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MFN 07-305 Supplement 1

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U.S. Nuclear Regulatory Commission
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Subject: **Response to Portion of NRC Request for Additional Information Letter No. 96 Related to ESBWR Design Certification Application - Technical Specifications - RAI Number 16.2-119 S01**

Enclosures 1, 2, and 3 contain the subject supplemental RAI response resulting from a July 3, 2007 e-mail from the NRC. The GE Hitachi Nuclear Energy (GEH) response to the original RAI was provided in the Reference 1 letter.

Verified DCD changes associated with this RAI response are identified in the enclosed DCD markups by enclosing the text within a black box. The marked-up pages may contain unverified changes in addition to the verified changes resulting from this RAI response. Other changes shown in the markup(s) may not be fully developed and approved for inclusion in DCD Revision 5.

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

Sincerely,

James C. Kinsey

James C. Kinsey
Vice President, ESBWR Licensing

D068
NRO

Reference:

1. MFN 07-305, Letter from James C. Kinsey to U.S. Nuclear Regulatory Commission, *Response to Portion of NRC Request for Additional Information Letter No. 96 Related to ESBWR Design Certification Application - Technical Specifications - RAI Number 16.2-119*, Jun 4, 2007

Enclosures:

1. MFN 07-305, Supplement 1 - Response to Portion of NRC Request for Additional Information Letter No. 96 Related to ESBWR Design Certification Application - Technical Specifications - RAI Number 16.2-119 S01
2. MFN 07-305, Supplement 1 – DCD Tier 2 Section 5.2 Markups for RAI Number 16.2-119 S01
3. MFN 07-305, Supplement 1 – DCD Tier 2 Section 15.5.1 Markups for RAI Number 16.2-119 S01

| | | |
|---|---|---|
| cc: | AE Cubbage | USNRC (with enclosures) |
| | DH Hinds | GEH (with enclosures) |
| | RE Brown | GEH (with enclosures) |
| | eDRFs | 71-8057/2 |

**Enclosure 1**

**MFN 07-305, Supplement 1**

**Response to Portion of NRC Request for**

**Additional Information Letter No. 96**

**Related to ESBWR Design Certification Application**

**- Technical Specifications -**

**RAI Numbers 16.2-119 S01**

**NRC RAI 16.2-119 S01**

(Received by e-mail from Chandu Patel - 07/03/07)

*Comment on response to RAI 16.2-119 (MFN 07- 305, June 4, 2007):*

*In the response to RAI 21.6-91 (MFN 07-256), GE states, "Changes to DCD Tier 2, Figure 5.2-4 will be made in response to this RAI. Figure 5.2-4 will be updated based on the result of a TRACG analysis that uses the following input files: MSIVF_EOC_NOFW.INP and SCRAM_PRESS_8GROUPS.TDT."*

*Since GE proposes to include only 1 SRV in the ESBWR Technical Specifications for overpressure protection, GE should verify that the input decks referenced above used to generate the Figures 5.2-4 in DCD Tier 2 only take credit for 1 SRV.*

**GEH Response**

The TRACG analysis that credits the capacity of only 1 SRV for over pressure protection has been performed with the input file MSIVF_EOC_NOFW.INP and kinetics file SCRAM_PRESS_8GROUPS.TDT. The input file was modified to simulate the SRV capacity change from approximately 3 SRVs to that of 1 SRV. A replacement for Figure 5.2-4 has been generated for inclusion in the DCD, but has been changed to Figure 15.5-1 as the analysis for the MSIV closure with flux scram event is now described in section 15.5.1.1 of the DCD.

The analysis resulted in no change in the maximum reactor vessel pressure and demonstrates that 1 SRV is sufficient to mitigate the reactor vessel pressure response.

**DCD Impact**

DCD Tier 2 Section 5.2 will be revised as noted in the Enclosure 2 markup.

DCD Tier 2 Section 15.5.1 will be replaced with the Enclosure 3 markup.

