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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 atmospheric and temperature greater than 212°F, all three safety valves and the safety modes of all relief valves shall be operable, except as specified in 3.6.D.2.

2.

a. From and after the date that the safety valve function of one relief valve is made or found to be inoperable, continued reactor operation is permissible only during the succeeding thirty days unless such valve function is sooner made operable.

b. From and after the date that the safety valve function of two relief valves is made or found to be inoperable, continued reactor operation is permissible only during the succeeding seven days unless such valve function is sooner made operable.

3. If Specification 3.6.D.1 is not met, an orderly shutdown shall be initiated and the reactor coolant pressure shall be reduced to a cool shutdown condition within 24 hours.

4. From and after the date that position indication on any one relief valve is made or found to be inoperable, continued reactor operation is permissible only during the succeeding thirty days unless such valve position indication is sooner made operable.

4.6.D Safety and Relief Valves

1. Approximately half of the safety valves and relief 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 valves shall be disassembled and inspected each refueling outage.

3. The integrity of the relief safety valve bellows on any three stage valve shall be continuously monitored.

4. The operability of the bellows monitoring system shall be demonstrated once every three months when three stage valves are installed.

5. Once per operating cycle, with the reactor pressure $\geq$ 100 psig, each relief valve shall be manually opened until the main turbine bypass valves have closed to compensate for relief valve opening.

6.

a. Operability of the relief valve position indicating pressure switches and the safety valve position indicating thermocouples shall be demonstrated once per operating cycle.

b. An Instrument Check of the safety and relief valve position indicating devices shall be performed monthly.

The relief and safety valves are bench tested every second operating cycle to ensure that their set points are within the $\pm$ 1 percent tolerance. Additionally, once per operating cycle, each relief valve is tested manually with reactor pressure above 100 psig to demonstrate its ability to pass steam.

The requirements established above apply when the nuclear system can be pressurized above ambient conditions. These requirements are applicable at nuclear system pressures below normal operating pressures because abnormal operational transients could possibly start at these conditions such that eventual overpressure relief would be needed. However, these transients are much less severe, in terms of pressure, than starting at rated conditions. The valves need not be functional when the vessel head is removed, since the nuclear system cannot be pressurized.

The position indicating pressure switches for the relief valves and the thermocouples for the safety valves serve as a diagnostic aid to the operator in the event of a safety/relief valve failure. If position indication is lost, alternate means are available to the operator to determine if a safety valve is leaking.

E. Jet Pumps

Failure of a jet pump nozzle assembly hold down mechanism, nozzle assembly and/or riser, would increase the cross-sectional flow area for blowdown following the design bases double-ended line break. Therefore, if a failure occurs, repairs must be made.

The detection technique is as follows. With the two recirculation pumps balanced in speed to within $\pm$ 5%, the flow rates in both recirculation loops will be verified by Control Room monitoring instruments. If the two flow rate values do not differ by more than 10%, riser and nozzle assembly integrity has been verified. If they do differ by 10% or more, the core flow rate measured by the jet pump diffuser differential pressure system must be checked against the core flow rate derived from the measured values of loop flow to core flow correlation. If the difference between measured and derived core flow rate is 10% or more (with the derived value higher) diffuser measurements will be taken to define the location within the vessel of failed jet pump nozzle (or riser) and the plant shut down for repairs. If the potential blowdown flow area is increased, the system resistance to the recirculation pump is also reduced; hence, the affected drive pump will "run out" to a substantially higher flow rate (approximately 115% to 120% for a single nozzle failure). If the two loops are balanced in flow at the same pump speed, the resistance characteristics cannot have changed. Any imbalance between drive loop flow rates would be indicated by the plant process instrumentation. In addition, the affected jet pump would provide a leakage path past the core thus reducing the core flow rate. The reverse flow through the inactive jet pump would still be indicated by a positive differential pressure but the net effect would be a slight decrease (3% to 6%) in the total core flow measured. This decrease, together with the loop flow increase, would result in a lack of correlation between measured and derived core flow rate. Finally, the affected jet pump diffuser differential pressure signal would be reduced because the backflow would be less than the normal forward flow.

A nozzle-riser system failure could also generate the coincident failure of a

G. A Fire Brigade of at least 5 members shall be maintained at all times. This excludes the 3 members of the minimum shift crew necessary for safe shutdowns, and other personnel required for other essential functions during a fire emergency. Three fire Brigade members shall be from the Operations Department and 2 support members may be from other departments inclusive of Security personnel.

Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence of Fire Brigade members provided immediate action is taken to restore the Fire Brigade to within the minimum requirements.

H. In order to perform the function of accident assessment an engineer from the normal plant engineering staff shall be assigned to each shift during reactor operation. If the lack of qualified engineers necessitates, an additional senior reactor operator assigned to each shift may substitute in the performance of the accident assessment function. This requirement is effective until January 1, 1981.

6.1.4 The minimum qualifications, training, replacement training, and retraining of plant personnel at the time of fuel loading or appointment to the active position shall meet the requirements as described in the American National Standards Institute N-18.1-1971, "Selection and Training of Personnel for Nuclear Power Plants". The Assistant to Station Superintendent qualifications shall comply with Section 4.2 of ANSI-N18.1-1971. The Chemistry and Health Physics Supervisor shall meet or exceed the qualifications of Regulatory Guide 1.8, Sept. 1975; personnel qualification equivalency as stated in the Regulatory Guide may be proposed in selected cases. The minimum frequency of the retraining program shall be every two years. The training program shall be under the direction of a designated member of the plant staff.

A. A training program for the fire brigade will be maintained under the direction of the plant training coordinator and shall meet or exceed the requirements of Section 27 of the NFPA Code 1976, except for Fire Brigade training sessions which shall be held at least quarterly.

The training program requirements will be provided by a qualified fire protection engineer, thru the Risk Manager.

6.8 Environmental Qualification

A. By no later than June 30, 1982 all safety-related electrical equipment in the facility shall be qualified in accordance with the provisions of: Division of Operating Reactors "Guidelines for Evaluating Environmental Qualification of Class IE Electrical Equipment in Operating Reactors" (DOR Guidelines); or, NUREG-0588 "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment", December 1979. Copies of these documents are attached to Order for Modification of License DPR-46 dated October 24, 1980.

B. By no later than December 1, 1980, complete and auditible records must be available and maintained at a central location which describe the environmental qualification method used for all safety-related electrical equipment in sufficient detail to document the degree of compliance with the DOR Guidelines or NUREG-0588. Thereafter, such records should be updated and maintained current as equipment is replaced, further tested, or otherwise further qualified.

6.9 Systems Integrity Monitoring Program

A program shall be established to reduce leakage from systems outside the primary containment that would or could contain highly radioactive fluids during a serious accident to as low as practical levels. This program shall include provisions establishing preventive maintenance and periodic visual inspection requirements, and leak testing requirements for each system at a frequency not to exceed refueling cycle intervals.

6.10 Iodine Monitoring Program

A program shall be established to ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include training of personnel, procedures for monitoring and provisions for maintenance of sampling and analysis equipment.