GE-Hitachi Nuclear Energy Americas LLC

James C. Kinsey
Project Manager, ESBWR Licensing

PO Box 780 M/C A-55
Wilmington, NC 28402-0780
USA

T 910 675 5057
F 910 362 5057
jim.kinsey@ge.com

MFN 07-457

Docket No. 52-010

August 17, 2007

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555-0001

Subject: **Response to Portion of NRC Request for Additional Information Letter No. 100 Related to ESBWR Design Certification Application - Safety Analyses - RAI Number 15.3-32**

Enclosure 1 contains GE-Hitachi Nuclear Energy Americas LLC (GEH) response to the subject NRC RAIs transmitted via Reference 1. Enclosure 2 contains the DCD Markups associated with this response.

If you have any questions or require additional information regarding the information provided here, please contact me.

Sincerely,

Kathy Sedney for

James C. Kinsey
Project Manager, ESBWR Licensing

D068

NRO

Reference:

1. MFN 07-327 - Letter from US Nuclear Regulatory Commission (NRC) to David H. Hinds, *Request for Additional Information Letter No. 100 Related to ESBWR Design Certification Application,* dated May 30, 2007

Enclosures:

1. Response to NRC Request for Additional Information Letter No. 100 Related to ESBWR Design Certification Application - Safety Analyses - RAI Number 15.3-32
2. DCD Markups

cc: AE Cubbage	USNRC (with enclosures)
GB Stramback	GEH /San Jose (with enclosures)
RE Brown	GEH /Wilmington (with enclosures)
eDRF	0000-0072-1576

Enclosure 1

MFN 07-457

Response to NRC Request for
Additional Information Letter No. 100
Related to ESBWR Design Certification Application

Safety Analyses

RAI Number 15.3-32

NRC RAI 15.3-32:

DCD Tier 2, Rev 3, Section 15.2.0 lists several COL applicant assumptions that are applied in the TRACG calculations. A COL information item was provided in DCD Section 15.2.7 for the COL applicant to confirm the applicability of these assumptions. Since the assumptions are also applied to DCD sections 15.3 and 15.5.5, similar COL information items should be added to DCD sections 15.3.17 and 15.5.8 for completeness.

GEH Response:

GEH has revised DCD, Tier 2, Subsection 15.2.7 to eliminate the COL applicant assumptions. The assumptions have been moved to Table 15.2-1 and/or event description subsections. System requirements are added to the appropriate system description. Note that some of these assumptions are already located in Table 15.2-1, event subsections or system descriptions. It will not be necessary for the COL Applicants to confirm the applicability of these assumptions.

Since these are no longer considered COL Applicant assumptions to be confirmed by the COL applicant, they do not need to be identified in Subsections 15.3.17 and 15.5.8.

DCD Impact:

DCD, Tier 2, Table 5.4-1, Subsections 10.4.7.1.2, 15.2.0, 15.2.1, 15.2.2, 15.2.7, 15.3.15.1 and Table 15.2-1 will be revised as noted on the attached markups.

