ENCLOSURE 4

H.B.ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
NRC DOCKET NO. 50-261/LICENSE NO. DPR-23
REQUEST FOR LICENSE AMENDMENT

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(1669RNP)

ENCLOSURE 5

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
NRC DOCKET NO. 50-261/LICENSE NO. DPR-23
REQUEST FOR LICENSE AMENDMENT

TECHNICAL SPECIFICATION PAGES

3.1.1.4 Reactor Coolant System (RCS) Vent Path

A. When the RCS temperature is greater than 200°F, the RCS vent paths consisting of at least two valves in series powered from emergency buses, shall be operable (except that valves RC-567, 568, 569, and 570 shall be closed with power removed from the valve actuators) from each of the following locations:

1. Reactor Vessel Head

2. Pressurizer Steam Space

B. When the RCS temperature is greater than 200°F, RCS vent path valves RC-571 and 572 shall be closed, except that they may be periodically cycled to depressurize the RCS vent system should leakage past RC-567, 568, 569, or 570 occur.

C. With less than the above required equipment operable, perform the following as applicable:

1. With the Reactor Vessel Head vent path inoperable, restore the vent path to operable status within 30 days or be in Hot Shutdown within 6 hours and Cold Shutdown within the following 30 hours.

2. With the Pressurizer Steam Space vent path inoperable, restore the vent path to operable status within 30 days, or prepare and submit a special report to the NRC within the following 14 days detailing the cause of the inoperable vent path, the action being taken to restore the vent path to operable status, the estimated date for completion of repairs, and any compensatory action being taken while the vent path is inoperable.

3. With both the Reactor Vessel Head and Pressurizer Steam Space vent paths inoperable, restore at least one vent path to operable status within 7 days or be in HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours.

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INSERT A

3.1.1.5 Relief Valves

Whenever T_{avg} is above 350 degrees F or the reactor is critical both power operated relief valves(PORVs) and their associated block valves shall be OPERABLE[1].

a. With one or both PORVs inoperable because of leakage through the PORV resulting in excessive RCS leakage, i.e., not in accordance with the leakage criteria in Technical Specification 3.1.5.2:

   1. Within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s); or

   2. Be in at least HOT SHUTDOWN condition using normal operating procedures within the next 12 hours and cool down the RCS below a T_{avg} of 350 degrees F within the following 12 hours[2].

b. With one PORV inoperable due to causes other than (1) leakage through the PORV resulting in excessive RCS leakage or (2) discretionary isolation to prevent minor leakage from becoming excessive:

   1. Within 1 hour either restore the PORV to OPERABLE status or close its associated block valve and remove power from the block valve; and

   2. Restore the PORV to OPERABLE status within the following 72 hours; or

   3. Be in at least HOT SHUTDOWN condition using normal operating procedures within the next 12 hours and cool down the RCS below a T_{avg} of 350 degrees F within the following 12 hours.

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1 PORV block valves shall not be considered inoperable solely because either their normal or emergency power source is inoperable.

2 Power operation may continue pursuant to the requirements of this specification with the associated block valve closed, as a precautionary measure, to isolate minor leakage prior to the RCS leakage exceeding the leakage criteria in Technical Specification 3.1.5.2, with power maintained to the block valve during the period of the discretionary isolation.

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c. With both PORVs inoperable due to causes other than (1) leakage through the PORV resulting in excessive RCS leakage or (2) discretionary isolation to prevent minor leakage from becoming excessive:

   1. Within 1 hour either restore at least one PORV to OPERABLE status; or close its associated block valve and remove power from the block valve; and

   2. Be in at least HOT SHUTDOWN condition using normal operating procedures within the next 12 hours and cool down the RCS below a T_{avg} of 350 degrees F within the following 12 hours.

d. With one or both block valves inoperable[1]:

   1. Within 1 hour restore the block valve(s) to OPERABLE status or place the associated PORV(s) in manual control; and

   2. Restore at least one block valve to OPERABLE status within the next hour if both block valves are inoperable; and

   3. Restore any remaining inoperable block valve to operable status within 72 hours; or

   4. Be in at least HOT SHUTDOWN condition using normal operating procedures within the next 12 hours and cool down the RCS below a T_{avg} of 350 degrees F within the following 12 hours.

e. For this specification, reactor startup, heatup and entry into operational conditions with T_{avg} greater than or equal to 350 degrees F may continue so long as the limits of the associated action statements are met.

f. During performance of the required surveillance testing of the PORVs and their associated Block Valves, the respective valve train need not be declared inoperable nor the associated action statements performed unless the associated valves are determined to be inoperable via this testing. Testing of no more than one train at a time may be performed and the time in the out of normal test configuration shall not exceed 24 hours.

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1 PORV block valves shall not be considered inoperable solely because either their normal or emergency power source is inoperable.

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Basis

At the conditions of the RCS temperature (T_{avg}) greater than 350 degrees or the reactor critical, the power-operated relief valves (PORVs) provide an RCS pressure boundary, manual RCS pressure control for mitigation of accidents, and automatic RCS pressure relief to minimize challenges to the safety valves.

Providing an RCS pressure boundary and manual RCS pressure control for mitigation of a steam generator tube rupture (SGTR) are the safety-related functions of the PORVs at the conditions noted above. The capability of the PORV to perform its function of providing an RCS pressure boundary requires that the PORV or its associated block valve is closed. The capability of the PORVs to perform manual RCS pressure control for mitigation of a SGTR accident is based on manual actuation and does not require the automatic RCS pressure control function. The automatic RCS pressure control function of the PORVs is not a safety-related function at the conditions noted above. The automatic pressure control function limits the number of challenges to the safety valves, while the safety valves perform the safety function of RCS overpressure protection. Therefore, the automatic RCS pressure control function of the PORVs does not have to be available for the PORVs to be OPERABLE.

Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable. Operation with the block valves open is preferred. This allows the PORVs to perform automatic RCS pressure relief should the RCS pressure actuation setpoint be reached. However, operation with the block valve closed to isolate PORV leakage is permissible since automatic RCS pressure relief is not a safety-related function of the PORVs.

The ability to operate with the block valve(s) closed with power maintained to the block valve(s) is only intended to permit operation of the plant for a limited period of time not to exceed the next refueling outage so that maintenance can be performed on the PORVs to eliminate the leakage condition. Power is maintained to the block valve(s) so that it is operable and may be subsequently opened to allow the PORV to be used to control reactor coolant system pressure. Closure of the block valve(s) establishes reactor coolant pressure boundary integrity for a PORV that has leakage resulting in excessive RCS leakage. (Reactor coolant pressure boundary integrity takes priority over the capability of the PORV to mitigate an overpressure event.) The PORVs should normally be available for automatic mitigation of overpressure events and should be returned to OPERABLE status prior to exceeding cold shutdown following the associated refueling outage.

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The OPERABILITY of the PORVs and block valves at the conditions noted above is based on their being capable of performing the following functions:

1. Maintaining the RCS pressure boundary,

2. Manual control of PORVs to control RCS pressure as required for SGTR mitigation,

3. Manual closing of a block valve to isolate a stuck open PORV,

4. Manual closing of a block valve to isolate a PORV with excessive seat leakage, and

5. Manual opening of a block valve to unblock an isolated PORV to allow it to be used to control RCS pressure for SGTR mitigation.

3.1.2 Heatup and Cooldown

3.1.2.1 The reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1-1a and Figure 3.1-2a (for vessel exposure up to 12.5 EFPY) or Figure 3.1-1b and Figure 3.1-2b (for vessel exposure up to 15 EFPY). The 15 EFPY curves may be used for operation prior to the end of 12.5 EFPY. These limitations are as follows:

a. Over the temperature range from cold shutdown to hot operating conditions, the heatup rate shall not exceed 60°F/hr. in any one hour.

b. Allowable combinations of pressure and temperature for a specific cooldown rate are below and to the right of the limit lines for that rate as shown on Figure 3.1-2a or 3.1-2b (as appropriate). This rate shall not exceed 100°F/hr. in any one hour. The limit lines for cooling rates between those shown in Figure 3.1-2a or Figure 3.1-2b may be obtained by interpolation.

c. Primary system hydrostatic leak tests may be performed as necessary, provided the temperature limitation as noted on Figure 3.1-1a or Figure 3.1-1b (as appropriate) is not violated. Maximum hydrostatic test pressure should remain below 2350 psia.

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d. ~~The overpressure protection system shall be operable whenever the RCS temperature is below 350°F and not vented to the containment. One PORV may be inoperable for seven days. If the inoperable PORV has not been returned to service within 7 days, or if at any time both PORVs become inoperable, then one of the following actions should be completed within 12 hours:~~

~~1. Cooldown and depressurize the RCS or~~

INSERT B

3.1.2.1 d. The overpressure protection system shall be OPERABLE[1], with both power operated relief valves OPERABLE with a lift setting of less than or equal to 420 psi whenever any RCS cold leg temperature is less than or equal to 350 degrees F and when the head is on the reactor vessel and the RCS is not vented to the containment.

1. With one PORV inoperable and T_{avg} greater than 200 degrees and any RCS cold leg temperature less than 350 degrees:

   A. Restore the inoperable PORV to OPERABLE status within 7 days; or

   B. Depressurize and vent the RCS to the CV within the next 12 hours.

2. With one PORV inoperable and T_{avg} less than or equal to 200 degrees F:

   A. Restore the inoperable PORV to OPERABLE status within 24 hours; or

   B. Complete depressurization and venting of the RCS to the CV within an additional 12 hours.

3. With both PORVs inoperable, complete depressurization and venting of the RCS to the CV within 12 hours.

4. With the RCS vented per 1, 2, or 3, verify the vent pathway:

   A. At least once per 31 days when the pathway is provided by a valve(s) that is locked, sealed, or otherwise secured in the open position; or

   B. At least once per shift.

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[1] The overpressure protection system shall not be considered inoperable solely because either the normal or emergency power source for the PORV block valves is inoperable.

<u>INSERT B - Con't</u>

5. In the event the PORVs or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.3 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or RCS vent(s) on the transient and any corrective action necessary to prevent recurrence.

6. For this specification, reactor startup, heatup and entry into operational conditions with T_{avg} greater than or equal to 350 degrees F may continue so long as the limits of the associated action statements are met.

~~2. Heatup the RCS to above 350°F.~~

e. Operation of the overpressure protection system to relieve a pressure transient must be reported as a Special Report to the NRC within 30 days of operation.

3.1.2.2 The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the vessel is below 120°F

3.1.2.3 The pressurizer shall neither exceed a maximum heatup rate of 100°F/hr nor a cooldown rate of 200°F/hr. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

3.1.2.4 Figures 3.1-1b and 3.1-2b shall be updated periodically in accordance with the following criteria and procedures before the calculated exposure of the vessel exceeds the exposure for which the figures apply.

a. At least 60 days before the end of the integrated power period for which Figures 3.1-1b and 3.1-2b apply, the limit lines on the figures shall be updated for a new integrated power period utilizing methods derived from the ASME Boiler and Pressure Vessel Code, Section III, Appendix G and in accordance with applicable appendices of 10CFR50. These limit lines shall reflect any changes in predicted vessel neutron fluence over the integrated power period or changes resulting from the irradiation specimen measurement program.

b. The results of the examinations of the irradiation specimens and the updated heatup and cooldown curves shall be reported to the Commission within 90 days of completion of the examinations.

