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Office of Nuclear Reactor Regulation
CRUTCHFIELD,D. Operating Reactors Branch 5

SUBJECT: Forwards response to NRC 800226 ltr requesting addl info re SEP design basis events.Topics cover steady state & instrument errors,single failure analysis,steam line break analyses & primary pump rotor seisure.

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LEON D. WHITE, JR.
VICE PRESIDENT

TELEPHONE
AREA CODE 716 546-2700

June 3, 1980

Director of Nuclear Reactor Regulation
Attention: Mr. Dennis M. Crutchfield, Chief
Operating Reactors Branch No. 5
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: SEP - Design Basis Events for Ginna
R. E. Ginna Nuclear Power Plant
Docket No. 50-244

Dera Mr. Crutchfield:

Enclosed is the additional information you requested in a letter from Mr. Dennis Ziemann dated February 26, 1980 which was received on March 5, 1980.

L. D. White, Jr.

L. D. White, Jr.

Enclosure

A035 5/1/1

8006090331

**SUBJECT:** SEP - Design Basis Events for Ginna

Question 1. The transient analyses provided in the topical report XN-NF-77-40 generally resulted in higher calculated MDNBR values than those calculated values corresponding to the reference cycle analysis. One apparent reason for this trend is the increased initial DNBR assumed for the two sets of analyses. Verify that the Cycle 8 analyses, based on an initial DNBR of 2.00, reflects the ratio at the assumed 102% power level and assumed pressure and temperature conditions.

Response: Page 5 of XN-NF-77-40 states the following assumptions are applied to all full power transients to account for steady state and instrument errors.

Reactor Power = 1.02 X 1520 MWt
$T_{ave}$ = 573.5 + 4°F
Pressure = 2250 - 30 psia

Question 2. Provide the following additional information regarding the analysis of a turbine trip event:

a. Provide a single failure analysis to determine the limiting failure concurrent with a turbine trip. Evaluate the effects of the worst single failure on the turbine trip analysis.

b. Justify why a range of reactivity feedback effects was not considered since this is normally analyzed for this type of plant.

c. Identify whether the scram characteristics include assuming the most reactive rod stuck out of the core.

Response 2.a. A specific single failure analysis is not available; however, several cases were analyzed by Westinghouse. The objective of performing analysis of this transient is to:

1. Show that the primary system pressure relieving devices can limit primary system pressure to acceptable levels.

2. Show that no core damage (DNB violations) occur during the transient.

SUBJECT: [illegible] Design Basis Events for Ginna

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1. Show that the primary system pressure relieving devices can limit primary system pressure to acceptable levels.

2. Show that no core damage (DNB violation) occurs during the transient.

Therefore, Westinghouse assumed the following:

1. Evaluated the transient at beginning of cycle (BOC) 0.0 moderator coefficient, and end of cycle (EOC), most negative value of moderator coefficient. For all cases, a large negative value of Doppler power coefficient ($-1.5 \times 10^{-4}$ $\Delta$K/%power) was used. This minimizes reduction in core power prior to trip.

2. No credit was taken for Steam Dump or Steam Generator PORV's.

Two cases were analyzed for both BOC and EOC conditions.

1. The reactor was assumed to be in normal automatic control (including pressurizer spray and PORV's) with control rods in the minimum incremental worth region.

2. The reactor was assumed to be in manual control. There was no control rod insertion following loss of load, and no credit taken for pressurizer spray and PORV's.

For the several cases evaluated, the DNBR during the transient never decreased below its initial steady state value. Only one case was done by Exxon. BOC conditions were assumed with 0.0 moderator coefficient, minimum Doppler, no pressurizer spray or PORV's and no steam dump or steam generator PORV's. During the Exxon transient reactor power, coolant temperature, and pressure all increased and DNBR decreased to a minimum of 1.83.

Response 2.b. A range of reactivity feedback effects were considered in the Westinghouse analysis. Moderator coefficients of 0.0 and $-3.5 \times 10^{-4}$ $\Delta K/^{o}F$ were used with the maximum negative Doppler power coefficient to minimize the reduction in core power prior to trip. Exxon analyzed only one case; BOC coefficients with 0.0 Moderator coefficient and minimum negative Doppler Temperature Coefficient.

Response 2.c. The FSAR states the following on page 14.1-1 for the Westinghouse analysis: "All reactor protection criteria are met presupposing the most reactive RCC assembly is in its fully withdrawn position...".

On page 10 of XN-NF-77-40 Exxon presents the value of scram worth used in its transient analysis.

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On page 9 of XN-NF-77-40 Exxon presents the value of [illegible] worth used in its transient analysis.

This value is less than the N-1 worth of Cycles 8,9, and 10; therefore, one can assume N-1 is used in the transient analysis.

Question 3 Provide the following additional information regarding the analyses of a main steam line break:

a. Discuss the main feedwater and auxiliary feedwater flow assumptions used in the analysis. These assumptions should conservatively maximize flow to the broken loop steam generator.

b. Discuss the potential for single failures in the auxiliary feedwater control system which may result in runout flow being continuously directed to the broken loop steam generator.

c. Specify the initial core flow assumed in the analysis and demonstrate the assumptions are conservative.

d. Specify how stored energy in the primary system (e.g., thick metal) was treated.