**Enclosure 2**

**MFN 07-305, Supplement 1**

**DCD Tier 2 Section 5.2 Markups for RAI Number 16.2-119 S01**

| |
|---|
| Verified DCD changes associated with this RAI response are identified in the enclosed DCD markups by enclosing the text within a black box. The marked-up pages may contain unverified changes in addition to the verified changes resulting from this RAI response. Other changes shown in the markup(s) may not be fully developed and approved for inclusion in DCD Revision 5. |

principal components of the reactor coolant system against environmental effects are presented in Section 3.11.

**Safety Relief Valves**

The design pressure and temperature of the valve inlet is 9.48 MPa gauge (1375 psig) at 307°C (585°F).

The valves have been designed to achieve the maximum practical number of actuations consistent with state-of-the-art technology.

#### 5.2.2.2.3 Mounting of Safety Relief Valves

The SRVs and SVsare installed vertically on the main steam piping. The design criteria and analysis methods for considering SRV discharge loads are contained in Section 3.9.

#### 5.2.2.2.4 Applicable Codes and Classification

The vessel overpressure protection system is designed to satisfy the requirements of Section III of the ASME Code. The general requirements for protection against overpressure of Section III of the Code recognize that reactor vessel overpressure protection is one function of the reactor protective systems and allows the integration of pressure-relief devices with the protective systems of the nuclear reactor. Hence, credit is taken for the scram protective system as a complementary pressure protection device. The NRC has also adopted the ASME Code as part of their requirements in the Code of Federal Regulations (10 CFR 50.55a).

#### 5.2.2.2.5 Material Specifications

Typical material specifications for pressure-retaining components of SRVs and SVs are listed in Table 5.2-4. All NBS relief and safety valve pressure-retaining materials comply with the requirements of the ASME Code, Section III, Article NB-2000.

### *5.2.2.3 Safety Evaluation*

<u>Results of the overpressure protection evaluation are provided in Subsection 15.5.1. The system is designed to satisfy the requirements of Section III of the ASME Code.</u>

~~**5.2.2.3.1 Method of Analysis**~~

~~The method of analysis is approved by the United States Nuclear Regulatory Commission (NRC) or developed using criteria approved by the NRC.~~

~~It is recognized that the protection of vessels in a nuclear power plant is dependent upon many protective systems to relieve or terminate pressure transients. Installation of pressure relieving devices may not independently provide complete protection. The safety valve sizing evaluation gives credit for operation of the scram protective system which may be tripped by either one of three sources: (1) a direct valve position signal, (2) a flux signal, or (3) a high vessel pressure signal. The direct scram trip signal is derived from position switches mounted on the MSIVs. The position switches are actuated when the valves are closing and following 15% travel of full stroke. The flux signal is derived from the Neutron Monitoring System and is actuated at 125% of rated nuclear boiler power. The pressure signal is derived from pressure transmitters piped to the vessel steam space.~~

~~Full account is taken of the pressure drop on both the inlet and discharge sides of the valves. All combination SRVs discharge into the suppression pool through a discharge pipe from each valve which is designed to achieve sonic flow conditions through the valve, thus providing flow independence to discharge piping losses.~~

#### ~~5.2.2.3.2 System Design~~

~~A parametric study was conducted to determine the required steam flow capacity of the SRVs based on the following assumptions.~~

**~~Operating Conditions~~**

- ~~Operating power = 4590 MWt (102% of nuclear boiler rated power);~~
- ~~Absolute vessel dome pressure ≤ 7.17 MPa (1040 psia); and~~
- ~~Steam flow = 2433 kg/s (19.31 Mlbm/hr).~~

~~These rated power conditions are the most severe because maximum stored energy exists at these conditions. At lower power conditions, the transients would be less severe.~~

**~~Pressurization Events~~**

~~The overpressure protection system is capable of accommodating the most severe pressurization event. The ESBWR pressurization is mild relative to previous other BWR designs because of the large steam volume in the chimney and vessel head, which mitigates the pressurization. The scram and initial pressurization drops the water level below the feedwater sparger; when the feedwater system performs as expected, the spray of subcooled water condenses steam in the vessel steam space and immediately terminates the pressurization. For purposes of overpressure protection analyses, the feedwater system is assumed to trip at the initiation of the event. The analyses of increase in reactor pressure events are evaluated Subsection 15.2.2, where the performance of the ICS is credited to prevent a lift of the SRVs or SVs. In order to evaluate the overpressure protection capability of the SRVs, no credit is taken in this evaluation for the ICS.~~

