PROPOSED REVISED TECHNICAL SPECIFICATIONS

4.0 OPERATING LIMITATIONS

4.0.1 DEFINITIONS

The following terms are defined so that uniform interpretation of these specifications may be achieved. When these terms appear in capitalized type, the following definitions apply in these Technical Specifications.

ACTION

ACTION shall be those additional requirements specified as corollary statements to each principle specification and shall be part of the specifications.

AVERAGE PLANAR EXPOSURE

The AVERAGE PLANAR EXPOSURE shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATES for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

CHANNEL CALIBRATION

A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, compairson of the channel indication and/or status with other indicating and/or status derived from independent instrument channels measuring the same parameter.

4.2.1.9 The containment building shall be isolated whenever the spent fuel storage well contains irradiated fuel which has decayed less than 43* days after exposure in a critical reactor and a shipping cask for irradiated fuel is being moved by the crane on the 701 foot level or located within one cask length of the top of the spent fuel storage well or is within the spent fuel storage well. During cask movement near or at the FESW the water level in the FESW must be at least 16 feet above the top of the fuel storage rack (no more than 7 feet below the top of the FESW).

4.2.2 Reactor Vessel, Coolant, and Auxiliary Systems

4.2.2.1 Additional penetrations to the systems containing reactor coolant shall be designed, manufactured, and tested according to the provisions of the ASME Boiler and Pressure Vessel Code and the ASA Code for Pressure Piping applicable as of June 1962. These additional penetrations shall be limited to instrument connections and piping connections, the latter being no larger than 1-in. inside diameter.

4.2.2.2 The reactor coolant shall be light water and shall conform to the following requirements. (However, these requirements can be exceeded for periods of no longer than 24 hr provided the maximum chloride concentration does not exceed 1.0 ppm or whenever there is cause to maintain boron solution in the reactor).

(a) maximum conductivity (micromhos/cm)	5
(b) maximum chlorides (ppm)	0.1

4.2.2.3 Deleted

* 43 days for off loading less than one half of the core, i.e. less than 36 fuel elements. 51 days for off loading more than 36 fuel elements.

REACTOR COOLANT SYSTEM

PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

4.2.2.4 The reactor coolant system pressure, measured at a level above the normal operating water level, and temperature above RT_{NDT} shall be limited in accordance with the limit lines shown on Figure 4.2.2.4-1 as Curve No. 1 for inservice hydrostatic or leak testing; Curve No. 2 for heatup and cooldown when the core is subcritical (except when the reactor vessel is vented) and; Curve No. 3 for operation with a critical core (except for low power physics tests or when the reactor vessel is vented) with:

a. A maximum heatup rate of 100°F per hour of reactor vessel,

b. A maximum cooldown rate of 150°F per hour of reactor vessel,

c. A maximum difference between the reactor vessel flange temperature and the closure head flange temperature of 50°F,

d. The forced circulation loop pressure:

1. Less than or equal to 280 psig unless forced circulation loop temperature is at least 130°F.

2. At atmospheric pressure unless loop temperature greater than 70°F.

APPLICABILITY: At all times

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering determination of the effects of the out-of-limit condition on the fracture toughness properties of the reactor coolant system; determine that the reactor coolant system remains acceptable for continued operations or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

