactor coolant (flow, temperature, and pressure) at the initiation of the transient. The time variation

* Coefficients for heat generation equation $Q = Q_0 e^{-\beta x}$

of these parameters during the transient shall be specified as well as the number of occurrences. A complete cycle consists of the heating and cooling during the transient until the plant is restored to the initial conditions at start of transient. If this information cannot be adequately described in tabular form, a plot of parameter versus time shall be provided.

4.5 NOZZLE CONDITIONS

The steady state flow conditions for the vessel nozzles are tabulated below:

Saturated Steam Outlet Nozzles (J) and (K)

steam pressure, psia	1,300
steam temperature, °F	577.5
steam flow, lb/hr, total	609,732

Forced Circulation Water Inlet Nozzles (A), (B), (C) and (D)

water pressure, psia	1300
water temperature, °F	560
water flow per nozzle, gpm	7500

Forced Circulation Water Outlet Nozzles (E), (F), (G) and (H)

water pressure, psia	1300
water temperature, °F	577.5
water flow per nozzle, gpm	7500

4.6 EXTERNAL PIPE REACTIONS

The vessel and nozzles shall be designed to withstand the following loads transmitted by the connecting piping:

nozzle size, in.	8	16
axial load, lb	8,000	11,000
bending moment, lb	90,000	160,000

Checks on nozzle and shell stresses shall be made at 70°F and at the specified design temperature.

4.7 BIMETAL WELDS

Pressure containing welds between materials possessing different mean coefficients of thermal expansion (where one is less than 85 percent of the other) shall be removed a distance of $2.5 \sqrt{rt}$ from severe changes in section or environmental conditions wherever possible. In this relationship "r" is the mean radius of the cylinder and "t" is its thickness.

5. STRUCTURAL DESIGN REQUIREMENTS

5.1 CODE REQUIREMENTS

The vessel shall be designed, constructed and tested in accordance with the requirements of this specification and the ASME Boiler and Pressure Vessel Code, Section VIII, 1962 Edition, and applicable Nuclear Code Cases in effect as of the date the order is placed. It shall be stamped in accordance with the latest revision of Code Case 1270 N in effect as of the date the order is placed.

5.2 DESIGN REQUIREMENTS

In addition to the requirements of (5.1) the vessel shall be stress analyzed in accordance with Section 2.3. The allowable stresses required by the ASME Code (Section VIII) in tables UCS-23 and UHA-23 shall be used in the analysis of primary membrane stresses. The allowable stresses for transient loadings and the methods of combination of primary, secondary and transient stresses shall be in accordance with Section (2.3). The dynamic loadings given in Section (4) shall be considered in the stress analysis by the Vendor.

Adequate stress analysis shall be made of all structural discontinuities. The procedure plan for the analysis shall be approved by the Purchaser.

The nil ductility transition (NDT) temperature of all materials in the reactor vessel after 20 yr of operation at 80 percent plant factor shall be 60 F below the lowest temperature at which the vessel is to be stressed in excess of 20 percent of the design stress.

5.3 DESIGN REPORT

A complete engineering design report in sufficient detail to allow independent checking shall be submitted to the Purchaser for approval prior to fabrication. As a minimum, the design report shall include calculations made to satisfy the requirements of (5.1) and (5.2) including a detailed analysis of critical areas. In general, the areas considered critical are those locations subjected to thermal transients or shock loads such as at nozzles, vessel support, and core support skirt.

In addition to the above, preliminary design reports shall be submitted for approval prior to ordering materials if the complete engineering design report is not ready for distribution. Approval of such a preliminary design report by the Purchaser in no way relieves the Vendor of the responsibility of conforming with this specification in its entirety. The preliminary design reports shall be submitted individually with the pertinent detailed drawings.

6. MATERIALS

All materials of construction shall be furnished in accordance with the requirements of the applicable ASTM specifications of (2.2) and as hereafter specified and so certified by the material manufacturer's mill test report, such material and certificates being subject to Purchaser approval before material use.

All materials in contact with primary cooling water must be austenitic stainless steel or Inconel for corrosion resistance and shall conform to Section (6). Carbon steel material may be used provided it is clad in accordance with Section (6.2.5).

All material shall be marked with indelible ink or electrolytic stencil in rows of constantly recurring symbols at intervals of not greater than 3 ft throughout the length of the piece. Adjacent rows shall be no more than 3 ft apart, and printing in adjacent rows shall be alternately staggered.

The recurring printing shall include the following:

- Vendor's name or symbol
- Specification number and grade
- Finish designation
- Thickness
- Heat number

If indelible ink is used for marking, the ink shall be a certified chloride-free type, similar and equal to MARSH STENCIL INK (supplied by the Abbott Company, Milwaukee, Wisconsin).

6.1 FERRITIC MATERIAL

6.1.1 Plate and Forgings

Ferritic plate and forging materials shall be any combination of the following materials: ASTM A212 (Gr. B) firebox quality plate, ASTM A105 (Gr. II) forgings, ASTM A302 (Gr. B) plate and ASTM A336 (modified in accordance with ASME Boiler Code, Case 1236) forgings.

The design and dimensions for forged pipe flanges shall conform with the requirements of ASA Standard B16.5. The faces of ring joint flanges shall conform with the requirements of ASA Standard B16.20. Special flanges, such as control rod nozzle flanges, shall be approved by the Purchaser.

6.1.2 Pipe

Ferritic pipe shall be in accordance with ASTM A335 (Gr. P22) or ASTM A106 Gr. C.

6.1.3 Welding Rods

Ferritic welding rods shall conform to the requirements of ASTM A316.

6.1.4 Bolting

Ferritic studs and bolts shall conform to the requirements of ASTM A193 (Gr. B6). Nuts and Washers shall comply with the requirements of ASTM A194 (Gr. 6), or other materials as approved by Purchaser.

6.1.5 Inspection and Test

All ferritic material used in the construction of the primary pressure boundary and highly stressed structural members shall have a minimum test impact strength conforming to (8.2). There shall be a minimum of one test made, three specimens per test, for each plate, forging, pipe and lot of bolting of such material. In addition the longitudinal weld seam of the vessel shall also have a minimum test impact strength the same as the base metal and in accordance with (8.2). There shall be a minimum of one test made for each heat of cylindrical shell course material and each heat of welding material. The test bar may be welded separately from the shell course. All ferritic forgings, pipe and plate materials to be used in the construction of the primary pressure boundary and highly stressed structural members shall be ultrasonically inspected by the Vendor prior to fabrication. Inspection and acceptance shall be in accordance with (8.3). In addition all ferritic forgings shall be fully inspected by the magnetic particle method in accordance with (8.6).

6.1.6 Submerged-arc Filler Metal and Flux

Submerged-arc filler metal and flux combinations shall be proven capable of producing weld deposits having chemical compositions compatible with and mechanical properties equivalent to the base material.

6.2 AUSTENITIC MATERIAL

6.2.1 Plate and Forgings

Austenitic plate, sheet, and strip shall conform to ASTM A240 (Type 304 or Type 304L) and austenitic forgings to ASTM A182 (Gr. F-304 or Gr. F-304L).

The design and dimensions for forged pipe flanges shall conform with the requirements of ASA Standard B16.5. The faces of ring joint flanges shall conform with the requirements of ASA Standard B16.20.

Material shall be furnished in the solution heat-treated (quench annealed) condition. The parts shall be heated in the range of 1850 F to 2050 F for 1 hr/in. of thickness, but no less than 1/2 hr; followed by cooling in air or by quenching to below 1000 F.

Sheet and strip may be re-rolled following heat treatment to produce the specified finish.

Reheat-treatment shall not be performed without the prior approval of the Purchaser. In no case shall more than one retreatment be permitted.

Plate, sheet, and strip shall be furnished with the following finish:

Plate:	Hot-rolled, annealed, descaled and pickled.
Sheet:	No. 1 finish; hot-rolled, annealed and pickled.
Strip:	No. 1 finish; cold-rolled, annealed and pickled.

Hydrochloric acid or other chloride or chlorine-bearing reagents shall not be used for pickling. Where the pickling medium does not contain nitric acid, the material shall be rinsed in a dilute solution of nitric acid and demineralized water following pickling.

Forgings shall be pickled free of scale to a smooth, even white-pickle finish. Exterior surfaces of the forgings, other than welding bevels, flange faces, or other machined surfaces may have a grit-blast finish equivalent to a No. 180 grit or finer. The blasting grit shall be iron, sulfur and chloride-free silica or aluminum oxide.

6.2.2 Bars and Billets

Austenitic hot-rolled and cold-finished bars and forged billets shall conform to ASTM A276 (Type 304 or Type 304L) except as modified herein. All material shall be furnished in the solution heat-treated (quench annealed) condition. The steel shall be heated in the range of 1850 F to 2050 F for 1 hr/in. of thickness but no less than 1/2 hr, followed by rapid cooling in air or by quenching to below 1000 F.

The finish of materials furnished shall be as follows:

- Hot-forged rounds: as forged and treated; scale not removed.
- Hot-rolled rounds: turned or centerless ground.
- Cold-finished rounds: as drawn or centerless ground.
- Hot-forged squares: as forged and treated; scale not removed.
- Hot-rolled squares and flats: pickled.
- Cold-finished squares and flats: as rolled or drawn.
- Hot-rolled and cold-finished materials shall be given a final rinse in a dilute solution of nitric acid and demineralized water to condition the surface.

6.2.3 Pipe

All austenitic pipe shall conform to ASTM A312 or ASTM A376 (Gr. TP 304 or TP 316); tubing shall conform to ASTM A213 (Gr. TP 304).

6.2.4 Welding Electrodes and Rods, Bare and Coated

Filler metal for the stainless steel shall be furnished under the following ASTM specifications:

(1) Covered electrodes: ASTM A298, Class E308, E308 ELC, or other approved classes.

(2) Bare filler metal including consumable inserts: ASTM A371, Class ER308, ER308L, or other approved classes. The chemistry of austenitic stainless steel electrodes bare filler metal and/or flux shall be such as to assure 4 to 10 percent ferrite in the weld deposit.

The deposited filler weld metal for carbon steel to austenitic stainless steel welding of strength joints shall be International Nickel Company Alloy 182-T or Inco A, or equal.

For cladding by weld deposit of austenitic stainless steel on ferritic base metal, dilution of the first layer and formation of martensitic structures shall be avoided by a first layer deposit from a Class E309 electrode or ER309 rod. Other high alloy first layer deposit alloys may be used subject to agreement between Purchaser and Vendor.

6.2.5 Clad

Austenitic stainless steel clad on the interior surfaces, except hardened gasket surface, shall conform to the requirements of ASTM A264, using ASTM A240 (Type 304 or Type 304L) corrosion resisting stainless steel cladding.

If the Vendor uses overlay welding techniques to provide stainless steel clad, the following requirements shall be met in addition to the other requirements of this section:

(1) Weld overlay procedures shall be in accordance with part UCL, Section VIII, of the ASME Code;

(2) The surface of the weld cladding shall be liquid penetrant tested in accordance with Section (8.5);

(3) The surface chemistry of the overlay shall meet all the requirements of ASTM A240 (Type 304 or Type 304L);

(4) The weld deposit cladding process shall result in no gross or continuous martensitic interlayer formation.

Reheat-treatment of clad plate shall not be performed without the approval of the Purchaser.

Before applying weld deposited cladding in the production weldment, the Vendor shall demonstrate that he has sufficient control over the variables of the process to insure a

ductile, crack-free overlay. Among problems that must be considered are:

- (1) "Hot shortness"
- (2) The effect of loss of ductility due to martensitic structure.
- (3) The effect of excessive ferrite.
- (4) The effect of intermediate ferrite which transfers to sigma on heat treatment.
- (5) The effect of excessive dilution with the base metal.
- (6) The effect of current, speed, weld wire or rod, flux and miscellaneous variables upon the items listed above.

Demonstration of procedure control shall be made in the qualification test and weldment as stipulated under Section (7.2.1) by means of metallographic examination of the qualification test weldment.

The cladding shall be of sufficient thickness to provide an adequate corrosion barrier for the base metal. This corrosion barrier shall be adequate for the design lifetime of the vessel. The cladding thickness shall be 0.125 in. minimum.

Following descaling, the stainless steel surfaces shall be pickled or brightened in a solution containing nitric acid. Hydrochloric acid or reagents containing chlorine or chloride-bearing compounds shall not be used. The stainless steel shall have a smooth, even, white-pickle finish, free of scale, laps, scabs and similar defects.

The minimum shear strength between the roll bonded stainless steel cladding and the base metal for the interior surfaces shall be 20,000 psi. Both the shear test and bend tests shall be required. Vendor shall furnish a method to check thickness of clad for Purchaser approval.

6.2.6 Bolting

Austenitic studs and bolts shall conform to the requirements of ASTM A193 (Gr. B8). Nuts and washers shall conform to the requirements of ASTM A194 (Gr. 8), or other materials as approved by Purchaser.

6.2.7 Inspection and Test

All austenitic material used to contain reactor pressure or used to form a highly stressed structural member shall be ultrasonically inspected by the Vendor prior to fabrication. Inspection and acceptance shall be in accordance with (8.3).

6.2.8 Surface Finish

The surface finish of all pressure containing welds shall be ground smooth and uniform, so the deposited metal blends smoothly into the parent material, with all evidence of weld ripple removed. The automatic weld overlay cladding shall be locally ground as necessary to eliminate any craters, fissures, or excessive weld ripple and obtain a 250 RMS finish or better suitable for liquid penetrant inspection.

6.3 MARTENSITIC MATERIAL

6.3.1 Bolting

Studs, bolts, and nuts shall conform to the requirements of ASTM A437 (Gr. B4B or B4C). Bolting of this material shall be used in a dry environment only to minimize stress corrosion. Thread shape and finish shall be designed to minimize stress localization.

6.4 OTHER MATERIALS

6.4.1 Pipe

Pipe or tubing conforming to the requirements of ASTM B167 may be used subject to Purchaser's approval of the design of the specific application. Caution shall be exercised in the design of the weld joining two dissimilar materials (4.7).

7. FABRICATION

7.1 PROCEDURES

All procedures in this section require the approval of the Purchaser and shall be submitted prior to intended use of the procedure and shall not be used until approved by the Purchaser.

Welding procedures including those for welding repair require qualification before Purchaser's approval will be granted. All non-destructive tests, inspection, and heat treatment required in actual fabrication shall be applicable to the test coupons.

7.2 WELDING

All welding methods, materials, techniques, and inspection shall comply with Section VIII, Part UW of ASME Code and weld details and welder qualification shall be in accordance with the requirements of Part A of Section IX of the ASME Code and as modified herein.

7.2.1 Qualification of Welding Procedures and Welders

The Vendor shall submit the following documents for Purchaser's approval:

(1) Detailed qualified welding procedures, including methods of joint preparation and cleaning, for each joint design, together with certified procedure qualification test reports.

(2) Certified performance qualification test reports for each welder or welding operator on the job.

The Vendor shall conduct procedure and performance qualification tests for each joint configuration under each procedure, and each welder and welding operator, in accordance with the provision of the ASME Code and the special requirements of this specification.

The Vendor shall permit only welders and welding operators qualified under this specification, using procedures qualified under this specification, to weld (including tack and attachment welds) on work covered by this specification.

The welding procedure specification shall include all applicable provisions of this specification and of the ASME Code, and shall specify the values for and the limits on such welding variables as preheat temperature, current, feed rate, filler size and type, postheat temperature, etc. Test welds for qualification shall be inspected in accordance with the requirements of the ASME Code and this specification.

The welder or welding operator shall not utilize a different welding technique than that covered by the procedure under which he is qualified. Performance qualifications shall be renewed if the welder or welding operator has not performed under a given procedure in a given position, for a period of 90 days.

The guided bend test specimens of qualification test welds shall have no more than three cracks or other open defects visible on the convex surface of the specimen. The maximum dimension at any crack or other open defect shall not exceed 1/8 in., except that larger cracks occurring on the corners of the specimen during testing are acceptable if the Vendor shows that these cracks do not result from weld defects.

The essential variables listed below shall be added to the qualification requirements of the ASME Code and apply to weld overlay cladding only.

(1) For qualification the minimum base metal thickness for the test assembly shall be 2-1/2 in.

(2) A reduction in the nominal thickness or number of layers of weld overlay cladding from that used on the test assembly shall require separate qualification.

(3) Two transverse side bend tests shall be performed as part of each qualification.

7.2.2 Joint Design

(1) Backup strips may be used to make full penetration welds providing they are removed and the points of attachment are inspected after removal by the magnetic particle or liquid penetrant method.

(2) The design shall provide for a continuous surface of cladding which meets the requirements of (6.2.5) except that Inconel may be substituted for austenitic stainless steel where necessary and approved by the Purchaser.

(3) Austenitic stainless steel weld overlay shall not be applied to Inconel surfaces.

7.2.3 Precautions

The following requirements shall be met in the preparation of the joint for welding:

(1) All irregularities, slag, previously molten weld metal, oxide and other defects shall be removed by mechanical methods approved by the Purchaser after thermal or mechanical cutting.

(2) The base metal for a distance of at least 2 in. on each side of the joint area shall be clean.

(3) Parts shall be prepared to obtain accurate fitup with negligible misalignment at the weld joint to assure complete joint penetration and complete fusion at the root of the joint.

(4) When clad material is used the cladding rather than the carbon steel material shall be accurately matched at the joint. In addition, cladding shall be removed by grinding with new, unused aluminum oxide wheels or machining only. In areas where the bond between the cladding and the base metal is broken, the loose cladding shall be removed, the adequacy of the remaining bond proven by liquid penetrant test and the corrosion resistant surface restored by overlaying with weld metal.

Where both sides of the welded joint are accessible for grinding or machining and complete inspection, any approved welding process may be utilized.

Where only one side of the joint is accessible, the root pass and one filler layer shall be made by the inert-gas metal-arc (tungsten-electrode) process; and any approved process may be used for the balance of the weld. Filler metal shall be deposited during all welding.

No crevices shall be permitted on either side of the joint.

Tack welds to stainless steel shall be made under the same procedure specification as the root pass or, as an alternate, by the inert-gas metal-arc (tungsten-electrode) process. Defective tack welds shall be removed in advance of the root pass. Tack welds shall be ground to facilitate fusion.

Peening of any weld deposit or of adjacent base metal shall be performed only with the prior approval of the Purchaser.

The weld reinforcement, including both surfaces of weld, shall conform to Section VIII Part UW-51 (b) of the ASME Code. A flush surface is preferred on the inner surface of all joints. Indentation marks for the purpose of welder identification shall not be permitted.

During welding, the welding area shall be reasonably clean and adequately lighted. Provisions shall be made to prevent air movement which might impair arc shielding. Control of welding variables consistent with the procedure in use shall be established.

Where only one side of the joint is accessible, inert gas protection shall be provided on the under side of stainless steel welds to assure that the TIG root pass is in accordance with this specification.

Arc-shielding gas for inert gas welding shall be welding grade argon or helium.

7.2.4 Joint Inspection

All full penetration pressure containing strength welds shall be radiographically inspected

according to (8.4) subsequent to final heat treatment. If it is not feasible to radiographically inspect a weld due to weld geometry, the weld shall be ultrasonically inspected according to (8.3).

The acceptance of partial penetration pressure containing strength welds shall be based on liquid penetrant or magnetic particle inspection of the first pass then every other layer until completion of the weld, the inspection being in accordance with (8.5) or (8.6).

All welds on the final layer prior to stress relief, shall be examined by liquid penetrant or magnetic particle inspection according to (8.5) and (8.6), respectively.

All welds shall be re-examined by liquid penetrant or magnetic particle inspection after final heat treatment.

All welds shall be examined by liquid penetrant or magnetic particle inspection according to (8.5) or (8.6) after the root pass and the final layer. All main seam welds shall be examined by liquid penetrant or magnetic particle inspection after the root pass, an intermediate layer and the final layer.

Cladding shall be liquid penetrant inspected in accordance with (8.5) after the first cladding layer and upon completion of the weld.

All welds shall be visually inspected in accordance with (8.1).

7.3 HEAT TREATMENT

A detailed shop heat treatment procedure including the temperature, holding time, heating and cooling rates, and areas to be heat treated, shall be submitted to the Purchaser for approval. Qualification of such procedure may be required.

