This supplement provides no additional changes to this page.

APPLICABILITY

SURVEILLANCE REQUIREMENTS (Continued)

4.0.5 (Continued)

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

f. The Inservice Inspection Program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with the NRC Staff positions on Schedule, methods and personnel, and sample expansion included in the generic letter.

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REACTOR COOLANT SYSTEM

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.3.1 The following reactor coolant system leakage detection systems shall be OPERABLE:

a. The drywell atmosphere particulate radioactivity monitoring system,

b. The drywell sump flow monitoring system, and

c. Either the drywell atmosphere gaseous radioactivity monitoring system or the drywell air coolers condensate flow rate monitoring system.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

REPLACE WITH ATTACHED

ACTION:

With only two of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days provided grab samples of the drywell atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous and/or particulate radioactive monitoring system is inoperable; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The reactor coolant system leakage detection systems shall be demonstrated OPERABLE by:

a. Drywell atmosphere particulate and gaseous monitoring systems-performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

b. Drywell sump flow monitoring system-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION TEST at least once per 18 months.

c. Drywell air cooler condensate flow rate monitoring system performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

d. Flow testing the drywell floor drain sump inlet piping for blockage at least once every 18 months during shutdown.

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ACTION:

With only two of the above required leakage detection systems OPERABLE,

a. operation may continue for up to 30 days when the drywell atmosphere particulate radioactivity monitoring system is inoperable provided grab samples of the drywell atmosphere are obtained and analyzed at least once per 24 hours;

b. operations may continue:

   1. with the drywell equipment drain sump flow monitoring subsystem inoperable provided the drywell equipment drain sump flow rate is monitored and determined by alternate means at least once per 12 hours,

   2. for up to 30 days with the drywell floor drain sump flow monitoring subsystem inoperable provided the drywell floor drain sump flow rate is monitored and determined by alterna means at least once per 8 hours,

c. operation may continue for up to 30 days when the drywell atmosphere gaseous radioactivity monitoring system and the drywell air coolers condensate flow rate monitoring system are inoperable provided grab samples of the drywell atmosphere are obtained and analyzed at least once per 24 hours.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.3.2 Reactor coolant system leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE.

b. 5 gpm UNIDENTIFIED LEAKAGE.

c. 25 gpm IDENTIFIED LEAKAGE (averaged over any 24-hour period).

d. 0.5 gpm leakage per nominal inch of valve size up to a maximum of 5 gpm from any reactor coolant system pressure isolation valve specified in ~~Table 3.4.3.2-1,~~ at rated reactor pressure.

e. INSERT ATTACHED

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With any reactor coolant system leakage greater than the limits in b and/or c, above, reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With any reactor coolant system pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two other closed manual or deactivated automatic valves, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. INSERT ATTACHED

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 The reactor coolant system leakage shall be demonstrated to be within each of the above limits by:

a. Monitoring the drywell atmospheric particulate and gaseous radioactivity at least once per 12 hours (not a means of quantifying leakage),

c. ~~b.~~ Monitoring the drywell ~~floor and~~ equipment drain sump flow rate at least once per 12 hours,

d. ~~c.~~ Monitoring the drywell air coolers condensate flow rate at least once per 12 hours, and

b. Monitoring the drywell floor drain sump flow rate at least once per 8 hours,

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Inserts for Technical Specification 3.4.3.2 (pg. 3/4 4-13)

(LIMITING CONDITION FOR OPERATION)

e. No greater than a 2 gpm increase in UNIDENTIFIED LEAKAGE within a 24-hour period or less during OPERATIONAL CONDITION 1.

(ACTION)

d. With any reactor coolant system UNIDENTIFIED LEAKAGE increase greater than 2 gpm within any 24-hour period or less (during OPERATIONAL CONDITION 1), within 4 hours from the time of discovery isolate the source of increased leakage or verify that the source of increased leakage is not associated with service sensitive Type 304 or 316 austenitic stainless steel; otherwise be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

SURVEILLANCE REQUIREMENTS (Continued)

4.4.3.2.1 (Continued)

e. ~~d.~~ Monitoring the reactor vessel head flange leak detection system at least once per 24 hours.

4.4.3.2.2 Each reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1 shall be demonstrated OPERABLE by leak testing pursuant to Specification 4.0.5 and verifying the leakage of each valve to be within the specified limit:

a. At least once per 18 months ~~*~~

b. Prior to returning the valve to service following maintenance, repair, or replacement work on the valve or its associated actuator.

c. As outlined in ASME Code, Section XI, paragraph IWV-3427(b).

The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 3.