Enclosure 2

MFN 07-457

DCD Markups

Table 5.4-1

Component and Subsystem Design Controls

Component/Subsystem	Control(s)
The main steamline flow restrictor	Limits the coolant blowdown rate from the reactor vessel in the event a main steamline break occurs outside the containment to a (choke) flow rate equal to or less than 200% of rated steam flow at 7.07 MPa gauge (1025 psig) upstream pressure. The throat diameter is ≤ 355 mm (14 in.).
The ratio of the main steamline flow restrictor venturi throat diameter to steamline inside diameter:	Approximately 0.5, which results in a maximum pressure differential (unrecovered pressure) of about 0.10 MPa (15 psi) at 100% of rated flow. This design limits the steam flow in a severed line to less than 200% rated flow, yet it results in negligible increase in steam moisture content during normal operation.
The main steamline flow restrictor duty:	Exposed to steam of about 0.10% moisture flowing at velocities of 53 m/sec (173.9 ft/sec) (steam piping ID) to 212 m/sec (695.5 ft/sec) (steam restrictor throat).
MSIV characteristics:	Nominally 700 mm (28 in.) diameter spring-loaded, pneumatic, piston-operated valves that fail closed on loss of pneumatic pressure to the valve actuator
MSIV rated steam flow:	2.19×10^6 kg/hr (4.82×10^6 lbm/hr).
MSIV pneumatic cylinder actuator can open the MSIV poppet with a maximum differential pressure of:	1.38 MPa (200 psi) across the isolation valve in a direction that tends to hold the MSIV closed.
The MSIV poppet travels approximately 90% of the valve stem travel to close the main steam port area;	Approximately the last 10% of the valve stem travel closes the pilot valve.
MSIV fast closing speed:	3.0 – 5.0 seconds when operating gas is admitted to the upper piston compartment and the lower piston compartment is vented to the atmosphere.
MSIV slow closing speed:	45 – 60 seconds, by admitting operating gas to both the upper and lower piston compartments.
MSIV steam design envelope:	Designed to accommodate saturated steam at plant operating conditions with moisture content of approximately 0.5%.
MSIV design life	60 years service at operating conditions.

Table 5.4-1

Component and Subsystem Design Controls

Component/Subsystem	Control(s)
MSIV corrosion allowance:	60 years service.
MSIVs are designed to remain closed under long-term post-accident environmental conditions:	≥ 100 days.
MSIV flow established by choked flow at the venturi flow restrictor installed in each main steamline reactor vessel nozzle:	200% of rated flow After the valve is approximately 75% closed, steam flow is further reduced as a function of the valve area versus travel characteristic
MSIVs longest design closing time:	5.0 seconds
MSIV shortest design closing time:	3.0 seconds
MSIV combined leakage	Combined leakage through the MSIVs for all four MSLs is less than or equal to 66.1 liters per minute (17.5 gallons per minute) at standard temperature of 20°C (68°F) and pressure (one atmosphere absolute pressure) with the differential pressure across the MSIV equal to or greater than 0.269 MpaD (39 psid).
ICS station blackout (i.e., unavailability of all AC power) capability:	≥72 hours
IC sizing:	Sized to remove post-reactor isolation decay heat with three out of four ICs operating and to reduce reactor pressure and temperature to safe shutdown conditions, in 36 hours, with occasional venting of radiolytically generated noncondensable gases to the suppression pool.
ICS Performance Requirements:	Heat removal capacity of the ICS (with 3 of 4 IC trains in service) is at least 101.25 MWt when reactor is above rated operating pressure.
Condensate return valve stroke-open time	**≥ 7.5 seconds and ≤ 30 seconds with a logic delay time not to exceed 1 second after the opening setpoint is reached.** ~~≤ 30 seconds with a logic delay time not to exceed 1 second after the opening setpoint is reached.~~

Table 5.4-1

Component and Subsystem Design Controls

Component/Subsystem	Control(s)
IC design capacity:	33.75 MWt each IC unit and is made of two identical modules.
ICS loop seal:	Assures that condensate valves do not have 285 °C (545 °F) water on one side of the disk and subcooled water [as low as 10 °C (50 °F)] on the other side during normal plant operation, thus affecting leakage during system standby conditions
The design temperature and pressure of the Class 1 portions of the main steam and feedwater lines (same as that of the RPV)	8.62 MPa gauge (1250 psig) and 302°C (576°F)
Number of MSLs	4
Nominal diameter of each MSL	700 mm (28 in)
The main steamline drain subsystem isolation valves are remote-manually operated from the main control room and are closed when reactor power exceeds:	40%.
Number of Feedwater lines and Branch lines to RPV	2 6
Diameter of each Feedwater line and Branch lines to RPV	550 mm (22 in) 300 mm (12 in)
Combined main steam line volume	Combined volume from RPV to the turbine stop valves and steam bypass valves is greater than or equal to 135 m^3 (4768 ft^3).