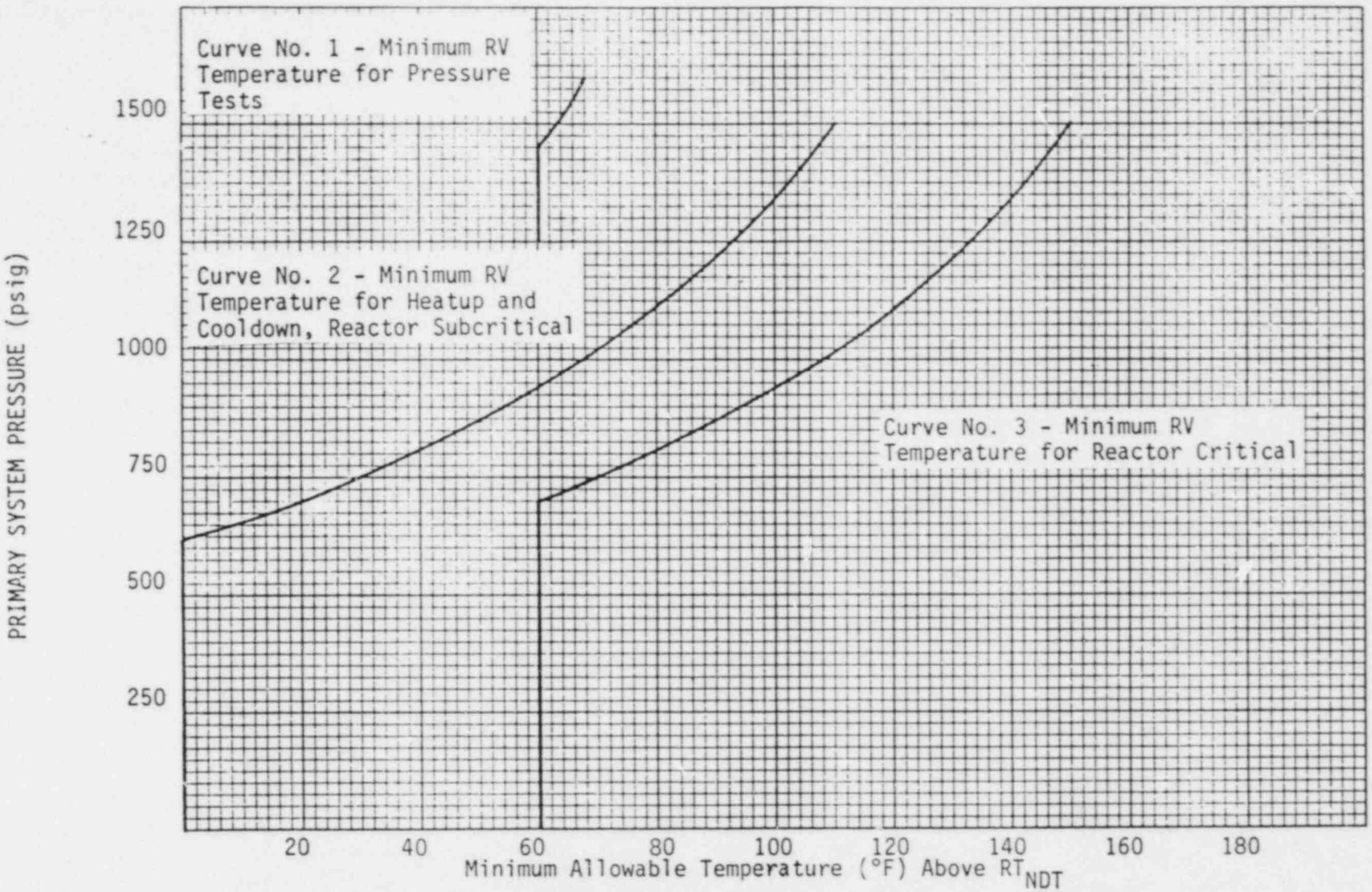
5.2.2.4.1 During system heatup, cooldown, and inservice leak and hydrostatic testing operations, at least once per 60 minutes determine:

SURVEILLANCE REQUIREMENTS (Continued)

a. The reactor vessel to be within the limits of LCO 4.2.2.4.