Response 3. Most of the information requested here was submitted to the NRC in our response to IE Bulletin 80-04 by letter dated April 30, 1980. Relevant information is resubmitted here.

Response 3.a. The original steam line break analysis was done by Westinghouse. Later, when Exxon fuel was used at Ginna, the most limiting steam line break was reanalyzed by Exxon.

In the Westinghouse analysis all auxiliary feedwater pumps are initially assumed to be operating, in addition to the main feedwater pumps. The flow is equivalent to the rated flow of all pumps at the steam generator design pressure. Feedwater is assumed to continue at its initial flow rate until feedwater isolation is complete, while auxiliary feedwater is assumed to continue at its initial flow rate. Main feedwater flow is completely terminated following feedwater isolation.

The most limiting steam line break determined by Westinghouse was analyzed by Exxon. This transient occurs at hot zero power with outside power available and the break occurring at the exit of the steam generator. The analysis does not specifically account for auxiliary feedwater. However, the steam generator

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heat transfer model, using constant heat transfer coefficients, continues to calculate heat transfer from the primary to the secondary side after the broken steam generator has been estimated to be empty. If auxiliary flow was specifically accounted for, its effect would be negligible during the initial portion of the transient and would have minimal effect during later portions of the transient since by the time the broken steam generator empties, the total system reactivity is negative and core power is decreasing. The additional reactivity addition associated with the slight cooldown due to runout flow is more than negated by the boron reactivity inserted by safety injection.

There is no need to consider the operation of the auxiliary feedwater pumps at runout flow because the core transient results are very insensitive to auxiliary feedwater flow. The first minute of the transient is dominated entirely by the steam flow contribution to primary - secondary heat transfer, which is the forcing function for both the reactivity and thermal - hydraulic transients in the core. The effect of auxiliary feedwater runout is minimal. The turbine-driven auxiliary feedwater pumps are controlled by a governor and will not exceed about 400 gpm. The motor driven pump flow is controlled by the AFW control valves, which receive an automatic throttle signal to 200 gpm from their flow controllers.

A potential single failure of the flow controller to control flow to 200 gpm is not considered a worst-case single failure since a failure that results in minimum safeguards will result in a more limiting core transient.

Greater main feedwater flow during the large steamline breaks would greater reduce secondary pressures, accelerating the automatic safeguards actions, i.e., steamline isolation, feedwater isolation and safety injection, which would terminate the transient sooner.

Response 3.b. Ginna does not have a system that terminates auxiliary feedwater flow to the broken steam generator. Therefore, there is no single failure that will result in runout flow being continuously directed to the broken loop steam generator. Auxiliary feedwater flow to the broken steam generator will eventually require operator action to realign flow to the intact generator or terminate flow to the broken generator. Positive information is available to the operator to determine

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which is the affected steam generator; through proper training and by use of the emergency procedures the operator will be capable of quickly recognizing the steamline break and perform the proper operations.

Response 3.c. Table 1.4-1 of "Technical Supplement Accompanying Application to Increase Power", February, 1971 lists the total primary flow as 68.0 x $10^6$ lbs/hr. XN-NF-77-40 Supp-1, March 1980 lists the total primary flow as 68.0 x $10^6$ lbs/hr. Westinghouse letter from L.B. Kincaid to E.U. Powell, January 29, 1970 provides the following preliminary measured data:

| | |
|---|---|
| loop A | 106.1% of design flow |
| loop B | 104.8% of design flow |

Therefore, the design flow is conservative.

Response 3.d. The analysis done by Exxon neglects the stored energy in thick metal. Neglecting stored energy would increase the cooldown rate resulting in a more limiting cooldown transient.

Question 4 Provide the following information regarding the analysis of a complete loss of forced coolant flow:

a. Provide an evaluation of various single failures and consider their impact on the consequences of this event.

b. Identify whether most reactive rod was assumed stuck out of the core following reactor scram.

Response 4.a. A single failure evaluation including their impact on the loss of flow transient is not available.

Response 4.b. See the response to 2.c.

Question 5 Provide the following information regarding the analysis of a primary pump rotor seizure event:

a. Identify whether the analysis considered a loss of offsite power, turbine trip, and coastdown of the remaining reactor coolant pump. If these assumptions were not made for the reference analysis provide an evaluation of the locked rotor event addressing those items.

b. Provide a single failure analysis in order to determine the most limiting failure for this

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b. Provide a single failure analysis in order to [illegible] most limiting [illegible]

event. Evaluate the effects of the limiting single failure.

c. Identify that non-safety grade equipment relied upon to mitigate the consequences of the accident. Specifically address the assumed operability of the pressurizer spray, relief and steam dump system.

d. Discuss the long term coolability of the core.

e. Specify whether the most reactive rod was assumed stuck out of the core following reactor trip.

