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GE Nuclear Energy

ABWR

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Subject SECY-91-355 Outstanding Issues 136 and 139

Message Attached are the modifications to GE letter (MFN No. 081-92) dated 4/6/92 resulting from the 5/22/92 conference call between Bob Huang and George Thomas.

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Table 15.0-2

RESULTS SUMMARY OF SYSTEM RESPONSE ANALYSIS TRANSIENT EVENTS

Sub Section I.D.	Figure I.D.	Description	Max. Neutron Flux % NBR	Max Dome Pressure (Kg/Cm^2g)	Max. Vessel Bottom Pressure (Kg/Cm^2g)	Max. Steam Line Pressure (Kg/Cm^2g)	Max Core Average Surface Heat Flux (% of Initial)	Δ in CPR	Freq. Category*	No. of Valves First Blowdown	Duration of Blowdown (seconds)
15.1		Decrease in core coolant temperature									
15.1.1		Loss of Feedwater heating	112.8	73.1	75.9	71.6	112.8	0.07	a	0	0
15.1.2	15.1-2	Runout of one feedwater pump	104.5	73.2	75.8	71.7	101.8	0.06	a	0	0
15.1.2	15.1-3	~~Runout of two feedwater pumps~~ Feedwater Controller Failure - Maximum Demand	139.0	83.3	84.9	82.8	105.9	0.10	a+	10	6
15.1.3	15.1-4	Opening of one Bypass Valve	102.1	73.1	75.6	71.6	100.0	**	a	0	0
15.1.3	15.1-5	Opening of all Control and Bypass Valves	102.0	80.4	81.8	80.1	100.0	**	a+	0	0
15.1.4		Inadvertent opening of One SRV				SEE	TEXT				
15.1.6		Inadvertent RHR Shutdown Cooling				SEE	TEXT				
15.2		Increase in Reactor Pressure									
15.2.1	15.2-1a	Fast Closure of One Turbine Control Valve	~~[illegible]~~ 129.4	~~[illegible]~~ 75.1	~~[illegible]~~ 77.6	~~[illegible]~~ 73.7	~~[illegible]~~ 103.6	0.10	a	0	0
15.2.1	15.2-1b	Slow Closure of One Turbine Control Valve	110.3	74.8	77.3	73.3	103.3	0.09			
15.2.1	15.2-2	Pres. Regulator Downscale Fail	154.8	85.8	87.4	85.1	103.0	*** ~~N/A~~	c	18	6
15.2.2	15.2-3	Generator Load Rejection, Bypass on	148.1	83.2	84.7	82.7	100.2	0.06	a	10	5

* Frequency definition is discussed in Subsection 15.0.4.1
** Not limiting (See Subsection 15.0.4.5.)
*** CPR Criterion does not apply.
a Moderate Frequency
b Infrequent
c Limiting Fault
N/A Not applicable
\+ This event should be classified as a limiting fault. However, criteria for moderate frequent incidents are conservatively applied.

expected for this transient.

15.1.1.3 Core and System Performance

15.1.1.3.1 Input Parameters and Initial Conditions

The transient is simulated by programming a change in feedwater enthalpy corresponding to a 55.6°C loss is feedwater heating. Another case with the ΔT setpoint in FWCS of 16.7°C is also analyzed.

15.1.1.3.2 Results

Because the power increase during this event is relatively slow, it can be treated as a quasi steady-state transient. The 3-D core simulator, PANACEA, has been used to evaluate this event for the equilibrium cycle. The results are summarized in Table 15.1-2 and 15.1-2a.

The MCPR response of this event is small due to the mild thermal power increase with shifting axial shape. The worst Δ CPR response is 0.07.

No scram is initiated in this event. The increased core inlet subcooling aids thermal margins. Nuclear system pressure does not change significantly (less than 0.4 Kg/Cm^2) and consequently, the reactor coolant pressure boundary is not threatened.

15.1.1.4 Barrier Performance

As noted previously the consequences of this event do not result in any temperature or pressure transient in excess of the criteria for which the fuel, pressure vessel or containment are designed; therefore, these barriers maintain their integrity and function as designed.

15.1.1.5 Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to either the secondary containment or to the environment, there are no radiological consequences associated with this event.

15.1.2 Feedwater Controller Failure--Maximum Demand

15.1.2.1 Identification of Causes and Frequency Classification

15.1.2.1.1 Identification of Causes

This event is postulated on the basis of a single failure of a control device, specifically one which can directly cause an increase in coolant inventory by increasing the feedwater flow.