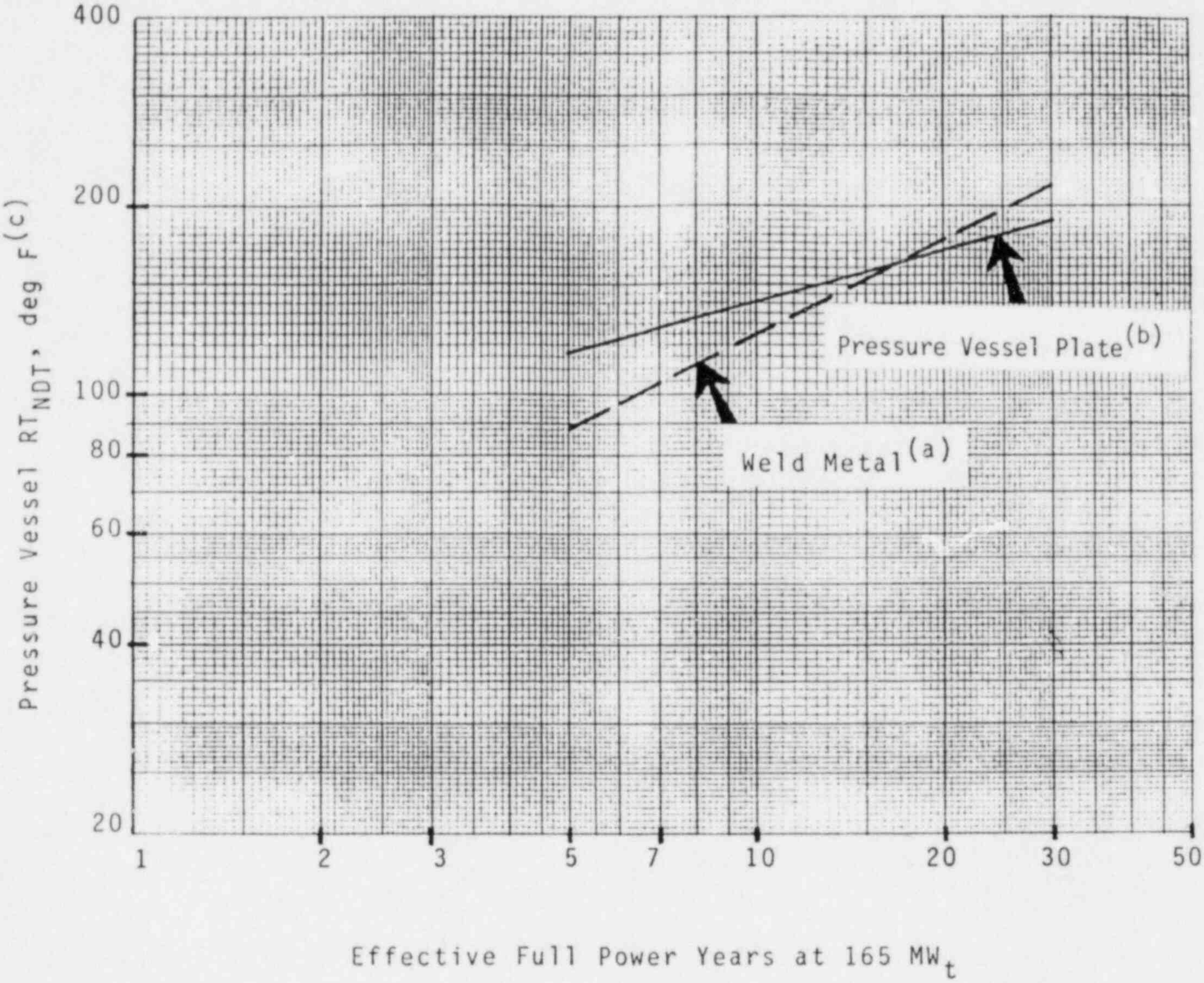
b. The forced circulation loop temperature to be within the limits of LCO 4.2.2.4.

5.2.2.4.2 The reactor material irradiation surveillance specimens shall be removed and examined to determine changes in material properties at the intervals shown in Table 4.2.2.4-1. The results of these examinations shall be used to update Figure 4.2.2.4-2, and as a factor to predict vessel lifetime.



REACTOR VESSEL MINIMUM ALLOWABLE TEMPERATURE ABOVE REFERENCE TRANSITION TEMPERATURE (RT_{NDT})

FIGURE 4.2.2.4-1



(a) Weld Metal in pressure vessel belt-line region.

(b) Material of pressure vessel Plate NP-1056.

(c) Minimum RT_{NDT} for vessel pressurization is the higher of the two.

PREDICTED REFERENCE TRANSITION TEMPERATURE VS PLANT OPERATIONAL LIFE

FIGURE 4.2.2.4-2

Table 4.2.2.4-1

REACTOR VESSEL MATERIAL IRRADIATION SURVEILLANCE SCHEDULE

REMOVAL SCHEDULE Full Power Years [a]	SURVEILLANCE CAPSULES Type	Quantity
1.4[b]	A	1
	B	1
2.5[c]	A	2
	B	2
	VW[e]	2
6[d]	A	2
	B	2
10	A	1
	B	1
Spare	A	2
	B	1

NOTES

(a) One full power year equals 60,200 $MwD_{(t)}$.
Withdrawals to be made during the nearest scheduled refueling outage.

(b) Capsules removed during August, 1972 outage.

(c) Capsules removed during May, 1975 outage.

(d) Capsules removed during November, 1980 outage.

(e) Vessel wall dosimeters.

4/5.2.2.4 PRESSURE/TEMPERATURE LIMITS

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 4 of the Safeguards Report. During heatup and cooldown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions, i.e., no thermal stresses, represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Consequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

The heatup and cooldown pressure-temperature limit, Curve #2 on Figure 4.2.2.4-1, is a composite curve. It was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 100°F per hour or with the inside wall always controlling where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall for any cooldown rate up to 150°F per hour.