Basis

The ability of the large steel pressure vessel that contains the reactor core and its primary coolant to resist fracture constitutes an important factor in ensuring safety in the nuclear industry. The beltline region of the reactor pressure vessel is the most critical region of the vessel because it is subjected to neutron bombardment. The overall effects of fast neutron irradiation on the mechanical properties of low alloy ferritic pressure vessel

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steels such as ASTM A302 Grade B parent material of the H. B. Robinson Unit No. 2 reactor pressure vessel are well documented in the literature. Generally, low alloy ferritic materials show an increase in hardness and other strength properties and a decrease in ductility and impact toughness under certain conditions of irradiation. Accompanying a decrease in impact strength is an increase in the temperature for the transition from brittle to ductile fracture.

A method for guarding against fast fracture in reactor pressure vessels has been presented in Appendix G, "Protection Against Non-Ductile Failure," to Section III of the ASME Boiler and Pressure Vessel Code. The method utilizes fracture mechanics concepts and is based on the reference nil-ductility temperature, RT_{NDT}.

RT_{NDT} is defined as the greater of: 1) the drop weight nil-ductility transition temperature (NDTT per ASTM E-208) or 2) the temperature 60°F less than the 50 ft-lb (and 35 mils lateral expansion) temperature as determined from Charpy specimens oriented in a direction normal to the major working direction of the material. The RT_{NDT} of a given material is used to index that material to a reference stress intensity factor curve (K_{IR} curve) which appears in Appendix G of the ASME Code. The K_{IR} curve is a lower bound of dynamic, crack arrest, and static fracture toughness results obtained from several heats of pressure vessel steel. When a given material is indexed to the K_{IR} curve, allowable stress intensity factors can be obtained for this material as a function of temperature. Allowable operating limits can then be determined utilizing these allowable stress intensity factors.

The Certified Material Test Reports (CMTRs) for the original steam generators provided records of Charpy V-notch tests performed at +10°F. Acceptable Charpy V-notch tests of +10°F indicate RT_{NDT} is at or below this temperature. The steam generator lower assemblies were replaced in 1984 and the material tests results indicate the highest RT_{NDT} is 60°F or below. The ASME code recommends that hydrostatic tests be performed at a temperature not lower than RT_{NDT} plus 60F°, thus the pressurizing temperature for the steam generator shell is established at 120°F to provide protection against nonductile failure at the test pressure.

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The value of RT_{NDT}, and in turn the operating limits of nuclear power plants, can be adjusted to account for the effects of radiation on the reactor vessel material properties. The radiation embrittlement or changes in mechanical properties of a given reactor pressure vessel still can be monitored by a surveillance program such as the Carolina Power & Light Company, H. B. Robinson Unit No. 2 Reactor Vessel Radiation Surveillance Program[1] where a surveillance capsule is periodically removed from the operating nuclear reactor and the encapsulated specimens tested. These data are compared to data from pertinent radiation effects studies and an increase in the Charpy

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V-notch 30 ft-lb temperature (ΔRT_{NDT}) due to irradiation is added to the original RT_{NDT} to adjust the RT_{NDT} for radiation embrittlement. This adjusted RT_{NDT} (RT_{NDT} initial + RT_{NDT}) is utilized to index the material to the K_{IR} curve and in turn to set operating limits for the nuclear power plant which take into account the effects of irradiation on the reactor vessel materials. Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods (2) derived from Appendix G to Section III of the ASME Boiler and Pressure Vessel Code. The approach specifies that the allowable total stress intensity factor (K_I) at any time during heatup or cooldown cannot be greater than that shown on the K_{IR} curve in Appendix G for the metal temperature at that time. Furthermore, the approach applies an explicit safety factor of 2.0 on the stress intensity factor induced by pressure gradients.

Following the generation of pressure-temperature curves for both the steady state and finite heatup rate situations, the final limit curves are produced in the following fashion. First, a composite curve is constructed based on a point-by-point comparison of the steady state and finite heatup rate data. At any given temperture, the allowable pressure is taken to be the lesser of the two values taken from the curves under consideration. The composite curve is then adjusted to allow for possible errors in the pressure and temperature sensing instruments.

The use of the composite curve is mandatory in setting heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling analysis switches from the O.D. to the I.D. location; and the pressure limit must, at all times, be based on the most conservative case. The cooldown analysis proceeds in the same fashion as that for heatup, with the exception that the controlling location is always at the I.D. position. The thermal gradients induced during cooldown tend to produce tensile stresses at the I.D. location and compressive stresses at the O.D. position. Thus, the I.D. flaw is clearly the worst case.

As in the case of heatup, allowable pressure temperature relations are generated for both steady state and finite cooldown rate situations. Composite limit curves are then constructed for each cooldown rate of

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interest. Again adjustments are made to account for pressure and temperature instrumentation error.

The overpressure protection system consists of two operable pressurizer Power Operated Relief Valves (PORV) connected to the station instrument air system, a backup nitrogen supply, and the associated electronics.

(INSERT C - SEE NEXT PAGE)

Pages 3.1-8 through 3.1-10 have been deleted.

References

1. S. E. Yanichko, "Carolina Power & Light Company, H. B. Robinson Unit No. 2 Reactor Vessel Radiation Surveillance Program," Westinghouse Nuclear Energy Systems - WCAP-7373 (January 1970).