Response 5.a. The analysis did not consider loss of offsite power or the coastdown of the remaining reactor coolant pump. The analysis assumed instantaneous seizure of one RCS pump with reactor trip generated by the low flow. A reactor trip causes a turbine trip. Therefore, the effect of a turbine trip should be included in the analysis.

An analysis of the locked pump rotor transient including the effect of loss of offsite power and pump coastdown is not available.

Response 5.b. A single failure analysis for this transient is not available. However, the analysis performed by Westinghouse neglected the pressure reducing effects of the pressurizer spray and PORV's. The analysis performed by Exxon also neglected the effect of pressurizer spray, PORV's and steam dump.

The sensitivity of the limiting single failure is not available.

Response 5.c. As stated above, pressurizer spray and PORV's were not used in the analysis. The steam dump was not used in the Exxon analysis.

Response 5.d. The DNBR for this transient and Exxon fuel is 1.23. The DNBR stays below 1.3 for about one second. A statistical analysis shows that fewer than one percent of the fuel rods are likely to experience DNB during this event. The fact that a fuel rod experiences DNB does not mean the structural integrity of the fuel rod is lost. Therefore, the long term coolability of the core should not be affected.

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Question 6 Provide the following additional information regarding the analysis of the inadvertent opening of a steam generator relief/safety valve:

a. Discuss the basis of the assumed steam vent flow; i.e., what valve fails open and why this particular case was chosen.

b. Discuss the main feedwater and auxiliary feedwater flow assumptions used in the analysis. These assumptions should conservatively maximize flow to the broken loop steam generator.

c. Discuss the potential for single failures in the auxiliary feedwater control system which may result in runout flow being continuously directed to the broken loop steam generator.

Response 6.a. The analysis done in the FSAR assumed a steam generator safety valve was stuck open. This assumption was also used by Exxon when they evaluated the small steamline break transient. There is no documentation as to why this particular case was chosen although this case is typically analyzed in current FSARs.

Response 6.b. This has been discussed in the response to IE Bulletin 80-04 and the response to Question 3.

Response 6.c. See response to 3.b.

Question 7 Provide the following additional information regarding the analysis of a continuous rod withdrawal at power:

a. Discuss the basis for the selection of the high reactivity insertion rate.

b. Identify whether the scram characteristics include assuming the most reactive rod stuck in its fully withdrawn position.

c. The results of the Cycle 8 analysis for a slow rod withdrawal show a marked increase over the calculated MDNBR for the previous analysis. Identify those input parameter differences between the analyses which contributed to the variation in the calculated results. If the change in

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input parameters is not in the conservative direction then justify the use of less conservative values for the Cycle 8 analysis. The reference (FSAR) analysis of a slow rod withdrawal assumed the most negative Doppler coefficient. Justify the conservatism of the assumed coefficient multiplier or provide an analysis assuming a Doppler coefficient of $-1.5 \times 10^{-5}$ $\Delta K/^{o}F$.

d. Current plants utilizing a Westinghouse NSSS routinely consider a low initial power case corresponding to 10% power. In light of the fact that the MDNBR decreases for the partload cases, provide an evaluation of assuming 10% power initially.

Response 7.a. The FSAR analysis illustrates on Figure 14.1.2-5 the effect of insertion rate on DNBR. This figure illustrates that two reactor trips cover the range of reactivity insertion rates; i.e., high nuclear power trip and overtemperature $\Delta T$ trip. The crossover point for these trips is approximately $5. \times 10^{-5}$ $\Delta K$/sec. Westinghouse then illustrated a rod withdrawal transient at power using an insertion rate of $6.0 \times 10^{-4}$ $\Delta K$/sec. and showed this transient not to be limiting. Exxon also analyzed the at power rod withdrawal transient for an insertion rate of $6.0 \times 10^{-4}$ $\Delta K$/sec. and showed this transient to be non-limiting. Exxon did not redo the sensitivity analysis to reactivity insertion rate because the plant response is mainly dependent on the protective system and not fuel.

Response 7.b. See response to 2.c.

Response 7.c. When the results of the Westinghouse slow rod withdrawal transient are plotted with the results of the Exxon analysis, a difference in the rate of power increase is noted. A comparison of the documented input parameters for this transient reveals Westinghouse used a maximum value for the Doppler Power Coefficient ($-1.5 \times 10^{-4}$ $\Delta K$/% power) where Exxon used a minimum value for the Doppler Temperature Coefficient ($-1.0 \times 10^{-5}$ $\Delta K/ ^{o}F$). The maximum Doppler used by Westinghouse would tend to slow the rate of power increase. The slower transient may just sneak under the rate portion of the overtemperature T trip resulting in a more limiting transient. Exxon is currently reviewing the assumption of maximum Doppler versus minimum Doppler. The results of that study

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will be made available when completed.

Response 7.d. A 10% analysis is not available. Analyses have been provided in the FSAR for rod withdrawal at subcritical conditions (0%power) and at 60%, 80% and 102% power.

It should also be noted that Ginna operates at low power levels such as 10% power very infrequently, only during startup and shutdowns.

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