2.2 REACTOR COOLANT SYSTEM

Applicability:

Applies to limits on reactor coolant system pressure.

Objective:

To establish a limit below which the integrity of the reactor coolant system is not threatened due to an overpressure condition.

Specification:

The reactor vessel pressure shall not exceed 1335 psig at any time when irradiated fuel is present in the reactor vessel.

2.4 REACTOR COOLANT SYSTEM

Applicability:

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objective:

To define the level of the process variables at which automatic protective action is initiated to prevent the safety limits from being exceeded.

Specification:

A. Reactor Coolant High Pressure Scram shall be $\leq$ 1075 psig.

B. The self-actuation function of at least seven Reactor Coolant System safety relief valves shall be operable. Valves shall be set as follows:

   8 valves at $\leq$ 1108 psig.

Bases Continued:

2.2 The normal operating pressure of the reactor coolant system is approximately 1025 psig. The turbine trip with failure of the bypass system represents the most severe primary system pressure increase resulting from an abnormal operational transient. The peak pressure in this transient is limited to 1207 psig. The safety/relief valves are sized assuming no direct scram during MSIV closure. The only scram assumed is from an indirect means (high flux) and the pressure at the bottom of the vessel is limited to 1248 psig in this case. The analysis assumed that only seven of the eight valves are operable and that they open at 1% over their setpoint with a 0.4 second delay. Reactor pressure is continuously monitored in the control room during operation on a 1500 psig full scale pressure recorder.

Bases:

2.4 The settings on the reactor high pressure scram, reactor coolant system safety/relief valves, turbine control valve fast closure scram, and turbine stop valve closure scram have been established to assure never reaching the reactor coolant system pressure safety limit as well as assuring the system pressure does not exceed the range of the fuel cladding integrity safety limit. The APRM neutron flux scram and the turbine bypass system also provide protection for these safety limits. In addition to preventing power operation above 1075 psig, the pressure scram backs up the APRM neutron flux scram for steam line isolation type transients.

The reactor coolant system safety/relief valves assure that the reactor coolant system pressure safety limit is never reached. In compliance with Section III of the ASME Boiler and Pressure Vessel Code, 1965 edition, the safety/relief valves must be set to open at a pressure no higher than 105 percent of design pressure, and they must limit the reactor pressure to no more than 110 percent of design pressure. The safety/relief valves are sized according to the Code for a condition of MSIV closure while operating at 1670 MWt, followed by no MSIV closure scram but scram from an indirect (high flux) means. With the safety/relief valves set as specified herein, the maximum vessel pressure (at the bottom of the pressure vessel) would be about 1248 psig. Only seven of the eight valves are assumed to be operable in this analysis and the valves are assumed to open at 1% above their setpoint with a 0.4 second delay.

The operator will set the reactor coolant high pressure scram trip setting at 1075 psig or lower. However, the actual setpoint can be as much as 10 psi above the 1075 psig indicated set point due to the deviations discussed in the basis of Specification 2.3 on Page 18. In a like manner, the operator will set the reactor coolant system safety/relief valve initiation trip setting at 1108 psig or lower. However, the actual set point can be as much as 11.1 psi above the 1108 psig indicated set point due to the deviations discussed in the basis of Specification 2.3 on Page 18.

A violation of this specification is assumed to occur only when a device is knowingly set outside of the limiting trip setting, or when a sufficient number of devices have been affected by any means

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
E. Safety/Relief Valves 1. During power operating conditions and whenever reactor coolant pressure is greater than 110 psig and temperature is greater than 345°F. a. The safety valve function (self-actuation) of seven safety/relief valves shall be operable. b. The solenoid activated relief function (Automatic Pressure Relief) shall be operable as required by Specification 3.5.E.	E. Safety/Relief Valves 1. a. A minimum of seven safety/relief valves shall be bench checked or replaced with a bench checked valve each refueling outage. The nominal setpoint of all operational safety/relief valves shall be 1108 psig. b. At least two of the safety/relief valves shall be disassembled and inspected each refueling outage. c. The integrity of the safety/relief valve bellows shall be continuously monitored. d. The operability of the bellows monitoring system shall be demonstrated at least once every three months.