2. E. B. Norris, "Reactor Vessel Material Surveillance Program for H. B. Robinson Unit No. 2, Analysis of Capsule V," Southwest Research Institute - Final Report SWRI Project No. 02-4397.

INSERT C

Basis

The TS requirements for LTOP apply when T_{avg} is less than 350 degrees F and the RCS is not vented to the containment. During these conditions, one train (or channel) of the LTOP system is capable of mitigating an LTOP event that is bounded by the largest mass addition to the RCS or by the largest increase in RCS temperature that can occur. The largest mass addition to the RCS is limited based upon the assumption that no more than a fixed number of pumps are capable of providing makeup or injection into the RCS. Hence, this is a matter important to safety that pumps in excess of this design basis assumption for LTOP not be capable of providing makeup or injection to the RCS. In this regard the SI Pump breakers are required to be racked out at less than 350 degrees F RCS temperature.

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Capsule No. 3 leads the vessel maximum exposure by a factor of 2.2 and is scheduled to be removed after twenty years. Thus, sample No. 3 will provide data for an exposure to the vessel of approximately forty years.

Capsule Nos. 4 and 5 lag the maximum vessel exposure by factors of 0.7 and 0.5, respectively. Thus, Capsule No. 4, which is scheduled to be removed after thirty years, provides data for a vessel exposure of twenty-one years and Capsule No. 5, which is scheduled to be removed at forty years, provides data for a vessel exposure of twenty years.

In addition to the capsules discussed above, there are three spares. Two are located at the same location as Capsule No. 5 and one is located at the same location as Capsule No. 4.

4.2.3 Primary Pump Flywheels

The flywheels shall be visually examined at the first refueling after each ten year inspection. At the fourth refueling after each ten tear inspection and at each fourth refueling thereafter, the outside surfaces shall be examined by ultrasonic methods.

INSERT D - SEE NEXT PAGE

References

(1) FSAR, Section 4.4
(2) FSAR, Volume 4, Tab VII, Question VI.C

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4.2.4 Relief Valves

4.2.4.1 In addition to the requirements of Specification 4.0.1, each PORV shall be demonstrated OPERABLE at each refueling by:

a. Performance of a CHANNEL CALIBRATION, and

b. Operating the PORV through one complete cycle of full travel while T_{avg} is greater than 350 degrees F and the reactor is subcritical.

c. Operating the solenoid air control valves and check valves for their associated accumulators in PORV control systems through one complete cycle of full travel or function testing of individual components.

4.2.4.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed in order to meet the requirements of Specification 3.1.1.5.a., b. or c.

4.2.4.3 The accumulator for the PORVs shall be demonstrated OPERABLE at each refueling by isolating the normal air and nitrogen supplies and operating the valves through a complete cycle of full travel.

4.2.5 Low Temperature Overpressure Protection

4.2.5.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE; and

b. Performance of a CHANNEL CALIBRATION at each refueling shutdown; and

c. Verifying the PORV Block Valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

Basis

The OPERABILITY of two PORVs for low temperature overpressure protection or an RCS vent ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 350°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle RCP with the secondary water temperature of the steam generator less than 50°F above the RCS cold leg temperatures, or (2) the start of three charging pumps with injection into a water-solid RCS.

The maximum allowed PORV setpoint for the Low Temperature Overpressure Protection system (LTOP) is derived by analyses which model the performance of the LTOP assuming various mass input and heat input transients. Operation with a PORV setpoint less than or equal to the maximum setpoint ensures that Appendix G criteria will not be violated with consideration for a maximum pressure over-shoot beyond the PORV setpoint which can occur as a result of time delays in signal processing and valve opening, instrument uncertainties, and single failure. To ensure that mass and heat input transients more severe than those assumed cannot occur, Technical Specifications require the power supply breakers of all three safety injection pumps be racked out while in hot shutdown and below 350°F with the reactor vessel head installed and the RCS is not vented to containment and disallow start of an RCP if secondary temperature is more than 50°F above primary temperature.

The maximum allowed PORV setpoint for the LTOP will be updated based on the results of examinations of reactor vessel material irradiation surveillance specimens performed as required by 10 CFR Part 50 Appendix H.

Surveillance Requirements provide the assurance that the PORVs and Block Valves can perform their required functions. Specification 4.2.4.1 addresses PORVs, 4.2.4.2 the Block Valves, and 4.2.4.3 the independent pneumatic power source. Specification 4.2.5.1 addresses the PORV overpressure protection functions and 4.2.5.2 addresses RCS vent pathways.

Surveillance Requirement 4.2.4.1.a provides assurance the actuation instrumentation for automatic PORV actuation is calibrated such that the automatic PORV actuation signal is within the required pressure range even though automatic actuation capability of the PORV is not necessary for he PORV to be OPERABLE in the power operating and hot shutdown conditions greater than 350 degrees F.

Surveillance Requirement 4.2.4.1.b provides assurance the PORV is capable of opening and closing. The associated block valve should be closed prior to stroke testing a PORV to preclude depressurization of the RCS. This test will be done at hot shutdown with T_{avg} greater than 350 degrees F before the PORV is required for overpressure protection in Technical Specification 3.1.2.1.d.

Surveillance Requirement 4.2.4.1.c. provides assurance that the mechanical and electrical aspects of the control system are functional.

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Surveillance Requirement 4.2.4.2 addresses the block valves. The block valves are exempt from the surveillance requirements to cycle the valves when they have been closed to comply with Technical Specification 3.1.1.5.a.,b. or c. This precludes the need to cycle the valves with a full system differential pressure or when maintenance is being performed to restore an inoperable PORV to OPERABLE status. Also, this limits the challenges to the primary function of the Block Valve which is to provide an RCS pressure boundary for a degraded PORV.

