Attachment

Proposed Technical Specification Change

Proposed Change

This proposed Technical Specification change addresses reactor coolant leak detection and leakage limits. New Limiting Condition for Operation (LCO) requirements are imposed for the rate of increase of reactor coolant leakage into the primary containment from unidentified sources. The proposed Technical Specification generally establishes a 2 gpm limit increase averaged over any 24-hour period. This LCO will apply only when the reactor has been in the RUN mode greater than 24 hours. This change places an additional restriction on plant operations because the present Technical Specifications do not have a provision for this type of situation.

New operability requirements are proposed for the Reactor Coolant Leakage Detection System and the Drywell Continuous Atmosphere Radioactivity Monitoring System. Greater specificity for the operational requirements is proposed to account for the redundancy of the systems and the redundancy of components within subsystems.

See the attached Technical Specification pages for proposed wording. Specific instructions for changes to the Technical Specifications are as follows:

Delete pages 125, 126, and 143

Add pages 125, 125a, 126, 126a, and 143

Reason for Change

These proposed Technical Specification changes are submitted to complete a licensing commitment related to Generic Letter 84-11: Inspections of BWR Stainless Steel Piping. Generic Letter 84-11 contains guidance for leak detection and leakage limits to ensure timely investigation of unidentified leakage that may be caused by throughwall cracks developed in austenitic stainless steel piping. The amount of austenitic stainless steel piping in the Pilgrim Station Reactor Coolant System has been minimized by replacement with Type 316 NG stainless steel during the most recent refueling outage. We have evaluated the guidance and how it should be applied to Pilgrim Station in light of the pipe replacement activities and in terms of the configuration of the Reactor Coolant Leakage Detection System and the Drywell Continuous Atmosphere Radioactivity Monitoring System. The proposed Technical Specification change meets the intent of Generic Letter 84-11 regarding leak detection and leakage limits.

Safety Considerations

This change does not present an unreviewed safety question as defined in 10 CFR 50.59. It has been reviewed and approved by the Operations Review Committee and reviewed by the Nuclear Safety Review and Audit Committee.

Significant Hazards Considerations

It has been determined that this amendment request involves no significant hazards consideration. Under the NRC's regulations in 10 CFR 50.92, this means that operation of the Pilgrim Nuclear Power Station in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The NRC has provided guidance concerning the application of standards for determining whether license amendments involve significant hazards considerations by providing certain examples (48 FR 14870). One example of an amendment that is considered not likely to involve a significant hazards consideration is "...(ii) A change that constitutes an additional limitation, restriction, or control not presently included in the technical specifications; for example, a more stringent surveillance requirement."

This proposed change imposes an additional limitation by imposing limits on the rate of increase of reactor coolant leakage into the primary containment from unidentified sources. In addition, the operability requirements for the leakage detection systems are more specifically detailed so as to apply more stringently.

Schedule of Change

This change will be put into effect in 30 days upon receipt of approval from NRC.

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.6.B Coolant Chemistry (Cont'd)	4.6.B Coolant Chemistry (Cont'd)
3. For reactor startups and for the first 24 hours after placing the reactor in the power operating condition, the following limits shall not be exceeded. Conductivity. . . 10 umho/cm Chloride ion. . . 0.1 ppm	3. a. With steaming rates of 100,000 pounds per hour or greater, a reactor coolant sample shall be taken at least every 96 hours and analyzed for chloride ion content. b. When all continuous conductivity monitors are inoperable, a reactor coolant sample shall be taken at least daily and analyzed for conductivity and chloride ion content.
4. Except as specified in 3.6.B.3 above, the reactor coolant water shall not exceed the following limits when operating with steaming rates greater than or equal to 100,000 pounds per hour. Conductivity. . . 10 umho/cm Chloride ion. . . 1.0 ppm	
5. If Specification 3.6.B cannot be met, an orderly shutdown shall be initiated and the reactor shall be in Hot Shutdown within 24 hrs. and Cold Shutdown within the next 8 hours.	
3.6.C Coolant Leakage	4.6.C Coolant Leakage
1. a. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212°F, reactor coolant leakage into the primary containment from unidentified sources shall not exceed 5 gpm. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm.	1. Reactor coolant system leakage shall be checked by the sump and air sampling system and recorded at least once per 8-hour shift.
b. Any time the reactor is in RUN mode, reactor coolant leakage into the primary containment from unidentified sources shall not increase by more than 2 gpm averaged over any 24 hour period in which the reactor is in the RUN mode except as defined in 3.6.C.1.c below.	

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.6.C Coolant Leakage (Cont'd) c. During the first 24 hours in the RUN mode following startup, an increase in reactor coolant leakage into the primary containment of >2 gpm is acceptable as long as the requirements of 3.6.C.1.a are met. d. If the condition of a or b above cannot be met, the reactor shall be placed in the Cold Condition within 24 hours. 2. Reactor Coolant Leakage Detection System shall be operable any time irradiated fuel is in the vessel and reactor coolant temperature is above 212°F. a. From and after the time that either the equipment drain sump or the floor drain sump subsystem is made or found to be inoperable for any reason, continued reactor operation is permissible for the succeeding seven days, provided that the Reactor Pressure Boundary Leak Detection System is operable. b. From and after the time that a redundant component of either subsystem is made or found to be inoperable for any reason, continued reactor operation is permissible for the succeeding 30 days, unless the component is sooner made operable. 3. The Reactor Pressure Boundary Leak Detection Systems shall be operable any time irradiated fuel is in the vessel and reactor coolant temperature is above 212°F. The system shall be considered operable if at least one of the monitoring systems is operable.	