10.4.7.1.2 Non-Safety Power Generation Design Bases

- The C&FS is designed to provide a dependable FW supply to the reactor at the required flow rate, pressure, and temperature under all anticipated steady-state and transient conditions.
- The C&FS is designed to supply at least 135% of the rated FW flow during anticipated operational occurrences.
- The C&FS is designed to permit long-term full power operation with one FW pump and/or one condensate pump out-of-service.
- The C&FS is designed to permit long-term operation with one LP heater string out of service at the maximum load permitted by the turbine manufacturer. This value is set by steam flow limitation on the affected LP turbine.
- The C&FS is designed to heat up the reactor FW to approximately 215.6°C (420°F) during full power operation.
- The C&FS is designed to cool the auxiliary condensers and support other auxiliary condensate loads.
- The C&FS is designed so that no single operator error or equipment failure shall cause more than a 55.6°C (100°F) decrease in final feedwater temperature.
- The C&FS, in conjunction with the Condensate Purification System, is designed to maintain water quality suitable for all plant conditions, including power operation, startup, shutdown and extended outages. The Condensate Purification System is discussed in Subsection 10.4.6.
- The C&FS is designed to allow for Final Feedwater Temperature Reduction (FFWTR) operation.
- During plant startups the C&FS is designed to pump preheated FW to the Reactor Pressure Vessel (RPV) for the purpose of RPV initial heating if sufficient core decay heat is not available.
- **The C&FS in conjunction with the feedwater control system provides inventory equivalent to 240 s of rated feedwater flow after main steam isolation valve (MSIV) isolation.**
- **The C&FS in combination with feedwater control system limits the maximum feedwater flow for a single pump to less than 75% of rated flow following a single active component failure or operator error in C&FS or feedwater control system.**
- All C&FS functions needed to support safe power operation use at least dual redundant controllers and triply redundant signals; a single control system failure does not cause an inadvertent pump trip or valve operation.

15.2 ANALYSIS OF ANTICIPATED OPERATIONAL OCCURRENCES

Each of the anticipated operational occurrences (AOOs) addressed in the Section 15.1, "Nuclear Safety Operations Analysis" (NSOA), is evaluated in the following subsections. Appendix 15A provides a determination of event frequency to categorize AOOs as defined in 10 CFR 50 Appendix A. Tables 15.2-1, 15.2-2 and 15.2-3 provide the important input parameters and initial conditions used/assumed in the AOO analyses.

In the analysis of AOOs and Infrequent Events in Section 15.3 Nonsafety-Related systems or components are considered to be operational in the following situations:

- When assumption of a Nonsafety-Related system results in a more limiting event;
- When a detectable and nonconsequential random, independent failure must occur in order to disable the system; and
- When Nonsafety-Related systems or components are used as backup protection (i.e. not the primary success path, included to illustrate the expected plant response to the event).

15.2.0 Assumptions

Assumptions are listed in the event discussions and in Table 15.2-1.

~~The following COL Applicant assumptions are applied in the TRACG calculations in Sections 15.2, 15.3 and 15.5.5:~~

- ~~The assumed initial suppression pool temperature is 43.3°C (110°F) and the scram set point 48.9°C (120°F) in the inadvertent safety relief valve (SRV) opening event analysis.~~
- ~~A bounding isolation condenser (IC) injection valve stroke time of 7.5 second is assumed in the inadvertent IC injection analysis. Injection of all ICs (4 ICs) is assumed. The temperature of the condensate in the IC system that is initially injected is assumed to be 10°C (50°F).~~
- ~~The turbine steam bypass system provides 50% of maximum bypass flow in the event of a single failure.~~
- ~~The feedwater controller failure analysis assumes a L8 feedwater runback signal. The L8 signal is backed up by a Safety-Related feedwater trip at L9.~~
- ~~The feedwater control system provides 240 s of rated feedwater flow (2434 kg/s) after main steam isolation valve (MSIV) isolation.~~
- ~~The Loss of feedwater heating setpoint is assumed to be 16.67°C (30°F) measured in the feedwater nozzle to maximize the SCRRI/SRI actuation delay.~~
- ~~The SCRRI/SRI control rod grouping was defined to be able to reduce the core power and reduce limit the DCPR/ICPR after a Loss of Feedwater Heating. It is divided in five subgroups. Four of them, SRI, with scattered insertion time (a separation of 10 s between~~

