3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITIONS FOR OPERATION

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

a. The primary containment airborne particulate radioactivity monitoring system,

b. The primary containment airborne gaseous radioactivity monitoring system,

c. The drywell floor drain tank fill rate monitoring system, and

d. Drywell equipment drain tank fill rate monitoring system.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the primary containment airborne particulate radioactivity monitoring system or the primary containment airborne gaseous radioactivity monitoring system inoperable, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 12 hours; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the drywell equipment drain tank fill rate monitoring system inoperable, operation may continue for up to 30 days provided that the drywell equipment drain tank fill rate is determined via alternate methods; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With the drywell floor drain tank fill rate monitoring system inoperable, operation may continue for up to 30 days provided that the drywell floor drain tank fill rate is determined via alternate methods; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With both drywell floor drain and the drywell equipment drain tank fill rate monitoring systems inoperable, restore either system to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

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RECIRCULATION SYSTEM

3/4.4.1 (Continued)

recirculation pump and recirculation nozzles. Sudden equalization of a temperature difference ≥ 145°F between the reactor vessel bottom head coolant and the coolant in the upper region of the reactor vessel by increasing core flow rate would cause undue stress in the reactor vessel bottom head.

3/4.4.2 SAFETY/RELIEF VALVES

The safety/relief valves operate during a postulated ATWS event to prevent the reactor coolant system from being pressurized above a design allowable value of 1375 psig in accordance with the ASME Code. A total of 16 OPERABLE safety/relief valves is required to limit local pressure at active components to within ASME III allowable design values (Service Level A). All other appropriate ASME III limits are also bounded by this requirement. Specified surveillance intervals have been determined in accordance with GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-Of-Service Times for Selected Instrumentation Technical Specification," as approved by the NRC and documented in the SER (letter to R. D. Binz IV from C. E. Rossi dated July 21, 1992).

The safety-relief valve lift settings will be demonstrated only during shutdown in accordance with the provisions of Specification 4.0.5.

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 are consistent with the recommendations of RG 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

Generic Letter 88-01, Supplement 1, allows for alternate methods to determine drywell leakage when the normal leakage monitoring systems are inoperable. These alternate methods may be used for up to 30 days. The alternate methods given in the Generic Letter Supplement are to manually pump down the drain tank, or to measure the differences in tank level. Other methods, which meet the requirements of Regulatory Guide 1.45 for accuracy and inspectability, may also be used.

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 background leakage normally expected to result from equipment design and the detection capability of the instrumentation for determining system leakage were also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for UNIDENTIFIED LEAKAGE, the probability is small

3/4.4.3.2 OPERATIONAL LEAKAGE (Continued)

that the imperfection or crack associated with such leakage would grow rapidly. An UNIDENTIFIED LEAKAGE increase of > 2 gpm within the previous 24 hour period indicates a potential flaw in the Reactor Coolant Pressure Boundary and must be quickly evaluated to determine the source and extent of the leakage. The increase is measured relative to the steady state value; temporary changes in leakage rate as a result of transient conditions (e.g., startup) are not considered. As such, the 2 gpm increase limit is only applicable in MODE 1 when operating pressures and temperatures are established. 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.

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.