~~No credit is taken for the first scram signal that would occur (e.g., valve position for MSIV isolation). This is in accordance with NUREG-0800, Subsection 5.2.2, which requires that the reactor scram be initiated by the second safety-related signal from the Reactor Protection System (neutron flux for MSIV isolation, turbine trip and load rejection).~~

~~The evaluation of event behavior, based on the core loading shown in Figure 4.3-1, demonstrates that MSIV closure, with scram occurring on high flux, (i.e., MSIV Closure With Flux Scram special event, MSIVF) is the most severe pressurization event, the result for this event is similar to the Turbine Trip With Total Turbine Bypass Failure event evaluated in Subsection 15.3.6. Other fuel designs and core loading patterns, including loading patterns similar to Figure 4.3-1, do not affect the conclusions of this evaluation. Table 5.2-3 lists the systems that could initiate during a MSIV Closure With Flux Scram special event.~~

**~~Evaluation Method~~**

~~The evaluation method for overpressure protection events is the TRACG computer code as described in Reference 15.2-1.~~

**~~SRV & Pressurization Event Analysis Specification~~**

- ~~Simulated valve groups:~~
  - ~~Spring-action safety mode (2 groups)~~
- ~~Opening pressure setpoint (maximum safety limit):~~
  - ~~Spring-action safety mode (Table 5.2-2)~~
- ~~Reclosure pressure setpoint (% of opening setpoint) both modes:~~
  - ~~Maximum safety limit (used in analysis) 96~~
  - ~~Minimum operational limit 90~~

~~The opening and reclosure setpoints are assumed at a conservatively high level above the nominal setpoints. This is to account for initial setpoint errors and any instrument setpoint drift that might occur during operation. Conservative SRV response characteristics are also assumed; therefore, the analysis conservatively bounds all SRV operating conditions.~~

**~~SRV Capacities~~**

~~SRV capacities are based on establishing an adequate margin from the peak vessel bottom pressure to the vessel code limit in response to pressurization events.~~

~~The analysis method assumes that whenever the system pressure increases to the valve mechanical lift set pressure of a group of valves, these valves begin opening and reach full open at 103% of set pressure. Only one SRV is required to open to prevent exceeding the ASME limit in the ASME overpressure protection event. Ten SRVs and eight SVs are included in the ESBWR design. The additional SRVs and SVs are needed for the ATWS event.~~

~~To demonstrate the margin for AOO pressurization events, an analysis model assuming only the capacity of a single SRV is evaluated below.~~

#### ~~5.2.2.3.3 Evaluation of Results~~

**~~Total SRV Capacity~~**

~~The adequacy of one SRV's capacity is demonstrated by analyzing the pressure rise from a MSIVF special event. Results of this analysis are given in Figure 5.2-4a through Figure 5.2-4f. The peak vessel bottom pressure calculated is below the acceptance limit of 9.481 MPa gauge (1375 psig). Figure 5.2-4a through Figure 5.2-4f show the MSIVF special event. The pressurization is not dynamic and does not significantly overshoot the relief valve setpoint. Vessel pressurization ceases to increase following a single relief valve opening when the steam discharge capacity exceeds the stored energy of the vessel plus rate of decay heat addition Figure 5.2-4d shows that peak vessel pressure is only a function of the valve setpoint. This is because the higher steam volume-to-power ratio of the ESBWR causes the pressure rate prior to scram to be much lower than operating BWRs. After a scram, the pressure rates due to core decay energy release are correspondingly lower.~~

**~~Statistical Evaluation of MSIV Special Event~~**

~~The statistical analysis performed to calculate the upper bound for the Maximum Vessel Pressure during MSIVF, perturbing the physical correlations and operating conditions (as explained in~~

~~Section 4 of Reference 5.2-9), results in a maximum vessel pressure below 9.481 MPa (1375 psia) with a 95/95 confidence. This value bounds the overpressure analysis peak pressure for one SRV opening at the setpoint specified for the analysis and is bounded by the calculation with no credit of feedwater flow shown in Figure 5.2-4a.~~