~~each subgroup) and another SCRRI group with a total insertion time of 110 s and activated simultaneously with the first SRI group..~~

- ~~The MSIV closure curve is (where time 0.0 s is the start of valve movement):~~
  ~~0.0 s 100% open.~~
  ~~0.6 s 100% open,~~
  ~~1.7 s 1% open &~~
  ~~3.0 s 0.0% open.~~
- ~~The automatic depressurization system (ADS) low water level setpoint analytical limit is 11.5 m above vessel 0.~~
- ~~The maximum feedwater pump runout for a single pump is 75% of rated flow.~~
- ~~For the transient representative of the loss of off site power with failure to transfer to internal power sources, it is assumed that initially a load rejection occurs, feedwater pumps trip and condensate pumps trip simultaneously.~~
- ~~The stuck open safety relief valve transient has been analyzed with 4 ICs available and with a bounding capacity, to observe the maximum possible depressurization rate.~~

15.2.1 Decrease In Core Coolant Temperature

15.2.1.1 Loss Of Feedwater Heating

15.2.1.1.1 Identification of Causes

A feedwater (FW) heater can be lost in at least two ways:

- Steam extraction line to heater is closed; or
- FW is bypassed around heater.

The first case produces a gradual cooling of the FW. In the second case, the FW bypasses the heater and no heating of the FW occurs. In either case, the reactor vessel receives colder FW. The maximum number of FW heaters that can be tripped or bypassed by a single event represents the most severe event for analysis considerations.

The ESBWR is designed such that no single operator error or equipment failure shall cause a loss of more than 55.6°C (100°F) FW heating. The reference steam and power conversion system shown in Section 10.1 meets this requirement.

The loss of FW heating causes an increase in core inlet subcooling. This increases core power due to the negative void reactivity coefficient. However, the power increase is slow.

The Feedwater Control System (FWCS) logic is provided in Subsection 7.7.3, and includes logic to mitigate the effects of a loss of FW heating capability. The system is constantly monitoring the actual FW temperature and comparing it with a reference temperature. When a loss of FW heating is detected [i.e., when the difference between the actual and reference temperatures exceeds a ΔT setpoint], the FWCS sends an alarm to the operator and sends a signal to the Rod Control and Information System (RC&IS) to initiate the Selected Control Rods Run-In and Select Rods Insertion (SCRRI/SRI) function to automatically reduce the reactor power and avoid a scram. This prevents the reactor from violating any thermal limits.

Control blade insertion is conservatively assumed to start only when the temperature difference setpoint is reached in the FW nozzle. The SCRRI/SRI is able to suppress totally the neutron power increase and the MCPR reduction is small.

The SCRRI/SRI control rod grouping was defined to be able to reduce the core power and limit the DCPR/ICPR after a Loss of Feedwater Heating. The SCRRI/SRI control rod pattern depends on the fuel cycle exposure and initiation event. The rod pattern analyzed in this event is divided into five control rod groups. Four SRI groups, with scattered insertion time (a separation of 10 seconds between each group) and a SCRRI group with a total insertion time of 110 seconds, initiated simultaneously with the first SRI group.

Events may exist where the SCRRI/SRI is not activated; because the loss of feedwater temperature is less than 16.67°C (30°F). These events have a ΔCPR/ICPR similar to the event studied here, however none of them will become limiting.

15.2.1.1.2 Sequence of Events and Systems Operation

Sequence of Events

Table 15.2-4 lists the sequence of events for Figure 15.2-1

Because no scram occurs during this event, no immediate operator action is required. As soon as possible, the operator should verify that no operating limits are being exceeded. Also, the operator should determine the cause of failure prior to returning the system to normal.

Systems Operation

In establishing the expected sequence of events and simulating the plant performance, it was assumed that normal functioning occurred in the plant instrumentation and controls, plant protection and reactor protection systems. The only assumed failure is for a single HCU, which actuates two control rods, avoiding the normal insertion of two rods.