Surveillance Requirement 4.2.4.3 provides assurance of operability of the accumulators and that the accumulators are capable of supplying sufficient Nitrogen to operate the PORV(s) if they are needed for RCS pressure control and normal Nitrogen and the backup Instrument Air systems are not available. Backup Instrument Air is supplied when the accumulator reaches its low pressure setpoint.

Surveillance Requirement 4.2.5.1 provides assurance that the instrumentation for the actuation of the LTOP function of PORVs is calibrated to provide automatic actuation of the PORVs for low temperature conditions. Also, the flow path to the PORV is assured to be open.

3.1.1.4 Reactor Coolant System (RCS) Vent Path

A. When the RCS temperature is greater than 200°F, the RCS vent paths consisting of at least two valves in series powered from emergency buses, shall be operable (except that valves RC-567, 568, 569, and 570 shall be closed with power removed from the valve actuators) from each of the following locations:

   1. Reactor Vessel Head

   2. Pressurizer Steam Space

B. When the RCS temperature is greater than 200°F, RCS vent path valves RC-571 and 572 shall be closed, except that they may be periodically cycled to depressurize the RCS vent system should leakage past RC-567, 568, 569, or 570 occur.

C. With less than the above required equipment operable, perform the following as applicable:

   1. With the Reactor Vessel Head vent path inoperable, restore the vent path to operable status within 30 days or be in HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours.

   2. With the Pressurizer Steam Space vent path inoperable, restore the vent path to operable status within 30 days, or prepare and submit a special report to the NRC within the following 14 days detailing the cause of the inoperable vent path, the action being taken to restore the vent path to operable status, the estimated date for completion of repairs, and any compensatory action being taken while the vent path is inoperable.

3. With both the Reactor Vessel Head and Pressurizer Steam Space vent paths inoperable, restore at least one vent path to operable status within 7 days or be in HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours.

3.1.1.5 Relief Valves

Whenever T_{avg} is above 350°F or the reactor is critical both power operated relief valves (PORVs) and their associated block valves shall be OPERABLE[1].

a. With one or both PORVs inoperable because of leakage through the PORV resulting in excessive RCS leakage, i.e., not in accordance with the leakage criteria in Technical Specification 3.1.5.2:

1. Within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s); or

2. Be in at least HOT SHUTDOWN condition using normal operating procedures within the next 12 hours and cool down the RCS below a T_{avg} of 350°F within the following 12 hours[2].

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1 PORV block valves shall not be considered inoperable solely because either their normal or emergency power source is inoperable.

2 Power operation may continue pursuant to the requirements of this specification with the associated block valve closed, as a precautionary measure, to isolate minor leakage prior to the RCS leakage exceeding the leakage criteria in Technical Specification 3.1.5.2, with power maintained to the block valve during the period of the discretionary isolation.

b. With one PORV inoperable due to causes other than (1) leakage through the PORV resulting in excessive RCS leakage or (2) discretionary isolation to prevent minor leakage from becoming excessive:

1. Within 1 hour either restore the PORV to OPERABLE status or close its associated block valve and remove power from the block valve; and

2. Restore the PORV to OPERABLE status within the following 72 hours; or

3. Be in at least HOT SHUTDOWN condition using normal operating procedures within the next 12 hours and cool down the RCS below a T_{avg} of 350°F within the following 12 hours.

c. With both PORVs inoperable due to causes other than (1) leakage through the PORV resulting in excessive RCS leakage or (2) discretionary isolation to prevent minor leakage from becoming excessive:

1. Within 1 hour either restore at least one PORV to OPERABLE status; or close its associated block valve and remove power from the block valve; and

2. Be in at least HOT SHUTDOWN condition using normal operating procedures within the next 12 hours and cool down the RCS below a T_{avg} of 350°F within the following 12 hours.

d. With one or both block valves inoperable[1]:

1. Within 1 hour restore the block valve(s) to OPERABLE status or place the associated PORV(s) in manual control; and

2. Restore at least one block valve to OPERABLE status within the next hour if both block valves are inoperable; and

3. Restore any remaining inoperable block valve to operable status within 72 hours; or

4. Be in at least HOT SHUTDOWN condition using normal operating procedures within the next 12 hours and cool down the RCS below a T_{avg} of 350°F within the following 12 hours.

e. For this specification, reactor startup, heatup and entry into operational conditions with T_{avg} greater than or equal to 350°F may continue so long as the limits of the associated action statements are met.

f. During performance of the required surveillance testing of the PORVs and their associated Block Valves, the respective valve train need not be declared inoperable nor the associated action statements performed unless the associated valves are determined to be inoperable via this testing. Testing of no more than one train at a time may be performed and the time in the out of normal test configuration shall not exceed 24 hours.

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1 PORV block valves shall not be considered inoperable solely because either their normal or emergency power source is inoperable.

Basis

At the conditions of the RCS temperature (T_{avg}) greater than 350°F or the reactor critical, the power-operated relief valves (PORVs) provide an RCS pressure boundary, manual RCS pressure control for mitigation of accidents, and automatic RCS pressure relief to minimize challenges to the safety valves.

Providing an RCS pressure boundary and manual RCS pressure control for mitigation of a steam generator tube rupture (SGTR) are the safety-related functions of the PORVs at the conditions noted above. The capability of the PORV to perform its function of providing an RCS pressure boundary requires that the PORV or its associated block valve is closed. The capability of the PORVs to perform manual RCS pressure control for mitigation of a SGTR accident is based on manual actuation and does not require the automatic RCS pressure control function. The automatic RCS pressure control function of the PORVs is not a safety-related function at the conditions noted above. The automatic pressure control function limits the number of challenges to the safety valves, while the safety valves perform the safety function of RCS overpressure protection. Therefore, the automatic RCS pressure control function of the PORVs does not have to be available for the PORVs to be OPERABLE.

Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable. Operation with the block valves open is preferred. This allows the PORVs to perform automatic RCS pressure relief should the RCS pressure actuation setpoint be reached. However, operation with the block valve closed to isolate PORV leakage is permissible since automatic RCS pressure relief is not a safety-related function of the PORVs.

The ability to operate with the block valve(s) closed with power maintained to the block valve(s) is only intended to permit operation of the plant for a limited period of time not to exceed the next refueling outage so that maintenance can be performed on the PORVs to eliminate the leakage condition. Power is maintained to the block valve(s) so that it is operable and may be

subsequently opened to allow the PORV to be used to control reactor coolant system pressure. Closure of the block valve(s) establishes reactor coolant pressure boundary integrity for a PORV that has leakage resulting in excessive RCS leakage. (Reactor coolant pressure boundary integrity takes priority over the capability of the PORV to mitigate an overpressure event.) The PORVs should normally be available for automatic mitigation of overpressure events and should be returned to OPERABLE status prior to exceeding cold shutdown following the associated refueling outage.

The OPERABILITY of the PORVs and block valves at the conditions noted above is based on their being capable of performing the following functions:

1. Maintaining the RCS pressure boundary,

2. Manual control of PORVs to control RCS pressure as required for SGTR mitigation,

3. Manual closing of a block valve to isolate a stuck open PORV,

4. Manual closing of a block valve to isolate a PORV with excessive seat leakage, and

5. Manual opening of a block valve to unblock an isolated PORV to allow it to be used to control RCS pressure for SGTR mitigation.

3.1.2 Heatup and Cooldown

3.1.2.1 The reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1-1a and Figure 3.1-2a (for vessel exposure up to 12.5 EFPY) or Figure 3.1-1b and Figure 3.1-2b (for vessel exposure up to 15 EFPY). The 15 EFPY curves may be used for operation prior to the end of 12.5 EFPY. These limitations are as follows:

a. Over the temperature range from cold shutdown to hot operating conditions, the heatup rate shall not exceed 60°F/hr. in any one hour.

b. Allowable combinations of pressure and temperature for a specific cooldown rate are below and to the right of the limit lines for that rate as shown on Figure 3.1-2a or 3.1-2b (as appropriate). This rate shall not exceed 100°F/hr. in any one hour. The limit lines for cooling rates between those shown in Figure 3.1-2a or Figure 3.1-2b may be obtained by interpolation.

c. Primary system hydrostatic leak tests may be performed as necessary, provided the temperature limitation as noted on Figure 3.1-1a or Figure 3.1-1b (as appropriate) is not violated. Maximum hydrostatic test pressure should remain below 2350 psia.

d. The overpressure protection system shall be OPERABLE[1], with both power operated relief valves OPERABLE with a lift setting of less than or equal to 420 psi whenever any RCS

[1] The overpressure protection system shall not be considered inoperable solely because either the normal or emergency power source for the PORV block valves is inoperable.

cold leg temperature is less than or equal to 350°F and when the head is on the reactor vessel and the RCS is not vented to the containment.

1. With one PORV inoperable and T_{avg} greater than 200°F and any RCS cold leg temperature less than 350°F:

   A. Restore the inoperable PORV to OPERABLE status within 7 days; or

   B. Depressurize and vent the RCS to the CV within the next 12 hours.

2. With one PORV inoperable and T_{avg} less than or equal to 200°F:

   A. Restore the inoperable PORV to OPERABLE status within 24 hours; or

   B. Complete depressurization and venting of the RCS to the CV within an additional 12 hours.

3. With both PORVs inoperable, complete depressurization and venting of the RCS to the CV within 12 hours.

4. With the RCS vented per 1, 2, or 3, verify the vent pathway:

   A. At least once per 31 days when the pathway is provided by a valve(s) that is locked, sealed, or otherwise secured in the open position; or

   B. At least once per shift.

5. In the event the PORVs or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.3 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or RCS vent(s) on the transient and any corrective action necessary to prevent recurrence.

6. For this specification, reactor startup, heatup and entry into operational conditions with T_{avg} greater than or equal to 350°F may continue so long as the limits of the associated action statements are met.

e. Operation of the overpressure protection system to relieve a pressure transient must be reported as a Special Report to the NRC within 30 days of operation.

3.1.2.2 The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the vessel is below 120°F.

3.1.2.3 The pressurizer shall neither exceed a maximum heatup rate of 100°F/hr nor a cooldown rate of 200°F/hr. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

3.1.2.4 Figures 3.1-1b and 3.1-2b shall be updated periodically in accordance with the following criteria and procedures before the calculated exposure of the vessel exceeds the exposures for which the figures apply.

a. At least 60 days before the end of the integrated power period for which Figures 3.1-1b and 3.1-2b apply, the limit lines on the figures shall be updated for a new integrated power period utilizing methods derived from the ASME Boiler and Pressure Vessel Code, Section III, Appendix G and in accordance with applicable appendices of 10CFR50. These limit lines shall reflect any changes in predicted vessel neutron fluence over the integrated power period or changes resulting from the irradiation specimen measurement program.

b. The results of the examinations of the irradiation specimens and the updated heatup and cooldown curves shall be reported to the Commission within 90 days of completion of the examinations.

