

1.1 FUEL CLADDING INTEGRITY

or core coolant flow is less than 10% of rated, the core thermal power shall not exceed 823 MWt (about 25% of rated thermal power).

- C. Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than 17.7 in. above the top of the normal active fuel zone.

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$$S_{RB} \leq (0.66W + 42\%)$$

where:

S_{RB} = Rod block setting in percent of rated thermal power (3293 MWt)

W = Loop recirculation flow rate in percent of rated (rated loop recirculation flow rate equals 34.2×10^6 lb/hr)

In the event of operation with the core maximum fraction of limiting power density (CMFLPD) greater than fraction of rated thermal power (FRP) the setting shall be modified as follows:

$$S_{RB} \leq (0.66W + 42\%) \frac{FRP}{CMFLPD}$$

- C. Scram and isolation-- ≥ 540 in. above reactor low water vessel zero level

- D. Scram--turbine stop ≤ 10 percent valve closure valve closure

- E. Scram--turbine control valve

1. Fast closure

When trip of the fast action shutdown valve

2. Loss of control ≥ 550 psig oil pressure

- F. Scram--low condenser vacuum ≥ 23 inches Hg vacuum

- G. Scram--main steam ≤ 10 percent line isolation valve closure

- H. Main steam isolation ≥ 800 psig valve closure--nuclear system low pressure

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1. J. & K. Practor low water level set point for initiation of HPCI and RCIC, closing main steam isolation valves, and starting LPCI and core spray pumps.

These systems maintain adequate coolant inventory and provide core cooling with the objective of preventing excessive clad temperatures. The design of these systems to adequately perform the intended function is based on the specified low level scram set point and initiation set points. Transient analyses reported in Section 14 of the FSAR demonstrate that these conditions result in adequate safety margins for both the fuel and the system pressure.

L. References

1. Linford, P. B., "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," NEDO-10802, Feb., 1973.
2. Generic Reload Fuel Application, Licensing Topical Report. NEDE-24011-P-A, and Addenda.

1.2 BASIS

pressure monitor higher in the vessel. Therefore, following any transient that is severe enough to cause concern that this safety limit was violated, a calculation will be performed using all available information to determine if the safety limit was violated.

REFERENCES

1. Plant Safety Analysis (BTRP FSAR Section 14.0)
2. ASME Boiler and Pressure Vessel Code Section III
3. UCSB Piping Code, Section B31.1
4. Reactor Vessel and Appurtenances Mechanical Design (BTRP FSAR Subsection 4.2)
5. Generic Reload Fuel Application, Licensing Topical Report, NEDE-24011-P-A and Addenda.

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REACTOR COOLANT SYSTEM INTEGRITY

To meet the safety design basis, thirteen relief valves have been installed on the unit with a total capacity of 82.6% of nuclear boiler rated steam flow. The analysis of the worst overpressure transient, (3-second closure of all main steamline isolation valves) neglecting the direct scram (valve position scram) results in a maximum vessel pressure which, if a neutron flux scram is assumed considering 12 valves operable, results in adequate margin to the code allowable overpressure limit of 1375 psig.

To meet the operational design, the analysis of the plant isolation transient (generator load reject with bypass valve failure to open) shows that 12 of the 13 relief valves limit peak system pressure to a value which is well below the allowable vessel overpressure of 1375 psig.

3.3/4.4 BASFS:D. Reactivity Anomalies

During each fuel cycle excess operative reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity may be inferred from the critical rod configuration. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of the critical rod pattern at selected base states to the predicted rod inventory at that state. Power operating base conditions provide the most sensitive and directly interpretable data relative to core reactivity. Furthermore, using power operating base conditions permits frequent reactivity comparisons.