The reactor vessel design specification required that the reactor vessel be designed for a maximum heatup and cooldown rate of 150°F per hour. The reactor vessel analysis for thermal stresses and fatigue indicated that the design heatup and cooldown rate of 150°F per hour would result in an excessive fatigue usage factor for the closure head boltings. However, the analysis showed that the cumulative usage factor for the bolting is satisfactory when the following limits are imposed: heatup rate not exceeding 100°F per hour; cooldown rate not exceeding 150°F per hour; and difference in temperature between the vessel shell flange and the closure head flange limited to 50°F.

PRESSURE/TEMPERATURE LIMITS (Continued)

The reactor vessel materials have been tested to determine their initial RT_{NDT}. Reactor operation and resultant fast neutron, E greater than 1 Mev, irradiation will cause an increase in the RT_{NDT}. The pressure/temperature limit curves on Figure 4.2.2.4-1 are used with Figure 4.2.2.4-2, the predicted RT_{NDT} change over the operational life of the reactor pressure vessel. The vessel plate NP-1056 will control the primary system RT_{NDT} for more than 15 EFPY of operation because of its high initial value of RT_{NDT}.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-79, reactor vessel material irradiation surveillance specimens. These specimens are installed inside the thermal shield of the reactor vessel in the core area. The neutron spectra at the irradiation specimens is accelerated by a factor of about 2 over the adjacent section of the reactor vessel. Figure 4.2.2.4-2 must be revised when the delta RT_{NDT} determined from the surveillance capsule is different from the predicted delta RT_{NDT}.

The pressure-temperature limit lines shown on Figure 4.2.2.4-1 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.2.2.4-1 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

REACTOR COOLANT SYSTEM

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

4.2.2.5 At least two reactor coolant system safety valves shall be OPERABLE with lift settings within ± 1% of the set pressure:

1 Safety valve with a set pressure of 1390 psig, and

1 Safety valve with a set pressure of 1390 or 1426 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3*.

ACTION:

With one or both of the above required reactor coolant system safety valves inoperable, either restore the inoperable valve to operable status, or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

*Except during performance of primary system hydrostatic or leak tests.

SURVEILLANCE REQUIREMENTS

5.2.2.5 The reactor coolant safety valves shall be demonstrated OPERABLE by verifying the set pressure in accordance with the schedule and requirements of Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition, with Summer 1975 addenda.

4.2.2.5 SAFETY VALVES

The safety valves are designed to meet the requirements of the ASME Boiler and Pressure Vessel Code. The reactor primary system overpressurization protection is sufficient to limit the pressure within the pressure-retaining boundaries to less than 1540 psig, which is less than 110% of the vessel design pressure of 1400 psig.

The safety valves have a minimum stamped relieving capacity of 294,612 lb per hour at a relief pressure of 1390 psig and 302,160 lb per hour at a relief pressure of 1426 psig. Three safety valves are installed. The relieving capacity with one valve inoperable is sufficient to limit the primary system pressure to less than 110 of the vessel design pressure during an abnormal limiting transient with the highest pressure, which is the MSIV closure. A high pressure scram is initiated at 1340 psia and no credit is taken for the MSIV closure scram signal, the power-flow scram, the overpower scram nor the pressure reduction due to automatic operation of the shutdown condenser heat sink.

During a normal most limiting pressure transient caused by a main steam isolation valve closure and full scram from 100% power, reactor pressure would not reach 1390 psig, the lowest safety valve set point. The safety valve function is therefore not expected to be required under the most limiting operational transient.

The testing frequency applicable to the safety valve function is provided to ensure operability and demonstrate reliability of the valves. The required testing interval varies with observed valve failures. Set point drift within ± 3% of the setpoint is not considered to be valve failure for the purposes of this test schedule. Setpoint drift > ± 3% of the setpoint will be cause to test additional valves in accordance with ASME Section XI test schedule. The popping setpoints are significantly below the 110% primary system design pressure safety limit. Therefore, adequate margin exists between the setpoint and the safety limit of 1555 psia.

For the purposes of establishing the test frequency, a valve shall be considered to have failed to function properly if the test relief pressure is determined to be outside of the allowable setpoint tolerance specified in the ASME Code to which the valve was constructed. For the LACBWR spring loaded valves which are constructed to the ASME Code Section VIII, 1962, and Nuclear Code Case N-1271, 1962, and which must be removed from the primary steam system to conduct the test, the allowable setpoint tolerance is ± 3% of the set pressure. However, when the safety valve relief pressure is set prior to installing the valve on the reactor system, the maximum deviation of the test relief pressure from the specified set pressure shall not exceed ± 1.0% of the required set pressure.