7.4 REPAIR

Any material containing a defect requiring repair or rejection of the material as established by visual or any of the inspections of Section (8) shall be treated in the following manner:

7.4.1 Repair of Plate, Pipe, Forgings, or Welds

These materials shall be repaired by either removing the defect or repair welding. Surface defects may be repaired without repair welding by removing material to a bottom radius at least three times the depth of the defect, provided the minimum metal thickness required by these specifications is maintained. The sides of such a repair shall be faired smoothly into the surrounding metal.

Any defect or deviation from the specification shall be immediately reported in writing to the Purchaser together with a proposed procedure for repair or replacement and a

statement of any corrective action taken by the fabricator to prevent a recurrence.

Prior approval of the repair procedure, written or verbal, is required from the Purchaser before any proposed repairs are initiated.

The only exceptions to the above requirements are the repair of minor defects which are normally repaired during welding, and the repair of minor defects that may be revealed by radiography or ultrasonic inspection after stress relieving. These repairs shall require only the prior approval of the Welding Inspector.

7.4.2 Repair of Clad

Defects in cladding shall be repaired by cutting out the defect and cladding. The procedure for repair shall be submitted to the Purchaser for approval.

7.4.3 Inspection of Repair

Inspection and acceptance of repairs shall be accomplished by the same inspection technique which located the defect.

7.5 FABRICATION PRECAUTIONS

Precautions shall be taken that all materials used in the fabrication of equipment retain identification throughout the fabrication process, and that only materials which are approved for fabrication are used in the fabrication process. Galvanized materials in contact with stainless steel shall not be used during and after fabrication. Materials containing graphite, copper, mercury, lead or chlorine, their alloys or compounds, shall not be used in the fabrication process.

All wire brushing, grinding, and disc sanding shall be done with tools which have not been used previously on any other type of material. All wire brushing of austenitic stainless steel shall be done with a new austenitic stainless steel wire brush.

7.6 CLEANING

All surfaces of the vessel shall be clean and free of iron particles, lubricant, weld spatter, chips, dirt and other foreign materials. Fluids used for cleaning, lubricating or testing which come in contact with austenitic steel surfaces shall be chloride and halogen free and careful inspection is to be performed regularly to detect chloride contamination of these surfaces from any source. After the vessel interior surface and components are cleaned and dried, the vessel shell and closures shall each be sealed off in such a manner that guarantees that the interior surfaces of both will remain clean until the seals are removed for installation of the vessel. The seals used on the vessel nozzles shall not affect the weld preparations or flange faces of the nozzles. Procedures for cleaning, including a chemical analysis of all cleaning agents and preservatives to

be used, shall be submitted to the Purchaser for approval. Subsequent to cleaning the reactor vessel and prior to drying it, all interior surfaces of the vessel are to be thoroughly rinsed with demineralized water to remove any residual cleaning agent.

7.7 PAINTING

After tests on the equipment have been completed and before shipment, the exterior surface of the vessel shall be prepared for painting by sandblasting in accordance with (2.6.1). Following this preparation, the exterior surfaces of the vessel are to receive two shop coats of paint conforming to Allis-Chalmers Specification ACP 5-46 or an equivalent paint subject to approval by Purchaser. Surfaces to be painted are to be clean and dry. Paint may be applied by brushing or spraying using suitable nozzles and air pressure. Care shall be taken to insure complete coverage.

8. INSPECTION AND TESTS

Test reports or inspection certificates shall be made for all tests and inspections conducted. One copy of all such reports and certificates shall be supplied to the Purchaser.

Certified test reports of all filler rod, consumable inserts, and welding electrodes, proposed for use shall be supplied to the Purchaser. Chemical analysis shall be made on deposits from each size and lot of filler metal except carbon steel. An all-weld-metal tension test shall be made on each lot of carbon steel electrodes per ASTM A316 for qualification purposes using the welding procedure or procedures to be used for the production weldment.

All procedures in this section require the approval of the Purchaser and shall be submitted prior to intended use of the procedure and shall not be used until approved by the Purchaser.

A complete inspection report shall be maintained of the heat number, chemical analysis, physical analysis, records of each weld, location of X-ray films with respect to the weld, and ultimate location within the finished reactor vessel of all materials used in construction. Duplicate radiographs shall be made, one of which shall be included in the inspection report. The report shall include the final "as-built" dimensions and weight of the reactor vessel and components.

The complete inspection report shall be submitted to the Purchaser prior to the shipment of the reactor vessel. The following specific requirements shall also be reported for the following materials:

ASTM A105

- Ladle Analysis
- Check Analysis
- Tensile Properties
- Ultrasonic Inspection as per (8.3)
- Impact Test as per (8.2)

ASTM A106

- Ladle Analysis
- Check Analysis
- Tensile Properties
- Bending Properties
- Flattening Tests
- Hydrostatic Test
- Ultrasonic Inspection as per (8.3)
- Impact Test as per (8.2)

ASTM A182

Ladle Analysis
Check Analysis
Tensile Properties
Hardness
Ultrasonic Inspection as per (8.3)

ASTM A193

Ladle Analysis
Tensile Properties

ASTM A194

Ladle Analysis
Hardness
Cone Stripping Test

ASTM A212

Ladle Analysis
Tensile Properties
Bending Properties
Ultrasonic Inspection as per (8.3)
Impact Test as per (8.2)

ASTM A213

Ladle Analysis
Check Analysis
Tensile Properties
Hardness
Flattening Test
Flaring Test
Hydrostatic Test

ASTM A240

Ladle Analysis
Check Analysis
Tensile Properties
Hardness
Bending Properties
Ultrasonic Inspection as per (8.3)

ASTM A264

Tensile Properties of Composite Plate

Bending Properties of Composite Plate
Test of Shear Strength between Clad and Base Metal
Ultrasonic Inspection as per (8.3)

ASTM A276

Ladle Analysis
Tensile Properties
Hardness
Ultrasonic Inspection as per (8.3)

ASTM A302

Ladle Analysis
Tensile Properties
Bending Properties
Ultrasonic Inspection as per (8.3)
Impact Test as per (8.2)

ASTM A335

Ladle Analysis
Check Analysis
Tensile Properties
Bending Properties
Flattening Test
Hydrostatic Test
Ultrasonic Inspection as per (8.3)
Impact Test as per (8.2)

ASTM A336

Ladle Analysis
Check Analysis
Tensile Properties
Ultrasonic Inspection as per (8.3)
Impact Test as per (8.2)

ASTM A312

Ladle Analysis
Check Analysis
Tensile Properties
Bending Properties or Flattening Test
Hydrostatic Test

ASTM A376

- Ladle Analysis
- Check Analysis
- Tensile Properties
- Bending Properties or Flattening Test
- Hydrostatic Test

ASTM A437

- Ladle Analysis
- Check Analysis
- Tensile Properties
- Impact Properties
- Hardness Test

ASTM B167

- Ladle Analysis
- Check Analysis
- Tensile Properties
- Bending Properties or Flattening Test
- Hydrostatic Test

Code inspection and stamping shall be accomplished in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section VIII, 1962 edition, and applicable Nuclear Code Cases in effect as of date the order is placed. The placement of the stamp shall not relieve the Vendor of the responsibility of satisfying the Purchaser that the provisions requisite to the placement of the stamp and the additional requirements of this specification have been met.

8.1 VISUAL INSPECTION

Before the start of any welding, the weld area and the base metal on each side shall be inspected using borescopes, mirrors or other visual aids as necessary. The base metal shall be clean and free of slag. Joint dimensions and fit-ups shall be verified for conformance with the drawings, the qualified procedure, and applicable provisions of the ASME Code. After each pass, and upon completion of welding, the root and face side of each weld deposit, including tack and attachment welds, in the as-welded condition and after cleaning or grinding, shall be examined, using mirrors, borescope and other visual aids as necessary. Welds having incomplete joint penetrations, cracks, incomplete fusion or overlap, pinholes, inclusions, roughness due to oxidation, undercut, improper bead contour with abrupt ridges or valleys or a bead that does not merge smoothly with base metal, weld reinforcement not conforming to (7.2.3), arc strikes or scratches, or similar defects shall be rejected. Surface finish for all surfaces of the vessel shall meet the requirements of (6.2.8).

8.2 IMPACT TESTS

The nil ductility transition (NDT) temperature shall be determined for all ferritic plate and forging material subject to accumulative irradiation during the service life of the reactor vessel in excess of 1×10^{18} n/cm² for neutrons above 1 Mev energy level. The NDT temperature shall be determined by the drop weight test method in accordance with an outline procedure acceptable to the Purchaser, on ferritic steel plate or forging material samples taken from the same heat and rolling or forging practice as the material used in the vessel assembly. Charpy impact tests shall be conducted to establish the dynamic energy absorption capacity of the ferritic materials employed in the vessel assembly. The specimens shall be of the standard size, vee-notch type complying with ASTM A370. Test procedures shall be in accordance with ASTM A370 employing not less than three specimens for each material and a test temperature of 10 F. The minimum average and the individual minimum test value shall not be less than the following:

Material	Minimum Set Average Impact Value (ft-lb)	Minimum Specimen Impact Value (ft-lb)
ASTM A212 (Gr. B)	15	10
ASTM A105 (Gr. II)	15	10
ASTM A106 (Gr. C)	15	10
ASTM A302 (Gr. B)	30	20
ASTM A336 (Mod. A302, Gr. B)	30	20
ASTM A335 (Gr. P22)	30	20

* * * * *

ASTM A193 (Gr. B6)	For information purposes only
ASTM A194 (Gr. 6)	For information purposes only
ASTM A437 (Gr. B4B)	For information purposes only
ASTM A437 (Gr. B4C)	For information purposes only

For plate, forgings, and pipe materials, the test specimen shall be removed from excess stock at a depth from the heat treated surface agreed upon by the Purchaser. For weld seams, the test specimens shall be taken from locations in accordance with Section VIII, Para. UG-84 of the ASME Code. The procedures and the orientation of the notches shall be approved by the Purchaser.

The Vendor shall supply to the Purchaser sufficient test specimens of all ferritic materials from which the vessel is fabricated, including weld metal, for irradiation in the LACBWR vessel to determine the effects of irradiation on material properties. For each ferritic material the test specimens shall consist of the following:

(1) Drop weight test specimens and standard Charpy V-notch specimens for determining initial nil ductility transition (NDT) temperature and correlating the Charpy test with the drop weight test.

(2) Standard Charpy V-notch specimens and standard tensile specimens contained in capsules to be mounted on the inside of the inner thermal shield. These specimen capsules will be periodically removed from the operating vessel to monitor the effects of irradiation on the material properties.

(3) Standard Charpy V-notch specimens and standard tensile specimens for use as control samples to insure consistent test results. Due to the periodic nature of the operation, the testing conditions may vary, hence, necessitating the use of control samples for consistent results.

The material specimens shall be obtained from the same heat and shall receive the same heat treatment as the vessel material. The Vendor shall supply the specimen capsules with provision for mounting on the inside of the thermal shield. Details of the test specimens and test specimen capsules shall be furnished to the Vendor at a later date.

8.3 ULTRASONIC INSPECTION

Ultrasonic inspection shall be performed in accordance with Allis-Chalmers Specifications No. ACP 6-31, ACP 6-32, ACP 6-60, ACP 6-61, 43-101-132, and 43-101-147. The inspection coverage shall be sufficient to insure that all defects of the sizes in (8.3.1) and (8.3.2) will be detected.

8.3.1 Acceptance of Plate, Pipe, Bars, and Forgings

Plate, pipe, or forging materials which contain unacceptable defects as classified in Allis-Chalmers Specifications No. ACP 6-31, ACP 6-32, ACP 6-60, ACP 6-61, 43-101-132 and 43-101-147, shall either be repaired or rejected unless the defect can be oriented so as to be removed during fabrication, see (7.4).

8.3.2 Acceptance of Clad

Non-bonded areas between the clad and base metal shall be repaired if permissible according to Allis-Chalmers Specification No. 43-101-132.

8.4 RADIOGRAPHIC INSPECTION

Radiographic inspection shall be the final step prior to acceptance of the weld. Radiographic practice shall conform with ASTM-E-94 and Para. UW-51 of Section VIII of the ASME Code.

The quality of radiography shall meet the requirements of ASTM Specification E142 for quality level 2-1T. Sufficient exposures shall be made to provide 100 percent weld

radiography to 1.4 percent sensitivity limits as determined by the placement of two penetrameters on each film (one for each end) to determine limits of such sensitivity. The use of one penetrameter is acceptable on seam welds 4 in. or less in diameter. All radiographs shall be subject to examination and approval by the Purchaser.

The inspection area shall include the weld and the base metal 1 in. either side of the weld. Insofar as possible, the penetrameter shall be placed on the source side of the joint and radiographs shall be made through only one wall of the plate or pipe being examined. Where it is necessary to make the radiograph through both walls of a pipe or vessel, the penetrameter thickness shall be based on a single wall thickness; only the image of the portion of the weld nearest the film shall be considered; images shall not be superimposed. The adequacy of techniques which deviate from the above must be demonstrated to the complete satisfaction of the Purchaser.

The standard of acceptance for radiographic inspection shall be in accordance with the ASME Code, Section VIII, Para. UW-51(m), except that the maximum elongated slag inclusion shall not be greater than 3/8 in.

8.5 LIQUID PENETRANT INSPECTION

Except as modified in this specification, the recommendations for inspection procedure as stated in ASTM E165 Tentative Methods for Liquid Penetrant Inspection shall be mandatory.

8.5.1 Method

The inspection method shall be either the Fluorescent Post Emulsified Penetrant Method (Procedure A-2 ASTM E165), or the Solvent Removable Visible Dye Penetrant Method (Procedure B-3 ASTM E165).

Where ASTM E165 contains alternative practices, recommendations, or suggestions, the recommended procedural details that produce the greatest sensitivity and reliability shall be used.

8.5.2 Preparation of Materials and Parts

All materials or parts to be inspected shall be cleaned before and after inspection. Abrasive blasting shall not be used. Slag and oxide shall be removed from welded joints by means that will not peen nor rough the metal surface. Wire brushes shall be used which have bristles which are compatible with the materials.

All materials or parts shall be completely free of pickling or cleaning solutions prior to drying before penetrant application.

Any specified visual inspection shall be performed before penetrant inspection.

A welded joint may be suitable for penetrant inspection without grinding provided the contour and surface finish of the weld conforms to this specification. Grinding or other methods of metal removal shall not mask defects nor otherwise produce a surface unsuitable for penetrant inspection.

8.5.3 Penetrant Application

For either penetrant, the penetration time shall be not less than 30 min.

For fluorescent post-emulsified penetrant, the emulsification time shall not exceed one minute.

For either penetrant, the developing time shall be not less than 15 min. Solvent removable penetrant shall be removed by first wiping with a clear, dry cloth, then by wiping with a clean cloth dampened with solvent.

8.5.4 Inspection

The inspection shall include the weld, the weld groove surface, the base metal for a distance of 1 in. minimum on either side of the joint and the underside of the weld where accessible. All surfaces shall be free of all cracks, laps, fissures, and other linear defects and free of linearly disposed rounded indications when there are four or more such rounded indications in a line and each is separated from the adjacent indications by less than 1/16 in. Rounded indications are any indications which are circular or elliptical with the long axis less than twice as long as the other axis and with no sharp corners. All rounded indications with maximum dimensions greater than 1/16 in. shall be removed. All defects shall be investigated, then removed by methods approved by the Purchaser. Re-inspection shall be performed to determine that all defects were removed.

After inspection and removal of defects, the material or part shall conform to all requirements of applicable specifications.

8.6 MAGNETIC PARTICLE INSPECTION

Magnetic particle inspection of welds shall be performed in accordance with ASTM E109 and forgings in accordance with ASTM A275. All surfaces of welds and forgings shall be free of linear or linearly-disposed defects.

8.7 CARBON CHECK

The carbon content of the austenitic stainless steel weld overlay clad surfaces shall be determined by chemical analysis of samples removed from no less than 0.020 in. below the surface of the finished clad. Sufficient samples shall be taken from representative locations on the flanges, nozzles, and weld seams to demonstrate the adequacy of the cladding processes. The carbon content of each sample shall be 0.080 maximum.

8.8 HYDROSTATIC TEST

The reactor vessel shall be hydrostatically tested, using demineralized water as specified in (8.9) after heat treatment and in accordance with Para. UG-99 of the ASME Pressure Vessel Code, Section VIII.

8.9 LEAK TEST

Both the inner and outer head closure gaskets are to be tested for leak tightness. The test shall be conducted with the vessel and its contents at a temperature at least 60 F above the vessel material nil ductility transition (NDT) temperature, with the gaskets which are to be used during vessel service. The vessel shall be placed in an upright position and shall be filled with demineralized water. The vessel is then to be pressurized to the vessel design pressure. There shall be a maximum of 60 cc/hr leakage from the inner gasket as measured by volume collection of water at the gasket leakoff connection. With both gaskets in place, the vessel shall be pressurized to design pressure and the gasket leakoff line pressurized to 350 psig. There shall be no detectable leakage from the outer gasket.

The water used for the hydrostatic test (8.8) and leak tests shall be commercially distilled or demineralized water having a total solids content not exceeding 1 ppm with less than 0.1 ppm of chloride. As an acceptable alternate, clean potable water having less than 5 ppm turbidity may be used providing that all surfaces exposed to the potable water are thoroughly cleaned and rinsed with demineralized or distilled water of the above specification following the test. All mechanical closures must be in place when the leak test and hydrostatic test are performed.

8.10 ADDITIONAL TESTS

Nothing in this specification shall be construed as prohibiting the Vendor from performing any additional checks, inspections or tests deemed necessary or advisable by the Vendor for satisfactory completion of the vessel or to guarantee the reliability and safety of the reactor vessel.

9. DRAWINGS AND TECHNICAL DATA

The Vendor shall prepare or cause to be prepared all drawings and technical data and submit to the Purchaser for information or approval as required by these specifications and as listed in Section (11).

10. PREPARATION FOR HANDLING AND DELIVERY

The Vendor shall provide permanent welded attachments for handling of the vessel and its components during fabrication, shipment, and subsequent installation and maintenance by the Purchaser which will protect against damage to the reactor pressure vessel and its components. The welded attachments shall provide for handling the pressure vessel assembly in the horizontal and vertical positions. The pressure vessel closure head shall be provided with welded attachments for handling in its normal position; i.e., with the flange face down. Measures shall be taken to prevent damage to the flange face during handling.

The pressure vessel shall be provided with adequate lifting lugs. The pressure vessel shall be blocked and braced to ensure safe delivery to destination in accordance with the requirements of Section (1), General Rules, Rule 1(b), Association of American Railroads Rules governing the loading of Commodities on Open Top Cars (2.7), which provides that originating carrier and shippers must confer as to appropriate blocking and bracing methods.

11. MISCELLANEOUS

11.1 ENGINEERING REPORTS AND TEST REPORTS REQUIRED

The Vendor shall supply copies of the following drawings and reports to the Purchaser:

Type of Information	Copies with Bid	Copies for Approval	Certified Copies After Approval		Copies Required Without Approval
			Prints	Reproducible Tracings	
1. Dimensioned Outline Dwgs.	1	6	6	2	
2. Assembly Dwgs.	1	6	6	2	
3. Detailed Assy. Dwgs.		6	6	2	
4. Design Report		6	6	1	
5. Fabrication Procedures		6	6	1	
6. Fabrication History			6	1	
7. Inspection & Test Pro- cedures		6	6	1	
8. Certified Mill Test Report		1 (where applicable)			1
9. Test Certificates & Inspection Report					1
10. Charts of Hydro- static Test Pressures					1
11. Charts of Stress Relief					1
12. Form U-1 ASME Boiler Code					6
13. As-built Shop Dwgs.			5	2	
14. Facsimile Rubbings of ASME Stamp and Nameplate Data					

Type of Information	Copies with Bid	Copies for Approval	Certified Copies After Approval		Copies Required Without Approval
			Prints	Reproducible Tracings	
15. Schedules		6	6	1	
16. Reports on any additional tests conducted by the Vendor		6	6	1	

General Notes:

1. "For Approval" drawings and data are required prior to intended use of such drawings and data.
2. "Certified" drawings and data are required within four weeks after return of "For Approval" drawings and data.