Basis

The ability of the large steel pressure vessel that contains the reactor core and its primary coolant to resist fracture constitutes an important factor in ensuring safety in the nuclear industry. The beltline region of the reactor pressure vessel is the most critical region of the vessel because it is subjected to neutron bombardment. The overall effects of fast neutron irradiation on the mechanical properties of low alloy ferritic pressure vessel steels such as ASTM A302 Grade B parent material of the H. B. Robinson Unit No. 2 reactor pressure vessel are well documented in the literature. Generally, low alloy ferritic materials show an increase in hardness and other strength properties and a decrease in ductility and impact toughness under certain conditions of irradiation. Accompanying a decrease in impact strength is an increase in the temperature for the transition from brittle to ductile fracture.

A method for guarding against fast fracture in reactor pressure vessels has been presented in Appendix G, "Protection Against Non-Ductile Failure," to Section III of the ASME Boiler and Pressure Vessel Code. The method utilizes fracture mechanics concepts and is based on the reference nil-ductility temperature, RT_{NDT}.

RT_{NDT} is defined as the greater of: 1) the drop weight nil-ductility transition temperature (NDTT per ASTM E-208) or 2) the temperature 60°F less than the 50 ft-lb (and 35 mils lateral expansion) temperature as determined from Charpy specimens oriented in a direction normal to the major working direction of the material. The RT_{NDT} of a given material is used to index that material to a reference stress intensity factor curve (K_{IR} curve) which appears in Appendix G of the ASME Code. The K_{IR} curve is a lower bound of dynamic, crack arrest, and static fracture toughness results obtained from several heats of pressure vessel steel. When a given material is indexed to the K_{IR} curve, allowable stress intensity factors can be obtained for this material as a function of temperature. Allowable operating limits can then be determined utilizing these allowable stress intensity factors.

The Certified Material Test Reports (CMTRs) for the original steam generators provided records of Charpy V-notch tests performed at +10°F. Acceptable Charpy V-notch tests of +10°F indicate RT_{NDT} is at or below this temperature. The steam generator lower assemblies were replaced in 1984 and the material tests results indicate that highest RT_{NDT} is 60°F or below. The ASME code recommends that hydrostatic tests be performed at a temperature not lower than RT_{NDT} plus 60°F, thus the pressurizing temperature for the steam generator shell is established at 120°F to provide protection against nonductile failure at the test pressure. The value of RT_{NDT}, and in turn the operating limits of nuclear power plants, can be adjusted to account for the effects of radiation on the reactor vessel material properties. The radiation embrittlement or changes in mechanical properties of a given reactor pressure vessel still can be monitored by a surveillance program such as the Carolina Power & Light Company, H. B. Robinson Unit No. 2 Reactor Vessel Radiation Surveillance Program[1] where a surveillance capsule is periodically removed from the operating nuclear reactor and the encapsulated specimens tested. These data are compared to data from pertinent radiation effects studies and an increase in the Charpy V-notch 30 ft-lb temperature (ΔRT_{NDT}) due to irradiation is added to the original RT_{NDT} to adjust the RT_{NDT} for radiation embrittlement. This adjusted RT_{NDT} (RT_{NDT} initial + RT_{NDT}) is utilized to index the material to the K_{IR} curve and in turn to set operating limits for the nuclear power plant which take into account the effects of irradiation on the reactor vessel materials. Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods (2) derived from Appendix G to Section III of the ASME Boiler and Pressure Vessel Code. The approach specifies that the allowable total stress intensity factor (K_I) at any time during heatup or cooldown cannot be greater than that shown on the K_{IR} curve in Appendix G for the metal temperature at that time. Furthermore, the approach applies an explicit safety factor of 2.0 on the stress intensity factor induced by pressure gradients.

Following the generation of pressure-temperature curves for both the steady state and finite heatup rate situations, the final limit curves are produced in the following fashion. First, a composite curve is constructed based on a

point-by-point comparison of the steady state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the two values taken from the curves under consideration. The composite curve is then adjusted to allow for possible errors in the pressure and temperature sensing instruments.

The use of the composite curve is mandatory in setting heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling analysis switches from the O.D. to the I.D. location; and the pressure limit must, at all times, be based on the most conservative case. The cooldown analysis proceeds in the same fashion as that for heatup, with the exception that the controlling location is always at the I.D. position. The thermal gradients induced during cooldown tend to produce tensile stresses at the I.D. location and compressive stresses at the O.D. position. Thus, the I.D. flaw is clearly the worst case.

As in the case of heatup, allowable pressure temperature relations are generated for both steady state and finite cooldown rate situations. Composite limit curves are then constructed for each cooldown rate of interest. Again adjustments are made to account for pressure and temperature instrumentation error.

The overpressure protection system consists of two operable pressurizer Power Operated Relief Valves (PORVs) connected to the station instrument air system, a backup nitrogen supply, and the associated electronics.

The TS requirements for LTOP apply when T_{avg} is less than 350°F and the RCS is not vented to the containment. During these conditions, one train (or channel) of the LTOP system is capable of mitigating an LTOP event that is bounded by the largest mass addition to the RCS or by the largest increase in RCS temperature that can occur. The largest mass addition to the RCS is limited based upon the assumption that no more than a fixed number of pumps are capable of providing makeup or injection into the RCS. Hence, this is a matter important to safety that pumps in excess of this design basis assumption for LTOP not be capable of providing makeup or injection to the

RCS. In this regard the SI Pump breakers are required to be racked out at less than 350°F RCS temperature.

Pages 3.1-8 through 3.1-10 deleted.