3.6.C Coolant Chemistry (Cont'd)

a. The monitoring systems may be removed from service for a period of 4 hours for calibration, function testing and maintenance without providing a temporary monitor.

b. From and after the time both Reactor Pressure Boundary Leak Detection Systems are made or found to be inoperable for any reason, continued reactor power operation is permissible during the succeeding seven days provided:

1. The Reactor Coolant Leakage Detection System is operable.

2. An appropriate grab sample is obtained and analyzed at least once per 96 hours.

c. In the event that both of the Reactor Pressure Boundary Leak Detection Systems are not operable and the unidentified reactor coolant leakage rate increases by more than 1 gpm when averaged over any 24 hour period, an appropriate grab sample will be obtained and analyzed at least once per 24 hours.

4. From and after the date that both the Reactor Coolant Leakage Detection System and the Reactor Pressure Boundary Leak Detection System are made or found to be inoperable, the reactor shall be placed in the Cold Condition within 24 hours.

3.6.D Safety and Relief Valves

1. During reactor power operating conditions and prior to reactor startup from a Cold Condition, or whenever reactor coolant pressure is greater than 104 psig and temperature greater than 340°F, both safety valves and the safety modes of all relief valves shall be operable.

2. If Specification 3.6.D.1 is not met, an orderly shutdown shall be initiated and the reactor coolant pressure shall be below 104 psig within 24 hours.
Note: Technical Specifications 3.6.D.2 - 3.6.D.5 apply only when two Stage Target Rock SRVs are installed.

3. If the temperature of any safety relief discharge pipe exceeds 212°F during normal reactor power operation for a period of greater than 24 hours, an engineering evaluation shall be performed justifying continued operation for the corresponding temp. increases, and a Report shall be issued per T.S. Section 6.9.B.1 which shall address the actions that have been taken or a schedule of actions to be taken.

4. Any safety relief valve whose discharge pipe temperature exceeds 212°F for 24 hours or more shall be removed at the next cold shutdown of 72 hours or more tested in the as-found condition, and recalibrated as necessary prior to reinstallation. Power operation shall not continue beyond 90 days

4.6.D Safety and Relief Valves

1. At least one safety valve and two relief/safety valves shall be checked or replaced with bench checked valves once per operating cycle. All valves will be tested every two cycles.

   The set point of the safety valves shall be as specified in Specification 2.2.

2. At least one of the relief/safety valves shall be disassembled and inspected each refueling outage.

3. Whenever the safety relief valves are required to be operable, the discharge pipe temperature of each safety relief valve shall be logged daily.

4. Instrumentation shall be calibrated and checked as indicated in Table 4.2.F.

5. Notwithstanding the above, as a minimum safety relief valves that have been in service shall be tested in the as-found condition during both Cycle 6 and Cycle 7.

BASES:

3.6.C and 4.6.C

Coolant Leakage

Allowable leakage rates of coolant from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes and on the ability to makeup coolant system leakage in the event of loss of offsite a-c power. The normally expected background leakage due to equipment design and the detection capability for determining coolant system leakage were also considered in establishing the limits. The behavior of cracks in piping systems has been experimentally and analytically investigated as part of the USAEC sponsored Reactor Primary Coolant System Rupture Study (the Pipe Rupture Study). Work utilizing the data obtained in this study indicates that leakage from a crack can be detected before the crack grows to a dangerous or critical size by mechanically or thermally induced cyclic loading, or stress corrosion cracking or some other mechanism characterized by gradual crack growth. This evidence suggests that for leakage somewhat greater than the limit specified for unidentified leakage, the probability is small that imperfections or cracks associated with such leakage would grow rapidly. However, the establishment of allowable unidentified leakage greater than that given in 3.6.C on the basis of the data presently available would be premature because of uncertainties associated with the data. For leakage of the order of 5 gpm, as specified in 3.6.C, the experimental and analytical data suggest a reasonable margin of safety that such leakage magnitude would not result from a crack approaching the critical size for rapid propogation. Leakage less than the magnitude specified can be detected reasonably in a matter of a few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time the plant should be shut down to allow further investigation and corrective action.

The 2 gpm limit for coolant leakage rate increase over any 24 hour period is a limit specified by the NRC (Reference 1.) This limit applies only during the RUN mode to accommodate the expected coolant leakage increase during pressurization.

The total leakage rate consists of all leakage, identified and unidentified, which flows to the drywell floor drain and equipment drain sumps.

The capacity of the drywell floor sump pumps is 100 gpm and the capacity of the drywell equipment sump pumps is also 100 gpm. Removal of 25 gpm from either of these sumps can be accomplished with considerable margin.

REFERENCE

1) US NRC Generic Letter 84-11: Inspections of BWR Stainless Steel Piping