~~*The requirements of this specification for valves 1E12F023, 1E51F066, and 1E51F013 will not be completed until prior to startup following the first refueling outage.~~

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BASES

3/4.4.1 RECIRCULATION SYSTEM (Continued)

The recirculation flow control valves provide regulation of individual recirculation loop drive flows; which, in turn, will vary the flow rate of coolant through the reactor core over a range consistent with the rod pattern and recirculation pump speed. The recirculation flow control system consists of the electronic and hydraulic components necessary for the positioning of the two hydraulically actuated flow control valves. Solid state control logic will generate a flow control valve "motion inhibit" signal in response to any one of several hydraulic power unit or analog control circuit failure signals. The "motion inhibit" signal causes hydraulic power unit shutdown and hydraulic isolation such that the flow control valve fails "as is." This design feature insures that the flow control valves do not respond to potentially erroneous control signals.

Electronic limiters exist in the position control loop of each flow control valve to limit the flow control valve stroking rate to 10±1% per second in opening and closing directions on a control signal failure. The analysis of the recirculation flow control failures on increasing and decreasing flow are presented in Sections 15.3 and 15.4 of the FSAR respectively.

The required surveillance interval is adequate to ensure that the flow control valves remain OPERABLE and not so frequent as to cause excessive wear on the system components.

3/4.4.2 SAFETY/RELIEF VALVES

The safety valve function of the safety/relief valves (SRV) operate to prevent the reactor coolant system from being pressurized above the Safety Limit of 1375 psig in accordance with the ASME Code. A total of 11 OPERABLE safety-relief valves is required to limit reactor pressure to within ASME III allowable values for the worst case upset transient. Any combination of 5 SRVs operating in the relief mode and 6 SRVs operating in the safety mode is acceptable.

Demonstration of the safety-relief valve lift settings will occur only during shutdown and will be performed in accordance with the provisions of Specification 4.0.5.

The low-low set system ensures that safety/relief valve discharges are minimized for a second opening of these valves, following any overpressure transient. This is achieved by automatically lowering the closing setpoint of 5 valves and lowering the opening setpoint of 2 valves following the initial opening. In this way, the frequency and magnitude of the containment blowdown duty cycle is substantially reduced. Sufficient redundancy is provided for the low-low set system such that failure of any one valve to open or close at its reduced setpoint does not violate the design basis.

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.3.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. ~~These detection systems meet the intent~~ of Regulatory Guide 1.45,

With certain exceptions as noted in the Clinton Power Station Updated Safety Analysis Report, these detection systems are consistent with the recommendations

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Except for the drywell particulate and gaseous radioactivity monitors, the Systems provide

REACTOR COOLANT SYSTEM

BASES

3/4.4.3.1 LEAKAGE DETECTION SYSTEMS (Continued)

"Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973, ~~and are consistent with the recommendations of ANSI N67.03, "Standard for Light Water Reactor Coolant Pressure Boundary Leak Detection," 1982. They provide~~ the ability to ~~detect and/or~~ measure leakage from fluid systems in the drywell.

INSERT PART I

3/4.4.3.2 OPERATIONAL LEAKAGE

The allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for UNIDENTIFIED LEAKAGE the probability is small that the imperfection or crack associated with such leakage would grow rapidly. ~~However,~~ In all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shut down to allow further investigation and corrective action.

INSERT PART II

INSERT PART III

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

3/4.4.4 CHEMISTRY

The water chemistry limits of the reactor coolant system are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel.

The effect of chloride is not as great when the oxygen concentration in the coolant is low, thus the 0.2 ppm limit on chlorides is permitted during POWER OPERATION. During shutdown and refueling operations, the temperature necessary for stress corrosion to occur is not present so a 0.5 ppm concentration of chlorides is not considered harmful during these periods.

Conductivity measurements are required on a continuous basis since changes in this parameter are an indication of abnormal conditions. When the conductivity is within limits, the pH, chlorides and other impurities affecting conductivity must also be within their acceptable limits. With the conductivity meter inoperable, additional samples must be analyzed to ensure that the chlorides are not exceeding the limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

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Part I

The drywell sump flow monitoring system consists of the drywell floor drain sump flow monitoring subsystem and the drywell equipment drain sump flow monitoring subsystem. OPERABILITY of each of these subsystems requires that the applicable portion of the monitoring subsystem associated with the v-notched weir box be OPERABLE. Other portions of the subsystem, including the sump pump control circuit and the associated timer, cycle counter and level switches, may be utilized as appropriate to provide an alternate means of monitoring and determining UNIDENTIFIED or IDENTIFIED leakage under the provisions of the associated ACTION statements for the respective subsystem.

Part II

With respect to IGSCC-related cracks in service sensitive austenitic stainless steel piping however, an additional limit on the allowed increase in UNIDENTIFIED LEAKAGE (within a 24-hour period or less) is imposed in accordance with Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping," since an abrupt increase in the UNIDENTIFIED LEAKAGE could be indicative of leakage from such a source.

Part III

The reactor will also be shut down if an increase in UNIDENTIFIED LEAKAGE exceeds the specified limit and the source of increased leakage cannot be isolated or it cannot be determined within a short period of time that the source of increased leakage is not associated with austenitic stainless steel.