Bases Continued 3.6 and 4.6:

D. Coolant Leakage

The former 15 gpm limit for leaks from unidentified sources was established assuming such a leakage was comming from the primary system. Tests have been conducted which demonstrate that a relationship exists between the size of a crack and the probability that the crack will propagate. From the crack size a leakage rate can be determined. For a crack size which gives a leakage of 5 gpm, the probability of rapid propagation is less than 10^{-5}. Thus, an unidentified leak of 5 gpm when assumed to be from the primary system had less than one chance in 100,000 of propagating, which provides adequate margin. A leakage of 5 gpm is detectable and measurable. The 24 hour period allowed for determination of leakage is also based on the low probability of the crack propagating.

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

E. Safety/Relief Valves

Testing of all required safety/relief valves each refueling outage ensures that any valve deterioration is detected. A tolerance value of 1% for safety/relief valve setpoints is specified in Section III of the ASME Boiler and Pressure Vessel Code. Analyses have been performed with all valves assumed set 1% higher (1108 psig + 1%) than the nominal setpoint; the 1375 psig code limit is not exceeded in any case.

The safety/relief valves are used to limit reactor vessel overpressure and fuel thermal duty.

The required safety/relief valve steam flow capacity is determined by analyzing the transient accompanying the main steam flow stoppage resulting from a postulated MSIV Closure from a power of 1670 Mwt. The analysis assumes a multiple-failure wherein direct scram (valve position) is neglected. Scram is assumed to be from indirect means (high flux). In this event, the safety/relief valve capacity is assumed to be 83.2% of the full power steam generation rate.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
C. Minimum Critical Power Ratio (MCPR) During power operation, the Operating MCPR Limit shall be $\geq$ 1.33 for 8x8 fuel and $\geq$ 1.33 for 8x8R fuel at rated power and flow. If at any time during operation it is determined that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours. For core flows other than rated the Operating MCPR Limit shall be the above applicable MCPR value times K_f where K_f is as shown in Figure 3.11.3.	C. Minimum Critical Power Ratio (MCPR) MCPR shall be determined daily during reactor power operation at $\geq$ 25% rated thermal power and following any change in power level or distribution which has the potential of bringing the core to its operating MCPR limit.

Bases Continued

C. Minimum Critical Power Ratio (MCPR)

The ECCS evaluation presented in Reference 4 assumed the steady state MCPR prior to the postulated loss-of-coolant accident to be 1.18 for all fuel types. In addition, the ECCS analysis presented in Reference 6 assumed an initial MCPR of 1.24 for reduced flow conditions. The Operating MCPR Limit of 1.33 for 8x8 fuel and 1.33 for 8x8R fuel is determined from the analysis of transients discussed in Bases Sections 2.1 and 2.3. By maintaining an operating MCPR above these limits, the Safety Limit (T.S. 2.1.A) is maintained in the event of the most limiting abnormal operational transient.

For operation with less than rated core flow the Operating MCPR Limit is adjusted by multiplying the above limit by K_f. Reference 5 discusses how the transient analysis done at rated conditions encompasses the reduced flow situation when the proper K_f factor is applied.

Those abnormal operational transients, analyzed in FSAR Section 14.5, which result in an automatic reactor scram are not considered a violation of the LCO. Exceeding MCPR limits in such cases need not be reported.

References

1. "Fuel Densification Effects in General Electric Boiling Water Reactor Fuel," Supplements 6, 7, and 8, NEDM-10735, August, 1973.

2. Supplement 1 to Technical Report on Densification of General Electric Reactor Fuels, December 14, 1974 (USAEC Regulatory Staff).

3. Communication: VA Moore to IS Mitchell, "Modified GE Model for Fuel Densification," Docket 50-321, March 27, 1974.

4. "Loss-of-coolant Accident Analysis Report for the Monticello Nuclear Generating Plant," NEDO-24050, September 1977, L O Mayer (NSP) to V Stello (USNRC), September 15, 1977.

5. "General Electric BWR Generic Reload Application for 8x8 Fuel," NEDO-20360, Revision 1, November 1974.

6. "Revision of Low Core Flow Effects on LOCA Analysis for Operating BWR's," R L Gridley (GE) to D G Eisenhut (USNRC), September 28, 1977.

EXHIBIT C

LICENSE AMENDMENT REQUEST

DATED MARCH 16, 1978

This exhibit consists of General Electric Report NEDO-24133-1 entitled, "Supplement 1 Monticello Reload 6 - Simmer Margin Evaluation." This report supplements the safety analysis for reload 6 contained in NEDO-24133. NEDO-24133 was submitted for NRC review on August 10, 1978.

NEDO-24133-1 provides the results of additional analyses which demonstrate the acceptability of increasing the maximum allowable safety/relief valve setpoint to 1108 psig.