4.2.2.6 At least one forced circulation loop shall be non-isolated, i.e. suction and discharge valves open, anytime the shutdown condenser is operating, or the Reactor Building Main Steam Isolation Valve (MSIV) and the Turbine Building MSIV are non-isolated.

4.2.2.7 The suction, discharge, and discharge bypass valves of the forced circulation pumps shall operate as described in Section 2.3.3.4.

4/5.3.2 POST-ACCIDENT RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

4.3.2 The post-accident radiation monitoring instrumentation channels shown in Table 4.3-2 shall be OPERABLE with their alarm setpoints within the specified limits.

APPLICABILITY: As shown in Table 4.3-2.

ACTION:

a. With a post-accident radiation monitoring channel alarm setpoint exceeding the value shown in Table 4.3-2, adjust the setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With less than the required number of post-accident radiation monitoring channels OPERABLE, take the ACTION shown in Table 4.3-2.

c. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

5.3.2 Each post-accident radiation monitoring instrumentation channel shall be demonstrated OPERABLE by the performance of a CHANNEL CHECK daily, CHANNEL FUNCTIONAL TEST monthly, and CHANNEL CALIBRATION at least once per 18 months during applicable conditions shown in Table 4.3-2.

TABLE 4.3-2

POST-ACCIDENT RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE CONDITIONS	ALARM SETPOINT	ACTION
1. Containment Building High Range Area Radiation Monitors	1	1, 2, 3	Hi Alarm 10^3-10^4 Rad/hr	(1)
2. Stack High Range Noble Gas Effluent Monitor	1	1, 2, 3	Hi Alarm $<1.25 \times 10^2$ µCi/cc	(1)
3. Stack Radioiodine Effluent Monitor	1	1, 2, 3	Hi Alarm $\leq$ 10 µCi	(1)

(1) With the number of OPERABLE Channels less than minimum required, either restore the inoperable Channel(s) to OPERABLE status within 72 hours, or:

a) Initiate an alternate method of monitoring containment building high range radiation levels, or stack releases using alternate channel of stack monitoring system (range 10^0-10^3 µCi/cc for noble gas) or sample every 8 hours, as applicable.

- and -

b) Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

4/5.3.2 POST-ACCIDENT RADIATION MONITORING INSTRUMENTATION

The operability of the post-accident radiation monitoring is required following accidents to indicate core damage as prescribed in NUREG 0578, Section 2.1.8.b and NUREG 0737, Section II.F.1-3 and post-accident indication of noble gas effluent as prescribed in NUREG 0578, Section 2.1.8.b and NUREG 0737, Section II.F.1-1.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability. Calibrations of Containment Building Area High Range Radiation Monitors may be performed under the requirements of NUREG 0578.

EMERGENCY CORE COOLING SYSTEMS

4/5.2.23 HIGH PRESSURE CORE SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

4.2.23.1 The high pressure core spray (HPCS) system shall be operable with:

a. For the high pressure core spray mode:

1. Two OPERABLE high pressure core spray pumps, and

2. An OPERABLE flow path capable of taking suction from the overhead storage tank and transferring the water through the core spray header to the reactor pressure vessel.

b. For the low pressure core spray mode:

1. An OPERABLE flow path capable of transferring water from the overhead storage tank to the reactor pressure vessel by gravity.

APPLICABILITY:

a. OPERATIONAL CONDITIONS 1[a], 2[a] and 3[a] for the high pressure core spray mode;

b. OPERATIONAL CONDITIONS 1, 2, 3, and 4[b] for the low pressure core spray mode.

[a]Except with the boron injection system operating.

[b]The low pressure core spray mode and the overhead storage tank are not required to be OPERABLE provided that the:

1. Reactor upper cavity is flooded or being flooded from, or drained to the overhead storage tank; or

2. The Alternate Core Spray system is OPERABLE.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

a. For the high pressure core spray mode:

   1. With one of the above required high pressure core spray pumps inoperable, POWER OPERATION may continue provided the manual depressurization system and the alternate core spray system are OPERABLE; restore two pumps to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. For the low pressure core spray mode:

   1. In OPERATIONAL CONDITION 1, 2, or 3, with the low pressure core spray mode inoperable, POWER OPERATION may continue provided that the alternate core spray system is OPERABLE; restore the low pressure core spray mode to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   2. In OPERATIONAL CONDITION 4[b], with the low pressure core spray mode inoperable, suspend all operations that have a potential for draining the reactor vessel.

