50-259
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TENNESSEE VALLEY AUTHORITY
CHATTANOOGA, TENNESSEE 37401
830 Power Building

November 30, 1978

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Denton:

In the Matter of the) Docket No. 50-259
Tennessee Valley Authority)

In my letters to you of September 8 and October 5, 1978, we submitted for NRC review an amendment request and corrections to the amendment request, respectively, to change the technical specifications of Browns Ferry Nuclear Plant unit 1. As discussed with members of your staff, TVA has requested that GE perform revised bundle loading error analysis for the rotated bundle utilizing the new method and to include mid-cycle (EOC - 2000 MWd/T) exposure MCPR points. Enclosed for your review are the results of the aforementioned analysis (Enclosure 2). Also enclosed are changes to the proposed technical specifications submitted by my September 8, 1978, letter (Enclosure 1). This letter and enclosures also serves to answer question 8.b) of T. A. Ippolito's letter to N. B. Hughes of November 22, 1978, concerning this subject.

Very truly yours,

J E Gilleland

J. E. Gilleland
Assistant Manager of Power

Subscribed and sworn to before
me this 30 day of Nov. 1978.

Ida B. Peckett
Notary Public

My Commission Expires March 30

Enclosures
cc: See page 2

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An Equal Opportunity Employer

cc (Enclosures):
Mr. Charles R. Christopher
Chairman, Limestone County Commission
P.O. Box 188
Athens, Alabama 35611

Dr. Ira L. Myers
State Health Officer
State Department of Public Health
State Office Building
Montgomery, Alabama 36104

ENCLOSURE 1

1.1 BASES

The safety limit has been established at 17.7 in. above the top of the irradiated fuel to provide a point which can be monitored and also provide adequate margin. This point corresponds approximately r the top of the actual fuel assemblies and also to the lower reactor low water level trip (378" above vessel zero).

REFERENCE

1. General Electric BWR Thermal Analysis Basis (GETAB) Data, Correlation and Design Application, NEDO 10958 and NEDE 10958.

2. GE BWR Reload 2 Licensing Amendment for BFNP unit 1 reload 2, NEDO-24136, August 1978 and Revision 1 dated November 1978.

2.1 BASES

I. J. & K. Reactor 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, R. B., "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor," NEDO-10802, Feb., 1973.

2. GE BWR Reload 2 Licensing Amendment for BFNP unit 1 reload 2, NEDO-24136, August 1978 and Revision 1 dated November 19 8.

1.2 BASES

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 (BFNP FSAR Section 14.0)

2. ASME Boiler and Pressure Vessel Code Section III

3. USAS Piping Code, Section B31.1

4. Reactor Vessel and Appurtenances Mechanical Design (BFNP FSAR Subsection 4.2)

5. GE BWR Reload 2 Licensing Amendment for BFNP unit 1 reload 2, NEDO-24136, August 1978 and Revision 1 dated November 1978.

3.3/4.4 BASES:

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% Δk. Deviations in core reactivity greater than 1% Δk 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. General Electric BWR Reload 2 Licensing Amendment for BFNP unit 1 reload 2, NEDO-24136, August 1978 and Revision 1 dated November 1978.

$LHGR_{max} \leq LHGR_d[1 - (\Delta P/P)_{max}(L/LT)]$

$LHGR_d$ = Design LHGR = 18.5 kW/ft. for 7x7fuel
= 13.4 kW/ft for 8x8fuel

$(\Delta P/P)_{max}$ = Maximum power spiking penalty
=0.026 for 7X7 fuel
= 0.022 for 8X8 fuel

LT* = Total core length = 12.0 feet for 7X7 fuel
= 12.2 feet for 8X8 fuel

L = 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)

The MCPR operating limit from the beginning of cycle 3 to the end of cycle 3 minus 2000 MWd/t is 1.20 for 7X7 fuel and 1.22 for 8X8 and 8X8R fuel; the limit from the end of cycle 3 minus 2000 MWd/t to the end of cycle 3 is 1.22 for 7X7 fuel and 1.27 for 8X8 and 8X8R fuel. These limits apply to steady state power operation at rated power and flow. For core flows other than rated, theMCPR shall be greater than the above limits times K_f. K_f is the value shown in Figure 3.5.2. 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 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. 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 remedial action is taken, the event shall be logged and reported in a 30-day written report.

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.

* 12.5 feet for 8x8 fuel

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 MTPF 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.I, 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.

M. References

1. "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel," Supplements 6, 7, and 8, NEDM-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. General Electric BWR Reload 2 Licensing Amendment for BFNP unit1, reload 2, NEDO-24136, August 1978 and Revision 1 dated November 1978.

3.6/4.6 BASES:

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. GE BWR Reload 2 Licensing Amendment for BFNP unit 1 reload 2, NEDO-24136, August 1978 and Revision 1 dated November 1978.

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.

ENCLOSURE 2