~~**Pressure Drop in Inlet and Discharge**~~

~~Pressure drop in the piping from the reactor vessel to the valves is taken into account in calculating the maximum vessel pressures. Pressure drop in the discharge piping to the suppression pool is limited by proper discharge line sizing to prevent backpressure on each SRV from exceeding 40% of the valve inlet pressure, thus assuring choked flow in the valve orifice and no reduction of valve capacity due to the discharge piping, for the 10 valves piped to the suppression pool or the 8 valves which discharge to the drywell.~~

~~**5.2.2.3.4 System Reliability**~~

~~The system is designed to satisfy the requirements of Section III of the ASME Code. Evaluation of events requiring a response by the NBS overpressure protection are provided in Subsections 15.2.4, 15.3.4, and 15.3.6. The special events evaluation of the ATWS scenario that also credits overpressure protection component responses is found Subsection 15.5.4. The potential failure events of inadvertent ICS initiation, inadvertent relief valve opening or stuck open relief valve are evaluated in Subsections 15.2.4.1, 15.3.13, and 15.3.15, respectively. The redundant divisions of the ICS combined with the number and location diversity of NBS pressure relief valves makes the likelihood of total NBS overpressure protection failure an extremely low probability.~~

### *5.2.2.4 Testing and Inspection Requirements*

The inspection and testing of applicable SRVs and SVs utilizes a quality assurance program, which complies with Appendix B of 10 CFR 50.

The SRVs and SVs are tested at a suitable test facility in accordance with quality control procedures to detect defects and to prove operability prior to installation. The following tests are conducted:

- Hydrostatic test at specified test conditions (ASME Code requirement based on design pressure and temperature).
- Thermally stabilize the valve to perform quantitative steam leakage testing at 1.03 MPaG (150 psig) below the nameplate value with an acceptance criterion not to exceed 0.45 kg/hr (1 lbm/hr) leakage.
- Full flow SRV test for set pressures and blowdown where the valve is pressurized with saturated steam, with the pressure rising to the valve set pressure (during production testing the SRV is adjusted to open at the nameplate set pressure ± 1%).
- Response time test where each valve is tested to demonstrate acceptable response time based on system requirements. The valves are installed as received from the factory. The valve manufacturer certifies that design and performance requirements have been met. This includes capacity and blowdown requirements. The setpoints are adjusted, verified, and indicated on the valves by the vendor. Specified manual and automatic

## 5.2.7 References

5.2-1 D. A. Hale, "The Effect of BWR Startup Environments on Crack Growth in Structural Alloys," Trans. of ASME, Vol. 108, January 1986.

5.2-2 F. P. Ford and M. J. Povich, "The Effect of Oxygen/Temperature Combinations on the Stress Corrosion Susceptibility of Sensitized T-304 Stainless Steel in High Purity Water," Paper 94 presented at Corrosion 79, Atlanta, GA, March 1979.

5.2-3 Electric Power Research Institute, "BWR Water Chemistry Guidelines - 2004 Revision," EPRI TR-1008192, October 2004.

5.2-4 B. M. Gordon, "The Effect of Chloride and Oxygen on the Stress Corrosion Cracking of Stainless Steels: Review of Literature," Material Performance, NACE, Vol. 19, No. 4, April 1980.

5.2-5 U.S.N.R.C, W. J. Shack, et. al., "Environmentally Assisted Cracking in Light Water Reactors: Annual Report, October 1983 – September 1984," NUREG/CR-4287, ANL-85-33, June 1985.

5.2-6 K. S. Brown and G. M. Gordon, "Effects of BWR Coolant Chemistry on the Propensity of IGSCC Initiation and Growth in Creviced Reactor Internal Components," paper presented at the Third International Symposium of Environmental Degradation of Materials in Nuclear Power Systems, ANS-NACE-TMS/AIME, Traverse City, MI, September 1987.

5.2-7 B. M. Gordon et al, "EAC Resistance of BWR Materials in HWC," Proceedings of Second International Symposium Environmental Degradation of Materials in Nuclear Power Systems, ANS, LaGrange Park, IL, 1986.

5.2-8 B. M. Gordon et al, "Hydrogen Water Chemistry for BWRs – Material Behavior," EPRI NP-5080, Palo Alto, CA, March 1987.