11.2 PHOTOGRAPHS

All unique or major parts (standard nuts, bolts, etc., excepted) are to be photographed before assembly. Sufficient views of such parts shall be furnished to completely describe the part. The photographs shall be captioned with the part name and corresponding construction drawing number. Two sets of prints and one set of negatives shall be supplied to the Purchaser.

11.3 SPARE PARTS AND TOOLS

The Vendor shall supply six spare closure studs, nuts and washers and three sets of closure gaskets. The Vendor shall submit a list of additional spare parts and tools which are to be furnished, the cost to be included in the cost of the vessel. The parts must be interchangeable with all other similar parts and shall receive the same design, fabrication and testing treatment as the original part.

11.4 INSTRUCTION BOOK

The Vendor shall prepare and supply to the Purchaser prior to shipment 30 copies of an instruction book containing instructions necessary for erection installation, operation, and maintenance of the vessel.

11.5 SCHEDULES

The Vendor shall submit within thirty (30) days of award of contract, a schedule for approval by the Purchaser. This schedule shall indicate dates at which all drawings

and procedures, material selections, and design calculations required by the specifications, will be submitted to Purchaser for approval. It shall also indicate a reasonable amount of time allowed Purchaser for review and approval.

LACBWR REACTOR VESSEL

FIG. A.1

CLOSURE HEAD- STEADY STATE PRESSURE DISTRIBUTION ($p = 1400$ psi)

FIG. A.2

CYLINDRICAL SHELL AND UPPER FLANGE -
STEADY STATE PRESSURE STRESS DISTRIBUTION ($p = 1400$ psi)

FIG. A.3

VESSEL CLOSURE AND UPPER FLANGE - STRESS DISTRIBUTION
RESULTING FROM BOLTING ONLY ($p = 0$ psi)

FIG. A.4

VESSEL CLOSURE AND UPPER FLANGE - STRESS
DISTRIBUTION RESULTING FROM BOLTING ONLY ($p = 0$ psi)

FIG. A.5

VESSEL UPPER CLOSURE HEAD PENETRATION - STEADY-
STATE INSIDE HOOP STRESS DISTRIBUTION ($p = 1400$ psi)

FIG. A.6

VESSEL UPPER CLOSURE PENETRATION - STEADY STATE
STRESS DISTRIBUTION ($p = 1400$ psi)

FIG. A.7

REACTOR VESSEL UPPER CLOSURE PENETRATION -
STEADY STATE FLAT COVER STRESS DISTRIBUTION
(24-in. bolt circumference; $p = 1400$ psi)

FIG. A.8

STEADY STATE PRESSURE INTENSITY STRESSES -
CYLINDRICAL SHELL AND BOTTOM CLOSURE HEAD ($p = 1400$ psi)

FIG. A.9

REACTOR VESSEL LOWER
SECTION STRESS INTENSITY

FIG. A.10

VESSEL CLOSURE AND UPPER FLANGE STRESS DISTRIBUTION ($p = 1400$ psi)

FIG. A.11

*Peak stress represents a mean temperature difference of 50 F between the flanges.

STRESS IN FLANGE REGION, BOLTS

FIG. A.12

STRESSES IN FLANGE REGION AT JUNCTION
OF HEAD AND CLOSURE FLANGE

FIG. A.13

CONTROL ROD NOZZLE REGION STRESSES - HEAD ADJACENT TO NOZZLE

FIG. A.15

RECIRCULATION OUTLET REGION STRESSES IN VESSEL SHELL AT NOZZLE

FIG. A.16

FIG. A.18

FIG. A.20

APPENDIX B
BURNOUT HEAT FLUX DESIGN
EQUATIONS FOR LACBWR

This Appendix was formerly issued in December 1964
as ACNP-64646 under the title BURNOUT HEAT
FLUX DESIGN EQUATIONS FOR LACBWR.

1. INTRODUCTION

Burnout heat flux data and correlations were reviewed during the design of the La Crosse Boiling-Water Reactor (LACBWR) to determine an appropriate correlation to use in the LACBWR thermal analyses. Burnout heat flux data for reactor design are obtained from electrically heated channels for the coolant conditions which approximate operating conditions in the reactor. Design correlations of the data are limited to the range of data tested, since the theoretical understanding is not adequate for a more general correlation which would cover a wider range of parameters.

At the time the Preliminary Hazards Report¹ was prepared for LACBWR (1962), the largest amount of experimented data and correlations were reported in WAPD-188². These empirical correlations were developed primarily for pressurized-water reactors. Therefore, the majority of the data were obtained with large subcooling at the inlet, small round tubes or rectangular channels, and high pressures. For a boiling-water reactor such as LACBWR, the inlet subcooling is smaller, the coolant channels are larger, and operating pressures are lower than in a pressurized-water reactor. In addition, an annular flow channel with the internal surface heated is considered to be more representative of the multirod LACBWR fuel elements than a round tube would be. Some experimental data³ for an internally heated annulus were available at the time that preliminary studies for LACBWR were made. The correlations of burnout heat flux given in Ref. 2 and 3 were used in the Preliminary Hazards Report to predict burnout heat flux at full power conditions in LACBWR, but it was stated that more applicable data would be reviewed during the design phase of LACBWR.

During the first part of the LACBWR design phase, two burnout heat flux correlations developed specifically for boiling-water reactors became available. Janssen and Levy⁴ developed design curves from experimental data for annuli at 600 to 1400 psia. Neusen and Kangas⁵ developed design curves for Pathfinder from A-C experimental data for annuli primarily at 600 psia but including some data at 200 and 400 psia. These reports and supporting data were reviewed for applicability to LACBWR design. Both design correlations adequately represent the data for the ranges of parameters for which they were developed. The design curves of Janssen and Levy were selected as the basis for LACBWR thermal analyses primarily because they include data near the LACBWR design pressure of 1300 psia and because they account for variations in mass flowrate and pressure.

During later stages of LACBWR design, additional burnout heat flux data from various sources became available. The data applicable to LACBWR (in particular, data for multirod test sections) were compared with the correlations used in the LACBWR design to test their validity against the more recent data.

The comparison of recent data with the design correlations used in LACBWR is the major purpose of this report. In addition, design curves are given for pressures and mass velocities less than those covered by the design correlations. Design curves for low pressure and low mass velocity are needed to calculate burnout heat flux for conditions encountered in reactor startup. The use of the design burnout heat flux correlations to calculate minimum burnout ratios for LACBWR under several operating conditions is described in Ref. 6.

2. SUMMARY

It is concluded that the design curves, as described in Sec. 3, represent the lower bound of experimental burnout heat flux data for LACBWR operating conditions. Recent experimental data for multirod test sections were found to lie near or above the design equations. From a total of 105 multirod data points, covering a mass velocity of 0.5×10^6 to 2.0×10^6 lb/hr ft² and pressure from 1000 to 1400 psia, only five points lie below the design curves. Of these, three are less than 10 percent and two are less than 17 percent below the design curves. For lower flows (to 0.2×10^6 lb/hr ft²) and lower pressures (to 200 psia), all data points lie above the design curves.

The application of the design curves to LACBWR with an appropriate safety margin, as described in Ref. 6, gives adequate assurance that the maximum fuel element heat flux will not closely approach or exceed the burnout heat flux. The criterion used⁶ is that the burnout heat flux as calculated from the design curves will be equal to or greater than 1.5 times the maximum fuel element heat flux for any operating conditions, including 120 percent overpower and transients. All applicable burnout heat flux data, including the few which lie below the design curves, are above the maximum fuel heat flux as determined by this criterion.

3. BURNOUT HEAT FLUX DESIGN EQUATIONS

The full power operating conditions for LACBWR are⁶: pressure, 1300 psia, core inlet subcooling, 14 F; maximum core exit steam quality, 0.11; and mass velocity in the hottest fuel element, 0.95×10^6 lb/hr ft². For power levels between 60 and 100 percent, the recirculating water flowrate varies from 43 to 100 percent, and pressure varies from 1250 to 1300 psia. The equations given in Sec. 3.1 are used to calculate burnout heat flux for significant power operation near full power design pressures. The equations given in Sec. 3.2 are used to calculate burnout heat flux for startup operations at reduced pressures. The equations should not be extrapolated beyond the specified ranges unless further experimental data confirm their validity.

3.1 OPERATING PRESSURE RANGE

The burnout heat flux design equations for pressures near reactor operating pressure are obtained from Janssen and Levy⁴ and are reproduced below. Reference 4 restricts the mass velocity between 0.5×10^6 and 2.0×10^6 . Data given in Sec. 4.2 show these equations can be extended to mass velocities of 0.2×10^6 .

For a pressure of 1000 psia and steam quality: X less than X_1 ,

$$q''_{bo}/10^6 = 0.705 + 0.237 (G/10^6). \quad (1)$$

For 1000 psia and steam quality: X between X_1 and X_2 ,

$$q''_{bo}/10^6 = 1.634 - 0.270 (G/10^6) - 4.710 X. \quad (2)$$

For 1000 psia and steam quality: X greater than X_2 ,

$$q''_{bo}/10^6 = 0.605 - 0.164 (G/10^6) - 0.653 X. \quad (3)$$

Where:

q''_{bo} = burnout heat flux, Btu/hr ft²;

G = coolant mass velocity, lb/hr ft²;

X = steam quality at the burnout point, weight fraction;

X_1 = $0.197 - 0.108 (G/10^6)$; and

X_2 = $0.254 - 0.026 (G/10^6)$.

For pressures other than 1000 psia but within the range of 600 to 1400 psia, a correction term is added to the burnout heat flux for 1000 psia as follows:

$$(q''_{boP}) = (q''_{bo})_{1000 \text{ psia}} + 440 (1000 - P) \quad (4)$$

Where:

P = reactor pressure, psia.

3.2 REDUCED PRESSURE RANGE

The burnout heat flux design equations for pressures between 200 and 600 psia are based on Allis-Chalmers data for internally heated annuli (Figs. B.1 and B.2) and Eindhoven data for seven-rod clusters.⁷ The data and equations are applicable to mass velocities between 0.2×10^6 and 2.0×10^6 lb/hr-ft² and steam qualities between 0 and 0.20.

For pressures between 400 and 600 psia,

$$q''_{bo}/10^6 = 0.95 - 3X. \quad (5)$$

For pressures between 200 and 400 psia,

$$q''_{bo}/10^6 = 0.95 - 4X. \quad (6)$$

4. DISCUSSION OF BURNOUT HEAT FLUX DATA

The LACBWR fuel elements, which are similar to those in most other boiling-water reactors, consist of fuel rods on a square pitch. Each LACBWR fuel element has 100 rods. The rods are 0.396 in. OD, and the pitch is 0.565 in., leaving an 0.169 in. space between the rods. The nominal space between the outer rods and the fuel element shroud is 0.085 in. The active fuel length is 83 in., and the total fuel rod length is 88 in. The equivalent diameter of the flow channels between rods is 0.63 in.

The heat generated in the fuel elements is removed by boiling water as it flows parallel to the fuel rods. The fuel surface temperatures are only slightly higher than water temperature as long as nucleate boiling occurs. If, however, the boiling mechanism changes to film boiling, the fuel surface temperature rises rapidly; this condition may cause fuel element failure. The heat flux from the fuel surface (Btu/hr ft²) at which this abrupt transition from nucleate boiling to film boiling occurs is commonly called "burnout heat flux." Other terms which have been used for this transition point are "critical heat flux" and "departure from nucleate boiling" (DNB). The reactor is designed so that the maximum fuel element heat flux for any operating conditions will not exceed the burnout heat flux.

The burnout heat flux data used in reactor design is obtained from electrically heated elements which simulate the geometry and coolant conditions in the fuel elements. Coolant conditions of pressure velocity and temperature can be duplicated closely, but fuel element geometry is more difficult to duplicate in an electrically heated and instrumented test section. Most of the tests have used one heated element to simulate

fuel element geometry. For boiling-water reactor fuel elements which are multirod assemblies, the geometry most frequently used^{4, 5} is a single heated rod inside a housing. Several boiling-water reactors have been successfully designed and operated based on burnout heat flux data for internally heated annuli. In the last few years, several burnout heat flux tests have been performed using multirod test sections to check correlations developed from the much more numerous single annulus data.

Theoretical methods for predicting burnout heat flux have been studied during the past few years. These studies have been useful in predicting trends of burnout heat flux with respect to test variables and in developing new correlations for a wider range of variables. However, the most reliable correlations for design purposes are empirical correlations for a limited range of variables which approximate the particular reactor conditions. The more limited empirical correlations, based on the data which are most applicable, are used as design equations for LACBWR (Sec. 3).

The coolant variables that have been investigated in burnout heat flux tests are steam quality at the burnout point, flowrate, inlet subcooling, and pressure. Of these variables, the local steam quality at the burnout point (burnout steam quality) most affects burnout heat flux. For this reason, burnout heat flux is usually plotted vs. burnout steam quality. The coolant mass velocity and pressure have shown a smaller but consistent effect on burnout heat flux. The inlet subcooling has a lesser effect, but it is not consistent. Inlet subcooling effect is most noticeable when data for large subcoolings on the order of 100 F are compared with that for small subcoolings of about 10 F. However, the data applicable to LACBWR (small subcooling) do not show an appreciable effect.

Geometry variables in the multirod tests have included heater length, rod diameter, spacing between rods, spacing between rods and channel wall, and arrangement of rods (square pitch, triangular pitch, and circular pitch). No appreciable effect is found for rod diameters close to those of the LACBWR fuel element. The test elements are also sufficiently long to present no length effect. For internally heated annuli, the eccentricity of the rod in the housing has been varied, as well as rod and housing diameters.

The most applicable burnout data reported in the past few years are compared with the LACBWR design equations (Sec. 3) in the following sections. Section 4.1 compares multirod data for a range of pressures and flowrates, which include the range to be encountered during significant power operation of LACBWR. Section 4.2 compares data for lower flowrates, and Sec. 4.3 compares data for lower pressures. Low flowrates and pressures will be encountered during startup.

4.1 DATA NEAR LACBWR OPERATING PRESSURE AND FLOWRATE

Multirod burnout heat flux data for pressures of 1000 to 1400 psia are plotted in Figs. B.3 through B.6. Most tests have been run at mass velocity increments of one-half million; therefore plots are made at 2.0×10^6 , 1.5×10^6 , 1.0×10^6 , and 0.5×10^6 lb/hr ft². Each figure shows the lower limit curves of Janssen and Levy,⁴ which were used in LACBWR design. The peak heat flux and corresponding steam quality for LACBWR⁶ are shown on Figs. B.5 and B.6. The significant dimensions of the multirod test sections and the LACBWR fuel dimensions are given in Table B-1.

The design equations give the lower bound of the large amount of data, which were available up to 1962,⁴ for annular, circular, and rectangular flow channels. Figures B.3 through B.6 show that the design equations also represent the lower bound for the burnout heat flux data for the recent multirod data applicable to LACBWR. The data for the four-rod and nine-rod geometry (which is closest to that for the LACBWR fuel) all lie above the design curves. As seen in Table B-1, the four-rod test section, the nine-rod test section, and the LACBWR fuel rods have a square array, and the spacing between the test rods is about the same as the spacing between the LACBWR fuel rods. The data for the circular arrays with a spacing about one-half that of LACBWR generally fall below the data for square arrays. Five of these data points for circular arrays with small spacings between rods lie below the design curves. In Fig. B.4, one point is 7 percent below the design curve; in Fig. B.5, one point is 10 percent below and two are 17 percent below the design curves at the pressure corresponding to the test pressure; in Fig. B.6, one point is 8 percent below the design curve. The data for the circular array with smaller spacings between rods are less significant in evaluating LACBWR burnout heat flux than the data for larger spacings. As discussed in Sec. 4.1.1, burnout heat flux decreases with smaller spacings between rods. The larger spacings of the square array are much closer to the LACBWR spacing.

The closest approach to burnout in LACBWR during steady state operating is at 100 percent flow ($G = 0.95 \times 10^6$ lb/hr ft²) and 120 percent power.⁶ Therefore, the burnout heat flux data for $G = 1.0 \times 10^6$ lb/hr ft², shown in Fig. B.5, are most significant in determining the minimum burnout safety margin. The peak operating flux and the corresponding steam quality are indicated in Fig. B.5 to show the relationship to individual data points. This peak operating point was determined by calculating heat flux steam quality and burnout heat flux as a function of length for the hottest fuel rod so that the closest approach to burnout heat flux⁶ could be determined. Figure B.5 shows that the peak operating flux is adequately below all individual data points, including those for the smaller rod-to-rod spacings. Figure B.6 shows that for 50 percent flow in LACBWR, the margin between burnout heat flux and the operating point is much larger than for 100 percent flow.

TABLE B-1

MULTIROD GEOMETRY OF THE LACBWR FUEL ELEMENT AND TEST DATA PLOTTED IN FIGS. B.3 THROUGH B.6

reference report	number of rods	type of spacer	array of rods	rod diameter, in.	rod-to-rod pitch, spacing, in.	rod-to-wall spacing, in.	length, in.
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LACBWR Fuel Element

ACNP-64626/ (Ref. 6)	100	web	square	0.396	0.565	0.169	0.085	83
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Test Sections

GEAP-3940 (Ref. 10)	9	flat rulon wafers	square	0.375	0.545	0.176	0.155	18
GEAP-4358 (Ref. 11)	4	barbells	square	0.438	0.625	0.187	0.135	36 and 48
Hw-77303 (Ref. 8)	19	wire wrap	circular	0.564	0.638	0.074	0.112	18
DP-857 (Ref. 9)	19	wire wrap	circular	0.550	0.633	0.083	0.166	36 and 72

B-7

4.1.1 Effect of Rod Spacing

The LACBWR fuel element, as well as multirod test sections, has two types of flow channels: internal flow channels within the bundle and edge flow channels formed by the outer rods and an unheated shroud or housing. The rod-to-rod spacing is the significant dimension in an internal flow channel, and the rod-to-wall spacing is the significant dimension for an edge flow channel. Rod diameters have not shown a significant effect on burnout heat flux for the small range of diameters tested (see Table B-1).

The multirod burnout data indicate that decreasing rod-to-rod spacing decreases burnout heat flux. As noted in a previous paragraph, the data of Waters⁸ (spacing = 0.074 in.) and Matzner⁹ (spacing = 0.083 in.) are slightly lower than the data of Polomick¹⁰ (spacing = 0.176 in.) and Hensch¹¹ (spacing = 0.187 in.). Waters⁸ also obtained data at 1200 psia with nineteen-rod bundles having spacings of 0.050 in. and 0.015 in. so that the effect of spacing on burnout heat flux could be determined. Waters⁸ concluded from these experiments that the nineteen-rod bundle with a 0.074 in. rod spacing has burnout heat flux comparable to the single flow channel (annular, circular, or rectangular) for outlet coolant conditions near saturated water and mass velocities of 0.5×10^6 to 4.0×10^6 lb/hr ft². For spacings of 0.050 in. and 0.015 in., the burnout heat flux was significantly below that for 0.074 in. spacing, particularly at mass velocities less than 2.0×10^6 lb/hr ft². Similarly, Matzner obtained significantly lower burnout heat flux for a 0.022 in. spacing between rods in a triangular array,¹² compared with 0.083 in. spacing in a circular array.⁹

The multirod data for small spacings between rods are not applicable to LACBWR and, therefore, are not plotted in Figs. B.3 through B.6. The minimum rod-to-rod spacing shown in the figures is the 0.074 in. data of Waters,⁸ and the maximum is the 0.187 in. data of Polomick.¹⁰ For rod-to-rod spacings within the range of those for the data plotted in Figs. B.3 through B.6, burnout heat flux is not significantly affected by rod spacing. LACBWR rod spacing (0.169 in.) falls within this range.

The burnout heat flux on an outer rod may be different from that on an inner rod for two reasons:

- (1) the outer rod has an unheated surface facing a portion of its circumference; and
- (2) the flow area associated with an outer rod may be different from that for an internal rod, causing a different mass velocity and steam quality for the outer rod. Since the corner rod of a fuel element usually has the highest heat flux, the burnout heat flux tests are designed to simulate conditions for this rod.