SURVEILLANCE REQUIREMENTS

5.2.23.1 The high pressure core spray system shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying the valve actuation nitrogen supply pressure from the regulator to be 30 $\pm$ 10 psig.

b. At least once per 31 days by verifying the valve actuation nitrogen supply bottle pressure to be greater than or equal to 100 psig.

c. For the high pressure core spray mode:

   1. Each COLD SHUTDOWN, if not performed within the previous 3 months, by cycling each power operated or automatic valve in the flow path through at least one complete cycle of full travel.

d. At least once per 18 months, during shutdown by performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence, and:

   1. Verifying that each automatic valve in the flow path actuates to its correct position on a:

      (a) Release of a boron injection actuation signal for the high pressure core spray mode, and

      (b) Low pressure core spray mode actuation signal.

   2. Verifying that both HPCS pumps start automatically upon receipt of a high pressure core spray mode actuation signal.

   3. Verifying that the valve actuation nitrogen supply pressure regulators operate to control valve actuation pressure at 30 $\pm$ 10 psig when cycling the associated valves.

EMERGENCY CORE COOLING SYSTEM

MANUAL DEPRESSURIZATION SYSTEM

LIMITING CONDITION FOR OPERATION

4.2.23.2 The manual depressurization system (MDS) shall be OPERABLE with:

a. Two OPERABLE shutdown condenser steam inlet valves, and

b. Two OPERABLE shutdown condenser condensate line reactor vent valves.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With one of the above required steam inlet valves and/or reactor vent valves inoperable, POWER OPERATION may continue provided the high pressure core spray system is OPERABLE; restore the inoperable valve(s) to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

5.2.23.2 The manual depressurization system shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying the valve actuation nitrogen supply pressure from the regulator to be 35 $\pm$ 5 psig.

b. Each COLD SHUTDOWN, if not performed within the previous 3 months, by verifying that each steam inlet valve and each reactor vent valve is manually OPERABLE from the control room by cycling each valve through at least one complete cycle of full travel.

EMERGENCY CORE COOLING SYSTEM

ALTERNATE CORE SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

4.2.23.3 The alternate core spray (ACS) system shall be OPERABLE with:

a. Two OPERABLE diesel driven ACS pumps, each with a separate fuel storage tank containing a minimum of 270 gallons of fuel for pump 1A and 108 gallons of fuel for pump 1B.

b. OPERABLE redundant control valves, and

c. An OPERABLE flow path capable of taking suction from the Mississippi River and transferring the water to the reactor pressure vessel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one of the above required diesel driven ACS pumps and/or redundant control valves inoperable, POWER OPERATION may continue provided that the high pressure core spray system is OPERABLE; restore two pumps and both redundant control valves to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In the event the ACS system is actuated and injects water into the reactor coolant system, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

SURVEILLANCE REQUIREMENTS

5.2.23.3 The alternate core spray system shall be demonstrated OPERABLE:

a. Each COLD SHUTDOWN, if not performed within the previous 3 months, by cycling each power operated or automatic valve in the flow path through at least one complete cycle of full travel.

b. At least once per 18 months, during shutdown, by:

1. Performing a system functional test which excludes actual injection of coolant into the reactor vessel, but which includes simulated automatic actuation of the system throughout its emergency operating sequence, and:

(a) Verifying that each:

(1) Automatic valve in the flow path actuates to its correct position upon actuation of a low reactor water level signal coincident with a high containment pressure signal,

(2) Automatic valve closes upon deactuation of the low reactor water level signal, and

(3) Automatic valve reopens upon reactuation of the low reactor water level signal.

(b) Verifying that each diesel driven ACS starts automatically upon receipt of a high containment pressure signal.

2. Verifying that each diesel driven ACS pump operates for greater than or equal to 2[illegible] minutes with a pressure greater than or equal to 90 psig, as measured by PI-38-35-801, at a flow rate greater than or equal to 900 gpm.

EMERGENCY CORE COOLING SYSTEM

OVERHEAD STORAGE TANK

LIMITING CONDITION FOR OPERATION

4.2.23.4 The overhead storage tank shall be OPERABLE with:

a. A minimum contained water volume of 15,000 gallons, equivalent to a level of 40 inches.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and 4*.