5.2-9 ~~GE Nuclear Energy, "TRACG Application for ESBWR," NEDC-33083P-A, Class III (Proprietary) March 2005 and NEDO-33083-A, Class I (Non-proprietary), October 2005.~~ (Deleted)

Table 5.2-3

~~Systems That May Initiate or Trip During Overpressure Event~~(Deleted)

| ~~Systems~~ | ~~Initiating/Trip Signal~~ |
|---|---|
| ~~Reactor Protection~~ | ~~Reactor shutdown on high flux~~ |
| ~~ICS~~ | ~~Initiated on high reactor pressure or reactor isolation or low reactor water level when mode switch is in "run"~~ |
| ~~CRD~~ | ~~ON when reactor water level is at L2~~ |
| ~~RWCU/SDC~~ | ~~OFF when reactor water level is at L2~~ |

Figure 5.2-4a. (Deleted)
~~MSIV Closure With Flux Scram (Deleted) (Obsolete Figure to be replaced Later)~~

Figure 5.2-4b. (Deleted)~~MSIV Closure With Flux Scram (Deleted)~~
~~(Obsolete Figure to be replaced Later)~~

Figure 5.2-4c. (Deleted)~~MSIV Closure With Flux Scram (Deleted)~~
~~(Obsolete Figure to be replaced Later)~~

Figure 5.2-4d. (Deleted)~~MSIV Closure With Flux Scram (Deleted)~~
~~(Obsolete Figure to be replaced Later)~~

Figure 5.2-4e. (Deleted)~~MSIV Closure With Flux Scram (Deleted)~~
~~(Obsolete Figure to be replaced Later)~~

Figure 5.2-4f. (Deleted)~~MSIV Closure With Flux Scram (Deleted)~~
~~(Obsolete Figure to be replaced Later)~~

## Enclosure 3

## MFN 07-305, Supplement 1

## DCD Tier 2 Section 15.5.1 Markups for RAI Number 16.2-119 S01

Verified DCD changes associated with this RAI response are identified in the enclosed DCD markups by enclosing the text within a black box. The marked-up pages may contain unverified changes in addition to the verified changes resulting from this RAI response. Other changes shown in the markup(s) may not be fully developed and approved for inclusion in DCD Revision 5.

## Table 15.0-2

## ESBWR Abnormal Event Classifications

| Abnormal Event | Event Classification | Relevant SRP(s) |
|---|---|---|
| Control Rod Withdrawal Error During Refueling | Infrequent Event | 15.4.1 |
| Control Rod Withdrawal Error During Startup | Infrequent Event | 15.4.1 |
| Control Rod Withdrawal Error During Power Operation with ATLM Failure | Infrequent Event | 15.4.2 |
| Fuel Assembly Loading Error, Mislocated Bundle | Infrequent Event | 15.4.7 |
| Fuel Assembly Loading Error, Misoriented Bundle | Infrequent Event | 15.4.7 |
| Inadvertent SDC Function Operation | Infrequent Event | ~~15.4.9~~15.1.1-15.1.4 |
| Inadvertent Opening of a Safety Relief Valve | Infrequent Event | 15.6.1 |
| Inadvertent Opening of a Depressurization Valve | Infrequent Event | 15.6.1, 15.6.5 |
| Stuck Open Safety Relief Valve | Infrequent Event | 15.6.1 |
| Liquid-Containing Tank Failure | Infrequent Event | 15.7.3 |
| Fuel Handling Accident | Accident | 15.7.4 |
| LOCA Inside Containment | Accident | 15.6.5 & 5a |
| Main Steamline Break Outside Containment | Accident | 15.6.4 |
| Control Rod Drop Accident | See Subsection 15.4.6 | |
| Feedwater Line Break Outside Containment | Accident | 15.3.5 |
| Failure of Small Line Carrying Primary Coolant Outside Containment | Accident | 15.6.2 |
| RWCU/SDC System Line Failure Outside Containment | Accident | 15.6.4, 15.6.5 |
| Spent Fuel Cask Drop Accident | Accident | 15.7.5 |
| MSIV Closure With Flux Scram (Overpressure Protection) | ~~Special Event~~*** | 5.2.2 |
| Shutdown Without Control Rods (i.e., SLC system shutdown capability) | Special Event | 9.3.5 |
| Shutdown from Outside Main Control Room | Special Event | 7.5 |
| Anticipated Transients Without Scram | Special Event | 15.8 |
| Station Blackout | Special Event | 8.2 (and RG 1.155) |
| Safe Shutdown Fire | Special Event | 9.5.1 |