The internally heated annulus was selected by Janssen and Levy⁴ for burnout heat flux tests to simulate multirod geometries because it most closely simulates the corner rod of a fuel bundle. This geometry has a ratio of unheated surface area to heated area which is larger than that of a corner rod in a fuel bundle. It has been demonstrated that burnout heat flux decreases as the ratio of unheated surface area to heated surface increases. Recent experiments by Becker and Hernborg¹³ have adequately demonstrated this effect for pressures up to 530 psia. In these tests, data were obtained for an annulus having internal heating only, simultaneous internal and external heating, and external heating only. From these tests, Becker and Hernborg concluded that with equal heating on both walls, the burnout heat flux data are comparable to those for round tubes. With internal heating only, the burnout heat flux is significantly lower than that for round tubes. Therefore, for LACBWR, the internally heated annulus is considered to be the most representative of possible single channel geometries to simulate corner rod conditions in a fuel element, since the effect of unheated surface area is taken into account.

The effect of outer rod-to-wall spacing on burnout heat flux has been investigated experimentally on a few multirod assemblies and in an annulus. Janssen¹⁴ investigated the effect of displacing outwardly a corner rod of a four-rod bundle at 30 psia. The rod-to-rod spacing was 0.125 in., and rod-to-wall spacing was 0.1875 in., for symmetric rod spacing. When one corner rod was displaced to 0.094 in. from the wall, there was no decrease in burnout heat flux compared to the symmetrical position of the rods.

Janssen et al¹⁵ measured burnout heat flux at 1000 psia in an internally heated annulus in which the heater rod was displaced in the housing. Measurements were made for the concentric position with a uniform annulus of ~ 0.100 in. and for eccentric positions with minimum clearances of 0.096 in., 0.061 in., 0.033 in., and 0 in. (touching). For a steam quality of 0.08 (which is near LACBWR hot spot conditions), the 0.096 in. and 0.061 in. clearances resulted in burnout heat flux which was 92 percent of that for the concentric position. For the clearances of 0.033 in. and 0 in., the burnout heat flux was decreased to 62 percent and 50 percent of that for the concentric positions. The annular data indicate that for very small spacings (less than 0.061 in.), the burnout heat flux may decrease significantly. Both the multirod and annular data indicate that for spacings above 0.060 in., burnout heat flux does not vary appreciably with the rod-to-wall spacing.

For the LACBWR fuel elements, the nominal rod-to-wall spacing is 0.085 in. The minimum rod-to-wall spacing is limited to 0.060 in. by raised bumps on the outside of the spacer grids to prevent a large decrease in the burnout heat flux on the outer rods as a result of smaller rod-to-wall clearances.

4.1.2 Effect of Flowrate

Several experimenters have reported the effect of flowrate on burnout heat flux. It is generally concluded that in the low quality and subcooled region, the burnout heat flux decreases with decreasing flowrate; and, in higher quality regions, the burnout heat flux increases with decreasing flowrate. The limit curves of Janssen and Levy,⁴ based primarily on annular data, properly account for this effect. Figures B.3 through B.6 illustrate that for the horizontal portion of the design curves (in the low quality region), the burnout heat flux decreases with decreasing mass flowrate; and, for the sloped portion of the lines (higher quality), burnout flux increases with decreasing flowrate. The multirod data plotted in Figs. B.3 through B.6 show a decreasing heat flux with decreasing mass flowrate in the low quality region and no definite trend in the higher quality region. The design curves adequately represent the data for the range of conditions applicable to LACBWR.

4.1.3 Effect of Pressure on Burnout Heat Flux

Burnout heat flux data from annular test sections for coolant conditions of interest in LACBWR have demonstrated that the burnout heat flux is highest in a pressure range of 600 to 1000 psia and that it decreases for pressures above and below this range.^{4, 5} The limit curves⁴ are based on data in the 600 to 1400 psia range. The multirod data plotted in Figs. B.3 through B.6 are adequately represented by the limit curves, although the amount of data is too small to note any definite trend with pressure.

4.2 DATA FOR LOW FLOWRATES

Data for a pressure of 1000 psia and mass velocities less than 0.5×10^6 lb/hr ft² are given in Fig. B.7. The design curve is plotted for the average mass velocity of the data, 0.2×10^6 lb/hr ft². In addition, the peak operating flux in LACBWR for a flow of 15 percent of full flow is shown.

The burnout heat flux data points shown are from GEAP-3899¹⁶ for annuli having dimensions shown in Table B-2. Additional data at mass velocities less than 0.9×10^6 lb/hr ft² have been correlated by Weatherhead¹⁷ for tubes and rectangular channels at 2000 psia and annuli at 500 to 1000 psia. Most of the data were correlated within ± 20 percent. The correlating line and data spread is shown in Fig. B.7 for 1000 psia and range of exit steam qualities tested. As with other empirical correlations, this correlation should not be extended beyond the range of data tested. The geometry of the test sections for data used in the correlation are shown in Table B-2.

The lower limit of the data of Fig. B.7 is adequately represented by the design curve for a mass velocity of 0.2×10^6 lb/hr ft². The data of GEAP-3899 clearly show the

trend of decreasing burnout heat flux with increasing mass velocity in the higher quality region (see Sec. 4.1.2). Weatherhead's correlation of data in the low quality region did not include a term for mass velocity. Since the data were correlated within ± 20 percent, the variation with mass velocity was apparently not large.

It is concluded from the comparison of data and the design curve in Fig. B.7 that Janssen and Levy's lower limit curves⁴ can be used in LACBWR for mass velocities down to 0.2×10^6 lb/hr ft². For this low flowrate, the burnout heat flux from the design curves is greater than four times the maximum heat flux to be encountered in LACBWR.

TABLE B-2

TEST SECTION GEOMETRY FOR DATA PLOTTED IN FIG. B. 7

<u>report no.</u>	<u>channel shape</u>	<u>cross section dimensions, in.</u>	<u>length, in.</u>	<u>mass velocity 10⁶ lb/hr ft²</u>	<u>psia</u>
GEAP-3899 (Ref. 16)	annulus	1.25 ODx0.875 ID	70	0.14 and 0.28	1000
	annulus	1.00 ODx0.500 ID	29	0.20	1000
ANL-6675 (Ref. 17)	tube	0.304 ID	18	< 0.9	2000
	tube	0.436 ID	18	< 0.9	2000
	rectangle	1.00x0.097	11.0625	< 0.9	2000
	rectangle	1.00x0.050	12.0625	< 0.9	2000
	annulus	5.76 ODx2.125 ID	70	< 0.9	500-650
	annulus	2.90 ODx2.25 ID	40	< 0.9	1000

4.3 DATA FOR LOW PRESSURES

As discussed in Sec. 4.1.3, burnout heat flux decreases with decreasing pressure below 600 psia. Allis-Chalmers has obtained data at 400 and 200 psia for annuli with mass velocities of 0.5×10^6 to 2.0×10^6 lb/hr ft². These data are plotted in Figs. B.1 and B.2. Geometry of the test sections is given in Table B-3. For the range of conditions tested, mass velocity does not appreciably affect burnout heat flux. A few additional points for seven-rod cluster⁷ at mass velocities down to 0.2×10^6 lb/hr ft² are also shown. These multirod points fall close to the annular data and indicate that mass velocities down to 0.2×10^6 lb/hr ft² do not appreciably affect burnout heat flux. The design equations are conservatively drawn below all data points.

TABLE B-3

TEST SECTION GEOMETRY FOR DATA PLOTTED IN FIGS. B.1 AND B.2

<u>report no.</u>	<u>channel shape</u>	<u>cross section dimensions, in.</u>	<u>length, in.</u>	<u>psia</u>
Allis-Chalmers annulus (this report)		solid symbols 0.864 OD x 0.401 ID	72	200 and 400
		open symbols 1.049 OD x 0.469 ID	72	
		open symbols 1.049 OD x 0.587 ID	72	
		flag symbols 0.864 OD x 0.284 ID	72	
EURAEC-855 (Ref. 7)	seven- rod	0.522 rod OD	67	220 and 410
		2.800 housing ID		
		0.400 rod-to-rod space		
		0.200 rod-to-wall space		

5. REFERENCES

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6. FIGURES

BURNOUT HEAT FLUX DATA NEAR 400 PSIA

FIG. B.1

BURNOUT HEAT FLUX DATA NEAR 200 PSIA

FIG. B.2

MULTI-ROD BURNOUT HEAT FLUX DATA NEAR 1300 PSIA AND $2.0 \times 10^6 \text{ LB/HR-FT}^2$

FIG. B.3

MULTI-ROD BURNOUT HEAT FLUX DATA NEAR 1300 PSIA AND $1.5 \times 10^6 \text{ LB/HR-FT}^2$

FIG. B.4

MULTI-ROD BURNOUT HEAT FLUX DATA NEAR 1300 PSIA AND $1.0 \times 10^6 \text{ LB/HR-FT}^2$

FIG. B.5

MULTI-ROD BURNOUT HEAT FLUX DATA NEAR 1300 PSIA AND $0.5 \times 10^6 \text{ LB/HR-FT}^2$

FIG. B.6

BURNOUT HEAT FLUX DATA NEAR 1300 PSIA AND $0.2 \times 10^6 \text{ LB/HR-FT}^2$

FIG. B.7

APPENDIX C

SHIELDING CALCULATIONAL METHODS AND MODELS

1. CALCULATIONAL METHODS

Preliminary calculations to determine shielding requirements were performed using hand computations¹ and various attenuation curves. All final shielding requirements were determined using the machine calculational methods described within this appendix.

1.1 SHIELDING PROGRAM 04-2

Shielding Program 04-2² computes fast-neutron dose rates and gamma dose rates for up to 10 different gamma energies. The program, with R, θ , Z coordinates, is coded for an IBM-704 computer having 8192 words of core memory. Source and shield geometries are described by combining regions that are formed by rotating rectangles or trapezoids about the source axis or by rectangles parallel to the source axis. Compositions are expressed as volume fractions of different materials. Each region is associated with a composition, by a code number, for as many as 20 compositions, 10 materials, and 120 different regions.

Program 04-2 uses trapezoidal integration over the source region adjusted to integrate over volume, surface, or line sources. The attenuation calculated by the program is along a straight line between a source point and a dose point.

1.1.1 Source Description

In Program 04-2, the source strength was assumed to be independent of the angle, θ , and is separable into the product of a radial and axial source strength. Axial and radial source distributions having either cosine or exponential form were used:

$$S(R) \text{ or } S(Z) = \tau_1 \cos \left[\tau_2 (Z_S - \tau_3) \right] + \tau_4$$

$$S(R) \text{ or } S(Z) = \tau_1 e^{\tau_2 (Z_S - \tau_3)} + \tau_4$$

The above source distributions may be the same or different for gammas and neutrons and may be described differently in as many as four ranges between the limits of integration. There may be from 1 to 15 equally spaced shells and planes and from 1 to 15 lines of point sources in each shell.

1.1.2 Neutron and Gamma Attenuation Functions

The neutron and gamma attenuation functions are a product of a material function and a spatial function, i.e., $1/4 \pi \rho^2$. The material function for fast neutrons, using a fast removal cross section is:

$$\psi_n(\sum_m \theta_m \rho) = \exp \left[-\rho \sum_{m=1}^m \theta_m \Sigma R(m) \right]$$

The fast neutron dose rate was then determined by the following equation:

$$D_n = \frac{C}{4\pi} \int_{Z_1}^{Z_2} S(z) dz \int_{R_1}^{R_2} S(R) R dR \int_{\phi_1}^{\phi_2} \frac{\psi_n(\sum_m \theta_m \rho)}{\rho^2} d\phi$$

where C is a constant (specified as input) used as a multiplying factor when source symmetry occurs.

The material attenuation function for gammas is:

$$\psi_\gamma(\mu_m(E_i) \theta_m \rho) = B(\mu(E_i) \theta_m \rho) e^{-\rho \sum_{m=1}^m \theta_m \mu_m(E_i)}$$

where: $B(\mu(E_i) \theta_m \rho)$ is a buildup factor.

For these calculations, a linear buildup factor was used of the form:

$$B = 1 + k \sum_{m=1}^m (\mu_m(E_i) \theta_m \rho)$$

where:

k = a constant equal to $(\mu_o - \mu_e)/\mu_e$,
 μ_o = the total absorption coefficient (cm^{-1}), and
 μ_e = the energy absorption coefficient (cm^{-1}).

The gamma dose rate is then determined by the following equation:

$$D_\gamma = \sum_{i=1}^J \frac{CK(E_i) \Gamma(E_i)}{4\pi} \int_{Z_1}^{Z_2} S(Z) dZ \int_{R_1}^{R_2} S(R) R dR \int_{\phi_1}^{\phi_2} \psi_\gamma \frac{(\mu_m(E_i) \theta_m \rho)}{\rho^2} d\phi$$

where:

$\Gamma(E_i)$ = the source intensity term for each source energy, and

$K(E_i)$ = a conversion factor to change $\gamma/\text{cm}^2 \cdot \text{sec}$ to any desired unit.

1.2 NEUTRON FLUX

Because it was desirable to have an accurate representation of the fast and thermal fluxes in the radial direction near the vessel wall and the nuclear instruments, a PIMG calculation was performed radially to a point in the concrete biological shield outside the pressure vessel. For greater penetrations, the fast flux from shielding code 04-2 was used.

1.2.1 Axial Thermal-Neutron Flux

After the fast neutron flux axially throughout the shield was known, the axial thermal-neutron flux was determined by the following relationship:

$$\phi_{th}(x) = \frac{\Sigma_R}{\Sigma_a} \phi_f(x-d)$$

where d represents the slowing-down distance for neutrons and is the ratio of the neutron age to the mean free path for slowing down (τ/λ_R), and Σ_R/Σ_a gives the asymptotic thermal-to-fast flux ratio. This selection of d assumed that the fast neutrons are directed axially away from the core and was conservative for the actual fast-neutron angular distribution, especially in the reflector region. For the axial case below the core, the grid plate was neglected, and the water reflector was assumed adjacent to the core.

1.3 SCATTERING CALCULATIONS

Scattering occurs around or through the many penetrations and doorways in the shielding walls. All scattering through ducts and pipes was calculated by the Simon-Clifford equations.¹

Figure C.1 shows the model used to calculate the scattering of gammas around doorways and shadow shielding. A slab source was formed for a section of the wall where the scattering takes place. The source strength is:

$$S_V = \mu_S \phi_1 \psi$$

where:

μ_S is the gamma scattering coefficient (cm^{-1}),

ϕ_1 is the incident gamma flux ($\gamma/\text{cm}^2\text{-sec}$)

ψ is $B_d(\mu_o X) e^{-\mu_o X}$, and

X is the thickness of slab (cm).

The source, $\mu_S \phi_1$ is assumed to be isotropic. Using shielding code 04-2 and the above source strength, the ratio of the reflected flux to the incident flux was determined at various distances from the slab. The ratio of reflected to incident gammas is shown in Fig. C.2.

The calculations were conservative, since the scattering around doorways is mostly straight ahead and not isotropic as was assumed and because no allowance was taken for gamma energy loss from scattering collisions. Since scattering sources did not dominate any of the dose levels, a more detailed consideration is not required.

1.4 GAMMA HEATING

The heat generation in various materials was determined by the following:

$$H_\gamma = \mu_e \phi_\gamma E \text{ mev/cm}^3\text{-sec}$$

where:

μ_e is the gamma-ray-energy absorption coefficient (cm^{-1}),

ϕ_γ is the gamma-ray flux ($\gamma/\text{cm}^2\text{-sec}$), and

E is the gamma-ray energy (mev).

1.5 FAST-NEUTRON HEATING

The fast-neutron heating was determined as follows:

$$H_n = \Sigma_R \phi_n E$$

where:

Σ_R is the fast-neutron-removal cross section (cm^{-1}),

ϕ_n is the fast neutron flux ($\text{n/cm}^2\text{-sec}$), and

E is the fast-neutron energy (mev).

1.6 SHIELDING CONSTANTS

Tables C-1 through C-5 are a tabulation of all constants (Refs. 1 and 3) used in the shielding calculations for LACBWR. The constant k is the constant used in the linear buildup factor ($B_d = 1 + k\mu_o\rho$).

2. ACTIVITY SOURCES AND CALCULATIONAL MODELS

The main sources of radioactivity in LACBWR and some aspects in the measurement and shielding requirement calculations are discussed in the following subsections.

2.1 CONTAINMENT BUILDING

The main sources of activity in the reactor building are: the operating core, capture gammas in the shield materials, the spent fuel, the activation product gammas and fission product gammas carried by the primary water and steam, and radioactivity from activated equipment. These sources are discussed below. All shrouds involved in the calculations were assumed to be of stainless steel. Because of the higher energy capture gammas resulting from neutron capture in the stainless steel as compared with gammas from capture in zirconium, this assumption is conservative.

2.1.1 Operating Core

2.1.1.1 Core Gammas. The core gamma sources are those gammas (described below) produced by fission, capture, and inelastic scattering reactions. For calculations, the gamma spectrum is divided into seven energy groups, which are used for operating core gammas, core capture-gammas, and capture gammas produced in the shielding materials surrounding the core.

2.1.1.1.1 Prompt Gammas. The prompt gammas are those emitted during fission in apparent coincidence with the fission event. The prompt-gamma ray spectrum is approximated by the following expression:

$$N(E) = 7.8 e^{-1.03E^{5.1}} \gamma/\text{fission} - \text{Mev (Ref. 4)}$$

TABLE C-1

GAMMA ATTENUATION COEFFICIENT (cm^{-1})

energy Mev	* μ_o H_2O	μ_o ordinary concrete	μ_o steel	μ_o UO_2	μ_o lead	μ_o heavy concrete
0.25	0.127	0.271	0.965	7.6	7.10	0.295
0.3	0.115	0.254	0.86	4.79	6.02	0.280
0.7	0.082	0.175	0.54	1.19	1.20	0.263
1.0	0.07	0.150	0.46	0.789	0.82	0.224
1.5	0.058	0.120	0.38	0.550	0.60	0.196
2.5	0.044	0.094	0.29	0.436	0.49	0.150
4.0	0.034	0.076	0.26	0.423	0.48	0.117
6.0	0.027	0.065	0.24	0.444	0.50	0.103
8.0	0.024	0.057	0.232	0.468	0.521	0.097
9.0	0.023	0.055	0.231	0.479	0.538	0.095

TABLE C-2

GAMMA ENERGY ABSORPTION COEFFICIENT (cm^{-1})

energy Mev	* μ_e H_2O	μ_e ordinary concrete	μ_e steel	μ_e UO_2	μ_e lead	μ_e heavy concrete
0.25	0.031	0.068	0.275	6.85	6.33	0.120
0.3	0.032	0.069	0.29	3.48	5.22	0.115
0.7	0.0325	0.068	0.215	0.735	0.76	0.105
1.0	0.031	0.065	0.20	0.456	0.51	0.098
1.5	0.028	0.060	0.175	0.340	0.365	0.087
2.5	0.0245	0.054	0.167	0.284	0.33	0.083
4.0	0.021	0.048	0.175	0.333	0.36	0.077
6.0	0.0185	0.044	0.180	0.361	0.42	0.076
8.0	0.0173	0.042	0.188	0.418	0.467	0.075
9.0	0.0171	0.041	0.192	0.426	0.488	0.076

*based on $\rho = 1 \text{ gm/cm}^3$ (μ is corrected for actual density of water when used in calculations)

TABLE C-3

CONSTANT FOR THE LINEAR BUILDUP FACTOR

energy Mev	k H ₂ O	k ordinary concrete	k iron	k UO ₂	k lead
0.25	3.11	2.95	1.99	0.13	0.15
0.3	2.6	2.7	1.97	0.38	0.15
0.7	1.52	1.57	1.51	0.62	0.58
1.0	1.26	1.31	1.30	0.73	0.61
1.5	1.07	1.0	1.17	0.62	0.64
2.5	0.80	0.74	0.76	0.53	0.48
4.0	0.62	0.58	0.49	0.27	0.33
6.0	0.46	0.48	0.33	0.23	0.19
8.0	0.39	0.35	0.23	0.12	0.12
9.0	0.36	0.31	0.19	0.13	0.10

TABLE C-4

GAMMA DOSE RATE CONVERSION
FACTORS

E(Mev)	C.F. (10 ⁻⁶) ^(a)
0.25	0.47
0.3	0.56
0.7	1.30
1.0	1.83
1.5	2.45
2.5	3.65
4.0	4.75
6.0	6.60
8.0	8.3
9.0	9.5

TABLE C-5

FAST NEUTRON REMOVAL
CROSS SECTIONS

material	R(cm ⁻¹)
H ₂ O (ρ = 1.0)	0.103
Fe	0.17
Pb	0.117
concrete (ordinary)	0.09
UO ₂	0.17

(a) converts γ flux (γ/cm²-sec)
to dose rate (R/hr)

2.1.1.1.2 Delayed Gammas. The delayed gammas are those emitted during the decay of the fission products. The delayed gamma ray spectrum during operation is approximated by the following expression:

$$N(E) = 8 e^{-1.33E^{5.2}} \gamma/\text{fission} - \text{Mev. (Ref. 5)}$$

2.1.1.1.3 Core Capture Gammas. The capture gammas⁶ in the core are mainly produced by the (N, γ) reactions with water, stainless steel, U-238, and U-235. The capture reaction rate, multiplied by the number of gammas emitted per capture, is divided by the fission rate to obtain the capture gammas per fission. Water has one 2.2-Mev gamma (a Group III gamma) per capture. Table C-6 gives the gamma disintegrations per capture⁷ for stainless steel and U-238. Uranium-235 capture gammas - 20 percent of all prompt gammas (capture-to-fission ratio - 0.20) - are assumed to have the same spectrum as the prompt gammas.