15.2.1.1.3 Core and System Performance

Input Parameters and Initial Conditions

The event is simulated by programming a change in FW enthalpy corresponding to the assumed loss in FW heating, shown in Table 15.2-1.

Results

Because the power increase during this event is controlled by the SCRRI/SRI insertion, the reduction of the MCPR is very small and is turned around when the SCRRI/SRI function takes effect. The results are summarized in Table 15.2-5.

No scram is assumed in this analysis. The increased core inlet subcooling aids thermal margins. Nuclear system pressure does not significantly change and consequently, the RCPB is not threatened.

This event is potentially limiting with respect to OLMCPR. The COL Applicant will provide reanalysis of this event for the specific initial and (COL Holder) reload core designs.

15.2.2.1 Generator Load Rejection With Turbine Bypass

15.2.2.2.1 Identification of Causes

Fast closure of the TCVs is initiated whenever electrical grid disturbances occur which result in significant loss of electrical load on the generator. The TCVs are required to close as rapidly as possible to prevent excessive over-speed of the turbine-generator (TG) rotor. Closure of the TCVs causes a sudden reduction in steam flow. To prevent an increase in system pressure, sufficient bypass capacity is provided to pass steam flow diverted from the turbine.

After sensing a significant loss of electrical load on the generator, the TCVs are commanded to close rapidly. At the same time, the turbine bypass valves are signaled to open in the "fast" opening mode by the SB&PC system, which uses a triplicated digital controller. As presented in Subsection 15.2.4.3, no single failure can cause all turbine bypass valves to fail to open on demand.

Assuming no single failure the plant has the full steam bypass capability available, the Reactor Protection System (RPS) will verify that the bypass valves are open. The fast closure of the TCVs produces a pressure increase that is negligible, because all the steam flow is bypassed through the turbine bypass valves. The reactor power decreases when the SCRRI/SRI actuates.

The SCRRI/SRI control rod grouping is defined to reduce the core power and limit the DCPR/ICPR after a generator load rejection with turbine bypass. The SCRRI/SRI rod pattern depends on the fuel cycle exposure and initiation event. The rod pattern analyzed in this event is divided into five control rod groups. Four SRI groups, with scattered insertion time (a separation of 10 seconds between each subgroup) and a SCRRI group with a total insertion time of 110 seconds, activated simultaneously with the first SRI group.

15.2.2.4 Turbine Trip With Turbine Bypass

15.2.2.4.1 Identification of Causes

A variety of turbine or nuclear system malfunctions can initiate a turbine trip. Some examples are high velocity separator drain tank high levels, large vibrations, operator lockout, loss of control fluid pressure, low condenser vacuum and reactor high water level.

After the main turbine is tripped, turbine bypass valves are opened in their fast opening mode by the SB&PC system. **The reactor power decreases when the SCRRI/SRI actuates.**

The SCRRI/SRI control rod grouping was defined to be able to reduce the core power and limit the DCPR/ICPR after a turbine trip with turbine bypass. The SCRRI/SRI rod pattern used in the turbine trip with turbine bypass is the same as the one used in the generator load rejection with turbine bypass discussed in Subsection 15.2.2.2.1.

are produced (with 30 s delay). In the case that HP_CRD is unavailable for level control, the system response is similar to the station blackout event described in Subsection 15.5.5, which demonstrates that level can be maintained above the top of active fuel with the ICS as the primary success path. In either case, the number of rods in boiling transition remains within the acceptance criterion for AOOs because increases in the heat flux are not experienced. Consequently, this event does not need to be reanalyzed for specific core configurations.

15.2.5.3.3 Barrier Performance

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.

15.2.5.3.4 Radiological Consequences

Because this event does not result in any fuel failures or any release of primary coolant to the environment, there is no radiological consequence associated with this event.

15.2.6 AOO Analysis Summary

The results of the system response analyses are presented in Table 15.2-5. Based on these results, the limiting AOO events have been identified. The potentially limiting events that establish the CPR operating limit are identified in Subsection 15.2.7.