## Table 15.0-2

## ESBWR Abnormal Event Classifications

| Abnormal Event | Event Classification | Relevant SRP(s) |
|---|---|---|
| Waste Gas System Leak or Failure | Special Event | 11.3 |

* An AOO in combination with an additional SACF or SOE, as discussed in SRP 15.1 and SRP 15.2.

** Both covered by the Loss of Non-Emergency AC Power to Station Auxiliaries event.

*** Event evaluated to demonstrate prevention of reactor coolant pressure boundary ASME Code Service Level B pressure limit(s) – Special Event.

# 15.5 SPECIAL EVENT EVALUATIONS

## 15.5.1 Overpressure Protection

~~Results of the overpressure protection evaluation are provided in Subsection 5.2.2.~~

### *15.5.1.1 Method of Analysis*

The method of analysis is approved by the United States Nuclear Regulatory Commission (NRC) or developed using criteria approved by the NRC. The acceptance criteria for overpressure protection are provided in Subsection 15.0.3.4.1. This analysis is required to demonstrate prevention of reactor coolant pressure boundary ASME Code Service Level B pressure limit(s).

It is recognized that the protection of vessels in a nuclear power plant is dependent upon many protective systems to relieve or terminate pressure transients. Installation of pressure-relieving devices may not independently provide complete protection. The safety valve sizing evaluation gives credit for operation of the scram protective system which may be tripped by either one of three sources: (1) a direct valve position signal, (2) a flux signal, or (3) a high vessel pressure signal. The direct scram trip signal is derived from position switches mounted on the MSIVs. The pressure signal is derived from pressure transmitters piped to the vessel steam space.

Full account is taken of the pressure drop on both the inlet and discharge sides of the valves. All combination SRVs discharge into the suppression pool through a discharge pipe from each valve which is designed to achieve sonic flow conditions through the valve, thus providing flow independence to discharge piping losses.

### *15.5.1.2 System Design*

A parametric study was conducted to determine the required steam flow capacity of the SRVs based on the following assumptions

**Operating Conditions**

- Operating power = 4590 MWt (102 % of nuclear boiler rated power);
- Absolute vessel dome pressure ≤ 7.17 MPa (1040 psia); and
- Steam flow = 2433 kg/s (19.31 Mlbm/hr).

These rated power conditions are the most severe because maximum stored energy exists at these conditions. At lower power conditions, the transients would be less severe.

**Pressurization Events**

The overpressure protection system is capable of accommodating the most severe pressurization event. The ESBWR pressurization is mild relative to previous other BWR designs because of the large steam volume in the chimney and vessel head, which mitigates the pressurization. The scram and initial pressurization drops the water level below the feedwater sparger; when the feedwater system performs as expected, the spray of subcooled water condenses steam in the vessel steam space and immediately terminates the pressurization. For purposes of overpressure protection analyses, the feedwater system is assumed to trip at the initiation of the event. The analyses of increase-in-reactor-pressure events are evaluated Subsection 15.2.2, where the performance of the ICS is credited to prevent a lift of the SRVs or SVs. In order to evaluate the overpressure protection capability of the SRVs, no credit is taken in this evaluation for the ICS.

No credit is taken for the first scram signal that would occur (e.g., valve position for MSIV isolation). This is in accordance with NUREG-0800, Subsection 5.2.2, which requires that the reactor scram be initiated by the second safety-related signal from the Reactor Protection System (neutron flux for MSIV isolation, turbine trip and load rejection).