References

1. S. E. Yanichko, "Carolina Power & Light Company, H. B. Robinson Unit No. 2 Reactor Vessel Radiation Surveillance Program," Westinghouse Nuclear Energy Systems - WCAP-7373 (January 1970)

2. E. B. Norris, "Reactor Vessel Material Surveillance Program for H. B. Robinson Unit No. 2, Analysis of Capsule V," Southwest Research Institute - Final Report SWRI Project No. 02-4397.

Capsule No. 3 leads the vessel maximum exposure by a factor of 2.2 and is scheduled to be removed after twenty years. Thus, sample No. 3 will provide data for an exposure to the vessel of approximately forty years.

Capsule Nos. 4 and 5 lag the maximum vessel exposure by factors of 0.7 and 0.5, respectively. Thus, Capsule No. 4, which is scheduled to be removed after thirty years, provides data for a vessel exposure of twenty-one years and Capsule No. 5, which is scheduled to be removed at forty years, provides data for a vessel exposure of twenty years.

In addition to the capsules discussed above, there are three spares. Two are located at the same location as Capsule No. 5 and one is located at the same location as Capsule No. 4.

4.2.3 Primary Pump Flywheels

The flywheels shall be visually examined at the first refueling after each ten year inspection. At the fourth refueling after each ten year inspection and at each fourth refueling thereafter, the outside surfaces shall be examined by ultrasonic methods.

4.2.4 Relief Valves

4.2.4.1 In addition to the requirements of Specification 4.0.1, each PORV shall be demonstrated OPERABLE at each refueling by:

a. Performance of a CHANNEL CALIBRATION, and

b. Operating the PORV through one complete cycle of full travel while T_{avg} is greater than 350°F and the reactor is subcritical.

c. Operating the solenoid air control valves and check valves for their associated accumulators in PORV control systems through one complete cycle of full travel or function testing of individual components.

4.2.4.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed in order to meet the requirements of Specification 3.1.1.5.a., b. or c.

4.2.4.3 The accumulator for the PORVs shall be demonstrated OPERABLE at each refueling by isolating the normal air and nitrogen supplies and operating the valves through a complete cycle of full travel.

4.2.5 Low Temperature Overpressure Protection

4.2.5.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST on the actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE; and

b. Performance of a CHANNEL CALIBRATION at each refueling shutdown; and

c. Verifying the PORV Block Valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

Basis

The OPERABILITY of two PORVs for low temperature overpressure protection or an RCS vent ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 350°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle RCP with the secondary water temperature of the steam generator less than 50°F above the RCS cold leg temperatures, or (2) the start of three charging pumps with injection into a water-solid RCS.

The maximum allowed PORV setpoint for the Low Temperature Overpressure Protection system (LTOP) is derived by analyses which model the performance of the LTOP assuming various mass input and heat input transients. Operation with a PORV setpoint less than or equal to the maximum setpoint ensures that Appendix G criteria will not be violated with consideration for a maximum pressure over-shoot beyond the PORV setpoint which can occur as a result of time delays in signal processing and valve opening, instrument uncertainties, and single failure. To ensure that mass and heat input transients more severe than those assumed cannot occur, Technical Specifications require the power supply breakers of all three safety injection pumps be racked out while in hot shutdown and below 350°F with the reactor vessel head installed and the RCS is not vented to containment and disallow start of an RCP if secondary temperature is more than 50°F above primary temperature.

The maximum allowed PORV setpoint for the LTOP will be updated based on the results of examinations of reactor vessel material irradiation surveillance specimens performed as required by 10 CFR Part 50 Appendix H.

Surveillance Requirements provide the assurance that the PORVs and Block Valves can perform their required functions. Specification 4.2.4.1 addresses PORVs, 4.2.4.2 the Block Valves, and 4.2.4.3 the independent pneumatic power source. Specification 4.2.5.1 addresses the PORV overpressure protection functions and 4.2.5.2 addresses RCS vent pathways.

Surveillance Requirement 4.2.4.1.a provides assurance the actuation instrumentation for automatic PORV actuation is calibrated such that the automatic PORV actuation signal is within the required pressure range even though automatic actuation capability of the PORV is not necessary for the PORV to be OPERABLE in the power operating and hot shutdown conditions greater than 350°F.

Surveillance Requirement 4.2.4.1.b provides assurance the PORV is capable of opening and closing. The associated block valve should be closed prior to stroke testing a PORV to preclude depressurization of the RCS. This test will be done at hot shutdown with T_{avg} greater than 350°F before the PORV is required for overpressure protection in Technical Specification 3.1.2.1.d.

Surveillance Requirement 4.2.4.1.c. provides assurance that the mechanical and electrical aspects of the control system are functional.

Surveillance Requirement 4.2.4.2 addresses the block valves. The block valves are exempt from the surveillance requirements to cycle the valves when they have been closed to comply with Technical Specification 3.1.1.5.a.,b. or c. This precludes the need to cycle the valves with a full system differential pressure or when maintenance is being performed to restore an inoperable PORV to OPERABLE status. Also, this limits the challenges to the primary function of the Block Valve which is to provide an RCS pressure boundary for a degraded PORV.

Surveillance Requirement 4.2.4.3 provides assurance of operability of the accumulators and that the accumulators are capable of supplying sufficient Nitrogen to operate the PORV(s) if they are needed for RCS pressure control and normal Nitrogen and the backup Instrument Air systems are not available. Backup Instrument Air is supplied when the accumulator reaches its low pressure setpoint.

Surveillance Requirement 4.2.5.1 provides assurance that the instrumentation for the actuation of the LTOP function of PORVs is calibrated to provide

automatic actuation of the PORVs for low temperature conditions. Also, the flow path to the PORV is assured to be open.

<u>References</u>

(1) FSAR, Section 4.4

(2) FSAR, Volume 4, Tab VII, Question VI.C