For the core loading in Figure 4.3-1, the resulting initial core MCPR operating limit is 1.30, using the methodologies listed in sections 4.4.3.1.3 and 4.4.2.1.3. The operating limit based on the initial core design will be provided by the COL Applicant.

15.2.7 COL Information

The following potentially limiting AOOs will be evaluated for the (COL Applicant) initial and (COL Holder) reload core designs:

- Loss of Feedwater Heating
- Closure of One Turbine Control Valve
- Generator Load Rejection with a Single Failure in the Turbine Bypass System
- Inadvertent Isolation Condenser Initiation

~~COL Applicant: Confirm the applicability of assumptions listed in Subsection 15.2.0.~~

15.2.8 References

15.2-1 GE Nuclear Energy, "TRACG Application for Anticipated Operational Occurrences Transient Analysis" NEDE-32906P-A, Revision 1, April 2003.

Table 15.2-1

Input Parameters And Initial Conditions Used In AOO and Infrequent Event Analyses

Parameter	Value
Thermal Power Level, MWt	4500
Core Flow, kg/s	10130
Steam Flow, kg/s (Mlbm/hr) Analysis Value	 2435 (19.3)
Feedwater Flow Rate, kg/s (Mlbm/hr) Analysis Value Total Flow For All Pumps Runout, % of rated at 1065 psig (At rated dome pressure, 1025 psig). **The condensate and feedwater system in conjunction with the feedwater control system provide inventory equivalent to 240 s of rated feedwater flow after MSIV isolation.** **The condensate and feedwater system in combination with feedwater control system limit the maximum feedwater flow for a single pump to 75% of rated flow following a single active component failure or operator error.**	 2429 (19.3) 155 (164)
Feedwater Temperature, °C (°F) Rated FW Heating Temperature Loss **Loss of FW Heating Setpoint (SCRRI/SRI Initiation)**	 216 (420) 55.6 (100) **16.67 (30)**
Vessel Dome Pressure, MPaG (psig)	7.07 (1025)
Vessel Core Pressure, MPaG (psig)	7.17 (1040)
Turbine Bypass Capacity, % of rated	110
Total Delay Time from TSV or TCV to the start of BPV Main Disc Motion	0.02
Total Delay Time from TSV or TCV to 80% of Total Capacity	0.17

Table 15.2-1

Input Parameters And Initial Conditions Used In AOO and Infrequent Event Analyses

Parameter	Value
TCV Closure Times, s Fast Closure Analysis Value (Bounding) Assumed Slow Closure Analysis Value	 0.08 2.5
TSV Closure Times, s	0.100
% of Rated Steam Flow That Can Pass Through 3 Turbine Control Valves	85 (Partial Arc)
Core Coolant Inlet Enthalpy, kJ/kg Rated Value Analysis Value	 1195 1193
Turbine Inlet Pressure, MPaG (psig)	6.57
Fuel Lattice	N
Core Leakage Flow, %	9.4
MCPR Operating Limit	1.30
Control Rod Drive Position versus Time	Table 15.2-2 & 3
Nuclear Characteristics Used in TRACG Simulations	Middle of Cycle and End of Cycle
Safety Relief Valve (SRV) capacity, %NBR (103% accumulation)(1) At design pressure, MPaG (psig) Quantity Installed	89.5 8.618 (1250) 18
Safety Function Delay, s	0.2
Safety Function Opening Time, s	1.5
Analysis values for SRV setpoints Low Setpoint, MPaG (psig) High Setpoint, MPaG (psig)	 8.618 (1250) 8.756 (1270)

Table 15.2-1

Input Parameters And Initial Conditions Used In AOO and Infrequent Event Analyses