Requiring a reactivity comparison at the specified frequency assures that a comparison will be made before the core reactivity change exceeds $1\% \Delta K$. Deviations in core reactivity greater than $1\% \Delta K$ are not expected and require thorough evaluation. One percent reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

References

1. Generic Reload Fuel Application, Licensing Topical Report, NEDE-24011-P-A and Addenda.

$LHGR \leq LHGR - (\Delta P/P) \quad (L/LT)$
 $\text{max} \quad \text{max}$
 $LHGR = \text{Design } LHGR = 18.5 \text{ kW/ft for } 7 \times 7 \text{ fuel}$
 d
 $= 13.4 \text{ kW/ for } 8 \times 8 \text{ fuel}$
 $8 \times 8R \text{ and } P8 \times 8R \text{ fuel}$
 $(\Delta P/P) = \text{Maximum power spiking penalty}$
 $= 0.026 \text{ for } 7 \times 7 \text{ fuel}$
 $= 0.022 \text{ for } 8 \times 8, 8 \times 8R \text{ and } P8 \times 8R \text{ fuel}$
 $* LT = \text{Total core length} = 12.0 \text{ ft for } 7 \times 7 \text{ fuel}$
 $\text{and } 8 \times 8$
 $= 12.5 \text{ ft for } 8 \times 8, 8 \times 8R \text{ \& } P8 \times 8R$

$L = \text{Axial position above bottom of core}$
 If at any time during operation it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR 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. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

K. Minimum Critical Power Ratio (MCPR)

From BOC to EOC-2000 $MW0/T$
 the MCPR operating limit for BFNP 1 cycle 4 is 1.23 for 7x7 fuel, 1.24 for 8x8 fuel, and 1.25 for 8x8R and P8x8R fuel. These limits apply to steady state power operation at rated power and flow. For core flows other than rated the MCPR shall be greater than the above limits times K_f . K_f is the value shown in Figure 3.5.2. From EOC-2000 to EOC the MCPR limits will be 1.23, 1.27, and 1.28 for 7x7, 8x8 and 8x8R/P8x8R respectively. If at any time during operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady 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, surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

L. Reporting Requirements

If any of the limiting values identified in Specifications 3.5.I, J, or K are exceeded and the specified action is taken, the event shall be logged and reported in a 30-day written report.

* 12.5 feet for 8x8R fuel

K. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during reactor power operation at 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.

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3.5 SAFES3.5.M Maintenance of Filled Discharge Line

If the discharge piping of the core spray, LPCIS, HPCIS, and RCICS are not filled, a water hammer can develop in this piping when the pump and/or pumps are started. To minimize damage to the discharge piping and to ensure added margin in the operation of these systems, this Technical Specification requires the discharge lines to be filled whenever the system is in an operable condition. If a discharge pipe is not filled, the pumps that supply that line must be assumed to be inoperable for Technical Specification purposes.

The core spray and RHR system discharge piping high point vent is visually checked for water flow once a month prior to testing to ensure that the lines are filled. The visual checking will avoid starting the core spray or RHR system with a discharge line not filled. In addition to the visual observation and to ensure a filled discharge line other than prior to testing, a pressure suppression chamber head tank is located approximately 20 feet above the discharge line highpoint to supply makeup water for these systems. The condensate head tank located approximately 100 feet above the discharge high point serves as a backup charging system when the pressure suppression chamber head tank is not in service. System discharge pressure indicators are used to determine the water level above the discharge line high point. The indicators will reflect approximately 30 psig for a water level at the high point and 45 psig for a water level in the pressure suppression chamber head tank and are monitored daily to ensure that the discharge lines are filled.

When in their normal standby condition, the suction for the HPCIS and RCIS pumps are aligned to the condensate storage tank, which is physically at a higher elevation than the HPCIS and RCIS piping. This assures that the HPCIS and RCIS discharge piping remains filled. Further assurance is provided by observing water flow from these systems high points monthly.

3.5.1. Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the LOCFR50, Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than $\pm 20^\circ\text{F}$ relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the LOCFR50 Appendix K limit. The limiting value for MAPLHGR is shown in Tables 3.5.1-1, -2, -3, -4, -5, -6, and -7 per reference 4.

3.5.J. Linear Heat Generation Rate (LHGR)

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation if fuel pellet densification is postulated. The power spike penalty specified is based on the analysis presented in Section 3.2.1 of Reference 1 as modified in References 2 and 3, and assumes a linearly increasing variation in axial gaps between core bottom and top, and assures with a 95% confidence, that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking. The LHGR as a function of core height shall be checked daily during reactor operation at $\geq 25\%$ power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the MTRF would have to be greater than 10 which is precluded by a considerable margin when employing any permissible control rod pattern.