2.1.1.1.4 Inelastically Scattered Gammas. Inelastically scattered gamma sources produce one Group I gamma per fission.

Table C-7 gives all core gamma contributions (in gammas per fission) by energy group and gives the core gamma volume sources, which are determined by the equation:

$$S_V = \frac{(\gamma/f) (P) 3.1 \times 10^{10}}{V}$$

where P is the power in watts and V is the volume ($3.9 \times 10^6 \text{ cm}^3$).

2.1.1.2 Core Neutrons. In determining the fast neutron volume source, it is assumed that 2.46 neutrons are emitted per fission.

The neutron volume source is determined by the equation:

$$S_V = \frac{(n/f) (P) 3.1 \times 10^{10}}{V}$$

2.1.2. Capture Gammas in Shielding Materials

2.1.2.1 Water Capture-Gammas. One gamma per capture is produced by an (n, γ) reaction in the water reflector. The water capture gamma, a Group III gamma, has a volume source (S_V) of $\Sigma_c \phi_{th}$, where ϕ_{th} is the thermal flux at the point of interest. The 2200 m/sec capture cross section (Ref. 8) for hydrogen is 0.35 barns.

TABLE C-6

GAMMA DISINTEGRATIONS: U-238 AND STAINLESS STEEL

group	I	II	III	IV	V	VI	VII
E (Mev)	1.0	1.5	2.5	4.0	6.0	8.0	9.0
s. s: γ /capture	0.8	0.54	0.27	0.22	0.22	0.39	0.0036
U-238: γ /capture	2.54	1.78	0.91	0.34	--	--	--

TABLE C-7

CORE GAMMAS AND CORE VOLUME SOURCES

group energy range Mev	I 0-1	II 1-2	III 2-3	IV 3-5	V 5-7	VI 7-9	VII 9
ave. E, Mev	1.0	1.5	2.5	4.0	6.0	8.0	9.0
prompt	4.88	1.73	0.62	0.31	0.037	0.0048	--
delayed	4.42	1.17	0.31	0.10	0.0073	0.00047	--
H ₂ O capture	--	--	0.082	--	--	--	--
stl. capture	0.25	0.167	0.084	0.068	0.068	0.121	0.0074
U-235 capture	0.94	0.33	0.12	0.06	0.0072	0.00095	--
U-238 capture	0.333	0.233	0.199	0.045	--	--	--
inelastically scattered	1.0	--	--	--	--	--	--
total (γ /f)	11.82	3.63	1.34	0.58	0.12	0.127	0.0074
S_V (γ /cm ³ sec)	1.67×10^{13}	5.4×10^{12}	1.97×10^{12}	8.6×10^{11}	1.74×10^{11}	1.85×10^{11}	1.1×10^{10}

2.1.2.2 Stainless-Steel Capture Gammas. These capture gammas are considered for the thermal shield and the vessel. The 2200 m/sec capture cross section⁸ for stainless steel is 3.15 barns, and the gamma disintegrations per capture are given in Table C-6.

2.1.2.3 Lead Capture Gammas. This source produces one Group VI gamma per capture. The 2200 m/sec capture cross section⁸ here is 0.17 barns.

2.1.3 Spent-Fuel Decay Gammas

Decay gammas become important during shutdown and in the transfer of spent fuel elements. The decay gammas are calculated from the table by Scoles⁹ for a decay time of 24 hr and a reactor operating time of 2.7 yr. The decay gamma groups and their respective gamma sources are given in Table C-8. The sources listed are for stored cores and spent fuel elements.

TABLE C-8

SPENT-FUEL SOURCES

<u>E, (Mev)</u>	<u>S_v (γ/cm³-sec) (1-day decay)</u>
1.0	5.8 × 10 ¹¹
1.5	1.2 × 10 ¹¹
2.5	2.2 × 10 ¹⁰
3.5	2.1 × 10 ⁷
4.5	5.4 × 10 ⁶

2.1.4 Primary Water Radioactivity Sources

The primary water radioactivity sources are N-16, O-19, activated corrosion products, and fission products from 44 failed-fuel pins.

2.1.4.1 Nitrogen-16. Nitrogen-16 is formed by an (n, p) reaction with oxygen, with a threshold energy of 9.6 Mev:

Eighty percent of the disintegrations emit a 6-Mev gamma and a 4-Mev beta; the rest emit betas to the ground state of O-16.

Nitrogen-16 has a half-life of 7.3 sec. The cross section for the reactor is 18.5 microbarns¹⁰, and, at full power, the reaction rate is 8.8×10^6 absorptions/cm³-sec.

2.1.4.1.1 Nitrogen-16 in Reactor Water. The circulation of reactor water is shown schematically in Fig. C.3. Using this sketch, the buildup of N-16 is calculated as follows:

$$\frac{dN}{dt} = \Sigma_{np} \phi - \lambda N \quad \text{from A to B}$$

$$N_A = N_B e^{-\lambda t_o} \quad \text{from B to A}$$

Combining and solving the above equations yields:

$$N_B = \frac{\Sigma_{np} \phi (1 - e^{-\lambda t_c})}{\lambda (1 - r e^{-\lambda t_t})}$$

where: N_A is the concentration of N-16 at the core inlet,
 N_B is the concentration of N-16 at the core outlet,
 λ is the decay constant for N-16 ($0.094, \text{sec}^{-1}$),
 t_c is the transit time in the core (1.11 sec),
 t_o is the recirculation time outside the core (21.79 sec),
 t_t is the total time ($t_c + t_o$)
 (based on a recirculation rate of 30,000 gpm and a total water volume of 1527 ft³),
 $\Sigma_{np} \phi$ is the energy and space average of production rate of N-16 (8.8×10^6),
 and
 r is the ratio of recirculation to core flow rates (0.943).

Evaluating this for full power operation gives:

$$N_B = 1.01 \times 10^7 \text{ atoms of N-16 per cm}^3 \text{ of water or } 6.45 \times 10^9 \text{ atoms of N-16 per pound of water.}$$

The volume source of N-16 in the reactor water is:

$$S_v = \lambda N_B N_Y = 7.6 \times 10^5 \text{ } \gamma/\text{cm}^3\text{-sec.}$$

The volume source of N-16 in the reactor water at the recirculation pump, assuming an 11.45 sec travel time to the pump, is $2.6 \times 10^5 \text{ } \gamma/\text{cm}^3\text{-sec.}$

2.1.4.2 Oxygen-19. Oxygen-19 is formed by an (n, γ) reaction with oxygen-18.

Oxygen-19 emits a 1.36 Mev gamma for 50 percent of the decays. The cross section for the reaction is 0.21 mb^{5.5}. The reaction rate is 1.94×10^5 absorptions/cm³-sec.

By utilizing the same equations that were developed for the N-16, the amount of O-19 in the reactor water is found to be 4.7×10^5 atoms/cm³ or 2.15×10^8 atoms per pound of water. The volume source for O-19 at the recirculation pump is $4.3 \times 10^3 \text{ } \gamma/\text{cm}^3\text{-sec.}$

2.1.4.3 Activated Corrosion Products: This source of radioactivity is a result of the irradiation of structural materials within the core and the subsequent corrosion-product release to the coolant. Short-lived isotopes are also present in the coolant, but, after holdup in the tanks of the waste disposal system, these are present only in negligible quantities and therefore were neglected. Also, coolant water must be cooled down before it is sent to the waste disposal system or the retention tanks. This cooling period would allow short-lived isotopes to decay. In addition to the isotopes listed in Table C-9, Cu-64 activity is present in the primary system because of out-of-core corrosion and subsequent irradiation. However, because of the relatively short half-life of Cu-64 (12.8 hr) and because Cu-64 emits a low energy gamma (~0.5 Mev), Cu-64 is not included in calculating the shielding requirements around the retention tanks. Therefore, Cu-64 is not included with the other isotopes in Table C-9.

The methods of determining the amount and types of primary-coolant activity are based to a large extent on the work performed for the SM-1 reactor^{11, 12}. The material used in the SM-1 for the fuel element cladding, reactor vessel cladding, vessel internals, piping, and other components of the primary system is Type-304 stainless steel, which

was also used for the fuel-element cladding, reactor-vessel internals of LACBWR. Because of the similarity of materials, the same types of activated corrosion products were expected in each system.

TABLE C-9
ACTIVATED CORROSION PRODUCT NUCLIDES

<u>active isotopes</u>	<u>half life</u>	<u>reaction</u>	<u>abundance %*</u>	<u>energy of γ Mev</u>	<u>no. of γ/dis</u>
Fe-59	46d	Fe-58 (n, γ)	0.31	1.0	1.0
Co-60	5.28y	Co-59 (n, γ)	100.0	1.0	2.0
Cr-51	26d	Cr-50 (n, γ)	4.31	0.3	0.1
Co-58	72d	Ni-58 (n, ρ)	67.8	1.0	1.0
Mn-54	300d	Fe-54 (n, ρ)	5.84	1.0	1.0

In the SM-1 reactor, it was found that nearly all of the long-lived coolant activity ($T_{1/2} > 24$ hrs) from neutron irradiation of stainless steel is accounted for by the nuclides given in Table C-9. Corrosion of the activated stainless steel results in the buildup of primary coolant activity.

Theoretical means of predicting the activity buildup in the SM-1 were derived, and the results were compared with measured values of activity. In all cases but from Mn-54, theoretical values were higher than measured values by a factor of two to three. For Mn-54, theoretical and actual values agreed well (Ref. 11, p. 103). Similar theoretical means of predicting activity buildup in LACBWR can be used.

The assumptions made for the analysis are as follows:

- (1) a constant corrosion rate of stainless steel causes transfer of nuclides from in-core stainless-steel surfaces to the coolant;
- (2) corrosion products are not released in proportion to their composition in the base metal but as measured in the SM-1;

*Ratio of parent isotope to all naturally occurring isotopes of the element.

(3) nuclides deposited on out-of-core surfaces are not re-released to the coolant;

(4) nuclides in the coolant are deposited on out-of-core surfaces only; and

(5) production of nuclides from irradiation of out-of-core corrosion products is neglected.

Based on these assumptions:

$$A_i = \frac{C A_c P \cdot \Sigma(\sigma \phi)_i}{V \cdot S} \left[\frac{1 - e^{-(DA + \alpha + \lambda_i)t}}{DA + \alpha + \lambda_i} + \frac{e^{-(DA + \alpha + \lambda_i)t} - e^{-\lambda_i t}}{DA + \alpha} \right]$$

where:

A_i is the activity of isotope i in coolant (dis/cm³-sec),

C is the corrosion rate (atoms/cm²-sec) (2.08×10^4),

A_c is the area of in-core surfaces (cm²) (7.17×10^6),

P is the atomic concentration of parent nuclide in stainless steel (atom/cm³) adjusted for proportions released in corrosion,

S is the atomic concentration of all nuclides in steel (atom/cm³),

$\Sigma(\sigma \phi)_i$ is cross section times flux for both fast and thermal fluxes (dis/sec),

V is the total coolant volume (cm³) (4.35×10^7),

λ_i is the decay constant of nuclide i (sec⁻¹),

A is the out-of-core surface area (cm²) (1.91×10^6),

D is the deposition probability (1/cm²-sec) (1.07×10^{-11}),

α is the purification constant (sec⁻¹) (6.10×10^{-5}), and

t is the time after startup (sec) (6.3×10^8).

The remaining values of constants used in solving the above equation are given in Table C-10.

TABLE C-10

CORROSION PRODUCT CONSTANTS^{13,14}

<u>nuclide</u>	<u>P/S</u>	<u>λ_i</u>	<u>$\Sigma(\sigma\phi)_i$</u>
Co-58	4.33×10^{-2}	1.11×10^{-7}	1.92×10^{-11}
Co-60	2.00×10^{-3}	4.14×10^{-9}	8.50×10^{-10}
Fe-59	2.14×10^{-3}	1.78×10^{-7}	1.26×10^{-10}
Mn-54	4.03×10^{-2}	2.67×10^{-8}	7.70×10^{-12}
Cr-51	2.60×10^{-5}	2.88×10^{-8}	1.26×10^{-10}

The primary water volume sources from corrosion products are then equal to $A_i N_\gamma$, where N_γ is the number of gammas per disintegration, and are given in Table C-11. (Since the 0.3 Mev gamma source from Cr-51 is small compared to the other water sources, it was neglected.)

TABLE C-11

CORROSION PRODUCT VOLUME SOURCES

<u>E, Mev</u>	<u>$S_v (\gamma/\text{cm}^3\text{-sec})$</u>
1.0	2.6×10^3

2.1.4.4 Fission-Product Activity. The total fission-product activity is a result of:

- (1) the fission-product activity released to the coolant from failed-fuel pins, and
- (2) fission-product activity from small amount of uranium contamination on exterior surfaces of the fuel-element cladding.

The fission product activity released from failed-fuel pins is given for the equivalent of 1 Mw of defective fuel, or approximately 44 failed-fuel pins. The isotopes considered as contributing to the fission product activity are given in Table C-12.

TABLE C-12
FISSION PRODUCT ISOTOPES

isotopes	Y_i	T 1/2	γ/Dis	Mev of gamma
I-131	2.9	8.05d	1.0	0.5
I-132	4.4	2.40h	2.0	0.7
I-133	6.5	20.8h	1.0	0.5
I-134	7.6	52.5m	1.0	1.0
I-135	5.9	6.68h	1.0	1.5
I-136	3.1	86.0s	1.0	1.5
			0.5	2.0
Br-84	1.1	30.0m	0.5	0.8
Br-87	2.7	55.6s	1.0	5.4

All other fission products were neglected in the shielding calculations, for the following reasons:

- (1) lack of gamma disintegrations, or low energy of gamma disintegrations,
- (2) low release rates from the fuel (Table C-13), and
- (3) steam carryoff of all the Xe and Kr isotopes (see Sec. 2.2.1.2).

2.1.4.4.1 Fission-Product Activity from 44 Failed-Fuel Pins (i.e., 1 Mw Failed Fuel.
It appears that fission products are released from UO_2 pellets by diffusion¹⁵. The amount released depends (theoretically) on the UO_2 temperature and density and the isotopic half-life, and, for the few isotopes for which diffusion coefficients have been measured, the amount released can be expressed analytically. Since, however, diffusion coefficients have not been determined for most fission products, theoretical-diffusion equations cannot be used to determine release. Therefore, release of fission products was expressed by experimentally determined values of an escape-rate coefficient, $\nu(\text{sec}^{-1})$, where ν is the fractional amount of an isotope, within the UO_2 , which escapes per second.

The following assumptions are made:

- (1) release of isotopes from the UO_2 may be expressed in terms of an escape-rate coefficient, ν ;
- (2) that, in a defective-fuel rod, all isotopes escaping from the UO_2 enter the coolant;
- (3) that no coolant isotopes are deposited in the system surfaces;
- (4) that fuel defects are present at the start of reactor operation; and
- (5) that defects have occurred in rods producing a total power of 1 Mw, the coolant fission product activity (Ref. 15, p. 692) may be expressed as:

$$A_i = \frac{\lambda_i \epsilon Y_i}{V (\nu_i + \lambda_i) (\alpha + \lambda_i)} \left[\frac{1}{(\nu_i + \lambda_i) (\alpha + \lambda_i)} - \frac{e^{-(\nu_i + \lambda_i)t}}{(\nu_i + \lambda_i)(\alpha - \nu_i)} + \frac{e^{-(\alpha + \lambda_i)t}}{(\alpha + \lambda_i)(\alpha - \nu_i)} \right]$$

where:

A_i is the activity of isotope i in coolant due to 1 Mw defective rods ($\text{dis/cm}^3/\text{sec}$),

ϵ is the fission rate/Mw (fission/sec-Mw),

Y_i is the fission yield of isotope i (atoms/fission),¹⁶

ν_i is the escape rate coefficient¹⁷ of isotope i (sec^{-1}), and

t is the operating time (sec) of the core (7.5×10^7).

The values of ν obtained in Test x-i-d, which were used in evaluating the fission-product activity in the PWR¹⁷ and in the nuclear ship Savannah, were used here. Higher values of ν have been used, but seem to represent worst cases rather than expected values. These values are given, according to group, in Table C-13.

2.1.4.4.2 Activity as a Result of Surface Contamination of Fuel Cladding. Fission-product activity in the coolant as a result of the presence of U-235 on cladding surfaces (deposited during fuel-element fabrication) has been detected in several operating reactors. Although improved fabrication procedures may eliminate cladding

contamination, the fuel element procurement specification gives a maximum contamination of 5 μg of U-235 ft^2 of surface area. The amount of coolant fission-product activity that is due here to this contamination has been determined and is expressed (Ref. 18, p.119) by:

$$A_i = \frac{FY_i \lambda_i}{(\alpha + \lambda_i) V} \left(1 - e^{-(\alpha + \lambda_i)t} \right)$$

where:

A_i is the coolant activity of isotope i that is due to surface contamination ($\text{dis}/\text{cm}^3\text{-sec}$).

F is the fission/sec on fuel-surface times the fraction of fission fragments which enter the coolant, or

$$\frac{\phi \sigma_F (\text{U-235})_s S}{2} \quad (1/2 \text{ for flat surface})$$

$\phi \sigma_F$ is the fission reactions/sec - atom U-235,

(U-235) is the atoms U-235/ cm^2 (1.38×10^{13} for a surface contamination of 5 $\mu\text{g}/\text{ft}^2$), and

S is the cladding surface-area ($4.80 \times 10^6 \text{ cm}^2$).

Utilizing the above equations and the data in Tables C-12 and C-13 gives the primary water volume sources due to fission products in Table C-14.

TABLE C-13

ESCAPE-RATE COEFFICIENTS

<u>group</u>	<u>elements</u>	<u>ν, sec^{-1}</u>
I	Cs, I, Xe, Kr, Rb, Br	1.3×10^{-8}
II	Sr, Ba	1.0×10^{-11}
III	Zr, Ce and other rare earths	1.6×10^{-12}
IV	Te	1.0×10^{-9}
V	Mo	2.0×10^{-9}

TABLE C-14
FISSION-PRODUCT VOLUME SOURCES

<u>E, Mev</u>	<u>S_V(γ/cm³-sec)</u>
0.5	1.32 × 10 ⁴
0.7	6.20 × 10 ³
1.0	2.60 × 10 ³
1.5	6.10 × 10 ³

2.1.5 Activated Equipment

The stainless steel and carbon steel in the control rods and vessel closure become activated by a (n, γ) reaction. The boron-carbide in the control rods was neglected in the calculations, since the active isotopes (gamma emitters) of carbon and boron are short-lived (less than 1 min) and decay before the control rods are handled. The active isotopes are given in Table C-15. At equilibrium, the reaction rate is $\Sigma_c \phi_{th}$, and the volume sources are thus:

$$S_V = \Sigma_c \phi_{th} N_\gamma$$

where N_γ is the number of gammas per disintegration.