ACTION:

a. With the overhead storage tank inoperable:

(1) In OPERATIONAL CONDITION 1, 2 or 3, declare the HPCS system high pressure core spray mode inoperable and be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 30 hours.

(2) In OPERATIONAL CONDITION 4*, declare the HPCS system low pressure core spray mode inoperable and suspend all operations that have a potential for draining the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.5.4.1 The overhead storage tank shall be demonstrated OPERABLE by:

a. At least once per 7 days, verifying the minimum contained water volume in the tank.

b. At least once per 18 months, verifying that the demineralized water makeup valve opens when tank level is:

(1) Greater than or equal to 80 inches with the makeup valve control switch in the open position, and

(2) Greater than or equal to 50 inches with the makeup valve control switch in the closed position.

---

*The overhead storage tank and low pressure core spray mode of the HPCS system are not required to be OPERABLE provided that the:

1. Reactor upper cavity is flooded, or being flooded from or drained to the overhead storage tank; or
2. The Alternate Core Spray System is OPERABLE.

4/5.2.23 EMERGENCY CORE COOLING SYSTEMS

The OPERABILITY of two independent ECCS systems, the high pressure core spray (HPCS) system and the alternate core spray (ACS) system with the manual depressurization system (MDS), ensures that sufficient emergency core cooling capability will be available in the event of a loss-of-coolant accident (LOCA) assuming the loss of one ECCS system through any single failure consideration. Either system is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest reactor coolant system cold leg pipe downward.

4/5.2.23.1 HIGH PRESSURE CORE SPRAY SYSTEM

The high pressure core spray (HPCS) system high pressure core spray mode is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the reactor coolant system and a loss-of-coolant which does not result in rapid depressurization of the reactor vessel. The HPCS system high pressure core spray mode permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HPCS system high pressure core spray mode continues to operate until reactor vessel pressure is below the pressure at which alternate core spray system operation maintains core cooling. The HPCS system high pressure core spray mode consists of two pumps, and associated valves and piping. The pumps each have the capacity to deliver 50 gallons per minute to the reactor at a pressure in excess of reactor operating pressure. The system is actuated by automatically starting the pumps on a signal from either one of two reactor water level sensing channels.

A function of the HPCS system is the low pressure core spray mode which provides, by gravity feed which bypasses the HPCS pumps, water from the OHST to the reactor vessel when the reactor is at low pressure or the reactor vessel head is removed, to provide a source for flooding of the core in case of accidental draining. The HPCS system low pressure core spray mode is actuated on a coincidence signal from the reactor pressure and reactor water level sensing channels.

4/5.2.23.2 MANUAL DEPRESSURIZATION SYSTEM AND 4/5.2.23.3 ALTERNATE CORE SPRAY SYSTEM

Along with the HPCS system, adequate core cooling is assured by the demonstrated OPERABILITY of the manual depressurization system (MDS) and the alternate core spray (ACS) system.

The MDS is manually initiated. It serves to reduce reactor pressure rapidly so that the ACS system can perform its function. The MDS provides the ACS system with the capability of performing its function in both long-term and short-term cooling modes.

The ACS system is provided to assure that the core is adequately cooled following a loss-of-coolant accident. The system is comprised of two diesel-driven pumps and associated valves and piping. The water supply for the ACS system is the Mississippi River. The ACS system is capable of starting to provide cooling water to the reactor only when reactor pressure drops to a pressure of approximately 150 psig.