Parameter	Value
Closure Scram Position of 2 or More MSIVs, % open Maximum delay time	85 0.06
MSIV Minimum Closure Time, s **MSIV Maximum Closure Time, s**	**3.0** **5.0**
MSIV Closure Profile used to bound minimum closure time, s **100% open,** **100% open,** **1% open.** **0% open**	 **0.0** **0.6** **1.7** **3.0**
High Flux Trip, % NBR, Sensor Time Constant	125.0 0.03
TSV Closure Scram Position of 2 or more TSV, % open Trip Time delay, s	85 0.06
TCV Fast Closure Scram Trip	0.08
High Pressure Scram, MPaG (psig). Máximum scram delay	7.619 (1105) 0.7
High Suppression Pool Temperature Scram trip, °C (°F), Maximum Delay Time	48.9(120) 1.05
High Suppression Pool Temperature FAPCS actuation, °C (°F)	43.3 (110)
Vessel level Trips (above bottom vessel) Level 9—(L9), m (in) Level 8—(L8), m (in) Level 4—(L4), m (in) Level 3—(L3), m (in) Level 2—(L2), m (in) Level 1—(L1), m (in) Level 0.5 – (L0.5) m (in)	 22.39 (881.5) 21.89 (861.8) 20.60 (811.2) 19.78 (778.7) 16.05 (631.9) 11.50 (452.8) 8.45 (332.7)

Table 15.2-1

Input Parameters And Initial Conditions Used In AOO and Infrequent Event Analyses

Parameter	Value
APRM Simulated Thermal Power Trip Scram, % NBR Time Constant, s	 115 7
Total Steamline Volume, m^3 (ft^3)	135 (4767)
CRD Hydraulic System minimum capacity, m^3/hr (gpm), Capacity in kg/s for 990 kg/m^3 density	235.1 (1035) 64.6 kg/s
Maximum time delay from Initiating Signal (Pump 1 & 2), s If offsite power is not available	10 & 25 s 145 s
Isolation Condensers Max Initial Temperature, °C (°F) Minimum Initial Temperature, °C (°F) Time To injection valve full open (Max) (2) Heat Removal Capacity for 4ICs, MW (% Rated Power) Isolation Condensers volume, 4 Units, from steam box to discharge at vessel m^3 (ft^3)	 40 (104) 10 (50) 31 (1) 135 (3%) 56.1 (1981)

established Technical Specifications; therefore, this event, at the worst, would only result in a small increase in the yearly integrated exposure level.

15.3.15 Stuck Open Safety Relief Valve

15.3.15.1 Identification of Causes

Cause of a stuck open safety relief valve is attributed to the malfunction of the valve after it has opened either inadvertently or in response to a high pressure signal. It is, therefore, simply postulated that a failure occurs and the event is analyzed accordingly. Detailed discussion of the valve design is provided in Chapter 5.

In this analysis, after any event which produces the scramming of the reactor, it is assumed that a SRV remains open without any possibility of closure. The operations of the ICs produce a depressurization, with the HP_CRD operating to recover the level after the Scram. **The event is analyzed with 4 ICs available and with bounding capacity, to observe the maximum possible depressurization rate.** Finally the reactor reaches pressure near atmospheric.

15.3.15.2 Sequence of Events and Systems Operation

Sequence of Events

Table 15.3-12 lists the sequence of events for this event. If auxiliary power is not available, the sequence of events is similar to the Main Steam Line Break sequence given in Table 6.3-8.

Identification of Operator Actions

The plant operator must re-close the valve as soon as possible and check that the reactor and TG output return to normal. If the valve cannot be closed and the reactor has scrammed because of some other reason (if the SRVs are in the open condition the plant must necessarily have scrammed previously, except for the Inadvertent SRV opening analyzed previously), manual activation of the IC and other systems must be initialized to reduce the amount of steam reaching the suppression pool.

Systems Operation

This event assumes normal functioning of the plant instrumentation and controls, specifically the operation of the pressure regulator and water level control systems.

15.3.15.3 Core and System Performance

The opening of one SRV allows steam to be discharged to the suppression pool. The sudden increase in the rate of steam flow leaving the reactor vessel causes a depressurization transient, with the vessel pressure slowly decreasing until reaching atmospheric pressure. The SRV steam discharge also results in a slight heating of the suppression pool.

Thermal margins decrease only slightly through the transient and no fuel damage is predicted for this event.