TABLE C-15
ACTIVE ISOTOPES IN ACTIVATED MATERIAL

<u>stainless steel</u>	<u>carbon steel</u>
Mn-56	Fe-59
Fe-59	Mn-56
Cr-51	
Co-60	
Ni-65	

2.1.6. Calculational Models

Considered below are the calculational models for the recirculation system, fuel storage pool, retention tank, demineralizer, main steamline, instrument ports, control rods, and vessel closure.

2.1.6.1 Primary Shield. In the calculations for the primary shield, the operating core was assumed to be a right circular cylinder with a diameter of 155 cm and an active length of 210 cm. The core materials and volume fractions are given in Table C-16, and the volume sources for the operating core are given in Table C-7.

TABLE C-16
CORE MATERIALS AND VOLUME FRACTIONS

<u>material</u>	<u>volume fraction</u>
UO ₂	0.23711
H ₂ O	0.55102
stainless steel	0.11918
void	0.09269

The axial and radial source-distributions used in the calculations for the primary shield are shown in Figs. C.4 and C.5. The discontinuity at the interface between the two axial ranges obtained the best overall fit to the actual axial power shape with the 04-2 source distributions. The shielding materials and thickness are shown in Table 10-1 and Fig. 10.4.

For determining the capture gammas in the primary-shield materials, a combination of exponential and cosine source distributions was used. The exponential source distributions have the same slope as the thermal neutron flux at the point of interest.

Since the lead-capture gamma dose rate in the concrete is small compared to the core gamma dose rate, the lead capture gammas are neglected in the shielding calculations.

2.1.6.2. Recirculation Pumps and Piping. The recirculation system consists of two pumps, their related inlet and outlet piping, and two recirculation header rings around the lower portion of the vessel. Since the two pumps are identical and each is in an identical shielded compartment, dose rates calculated for one pump apply to both. The calculational model was further simplified by using one section of pipe and increasing the dose rate accordingly to account for all inlet and outlet piping. The two models were then:

- (1) the main pump volume, and
- (2) the inlet pipe.

2.1.6.2.1 Pump Model. The recirculation pump was assumed to be a right circular cylinder 86 cm in diameter and 90 cm long. There is a 7.6-cm steel casing as

integral shielding. For purposes of calculation, the source material was assumed to be water. The pump sources considered were crud, fission products from 44 failed pins, and N-16 and O-19 carried by the water. These sources are given in Table C-17 and explained in Sec. 2.1.4. Since sources caused by plate-out of crud¹⁹ are small by comparison, they were neglected in the calculations.

TABLE C-17
RECIRCULATION PUMP VOLUME SOURCE

<u>E, Mev</u>	<u>$S_{V,r} (\gamma/\text{cm}^3\text{-sec})$</u>
0.5	1.32×10^4
0.7	6.2×10^3
1.0	4.7×10^3
1.5	1×10^4
6.0	2.75×10^5

2.1.6.2.2. Pipe Model. The pipe model was a right circular cylinder 44.5 cm in diameter and 365 cm long. A 2-cm steel wall serves as integral shielding. The source material was assumed to be water, and the volume sources are those given for the pump model.

2.1.6.3. Fuel Storage Pool. Although the spent-fuel storage area is rectangular, the calculational model was a right circular cylinder with the spent fuel around the periphery. The central cylinder is water, 310 cm in diameter and 636 cm long. The spent fuel located at the bottom of the water cylinder was assumed to be an annulus with an inner diameter of 260 cm, an outer diameter of 310 cm, and a length of 210 cm. The source distributions for the spent fuel were assumed to be flat. The volume sources are given in Table C-9.

The source model for a single element is a right circular cylinder 9.2 cm in diameter and 210 cm long. Other parameters are as in the fuel-storage-pool model.

2.1.6.4. Retention Tank. The retention tank was assumed to be a right-circular cylinder 290 cm in diameter and 430 cm long. The source material was assumed to be water, and the source distributions flat. Assuming there is sufficient decay time before the water reaches the tank, the only sources of activity are crud and fission products from 44 failed pins. The volume sources are those given for the recirculation pump, minus the O-19 and N-16 volume sources.

2.1.6.5. Demineralizer. The demineralizer removes active particles from the reactor water. For purposes of calculation, the demineralizer was assumed to be a right circular cylinder 45 cm in diameter and 155 cm long, and the source material was assumed to be water. The source distributions are flat, and the volume sources are determined by the following:

$$S_v = \frac{N_i F (1 - e^{-\lambda t})}{V}$$

where:

N_i is the atoms/cm³ in the coolant (A_i/λ_i),

F is the flow rate into the demineralizer (40 gpm),

V is the volume, and

t is the operating time (53 days).

Again, because of the transit time to the demineralizers, only corrosion products and fission products are sources. The volume sources are given in Table C-18.

TABLE C-18

DEMINERALIZER VOLUME SOURCES

<u>E, Mev</u>	<u>$S_v, (\gamma/\text{cm}^3\text{-sec})$</u>
0.6	2.4×10^8
0.8	2.1×10^8
1.0	1.9×10^7
1.3	1.7×10^7
1.8	2.3×10^6

2.1.6.6 Main Steam Line. The main-steam-line source model and related items are described in Sec. 2.2.2.2.

2.1.6.7 Gamma Streaming Through the Instrument Ports. Here, two sources of gamma radiation were considered:

- (1) gammas originating in the core, and
- (2) gammas due to neutron capture in the stainless-steel port liner.

2.1.6.7.1 Core Gammas. The core-gamma dose rate at the inside of the instrument-port tube was 5.4×10^3 r/hr, but only one-tenth of this rate was assumed to stream through the ports. However, this estimate is probably liberal. Less than one-tenth of the rate would stream because gamma scattering is predominantly in the forward direction, and the ports approach the pressure vessel tangentially, requiring a scattering angle of about 90 degrees for gammas to stream.

2.1.6.7.2 Capture Gammas. The dose rate from capture gammas⁶ in the stainless-steel port liner is 9.07×10^4 r/hr at the center of the port, and half of this dose was assumed to stream through the port.

Dose rates from both sources were combined and inserted into the streaming equation, showing a dose rate at the external port mouth of 7.4 r/hr.

In order to reduce the streaming dose rate to a safe level, a concrete shield plug was inserted into the instrument port. The thickness required for this plug was determined by considering the gammas available for streaming to be from a plane, monodirectional source. Thus, the gamma flux on the receiver side of the concrete plug is:

$$\phi = B \phi_0 e^{-\mu t}$$

Allowing for the spectrum of the scattered photons, the required thickness of heavy concrete ($\rho = 3.65$ g/cm³) was found to be 45 in.

2.1.6.8 Control Rods. The actual control rod is a cruciform, stainless-steel clad, boron-carbide element. The control-rod calculational model is an equivalent right circular cylinder, 9 cm in diameter and 213 cm long. The source material is steel, and the source distributions are assumed flat.

Assuming an average thermal flux of 1.5×10^{13} n/cm²-sec in the core and using the data in Table C-15, the volume sources for the control rod are those given in Table C-19.

2.1.6.9 Vessel Closure. The vessel-closure source model was a right circular cylinder 275 cm in diameter and 9.21 cm high. Separate integrations were performed for carbon-steel sources and stainless-steel sources. The source material is iron, and all other material was assumed to be void. Assuming an incident thermal flux of 2×10^5 n/cm²-sec during

operation, the carbon-steel and stainless-steel sources are those given in Table C-20. The axial source-distribution for the carbon-steel and stainless-steel sources is exponential and has the same slope in the closure as the thermal flux. The radial source distribution for both was assumed flat.

TABLE C-19

CONTROL-ROD VOLUME SOURCES

<u>E, Mev</u>	<u>$S_{V,r} (\gamma/\text{cm}^3\text{-sec})$</u>
0.3	1.3×10^{10}
1.0	5.0×10^{11}
1.5	1.1×10^9
2.0	1.2×10^{11}

TABLE C-20

VESSEL CLOSURE VOLUME SOURCES

<u>E, Mev</u>	<u>$S_{V,r} (\gamma/\text{cm}^3\text{-sec})$</u>	
	<u>carbon steel</u>	<u>stainless steel</u>
0.3	--	1.7×10^2
1.0	1.8×10^3	6.5×10^3
1.5	--	1.6×10^1
2.0	7.2×10^3	1.6×10^3

2.2 TURBINE BUILDING

This section describes the sources of activity in the turbine building, the calculational models used for each source, and the integral shielding. The primary sources of activity in the turbine building during normal plant operation are N-16, O-19, and the gaseous fission products from 44 failed-fuel pins.

2.2.1 Sources of Activity

2.2.1.1 Nitrogen-16 and Oxygen-19. For the N-16 and O-19 isotopes, it was assumed that the number of N-16 and O-19 atoms is the same in a pound of steam as in a pound of water: N-16 = 6.45×10^9 , O-19 = 2.15×10^8 .

2.2.1.2 Gaseous Fission Products. For the shielding calculations in the turbine building, it was assumed that all the Kr and Xe fission-product isotopes are released to the steam. The Br and I isotopes, soluble in water, are a small portion of the gaseous fission-product activity in the steam (compared to the Xe and Kr activity) and were therefore neglected. The atoms (N_B) of Xe and Kr per pound of steam is as follows:

$$N_B \text{ is } \frac{\Sigma f \phi Y_i}{FR} \frac{\nu}{(\lambda + \nu)}$$

where:

$\Sigma f \phi$ is the fissions per sec in 44 fuel pins (3.1×10^{16}),

Y_i is the yield, and

FR is the flow rate in lb/sec (169 lb/sec).

2.2.2 Calculational Models and Volume Sources

2.2.2.1 Volume Source Calculation - Methods. The volume sources are calculated by two methods. Method 1 is for equipment in which there is a buildup of active isotopes (such as the feedwater heaters and condenser), and Method 2 is for equipment in which there is no buildup of active isotopes (such as the main steam line, the turbine, the separator, reheater, and the air ejector).

Method 1

The volume source where there is a buildup of active isotopes is:

$$S_V = \frac{F_i N_i (1 - e^{-\lambda t}) N_Y}{V}$$

where:

$F_i N_i$ is the rate of active isotopes entering the components (atoms/sec),

t is the buildup time (sec) for the components, and

V is the volume of the component.

TABLE C-21

Xe AND Kr ISOTOPES IN THE STEAM

isotope	$Y_i(\%)$	$T_{1/2}$	$N_Y (\gamma/\text{dis})$	energy of $\gamma(\text{Mev})$
Kr-83m	0.48	1.86h	2.0	0.03
Kr-85m	1.5	4.40h	1.0	0.30
Kr-85	0.3	10.40y	0.01	0.50
Kr-87	2.7	78.00m	0.5	2.50
Kr-88	3.7	2.80h	1.0	2.40
Xe-131m	0.03	12.0d	1.0	0.16
Xe-133m	0.16	2.30d	1.0	0.23
Xe-133	6.5	5.27d	1.0	0.08
Xe-135	6.2	9.13h	1.0	0.25
Xe-138	5.5	17.0m	1.0	0.50

The buildup time, t , is determined by the following:

$$t \text{ is } \frac{V}{v_F (1 - f)} \ln \left[\frac{1}{f} \right]$$

where:

V is the volume of the component,

v is the specific volume of the steam,

F is the steam flow rate into the component, and

f is the ratio of inlet to outlet steam-flow.

In the case of the feedwater heaters, f is the ratio of the amount of gas in the steam to that in the condensate, according to Henry's Gas Law.

Method 2

The volume source where there is no buildup of steam is:

$$S_V = N_B (N_{Y/v})$$

where:

v is the specific volume of steam (ft^3/lb), and

N_γ is the gammas per disintegration.

2.2.2.2 Main Steam Line. The main steam line (MSL) calculational model was a right circular cylinder 25.4 cm in diameter and 600 cm long. There is a 2-cm pipe wall as integral shielding. The source material is steam and for purposes of calculation was assumed to be void. The volume sources given in Table C-22 comprise N-16, O-19, and Xe and Kr from 44 failed-fuel pins.

TABLE C-22
MAIN STEAM LINE VOLUME SOURCES

E, Mev	$S_v, (\gamma/\text{cm}^3\text{-sec})$
0.25	1.9×10^1
0.50	1.5×10^1
2.5	9.1×10^0
6.0	1.3×10^4

2.2.2.3 Turbine. The turbine includes four different sources:

- (1) the high-pressure turbine,
- (2) the two crossover pipes,
- (3) the low-pressure turbine, and
- (4) the exhaust flanges on each end of the low pressure section.

Different integration limits were used to account for each source. The models used for each were as follows.

2.2.2.3.1 High-Pressure Turbine. The high-pressure turbine model was a right circular cylinder 172 cm in diameter and 306-cm long. The steel shaft 50 cm in diameter, running the length of the turbine, was included in the model, and a 7.6-cm steel casing was included as integral shielding. The radial source distribution was assumed flat. The axial source distribution varies exponentially from the front to the back, with the maximum at the front because of the expansion of steam.

2.2.2.3.2 Crossover Pipes. The crossover pipes, one on each side of the turbine, carry steam from the high-pressure section to the center of the low-pressure section. The calculational model was an equivalent right-circular cylinder, 600 cm in diameter and 365-cm long, and there was a 2-cm steel wall as integral shielding. Since both pipes are the same, dose rates calculated for one pipe (and its related shielding) applied to both.

2.2.2.3.3 Low-Pressure Turbine. The low-pressure turbine model was a right circular cylinder 230 cm in diameter and 180-cm long. The 50-cm diameter shaft ran through the center of the model, and a 3.2-cm steel casing served as integral shielding. The radial source distribution was assumed flat. The axial source distribution is exponential, with the maximum at the center (where the steam is fed into the turbine) and minimums at both ends (toward which the steam expands).

2.2.2.3.4 Exhaust Flanges. The exhaust flanges, one on each end of the low-pressure section, are each one-half of a right circular cylinder, 460 cm in diameter and 90-cm long. The integral shielding consists of a 50-cm steel shaft and a 3.2-cm steel casing.

All these sources consist of steam (assumed void) and steel homogenized into an equivalent material. Table C-23 lists the volume fractions used for each source.

TABLE C-23

TURBINE-MODEL VOLUME FRACTIONS

<u>model</u>	<u>steel</u>	<u>void</u>
high-pressure section	0.8	0.2
crossover pipe	0.0	1.0
low-pressure section	0.68	0.32
exhaust	0.0	1.0

Table C-24 lists the volume sources for each source model, assuming specific volumes of 0.329 ft³/lb, 7.5 ft³/lb, 7.5 ft³/lb, and 652.3 ft³/lb for the high-pressure turbine, the crossover pipes, the low pressure turbine, and the exhaust flanges, respectively.

TABLE C-24

TURBINE VOLUME SOURCES S_v ($\gamma/\text{cm}^3\text{-sec}$)

<u>E, Mev</u>	<u>HP turbine*</u>	<u>crossover pipes</u>	<u>LP turbine*</u>	<u>exhaust flange</u>
0.25	1.9×10^1	8.5×10^{-1}	8.5×10^{-1}	1.0×10^{-2}
0.50	1.5×10^1	1.6×10^{-1}	1.6×10^{-1}	7.4×10^{-3}
2.5	9.1×10^0	4.0×10^{-1}	4.0×10^{-1}	4.6×10^{-3}
6.0	1.30×10^4	5.6×10^2	5.6×10^2	1.5×10^1

2.2.2.4 Separator and Reheater. The reheater and separator are located directly under the high-pressure turbine, between the turbine foundation. The moisture from the steam extracted from the high-pressure section is removed in the separator, reheated by primary steam in the reheater, and returned to the high-pressure section of the turbine.

2.2.2.4.1 Separator. The separator model was a right circular cylinder, 140 cm in diameter and 760 cm long. The source material was assumed void, and there was a 1.3-cm steel shell as integral shielding. The volume sources given in Table C-25 assume a specific volume of $2.18 \text{ ft}^3/\text{lb}$ for the steam in the separator, and flat source distributions.

TABLE C-25

SEPARATOR VOLUME SOURCES

<u>E, Mev</u>	<u>S_v, ($\gamma/\text{cm}^3\text{-sec}$)</u>
0.25	2.9×10^0
0.50	2.2×10^0
2.5	1.37×10^0
6.0	1.90×10^3

*calculated at the entrances to the high-pressure section and the low-pressure section, respectively.

2.2.2.4.2 Reheater. The reheater model was also a right circular cylinder, 170 cm in diameter and 760 cm long. The source distributions and integral shielding were assumed to be those of the separator. By taking the flow rates of steam from the main steam line and the turbine, an equivalent specific volume of 1.3 ft³/lb was calculated and was used to determine the volume sources given in Table C-26.

TABLE C-26
REHEATER VOLUME SOURCES

<u>E, Mev</u>	<u>S_v, (γ/cm³-sec)</u>
0.25	5.0 × 10 ⁰
0.50	3.8 × 10 ⁰
2.5	2.3 × 10 ⁰
6.0	2.22 × 10 ³

A volume fraction of 0.9 for steam (void) and 0.1 for steel was used to determine an equivalent source material.

2.2.2.5 Condenser. The condenser calculational model was divided into three sections:

- (1) the transition volume between the turbine exhaust flanges and the condenser proper,
- (2) the main condensing volume, and
- (3) the two air baffles or air-collection sections.

Figure C.6 shows the location and integral shielding of each section.

2.2.2.5.1 Transition Volume. Feedwater heaters No. 1 and No. 2 are also located in this section, and, since the feedwater heaters are the main source of activity, the transition volume was neglected in the shielding calculations.

2.2.2.5.2 Main Condensing Volume. For purposes of calculation, the main condensing section was made into an equivalent right circular cylinder 320 cm in diameter and 425 cm long. Two large thicknesses of concrete were placed at the end of the cylinder, 153 cm above and below the axis of the cylinder, to make source volume contributing to the dose in the model like that of the actual main condensing section (see Fig. C.6).

The water boxes, on each end, and a 2-cm steel shell served as integral shielding. The source distributions were assumed flat. The source materials and volume fractions used for the LACBWR condenser were those used for the Pathfinder condenser; they are given in Table C-27.

TABLE C-27

MAIN CONDENSING SECTION SOURCE MATERIALS

<u>material</u>	<u>volume fraction</u>
water	0.19
void	0.76
Admiralty metal	0.05

2.2.2.5.3 Air Baffles. The two air baffles run lengthwise in the main section and are centrally located in both halves of the main section (see Fig. C.6). The integral shielding, source distributions, and source materials are as in the main section. The actual air collection section is rectangular (425 cm x 30 cm x 305 cm). Using appropriate integration limits and the same techniques as for the main section, the source can be described in cylindrical geometry. Owing to the condensation in the main section and the air collection sections, there is a buildup of active isotopes. The average buildup times are 0.17 sec for the main section and 0.43 sec for the air collection section, and the specific volume for both is 652.3 ft³/lb. The flow rates are 112.3 lb/sec for the main section and 11.2 lb/sec for the air collection section, and the volume sources for each are given in Table C-28.

TABLE C-28
CONDENSER VOLUME SOURCES ($\gamma/\text{cm}^3\text{-sec}$)

<u>E, Mev</u>	<u>S_v, main volume</u>	<u>S_v, air collection vol</u>
0.25	2.5×10^{-2}	6.5×10^{-1}
0.50	1.9×10^{-2}	4.8×10^{-1}
2.5	1.2×10^{-2}	3.0×10^{-1}
6.0	1.6×10^1	4.1×10^2

2.2.2.6 Feedwater Heaters. There are three feedwater heaters (FWH) that are sources of radiation: FWH 1, FWH 2, and FWH 3. Feedwater Heaters 1 and 2 are in the transition section of the condenser, and FWH 3 is in the southeast corner of the building on the grade floor. The source distributions for all heaters are assumed to be flat. Table C-29 lists the specifications and other pertinent data for each FWH model, and Fig. C.7 shows a typical FWH model with integral shielding.