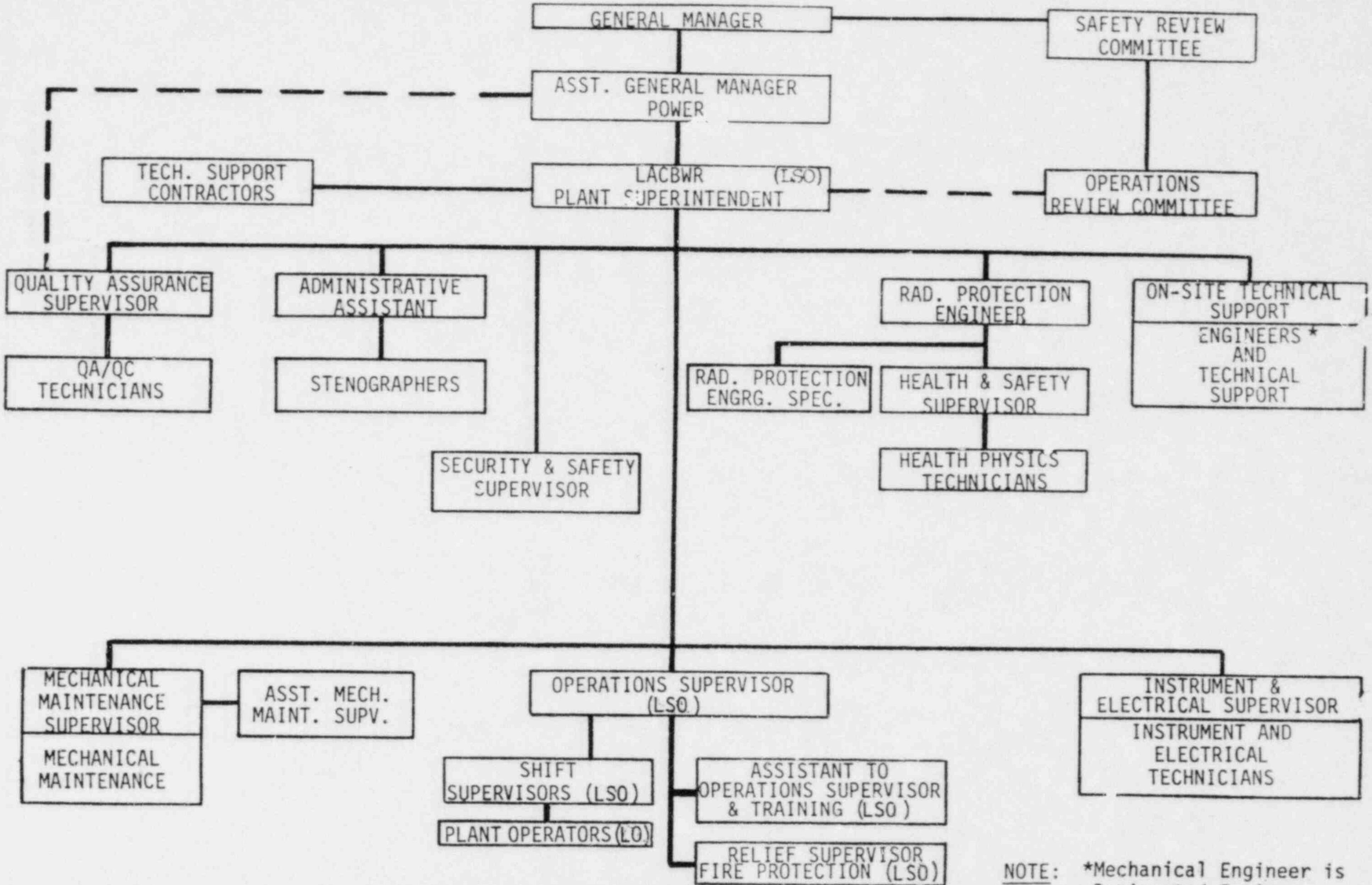
Two Containment Building pressure sensors and two reactor water level sensors provide the signals to actuate operation of the ACS system. A containment pressure of 5 psig will cause a sensor to generate a signal to actuate and automatically start its respective pump. A motor operated valve associated with each pump will be opened on a low reactor water level signal from either of two reactor water level sensors coincident with high Containment Building pressure. Similarly the second pump will start and the valve will open when the respective set of instruments generate the required signals.

The surveillance requirements provide adequate assurance that MDS will be OPERABLE when required. A complete functional test results in reactor blowdown and therefore is only performed during shutdown.

The surveillance requirements provide adequate assurance that the ACS system will be OPERABLE when required. All active components are not testable and a full functional test requires reactor shutdown.

4/5.2.23.4 OVERHEAD STORAGE TANK

The OPERABILITY of the Containment Building overhead storage tank (OHST) as part of the ECCS ensures that a sufficient supply of water is available for injection by the HPCS system in the event of a LOCA. Demineralized water for the high pressure core spray system is supplied from the 42,000-gallon overhead storage tank, located in the dome of the Containment Building. The high pressure core spray cooling system is connected near the bottom of the tank. The containment building spray system is connected to a standpipe within the tank with the top of the standpipe located so that 15,000 gallons are reserved for the high pressure core spray system. The demineralized water system replenishes the OHST for long term cooling. The contained water volume limit includes an allowance for water not usable because of tank dis harge line location or other physical characteristics.



DAIRYLAND POWER COOPERATIVE
LACBWR FACILITY ORGANIZATION

FIGURE 6.2.2-1

NOTE: *Mechanical Engineer is Designated Engineer Responsible for Fire Protection Program.