The ABWR feedwater control system uses a triplicated digital control system, instead of a single-channel analog system as used in current BWR designs (BWR 2-6). The digital systems consist of a triplicated fault-tolerant digital controller, the operator control stations and displays. The digital controller contains three parallel processing channels, each containing the microprocessor-based hardware and associated software necessary to perform all the control calculations. The operator interface provides information regarding system status and the required control functions.

Redundant transmitters are provided for key process inputs, and input voting and validation are provided such that faults can be identified and isolated. Each system input is triplicated internally and sent to the three processing channels. (See Figure 15.1-1) The channels will produce the same output during normal operation. Interprocessor communication provides self-diagnostic capability. A two-out-of-three voter compares the processor outputs to generate a validated output to the control actuator. A separate voter is provided for each actuator. A "ringback" feature feeds back the final voter output to the processors. A voter failure will thereby be detected and alarmed. In some cases a protection circuit will lock the actuator into its existing position promptly after the failure is detected.

their integrity and function as designed.

15.1.2.5 Radiological Consequences

While the consequences of this event do not result in any fuel failures, radioactivity is nevertheless discharged to the suppression pool as a result of SRV actuation. However, the mass input, and hence activity input, for this event is much less than those consequences identified in Subsection 15.2.4.5 for Type 2 events. Therefore, the radiological exposures noted in Subsection 15.2.4.5 cover the consequences of this event.

15.1.3 Pressure Regulator Failure--Open

15.1.3.1 Identification of Causes and Frequency Classifications

15.1.3.1.1 Identification of Causes

The ABWR steam bypass and pressure control system (SB&PCS) uses a triplicated digital control system instead of an analog system as used in current BWR designs (BWR 2-6). The SB&PCS controls turbine control valves and turbine bypass valves to maintain reactor pressure. As presented in Section 15.1.2.1.1, no credible single failure in the control system will result in a maximum demand to all actuators for all turbine control valves and bypass valves. A voter or actuator failure may result in an inadvertent opening of one turbine control valve or one turbine bypass valve. In this case, the SB&PCS will sense the pressure change and command the remaining control valves to close, and thereby automatically mitigate the transient and maintain reactor power and pressure.

Because the effect of sudden opening of one bypass valve, which bypasses about 11% of rated steam flow when full opened is more severe than sudden opening of one turbine control valve, which is almost wide open at rated power, it is assumed for purposes of this transient analysis that a single failure causes a single bypass valve to fail open.

As presented in Section 15.1.2.1.1, multiple failures might cause the SB&PCS to erroneously issue a maximum demand to all turbine control valves and bypass valves. Should this occur, all turbine control valves and bypass valves could be fully opened. However, the probability of this event to occur is extremely low (less than 7×10^{-6} failure per reactor year), and hence the event is considered as a limiting fault. However, the criteria of moderate frequent incidents are conservatively applied to this event.

15.1.3.1.2 Frequency Classification

15.1.3.1.2.1 Inadvertent Opening of One Turbine Bypass Valve

This transient disturbance, estimated to occur less than 0.0088 times per year, is conservatively categorized as one of moderate frequency.

15.1.3.1.2.2 Inadvertent Opening of all Turbine Control Valves and Bypass Valves

The frequency of occurance for this event is estimated to be less than once per 10000 years. It should be classified as a limiting fault as specified in Chapter 15 of Regulatory Guide 1.70. Nonetheless, since the consequence of this event has no significant impact on the operating CPR limit, the criteria of moderate frequent incidents are conservatively applied to this event.

15.1.3.2 Sequence of Events and Systems Operation

15.1.3.2.1 Sequence of Events

15.1.3.2.1.1 Inadvertent Opening of One Turbine Bypass Valve

Table 15.1-6 lists the sequence of events for Figure 15.1-4.

15.1.3.2.1.2 Inadvertent Opening of All Turbine Control Valves and Bypass Valves

Table 15.1-7 lists the sequence of events for Figure 15.1-5.

15.1.3.2.1.3 Identification of Operator Actions

15.1.3.2.1.3.1 Inadvertent Opening of One Turbine Bypass Valves

Because no scram occurs during this event, no

Insertion ©

15.2.1.5.2 Pressure Regulator Downscale Failure

During ~~expected~~ this event, less than 0.2% of fuel rods ~~is estimated to~~ get into transition boiling. ~~It is expected that~~ No fuel failures are expected. However, it is conservatively assumed that 0.2% of fuel rods fail in the radiological dose calculation. The results shows that both the whole ~~body~~ body dose and thyroid dose are well within 10% of 10CFR100 requirements. Therefore, the acceptance criteria are met.