TABLE C-29
FEEDWATER HEATER SPECIFICATIONS

<u>parameter</u>	<u>FWH 1</u>	<u>FWH 2</u>	<u>FWH 3</u>
source diameter, cm	109	88	86
source length, cm	880	880	660
total length, cm	960	960	737
Source vol-fractions*			
tube	0.1	0.1	0.1
steam	0.6	0.6	0.6
H ₂ O	0.3	0.3	0.3

*these are assumed to be the same as in a typical FWH used in Pathfinder.

Table C-29 (cont.)

<u>parameter</u>	<u>FWH 1</u>	<u>FWH 2</u>	<u>FWH 3</u>
shell thickness, cm**	7.6	7.6	7.6
water box depth, cm	71.0	73.0	69.0
water box cover, cm	10.2	8.4	7.6
flow, lb/sec	11.23	9.97	16.4
volume, cm ³	8.15×10^6	5.37×10^6	3.82×10^6
f (flow out/flow in), ***	10^{-7}	10^{-7}	10^{-7}
t, sec	6.1	15.9	21.0
v, ft ³ /lb	67.24	19.19	6.29

Table C-30 lists the volume sources for each of the FWH models.

TABLE C-30

FEEDWATER HEATER VOLUME SOURCES ($\gamma/\text{cm}^3\text{-sec}$)

<u>E, Mev</u>	<u>FWH 1</u>	<u>FWH 2</u>	<u>FWH 3</u>
0.25	4.8×10^{-1}	1.7×10^0	5.0×10^1
0.50	4.6×10^{-1}	1.6×10^0	5.0×10^0
2.5	7.0×10^{-2}	2.5×10^{-1}	7.6×10^{-1}
6.0	7.8×10^2	2.1×10^3	4.8×10^3

2.2.2.7 Air Ejector. The air ejector is composed of two sections, the intercooler and the aftercooler. Each is the same size and has the same physical characteristics. The source model for both was a right circular cylinder, 50 cm in diameter and 122 cm in length, and the material volume fractions for the source models are given in Table C-31.

**Because of the thinness of the steel shell, an equivalent amount of water is substituted for the steel.

***Henry's Gas Law - fraction is the ratio of gas in air above a liquid to that in the liquid.

TABLE C-31

AIR EJECTOR VOLUME FRACTIONS

<u>material</u>	<u>volume fraction</u>
water	0.14
steam	0.83
steel	0.03

Each source has a 1-cm steel shell and a 27-cm-thick water box on each end as integral shielding. Centerlines of the inter- and aftercoolers are 107 and 190 cm, respectively, above the mezzanine floor, and 183 cm from the west shielding wall. The source distributions for each were assumed to be flat.

The volume sources, given in Table C-32, assume a flow rate of 200 lb/hr to the air ejector and specific volumes of 177 ft³/lb for the intercooler and 26.8 ft³/lb for the aftercooler.

TABLE C-32

AIR EJECTOR VOLUME SOURCES ($\gamma/\text{cm}^3\text{-sec}$)

<u>E, Mev</u>	<u>S_v, intercooler</u>	<u>S_v, aftercooler</u>
0.25	6.5×10^2	4.3×10^3
0.50	5.2×10^2	3.4×10^3
2.5	3.2×10^2	2.1×10^3
6.0	4.5×10^5	3.0×10^6

2.2.2.7.1 Air Suction Line. There is an 8-in. air suction line from the condenser to the air ejector. The source model was a right circular cylinder 20.3 cm in diameter and 305 cm long. The steel pipe-wall was assumed to be 1 cm thick. Assuming a specific volume of 652.3 ft³/lb, the volume sources for the air suction line are those in Table C-33.

TABLE C-33

AIR SUCTION LINE VOLUME SOURCES

<u>E, Mev</u>	<u>$S_v (\gamma/\text{cm}^3\text{-sec})$</u>
0.25	1.8×10^2
0.50	1.4×10^2
2.5	8.6×10^1
6.0	1.2×10^5

2.2.2.8 Flash Tank. The flash tank is in the southeast corner of the building near the high-pressure heater and main steam line. The calculational model for the flash tank was a right circular cylinder 91 cm in diameter and 244 cm long. The volume sources, assuming a specific volume of 629 ft³/lb, are those in Table C-34.

TABLE C-34

FLASH TANK VOLUME SOURCES

<u>E, Mev</u>	<u>$S_v (\gamma/\text{cm}^3\text{-sec})$</u>
0.25	1.0×10^0
0.50	7.6×10^{-1}
2.5	4.8×10^{-1}
6.0	6.6×10^2

The source material was assumed to be void and the source distributions flat. A 1.3-cm steel shell was included as integral shielding.

2.2.2.9 Full-Flow Demineralizer. There are three demineralizers located on the grade floor in the southwest corner of the building. The calculational model (the same for all three) was a right circular cylinder 137 cm in diameter and 213 cm long. The source material was assumed to be water and the source distributions flat.

The volume sources, assuming crud and fission products from 44 failed-fuel pins, are those in Table C-35. Only the crud and fission products that are contained in the moisture carryover (0.1 percent) contribute to demineralizer activity. A flow rate is determined by taking the 0.1 percent carryover, and the volume sources are determined by the method used for the primary system demineralizers, assuming a bed life of 14 days.

TABLE C-35

FULL FLOW DEMINERALIZER VOLUME SOURCES

<u>E, Mev</u>	<u>$S_v, (\gamma/\text{cm}^3\text{-sec})$</u>
0.6	7.75×10^6
0.8	3.0×10^5
1.0	2.15×10^5
1.3	1.87×10^5
1.8	4.1×10^4

2.2.2.10. Off-Gas Holdup Tank. The off-gas holdup tank, located between the reactor building and the turbine building at el 629 ft, is designed to hold up the off-gas from the air ejector for 10 min to allow decay of N-16 and O-19 activity before the gas is released from the stack.

The calculational model was a right circular cylinder 183 cm in diameter and 163 cm long. The source distributions are assumed flat and the source material void. The volume sources (C-36) for the off-gas holdup tank were determined utilizing the 10-min holdup time and the volume sources from the aftercooler of the air ejector.

TABLE C-36

OFF-GAS HOLDUP TANK VOLUME SOURCES

<u>E, Mev</u>	<u>$S_v, (\gamma/\text{cm}^3\text{-sec})$</u>
0.25	4.3×10^3
0.50	1.6×10^3
2.5	2.1×10^3
6.0	1.6×10^3

2.2.2.11 Liquid-Waste Storage Tank. The liquid-waste storage tank source-model is a right circular cylinder 183 cm in diameter and 670 cm long. The source material is water and the source distributions were assumed flat.

The tank handles many different types of liquid waste during operation. For purposes of calculation, however, the volume sources are assumed to be the same as those for the reactor water, minus the short-lived O-19 and N-16 isotopes. The water sources represent the maximum activity that the tank handles.

2.2.2.12 Maximum Credible Accident. The maximum credible accident (MCA) is discussed fully in Sec. 14.

2.2.2.13 Calculational Model for MCA—Direct Dose. The calculational source model for the control room during MCA was the containment building, which was assumed to be 9.14 m in diameter and 33.68 m long. For purposes of calculation, the source material was assumed to be void. There is a 3-cm steel shell and a 22.9-cm ordinary concrete wall around the periphery of the containment building as integral shielding. The other ordinary concrete shielding walls around the control room are shown in Fig. 10.7.

It was assumed that 100 percent of the noble gases, 50 percent of the halogens, and 1 percent of the solid fission products from 100 percent of the fuel pins are uniformly distributed throughout the reactor building, and, utilizing these fission products and the reactor building volume, the volume sources are as given in Table C-37. The maximum direct dose rate is 4.1 r/hr.

TABLE C-37

REACTOR BUILDING MCA VOLUME SOURCES

<u>E, Mev</u>	<u>S_v, Mev/m³-sec</u>
0.081	3.14×10^{12}
0.242	1.06×10^{13}
0.516	1.96×10^{13}
0.78	3.34×10^{13}
1.3	2.91×10^{13}
1.5	2.72×10^{13}
2.0	5.45×10^{13}

2.2.2.14. Calculational Model for Air Scattering. In addition to the direct dose rate from MCA there is a contribution from air-scattered gammas. The calculational model used for the air scattering is shown in Fig. C.8. It was assumed that the gammas pass through the top of the reactor building; the steel shell was the only integral shielding considered for the top of the containment building. The gammas passing through the 9-in. concrete sides were neglected in the scattering calculations, since they are less than one-tenth of the gammas that pass through the top.

The air-scattered gamma dose rate is determined by the following:

$$\dot{\phi} = \frac{N S_{ow}}{4\pi X} \frac{d\sigma}{d\Omega} (\psi_2 - \psi_1) \left(\pi - \frac{\psi_2}{2} - \frac{\psi_1}{2} - \phi_1 \right) \left[B_d e^{-\mu X} CF \right]$$

where:

N is the number density of air (3.6×10^{20}),

S_o is the source strength ($S_v(9898M^3)$),

$\frac{d\sigma}{d\Omega}$ is the Klein - Nishina scattering cross section,²⁰

ω , ψ_2 , ψ_1 , and ϕ_1 are as shown in Fig. C.8.

X is the distance from source point to receiver point,

$B_d e^{-\mu X}$ is the attenuation in the steel containment shell and the concrete walls around the control room, and

CF is the flux-to-dose-rate conversion factor.

The dose rate for single scattering is then increased by a factor of 1.5 to include multiple scattering contributions. The point sources are obtained by multiplying each energy group in Table C-21 by the volume of the containment building. The control room shielding was assumed to be a minimum, i.e., the 1-ft thick west and north walls. Combining the contribution from each energy group (Table C-37), the maximum scattering dose rate in the control room is 0.064 r/hr.

2.3 WASTE TREATMENT BUILDING

This section describes the sources of activity in the waste treatment building and the calculational models used for each source.

The primary sources of activity during normal plant operation are activated crud and fission products from 44 failed-fuel pins; these sources were described in Secs. 2.1.4.3 and 2.1.5.4. The N-16 and O-19 activity is negligible because of decay. A description of the calculational models follows.

2.3.1 Evaporator Feed Tank

The calculational model for the evaporator feed tank was a right circular cylinder 137 cm in diameter and 256 cm long. The source material is water and the source distributions were assumed to be flat.

The sources for the evaporator feed tank were those of the reactor water, minus the short-lived isotopes, and are given in Table C-38.

TABLE C-38

EVAPORATOR FEED TANK VOLUME SOURCES

<u>E, Mev</u>	<u>$S_v, (\gamma/\text{cm}^3\text{-sec})$</u>
0.5	8×10^3
0.7	3.1×10^3
1.0	5.4×10^3
1.5	1.3×10^4

2.3.2 Concentrated-Waste Tank

After leaving the feed tank and passing through the evaporator, the residue is concentrated by a factor of 50 before being sent to the concentrated-waste tank. Therefore, sources for the concentrated-waste tank were a factor of 50 greater than those of the evaporator feed tank (see Table C-38).

The calculation model for the concentrated-waste tank is a right circular cylinder 107 cm in diameter and 213 cm long. The source material was assumed to be water and the source distributions are flat.

2.3.3 Evaporator

The dose rates from the evaporator were assumed to be the same as those from the concentrated-waste storage tank.

Condensate from the evaporator directed to the 1000-gal. water-collection tank has undergone a reduction in activity of between 10^{-3} and 10^{-4} , and no calculation was performed for the water-collection tank since the shielding provided for the spent-resin and evaporator-feed tanks is more than adequate.

2.3.4 Spent-Resin Tank

The calculational model for the spent-resin tank is a right circular cylinder 137 cm in diameter and 174 cm long. The source material is assumed to be water, and the source distributions flat. The sources for the spent-resin tank were assumed to be those for the primary system demineralizer (see Table C-18).

2.3.5 Off-Gas Storage Tanks

The calculational model for the off-gas storage tank is a right circular cylinder 360 cm in diameter and 460 cm long. The source material is assumed to be void, and the source distributions are flat. The sources for the tank are the gaseous fission products from 44 failed-fuel pins in the primary steam, and, assuming that the tanks are capable of retaining all gases produced during three days of plant operation at full power, the volume sources given in Table C-39 are determined by the following:

$$S_v = \frac{N_i^{21} (1 - e^{-\lambda t})}{V}$$

where:

N_i is the atoms per sec of nuclide i entering the tank,

V is the volume of the tank, and

t is the time (sec) (2.6×10^5).

TABLE C-39

OFF-GAS STORAGE TANK VOLUME SOURCES

$E, \text{ Mev}$	$S_v (\gamma/\text{cm}^3\text{-sec})$
0.5	2.44×10^4
1.5	9.20×10^2
2.5	2.97×10^3

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4. FIGURES

GAMMA SCATTERING MODEL

FIG. C.1

GAMMAS SCATTERED BY A CONCRETE SLAB

FIG. C.2

CIRCULATION OF STEAM FLOW AND WATER

FIG. C.3

AXIAL SOURCE DISTRIBUTION

FIG. C.4

RADIAL SOURCE DISTRIBUTION

FIG. C.5

CONDENSER SOURCE MODEL

FIG. C.6

TYPICAL FEEDWATER HEATER MODEL

FIG. C.7

CONTROL ROOM AIR-SCATTERING MODEL

FIG. C.8

APPENDIX D

JOB DESCRIPTIONS OF KEY PERSONNEL
FOR LACBWR STARTUP AND OPERATION

1. JOB DESCRIPTIONS FOR REACTOR PLANT SUPERVISORY PERSONNEL

Operations Supervisor

The Operations Supervisor reports directly to the Project Operations Manager and is responsible for the security, operation, and preventive maintenance of the reactor, the reactor auxiliary systems, and the waste treatment facility. He is responsible for all instrument maintenance.

He will direct and coordinate the activities of the reactor shift supervisors, the reactor operators, and the instrument technicians. During power testing, he is responsible for continuous liaison with his counterpart in the generator plant to ensure that all activities are compatible.

He provides operating personnel and direction for each test during the entire test program. He sees that the tests are performed safely and efficiently and in accordance with written test procedures. He is also responsible for collecting the data prescribed in the written procedures.

At all times his primary concern is the safety of the plant and personnel.

After the various systems, components, and procedures are tested and proven acceptable, the Operations Supervisor assures that the systems are operated and maintained properly. He is to advise the Plant Engineering Supervisor of any need for heavy maintenance or repairs.

The Operations Supervisor also sees that the appropriate service records for the various components and systems are maintained from the time of the preoperational test. He is responsible for the reactor operating log and for keeping the detailed procedures of the operating manual current.

Test Coordinator

The Test Coordinator reports directly to the Project Operations Manager and is responsible for test scheduling, coordination, consolidation of data, and the preparation of test reports.

The scheduling function is done in consultation with the Operations Supervisor to assure that operations personnel are available to perform the test and to collect the test data.

Prior to each test the Test Coordinator reviews the written procedures to assure that the test instructions are clear and the data to be taken are adequate. He ascertains that the test prerequisites are fulfilled and sees that any special equipment or instrumentation required is available for the test.

After the tests he consolidates and reviews the data to see that they are consistent and adequate for the intended purpose. He is also responsible for applying whatever correction factors, conversion factors, etc., are necessary to convert the data to usable form.

He is responsible for the preparation of test reports which describe the test and present both raw data and corrected data.

Allis-Chalmers personnel from the analysis or engineering departments will be at the site for certain special tests and will coordinate their program requirements through the Test Coordinator. Just as for the normal tests, the Test Coordinator is responsible for the preparation and distribution of a test report.

The Test Coordinator maintains comprehensive records of all measured performance data and periodically updates applicable portions of such control documents as the operating manual.

Health Physicist

The Health Physicist reports directly to the Project Operations Manager and is responsible for conventional safety and industrial hygiene, as well as radiation safety.

He implements a surveillance program to assure that personnel radiation exposure is minimized and is within acceptable limits so as to ensure safety of all personnel at LACBWR.

The Health Physicist informs the Project Operations Manager, the Operations Supervisor, or (in their absence) the shift supervisor regarding any operations (including reactor operations) that he judges must be halted for safety reasons.

He is responsible for maintaining protective equipment stock and radiation detection devices and designates when they are to be used. He installs safety warning signs and labels denoting hazardous or potentially hazardous conditions, and keeps appropriate records of personnel radiation exposure and of radioactive waste disposal.

The Health Physicist trains operating personnel in health physics and is responsible for keeping the Health Physics Manual current.

Plant Engineering Supervisor

The Plant Engineering Supervisor reports directly to the Project Operations Manager and is responsible for heavy maintenance, for all mechanical repairs, and for correction of any deficiencies discovered during testing.

He directs and coordinates the activities of all mechanical maintenance personnel and any personnel supplied by the construction contractor. Before preoperational testing begins, he organizes and subsequently maintains a check list of items to be completed or corrected.

In consultation with the Operations Supervisor and the Test Coordinator he schedules the necessary repairs or corrections. He sees that the necessary labor, material, and equipment are available to complete these tasks on schedule. He is also responsible for updating all drawings to include modifications, so that they represent the as-built condition.

The Plant Engineering Supervisor is responsible for procurement of all purchased material, for maintaining a list of spare parts on hand, and for fuel accountability.

Since the Plant Engineering Supervisor is at the site during construction, he is responsible for maintaining service records on equipment and components until this function is assumed by the Operations Supervisor.

Shift Supervisor

The Shift Supervisor reports to the Operations Supervisor. He personally supervises the actions of the reactor operators and is responsible for the startup, operation, and shut-down of the nuclear plant in accordance with established operating rules and procedures. He is also responsible for performance of nuclear plant preventative maintenance and for the fuel element changes directed.

He directs and helps prepare all logs and reports concerning operation, maintenance, fuel elements, instruments, and equipment. He operates or directs the operation of all nuclear plant facilities (controls, switches, valves, pumps, etc.).

The Shift Supervisor reports any suspected malfunctions of equipment or instrumentation to the Operations Supervisor as soon as practical. He advises the Health Physicist of potentially hazardous operations and obtains his approval as required. He prevents unauthorized personnel from entering restricted areas and reports all violations.

Safety Committee

The Safety Committee consists of at least one specialist in reactor performance, one in power-plant operations, and one in reactor mechanical components.

Each member, or his assigned alternate, is to be available at the site within 24 hr.

The Committee normally meets as scheduled by the Committee Chairman. However, any member of the Committee or the Project Operations Manager or Operations Supervisor can at any time request the Chairman to schedule a special meeting.

The Committee reviews and approves all nuclear test procedures prior to performance and all changes in operating or health physics procedures. The Committee has access to all test results and observes tests which it considers to be of particular significance.

It informs the Project Operations Manager of any phase of the test program which it considers is not being or cannot be conducted safely.

The AEC Site Representative is a permanent observer at all Safety Committee meetings.

2. JOB DESCRIPTIONS FOR DAIRYLAND POWER KEY PERSONNEL FOR OPERATION OF LACBWR

LACBWR Plant Superintendent

Authorities:

- (1) The Plant Superintendent is in charge of the nuclear power plant.
- (2) The Plant Superintendent has authority to direct, coordinate and control all plant activities required for the safe, reliable and economic operation and maintenance of the plant. He has authority to shut down the reactor and generator plants or to order necessary actions to ensure compliance with directives issued by higher authority, including established regulations, controls, authorizations, licenses, approved procedures and specifications for operation and maintenance.

Duties:

He shall be responsible for the following:

- (1) Directing all plant activities, ensuring that all subordinate or delegated responsibilities, authorities and relationships are fully understood and implemented.
- (2) Compliance by plant personnel with established regulations, authorizations, licenses, approved procedures, and specifications for operation and maintenance of the plant.
- (3) Selection and hiring of personnel for the plant organization.
- (4) Development and performance of technical training plans and programs to assure a trained and qualified organization for plant operation and maintenance.

(5) Development and implementation of administrative, operational, and maintenance plans and procedures.

(6) Control and procurement of materials, supplies, equipment, or services required for plant operation and maintenance.

(7) Development of operating budgets and control of expenditures.

(8) Development of an efficient system of records and reports to facilitate reviews of plant performance and economic factors.

