# INDEX

BASES

REACTOR COOLANT SYSTEM

ACTION (Continued)

b) Place the following reactor trip system and ESFAS instrumentation channels, associated with the loop not in operation, in their tripped conditions:

1) Overpower ΔT channel.

2) Overtemperature ΔT channel.

3) $T_{avg}$ -- Low-Low channel used in the coincidence circuit with steam Flow - High for Safety Injection.

4) Steam Line Pressure - Low channel used in the coincidence circuit with Steam Flow - High for Safety Injection.

5) Steam Flow-High channel used for Safety Injection.

6) Differential Pressure Between Steam Lines - High Channel used for Safety Injection (trip all bistables which indicate low active loop steam pressure with respect to the idle loop steam pressure).

c) Change the P-8 interlock setpoint from the value specified in Table 3.3-1 to $\leq$75% of RATED THERMAL POWER.

2. THERMAL POWER is restricted to $\leq$70% of RATED THERMAL POWER.

Below P-7:

a. With $K_{eff} \geq 1.0$, operation below P-7 may proceed provided at least four reactor coolant loops and associated pumps are in operation.

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

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 a. With any control rod drive mechanism (CRDM) energized, all four reactor coolant loops shall be in operation.

b. With no CRDMs energized, at least two reactor coolant loops shall be OPERABLE with one reactor coolant loop in operation.*

APPLICABILITY: MODE 3

ACTION:

a. With less than four reactor coolant loops in operation, immediately de-energize all CRDMs.

b. With less than two reactor coolant loops OPERABLE when the CRDMs are de-energized, restore the required loops to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

c. With no reactor coolant loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.2.1 All CRDMs shall be verified de-energized at least once per 12 hours if less than four reactor coolant loops are in operation.

4.4.1.2.2 At least two reactor coolant pumps, if not in operation, shall be determined to be OPERABLE once per 7 days by verifying correct breaker alignments and indicated power availability.

4.4.1.2.3 At least one cooling loop shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

* All reactor coolant pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause dilution of the reactor coolant system boron concentration, and (2) core outlet temperature is maintained at 10°F below saturation temperature.

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with all reactor coolant loops in operation, and maintain DNBR above 1.73 during all normal operations and anticipated transients. With one reactor coolant loop not in operation, THERMAL POWER is restricted to <58 percent of RATED THERMAL POWER until the Overtemperature ΔT trip is reset. Either action ensures that the DNBR will be maintained above 1.73. A loss of flow in two loops will cause a reactor trip if operating above P-7 (10 percent of RATED THERMAL POWER) while a loss of flow in one loop will cause a reactor trip if operating above P-8 (39 percent of RATED THERMAL POWER).

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE. Four loops in operation while control rod drive mechanisms are energized ensures that the DNB design basis can be met for a bank withdrawal from subcritical or low power accident.

In MODES 4 and 5, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two RHR loops to be OPERABLE.

The operation of one Reactor Coolant Pump or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump with one or more RCS cold legs less than or equal to 290°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into, or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures, or (3) by restricting starting of an RCP unless another RCP is running.

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.2 SAFETY VALVES and 3/4.4.3 SAFETY AND RELIEF VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at 110% of the valve's setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip set point is reached (ie, no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power-operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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| In the Matter of | ) | |
| | ) | |
| PORTLAND GENERAL ELECTRIC COMPANY, | ) | Docket 50-344 |
| THE CITY OF EUGENE, OREGON, AND | ) | Operating License NPF-1 |
| PACIFIC POWER & LIGHT COMPANY | ) | |
| | ) | |
| (TROJAN NUCLEAR PLANT) | ) | |

CERTIFICATE OF SERVICE

I hereby certify that copies of License Change Application 113, Revision 1 to the Operating License for Trojan Nuclear Plant, dated December 27, 1985, have been served on the following by hand delivery or by deposit in the United States mail, first class, this 27th day of December 1985:

Mr. Lynn Frank, Director
State of Oregon
Department of Energy
Labor & Industries Bldg, Rm 102
Salem OR 97310

Mr. Robert L. King
Chairman of County Commissioners
Columbia County Courthouse
St. Helens OR 97051

D A Zimmerman

D. A. Zimmerman, Manager
Nuclear Regulation Branch
Nuclear Safety & Regulation

Subscribed and sworn to before me this 27th day of December 1985.

Betty J. Simich
BETTY J. SIMICH
NOTARY PUBLIC – OREGON
My Commission Expires 10-24-86

Betty J. Simich
Notary Public of Oregon

My Commission Expires: 10-24-86