The evaluation of event behavior, based on the equilibrium core in Reference 15.5-6, demonstrates that MSIV closure, with scram occurring on high flux, (i.e., MSIV Closure With Flux Scram special event, MSIVF) is the most severe pressurization AOO event, the result for this event is similar to the Turbine Trip With Total Turbine Bypass Failure event evaluated in Subsection 15.3.6. Other fuel designs and core loading patterns, including loading patterns similar to Reference 15.5-6, do not affect the conclusions of this evaluation. Table 15.5-1a lists the systems that could initiate during a MSIV Closure With Flux Scram special event

The results of the overpressure protection analysis for the initial core loading documented in Reference 15.5-3 are provided in Reference 15.5-4. Overpressure protection analysis bounding operation in the feedwater temperature operating domain are documented in Reference 15.5-5.

**Evaluation Method**

The evaluation method for overpressure protection events is the TRACG computer code as described in Reference 15.5-7.

**SRV & Pressurization Event Analysis Specification**

- Simulated valve group:
  - Spring-action safety mode – 1 valve credited in analysis
- Opening pressure setpoint (maximum safety limit):
  - Spring-action safety mode – Low Setpoint, Table 15.2-1
- Reclosure pressure setpoint (% of opening setpoint) both modes:
  - Maximum safety limit (used in analysis) 96
  - Minimum operational limit 90

The opening and reclosure setpoints are assumed at a conservatively high level above the nominal setpoints. This is to account for initial setpoint errors and any instrument setpoint drift that might occur during operation. Conservative SRV response characteristics (Table 15.2-1) are also assumed; therefore, the analysis conservatively bounds all SRV operating conditions.

The RPS high flux scram settings assumed are provided in Table 15.2-1.

The MSIV design closure time range and the worst case (bounding) closure time assumed in this analysis are provided in Table 15.2-1.

### *15.5.1.3 Evaluation of Results*

**Total SRV Capacity**

SRV capacities are based on establishing an adequate margin from the peak vessel bottom pressure to the vessel code limit in response to pressurization events.

The analysis method assumes that whenever the system pressure increases to the valve mechanical lift set pressure of a valve, the valve begins opening and reaches full open at 103% of set pressure. Only one SRV is required to open to prevent exceeding the ASME limit in the ASME overpressure protection event. Ten SRVs and eight SVs are included in the ESBWR design. The additional SRVs and SVs are needed for the ATWS event.

The adequacy of one SRV's capacity is demonstrated by analyzing the pressure rise from a MSIVF special event. Results of this analysis are given in Figure 15.5-11a through Figure 15.5-11f. Table 15.5-1b lists the sequence of events for Figure 15.5-11. The calculated peak vessel bottom pressure is less than the acceptance limit of 9.481 MPa gauge (1375 psig). The pressurization is not dynamic and does not significantly overshoot the relief valve setpoint. Vessel pressurization ceases to increase following a single relief valve opening when the steam discharge capacity exceeds the stored energy of the vessel plus rate of decay heat. The peak vessel pressure is only a function of the valve setpoint. This is because the higher steam volume-to-power ratio of the ESBWR causes the pressure rate prior to scram to be much lower than operating BWRs. After a scram, the pressure rates due to core decay energy release are correspondingly lower.

**Statistical Evaluation of MSIV Special Event**

As explained in Section 8.4 of Reference 15.5-7, the assumption of feedwater trip and no IC function has the effect of reducing TRACG model uncertainty, and the SRV setpoint (plant parameter) uncertainty dominates. This results in the analysis done herein being a bounding analysis with no further statistical analysis required.

**Pressure Drop in Inlet and Discharge**

Pressure drop in the piping from the reactor vessel to the valves is taken into account in calculating the maximum vessel pressures. Pressure drop in the discharge piping to the suppression pool is limited by proper discharge line sizing to prevent backpressure on each SRV from exceeding 40% of the valve inlet pressure, thus assuring choked flow in the valve orifice and no reduction of valve capacity due to the discharge piping, for the 10 valves piped to the suppression pool or the 8 valves which discharge to the drywell.

### *15.5.1.4 System Reliability*

The system is designed to satisfy the requirements of Section III of the ASME Code. Evaluations of events requiring a response by the NBS overpressure protection are provided in Subsections 15.2.2, 15.2.5, 15.3.2, 15.3.4, 15.3.5 and 15.3.6. The special events evaluation of the ATWS scenario that also credits overpressure protection component responses is found Subsection 15.5.4. The potential failure events of inadvertent ICS initiation, inadvertent relief valve opening or stuck open relief valve are evaluated in Subsections 15.2.4.1, 15.3.13, and 15.3.15, respectively. The redundant divisions of the ICS combined with the number and location diversity of NBS pressure relief valves makes the likelihood of total NBS overpressure protection failure an extremely low probability.