(9) Reception, safety, and courteous handling of visitors to the site and plant.

(10) Coordination of test or development programs with operation of the plant.

(11) Review of plant performance and submission of essential reports and recommendations to the AEC and DPC management.

LACBWR Assistant Superintendent

Authorities:

(1) During the absence of the Plant Superintendent, the Assistant Superintendent has the authority and responsibility to perform all operational duties of the Superintendent and to be in complete charge of the plant.

(2) The Assistant Superintendent has the authority and responsibility to shut down the reactor and generator plants or to order appropriate action to correct unsafe or abnormal conditions if, in his opinion, failure to issue the orders could possibly result in a significant hazard to equipment, personnel, or the public.

(3) The Assistant Superintendent has the authority and responsibility to order reactor and/or generator plant operating conditions or methods changed if established regulations or procedures are not being followed or established limits are being exceeded.

Duties:

(1) The Assistant Superintendent is responsible to the Plant Superintendent for the safe and efficient operation of the nuclear reactor plant and for the technical and functional direction of the Shift Supervisors, Operations Supervisor, Health and Safety Engineer, Process Engineer, Instrument and Electrical Supervisor, and the Mechanical Maintenance Supervisor.

(2) Routinely audit and analyze plant operation and personnel performance to ensure safe and efficient operation as specified by approved regulations and procedures and according to Dairyland's power generation requirements and contractual agreements. Maintain complete, accurate, and legible records of results of audits and inspections.

(3) Assist in planning and coordinating all phases of operations, maintenance, modifications, and performance and engineering tests in order to obtain the desired operating performances.

(4) Recommend revisions and modifications of equipment and procedures for improving plant operations.

(5) Review proposed changes in equipment and procedures and make recommendations from the standpoint of economy and technical and operating feasibility and safety.

(6) Assist in the review, evaluation, and preparation of proposals, plans, and requests for appropriations for revisions, modifications, or additions to plant equipment and facilities.

(7) Assist in training plant personnel and preparing training materials.

(8) Assist in selecting and hiring plant personnel.

(9) Provide information and instructions to operating and maintenance personnel and aid in maintaining adequate plant communications.

(10) Prepare routine and special reports as required.

(11) Assist in the establishment and maintenance of an effective program for the purpose of keeping the plant clean, orderly, and reasonably free from hazards.

LACBWR Operations Supervisor

Duties:

(1) The Operations Supervisor is responsible to the Plant Superintendent, through the Assistant Superintendent, for the safe and efficient planning, scheduling, and coordination of plant operations; maintenance and testing activities during operation and shut-down periods; and for the technical and functional supervision of operating personnel assigned to him.

(2) Coordinate planning and scheduling with the maintenance supervisors, Process Engineer, and Health and Safety Engineer.

(3) Routinely review and evaluate operational information and recommend to the Plant Superintendent and/or Assistant Superintendent changes in procedures or practices for improvement of plant operation.

(4) Assist in providing necessary operational information to shift personnel.

(5) Maintain adequate and economic inventories of materials required for the operation of the plant. Maintain complete and accurate records of material costs, inventories, and usage.

(6) Enforce strict compliance with all established safety and radiation protection rules and regulations. Advise the LACBWR Health and Safety Engineer of potential hazardous operations and obtain his approval as required.

(7) Conduct a continuous training program for the operating personnel assigned to him. Assist in training other personnel as assigned.

(8) Maintain complete, accurate, and legible records of required information.

(9) Minimize and control within established limits, the radiation exposure of all personnel. Direct necessary decontamination efforts to avoid uncontrolled spread of radioactive contamination and resulting hazard to personnel.

(10) Direct and supervise material handling associated with plant operations.

(11) Prepare, revise, and maintain operating procedures, training materials, maintenance procedures, results manuals, reports, reviews, and other written materials as assigned.

(12) Establish and maintain an effective program for the purpose of keeping the plant clean, orderly, and reasonably free from hazards.

(13) Perform relief for the Shift Supervisors during vacations and other absences.

(14) Perform all duties of the Shift Supervisors during relief periods.

LACBWR Shift Supervisor

Duties:

(1) The Shift Supervisor is responsible to the Plant Superintendent through the Assistant Superintendent for the safe and efficient operation of the plant, for the coordination of normal plant activities, and for the technical and functional supervision of operating personnel on his assigned shift.

(2) Direct the operation of the plant on his shift and ensure that all operations are conducted safely and according to approved procedures and specifications.

(3) Enforce strict compliance with all established safety and radiation protection rules and regulations. Advise the LACBWR Health and Safety Engineer of potential hazardous operations and obtain his approval as required.

(4) Direct all equipment and system checkoffs for plant startup operation and shutdown.

(5) Direct plant operation according to electrical load demands of Dairyland Power Cooperative power system.

(6) Evaluate the magnitude of any incident that may occur and take necessary emergency action to ensure safety of personnel and minimize equipment damage.

(7) Conduct a continuous training program for the operating personnel on his shift. Assist in training other personnel as assigned.

(8) Maintain complete, accurate, and legible records of all plant data and significant happenings on his assigned shift.

(9) Initiate and control necessary maintenance for safety and continuity of plant operation. Provide guidance and controls for other planned maintenance performed on his shift.

(10) Minimize and control within established limits, the radiation exposure of all personnel. Direct necessary decontamination efforts to avoid uncontrolled spread of radioactive contamination and resulting hazard to personnel.

(11) Direct and supervise material handling associated with plant operations, including receipt, storage, loading, and discharging of reactor fuel, shipment of irradiated fuel and disposal of radioactive and non-radioactive wastes.

(12) Prepare, revise, and maintain operating procedures, training materials, maintenance procedures, results manuals, reports, reviews and other written materials as assigned.

(13) Evaluate plant performance and make recommendations regarding improvements and revisions in procedures, equipment, safety practices, and operating economics.

(14) Establish and maintain an effective program on his assigned shift for the purpose of keeping the plant clean, orderly, and reasonably free from hazards.

LACBWR Plant Operator

Duties:

- (1) The Plant Operator is responsible to the Shift Supervisor for the operation of the reactor, power generating equipment, and all other auxiliary and miscellaneous plant equipment according to established operating procedures and instructions and as directed.
- (2) Assignment A: Control the reactor and related equipment from the reactor control board during startup, equilibrium operation, shutdown, and emergency operations in a safe and efficient manner according to approved instructions and procedures. Work closely with the operator on the turbine-generator control board and with the operator on the auxiliary equipment assignment to ensure proper operation of the total plant.
- (3) Assignment B: Control the steam turbine, turbine generator, and related equipment from the turbine-generator control board during startup, steady operation, shutdown, and emergency operations in a safe and efficient manner according to approved instructions and procedures. Work closely with the operator on the reactor control board and the operator on the auxiliary equipment assignment.
- (4) Assignment C: Operate all auxiliary equipment and perform routine responsibilities and non-routine functions as directed. These duties shall be performed outside the control room. Cooperate with the control room operators and closely follow all instructions originating from the Shift Supervisor and/or the control room operators to ensure proper operation of the total plant. Notify the control room before performing duties that will or may affect plant operation and of any abnormal situations.
- (5) Perform all equipment and system checkoffs for plant startup and shutdown.
- (6) Perform plant equipment installation, repairs, and modifications as directed, utilizing the proper tools proficiently.
- (7) Operate monitoring and control instruments, including health physics instruments, as required for the safe and efficient operation of the plant. Instrument operation includes making minor adjustments, changing charts, and inking.
- (8) Record all required plant data in a complete, accurate, and legible manner.
- (9) Perform all responsibilities with strict compliance to established safety and radiation protection rules and regulations. Perform necessary health physics responsibilities as directed.

(10) Perform responsibilities in such a manner so as to minimize personal exposure to radiation and health hazards.

(11) Perform normal and special housekeeping activities as necessary to keep the plant clean, orderly, and reasonably free from hazards.

(12) Perform nuclear fuel handling as directed.

(13) Maintain a thorough working knowledge of plant equipment, procedures, policies, and safety rules by active participation in training programs and continuous review of the necessary materials.

(14) Assist in instructing and training new plant operators and other personnel as directed.

(15) Recommend changes in procedures, equipment design, and operation and work methods which may simplify an operation, eliminate hazards, improve operating economy and/or generally improve plant operation. Recommend necessary maintenance of equipment.

(16) Assist in preparing and revising operating procedures, maintenance procedures, reports, reviews, and other required written materials.

(17) Perform routine calculations.

(18) Participate in and lead safety programs, including meetings, preparations of safety instructions, and plant inspections.

(19) Exercise initiative and judgment in performing duties and know when to refer matters to supervision for decision.

(20) Perform necessary decontamination activities to prevent uncontrolled spread of radioactive contamination and to maintain contamination levels on tools and equipment below the specified maximums for handling, storage, and/or off-site shipment.

LACBWR Health and Safety Engineer

Duties:

(1) The Health and Safety Engineer is responsible to the Plant Superintendent through the Assistant Superintendent for the planning and direction of the plant health and safety programs, for the operation of the chemical laboratory, and for the technical and functional supervision of the health physics and chemical laboratory technicians.

(2) Plan, recommend, schedule, and direct all LACBWR health physics, industrial safety, and fire protection programs and activities.

- (3) Make evaluations, interpretations, and recommendations for plant and local matters concerning health physics and industrial safety.
- (4) Plan, recommend, and direct the functions performed by the chemical laboratory.
- (5) Recommend radiation exposure limits for plant personnel based on established regulations and policies.
- (6) Minimize and control, within established limits, radiation exposure of all personnel under his supervision.
- (7) Evaluate plant health and safety performance, and make recommendations regarding improvements and revisions.
- (8) Enforce strict compliance with all established safety and radiation protection rules and regulations. He has the authority and responsibility to order hazardous work to cease until the Shift Supervisor and/or the Plant Superintendent are apprised of the situation.
- (9) Maintain complete, accurate, and legible records of personnel radiation exposure, radiological surveys, environmental monitoring, radiochemical analyses, radioactive waste disposal, personnel industrial hygiene, and of all other necessary plant health and safety activities.
- (10) Prepare, revise, and maintain health and safety procedures, training materials, laboratory procedures, reports, reviews, decontamination procedures, and other required health and safety written materials.
- (11) Conduct a continuous training program for the health physics and laboratory technicians in order to establish and maintain a high degree of proficiency. Assist in training other plant personnel on subjects of health physics, industrial safety, and laboratory techniques.
- (12) Perform liaison and coordination responsibilities with local, state, and federal agencies concerning health physics and industrial safety activities.
- (13) Ensure that all applicable regulations and safety requirements for shipments of radioactive materials are met.
- (14) Plan, recommend, and direct the protective clothing program including acquisition, storage, issuance, handling, cleaning, and disposal.

Health Physics Technician

Duties:

- (1) The Health Physics Technician is responsible to the Health and Safety Engineer for providing contamination and radiation exposure control, planning and executing the

health and safety program, and performing chemical and radiochemical analysis as required.

(2) Cooperate with and assist all plant personnel in maintaining contamination, radiation, and safety control for the safe and efficient operation of the LACBWR plant.

(3) Maintain his own radiation exposure within permissible limits.

(4) Must have a general working knowledge of the reactor and generator plant systems and operating procedures.

(5) Prescribe time limits and clothing requirements for entry into temporary and permanent radiation zones and for work within these zones on contaminated equipment and radioactive materials within established regulations and procedures.

(6) Investigate the circumstances leading to clothing or skin contamination or uncontrolled contamination spreads in, or associated with, the LACBWR plant.

(7) In the event of an evacuation because of radiation hazards or other causes, and in the absence of the Health and Safety Engineer, the Health Physics Technicians will assume full radiation monitoring responsibility, rendering assistance and advice pertaining to radiation aspects of the situation to the operating supervisor in charge.

(8) Make routine radiation surveys both of radiation zones and non-radiation zones and report unsatisfactory conditions to supervision for action.

(9) Make necessary radiation surveys to classify conditional and unconditional releases of equipment or materials prepared for off-plant shipment.

(10) Utilize air sample results to recommend necessary respiratory protection.

(11) Assist in handling contaminated or injured personnel.

(12) Check the operation of personnel-monitoring equipment according to established procedures, make routine chart inspections of radiation monitoring equipment, and investigate any suspicious readings revealed by such inspections or reported by others, reporting off-standard conditions to supervision.

(13) Provide either constant or intermittent monitoring of radiation-zone work guided by written permits or procedures assigned by the Health and Safety Engineer.

- (14) Recognize and evaluate changing radiological conditions that may affect the progress of the job being monitored. Assess results to assure continuance of monitored work or occupancy in a controlled and safe manner.
- (15) Assist personnel to maintain accurate and complete exposure records correctly.
- (16) Recommend steps to responsible persons to minimize exposures and to control contamination spreads.
- (17) Recognize and report unsafe conditions of either a physical or radioactive nature.
- (18) Make legible, concise, accurate, and complete records of all activities.
- (19) Collect and analyze samples for evaluation of radiation hazards to internal or external plant environments.
- (20) Operate counting equipment and interpret the results.
- (21) Operate or assist in the operation of waste disposal facilities according to established procedures and regulations.
- (22) Perform or assist in the performance of decontaminating building surfaces, tools, or equipment following established procedures or as directed.
- (23) Package dry radioactive waste material for shipment to designated disposal areas.
- (24) Prepare and change film in badges for LACBWR employees and visitors and mail exposed film to contractor.
- (25) Maintain record of all containment vessel entries.
- (26) Keep continually aware of all injuries and report those occurring in controlled areas and complete appropriate forms.
- (27) Prescribe industrial health and safety requirements for areas within the LACBWR plant.
- (28) Keep all information related to industrial health and safety updated and available to all LACBWR personnel.
- (29) Physically identify hazardous areas within the LACBWR plant as related to industrial health and safety.
- (30) Perform chemical and radiochemical analyses to support nuclear power plant operations.
- (31) Submit prompt reports on all sample results to Shift Supervisor on duty.
- (32) Maintain good housekeeping practices within the nuclear power plant, particularly in areas assigned to the health and safety department.
- (33) Possess qualifications to administer first aid treatment.
- (34) Ensure that designated first aid supply stations are well stocked.

LACBWR Process Engineer

Duties:

- (1) The Process Engineer is responsible to the Plant Superintendent through the Assistant Superintendent for the reactor and generator plant process engineering, reactor physics functions, and for the technical and functional supervision of personnel assigned to plant engineering and other activities under his direction.
- (2) Advise the Plant Superintendent, Assistant Superintendent, and Shift Supervisors of any unsafe reactor or generator plant condition; order appropriate action to correct this condition if, in his opinion, failure to issue the orders immediately could possibly result in a serious hazard to equipment, personnel, or the public. In cases where immediate action is not required, the potentially unsafe conditions will be reviewed with the Plant Superintendent, and/or the Assistant Superintendent whose decisions will govern.
- (3) Plan, recommend, direct, and evaluate the nuclear fuel management program. Perform fuel burn-up calculations and provide all necessary information for determining fuel costs, including fabrication, use charges, and chemical processing costs. Research and advise on improvements in reactor-fuel technology.
- (4) Perform nuclear analyses of the reactor core, reactor control systems, and other systems or conditions affecting core reactivity. Provide reactivity and refueling information for plant operation and administration.
- (5) Plan, recommend, direct, and evaluate changes in process systems and operating and maintenance procedures for the primary purpose of improving plant performance.
- (6) Maintain complete, accurate, and legible records of reactor performance, fuel management, core reactivity, changes to process systems, equipment and all other necessary plant engineering information.
- (7) Ensure that all applicable pressure-system construction and operating codes are followed and necessary records maintained.
- (8) Coordinate his activities with operations and maintenance supervision and other staff personnel.
- (9) Prepare necessary reports, reviews, procedures, etc. related to the process-engineering function.
- (10) Assist in training plant personnel in the area of process technology.

LACBWR Instrument and Electrical Supervisor

Duties:

(1) The Instrument and Electrical Supervisor is responsible to the Plant Superintendent through the Assistant Superintendent for the repair, maintenance, and modification of all plant instrument, control, and electrical equipment and for the technical and functional supervision of instrument technicians and electricians assigned to him.

(2) Plan, recommend, schedule, and direct all plant instrument and electrical maintenance. Recommend improvements and modifications to the Plant Superintendent. Coordinate planning and scheduling with the Operations Supervisor, Mechanical Maintenance Supervisor, Process Engineer, and Health and Safety Engineer.

(3) Enforce strict compliance with all established safety and radiation protection rules and regulations.

(4) Plan and direct an adequate instrument and electrical preventative maintenance program. Maintain complete, accurate, and legible preventative maintenance records.

(5) Maintain complete, accurate, and legible records of significant instrument and electrical tests, calibrations, modifications, and repairs.

(6) Prepare, revise, and maintain instrument and electrical maintenance procedures, training materials, equipment vendor information files, reports and other written materials as required.

(7) Conduct a continuous training program for the plant instrument technicians and electricians in order to establish and maintain a high degree of proficiency. Assist in training other personnel as assigned.

(8) Plan, recommend, and maintain an adequate and economic instrument and electrical spare parts and equipment inventory.

(9) Establish and maintain an effective program for the purpose of keeping the plant clean, orderly, and reasonably free from hazards.

(10) Minimize and control, within established limits, radiation exposure of all personnel under his supervision.

(11) Maintain and control communications equipment, where applicable, to ensure compliance with FCC requirements.

LACBWR Instrument Technician

Duties:

(1) The Instrument Technician is responsible to the Instrument and Electrical Supervisor for the repair, maintenance, and approved modification of all plant instrument and plant control systems.

(2) Recommend improvements and modifications of the instrument and control systems to the Instrument Supervisor for review.

(3) Comply with all established safety and radiation protection rules and regulations.

(4) Conduct an adequate instrument and control preventative maintenance program. Accurately complete the preventative maintenance records provided.

(5) Accurately complete instrument and control records related to tests, calibrations, modifications, and repair.

(6) Become familiar with equipment vendor information files, reports, instrument maintenance procedures, and other written materials assigned.

(7) Maintain a high degree of proficiency by attending training programs offered. Assist in training other instrument trainees assigned to him.

(8) Maintain an adequate and orderly instrument and control spare-parts and equipment inventory.

(9) Maintain electronic and pneumatic test equipment in proper operating order.

(10) Maintain a clean, orderly, and hazard-free work area.

(11) Maintain radiation exposure within permissible limits.

(12) Maintain communications equipment, where applicable, to ensure compliance to FCC requirements.

(13) Must have a general working knowledge of the reactor and generator plant systems and operating procedures.

(14) Prepare maintenance procedures for the checkout of complete instrument and control systems.

LACBWR Mechanical Maintenance Supervisor

Duties:

(1) The Mechanical Maintenance Supervisor is responsible to the Plant Superintendent through the Assistant Superintendent for the repair, maintenance, and modification of all plant mechanical process equipment, operation and maintenance of machine shop equipment, and for the technical and functional supervision of mechanics, repairmen, storekeeper, and janitor.

(2) Plan, recommend, schedule, and direct all plant mechanical maintenance, yards and grounds care and improvements, machine shop operations, and normal and special tooling functions. Recommend improvements and modifications to the Plant Superintendent. Coordinate planning and scheduling with the Operations Supervisor, Instrument and Electrical Supervisor, Process Engineer, and Health and Safety Engineer.

(3) Enforce strict compliance with all established safety and radiation protection rules and regulations.

(4) Plan and direct an adequate preventative maintenance program, including the lubrication program for all plant mechanical equipment. Maintain complete, accurate, and legible preventative maintenance and lubrication records.

(5) Maintain complete, accurate, and legible records of significant modifications and repairs.

(6) Prepare, revise, and maintain mechanical maintenance procedures, training materials, equipment vendor information files, and reports and other written materials as required.

(7) Conduct a continuous training program for the mechanical maintenance personnel in order to establish and maintain a high degree of proficiency. Assist in training other personnel as assigned.

(8) Plan and maintain an adequate and economical mechanical equipment spare parts and equipment inventory.

(9) Establish and maintain an effective program for the purpose of keeping the plant clean, orderly, and reasonably free from hazards.

(10) Minimize and control, within established limits, radiation exposure of all personnel under his supervision.