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(c) The radiological doses shall be less than 10% of 10CFR100 requirements.

As discussed in Subsection 15.3.1.3.2.2, the PCT during this event is less than 600°C and the time at high temperature is less than 60 seconds. Therefore, no fuel failures need to be assumed for fuel rods with less than or equal to 20 GWD/T exposure.

In general, fuel rods with more than 20 GWD/T exposure are those stay in the core for at least 2 fuel cycles. In the equilibrium cycle, these fuel bundles only account for about 45% of total bundles. The power generated by these bundles are usually 20% less than the hottest bundles. Therefore, less than 0.2% of these rods ~~is expected to~~ gets into transition boiling. Therefore, the requirements of 10% of 10CFR100 are met.

15E.6 TRANSIENT RESPONSES

For every event selected for analysis, three cases were analyzed. The first one shows the ATWS performance with ARI. This case is intended to show the effectiveness of the ARI design. The second case, which uses FMCRD run-in, assuming a total failure of ARI, was performed to show the backup capability of FMCRD run-in. The third case was analyzed to show the in-depth ATWS mitigation capability of the ABWR. In this case, both ARI and FMCRD run in are assumed to fail. Automatic boron injection with a 180 seconds delay, are relied upon to mitigate the transient event.

If the ARI and FMCRD run-in fail at the same time, which has extremely low probability of occurrence, the peak reactor Pressure would still be controlled by the Recirculation runback and relief valves. However, the nuclear shutdown will then rely on the automatic SLCS injection. The boron would reach the core 60 seconds after the initiation. The operation of both SLCS pumps generate a 100 gpm volumetric flow rate of sodium pentaborate. The nuclear shutdown would begin when boron reaches the core.

15E.6.1 Main Steam Isolation Valve Closure

This transient is considered an initiating event caused by either operator action or instrument failure. Scram signal paths that are assumed to fail include valve position, high neutron flux, high vessel pressure, and all manual attempts. A short time after the MSIVs have closed completely, the ATWS high pressure setpoint is reached, which initiates four of the ten recirculation pumps to trip and the rest start to runback. The combined effect of the trip and runback reduces the core flow and increases core voids, thereby reducing power generation which limits pressure increase and steam discharge to the suppression pool. The ATWS high pressure signal causes the actuation of ARI and the electric insertion of the FMCRDs. The insertion of the control rods is successful in bringing the reactor to hot shutdown. Peak values of key parameters are shown in Table 15E.6.1-1 for the ARI case and Table 15E.6.1-2 for the FMCRD run-in case. In the case that control rods fail to insert, the reactor will be brought to hot shutdown by automatic boron injection in about 19.4 minutes from the beginning of the event. The transient behavior of this case is listed in Table 15E.6.1-3. The reactor system response is presented by Figures 15E.6.1-1 to 15E.6.1-4 for ARI activated, Figures 15E.6.1-5 to 15E.6.1-8 for FMCRD run-in case and Figures 15E.6.1-9 to 15E.6.1-12 as SLCS operating, respectively. The normalized axial power shape change during FMCRD run-in are presented in Figure 15E.6.1-13. The increase of the local power density ~~does not violate~~ is not expected to damage the fuel. Therefore the performance criteria are met.

15E.6.2 Loss of AC Power

In this event, all scram signal paths, including valve position, high flux, high pressure, low level, and all manual attempts have been assumed to fail.

The loss of AC power has the following effects:

(1) An immediate load rejection will occur. This will cause the turbine control valves to close.

(2) As a result of the load rejection, four of the ten recirculation pumps will trip.

(3) Due to the loss of power to the condensate pumps, feedwater will be lost.

(4) The reactor will be isolated after loss of main condenser vacuum.

Figures 15E.6.2-1 to 15E.6.2-4 show the transient behavior under ARI activation, Figures 15E.6.2-5 to 15E.6.2-8 for FMCRD run-in and Figures 15E.6.2-9 to 15E.6.2-12 for automatic SLCS, respectively.

The fast closure of the turbine control valves causes a rapid increase of pressure, and the ATWS high pressure setpoint is reached shortly after the control valves have closed. Because the four pumps have already tripped at this time on the load rejection