3.5.K. Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25%, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience and thermal hydraulic analysis indicated that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR. The daily requirement for calculating MCPR above 25% rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.

3.5.L. Reporting Requirements

The LCO's associated with monitoring the fuel rod operating conditions are required to be met at all times, i.e., there is no allowable time in which the plant can knowingly exceed the limiting values for MAPLHGR, LHGR, and MCPR. It is a requirement, as stated in Specifications 3.5.1.J, and .K, that if at any time during steady state power operation, it is determined that the limiting values for MAPLHGR, LHGR, or MCPR are exceeded action is then initiated to restore operation to within the prescribed limits. This action is initiated as soon as normal surveillance indicates that an operating limit has been reached. Each event involving steady state operation beyond a specified limit shall be logged and reported quarterly. It must be recognized that there is always an action which would return any of the parameters (MAPLHGR, LHGR, or MCPR) to within prescribed limits, namely power reduction. Under most circumstances, this will not be the only alternative.

H. References

1. "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel," Supplements 6, 7, and 8, NEEM-10735, August 1973.
2. Supplement 1 to Technical Report on Densifications of General Electric Reactor Fuels, December 14, 1974 (USA Regulatory Staff).
3. Communication: V. A. Moore to I. S. Mitchell, "Modified GE Model for Fuel Densification," Docket 50-321, March 27, 1974.
4. Generic Reload Fuel Application, Licensing Topical Report, NEDE-24011-P-A and addenda.

3.6/4.6 BASES

detected reasonably in a matter of few hours utilizing the available leakage detection schemes, and if the origin cannot be determined in a reasonably short time the unit should be shut down to allow further investigation and corrective action.

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 pump is 50 gpm and the capacity of the drywell equipment sump pump is also 50 gpm. Removal of 25 gpm from either of these sumps can be accomplished with considerable margin.

REFERENCES

1. Nuclear System Leakage Rate Limits (BNPP FSAR Subsection 4.10)

3.6.D/4.6.DRelief Valves

To meet the safety basis thirteen relief valves have been installed on the unit with a total capacity of 82.6% of nuclear boiler rated steam flow. The analysis of the worst overpressure transient, (3-second closure of all main steam line isolation valves) neglecting the direct scram (valve position scram) results in a maximum vessel pressure which, if a neutron flux scram is assumed considering 12 valves inoperable, results in adequate margin to the code allowable overpressure limit of 1375 psig.

To meet operational design, the analysis of the plant isolation transient (generator load reject with bypass valve failure to open) shows that 12 of the 13 relief valves limit peak system pressure to a value which is well below the allowed vessel overpressure of 1375 psig.

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Experience in relief valve operation shows that a testing of 50 percent of the valves per year is adequate to detect failures or deteriorations. The relief valves are benchtested every second operating cycle to ensure that their set points are within the ± 1 percent tolerance. The relief valves are tested in place once per operating cycle to establish that they will open and 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 those starting at rated conditions. The valves need not be functional when the vessel head is removed, since the nuclear system cannot be pressurized.

REFERENCES

1. Nuclear System Pressure Relief System (BFNP FSAR Subsection 4.4)
2. Amendment 22 in response to AEC Question 4.2 of December 6, 1971.
3. "Protection Against Overpressure" (ASME Boiler and Pressure Vessel Code, Section III, Article 9)
4. Browns Ferry Nuclear Plant Design Deficiency Report--Target Rock Safety-Relief Valves, transmitted by J. E. Gilleland to F. E. Kruesi, August 29, 1973.
5. Generic Reload Fuel Application, Licensing Topical Report, NEDE-24011-P-A and Addenda.

3.6.E/4.6.E Jet Pumps

Failure of a jet pump nozzle assembly holddown mechanism, nozzle assembly and/or riser, would increase the cross-sectional flow area for blowdown following the design basis double-ended line break. Also, failure of the diffuser would eliminate the capability to reflood the core to two-thirds height level following a recirculation line break. Therefore, if a failure occurred, repairs must be made.

The detection technique is as follows. With the two recirculation pumps balanced in speed to within ± 5 percent, 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 percent, riser and nozzle assembly integrity has been verified.