**Table 15.5-1a**

**Systems That May Initiate or Trip During Overpressure Event**

| **Systems** | **Initiating/Trip Signal** |
|---|---|
| Reactor Protection | Reactor shutdown on high flux |
| IC (not credited in analysis) | Initiated on high reactor pressure or reactor isolation or low reactor water level when mode switch is in "run" |
| HP_CRD (not credited in analysis) | ON when reactor water level is at L2 |
| RWCU/SDC | OFF when reactor water level is at L2 |

## Table 15.5-1b

## Sequence of Events for Closure of all MSIV with Flux Trip

| Time (s) | Event * |
|---|---|
| 0.0 | Closure of all MSIVs. |
| 0.78 | MSIVs reach 85% open. |
| 1.6 | High flux trip scram initiated. |
| 2.83 | L3 is reached |
| 2.88 | MSIVs are closed |
| 9.00 | L2 is reached |
| 37.81 | SRV setpoint reached |
| 37.81 | Reactor pressure reaches its peak value. |
| 37.95 | SRV full Open |

* See Figure 15.5-11.

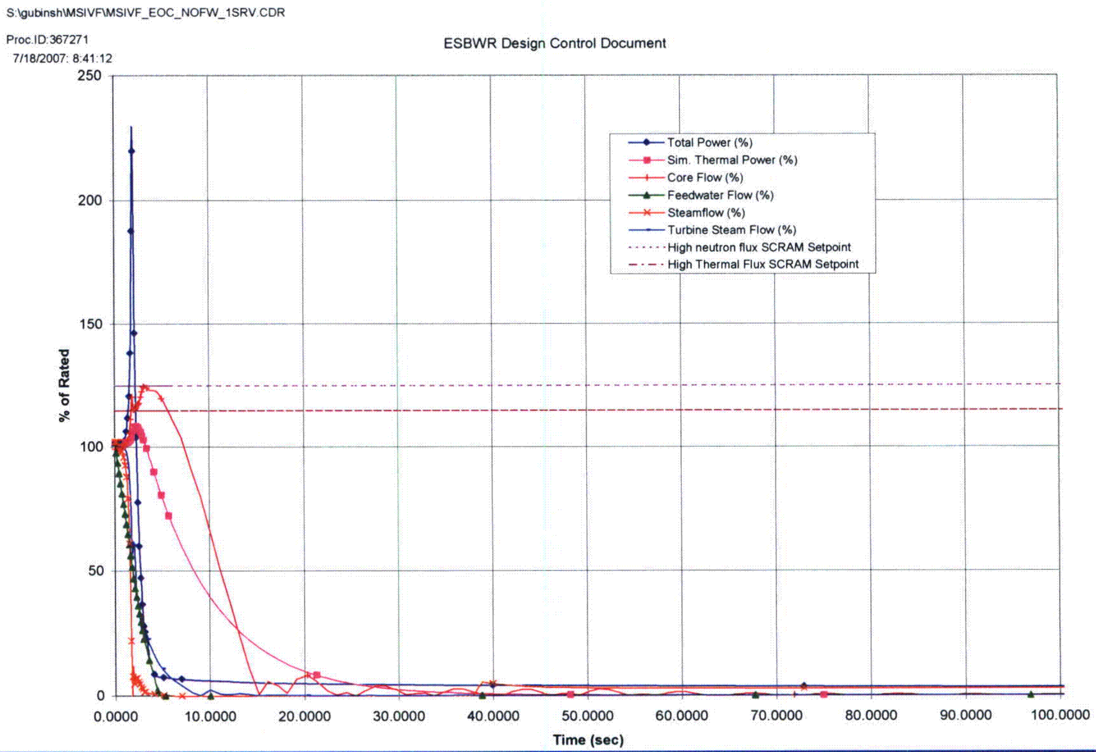


Figure 15.5-11a. MSIV Closure With Flux Scram

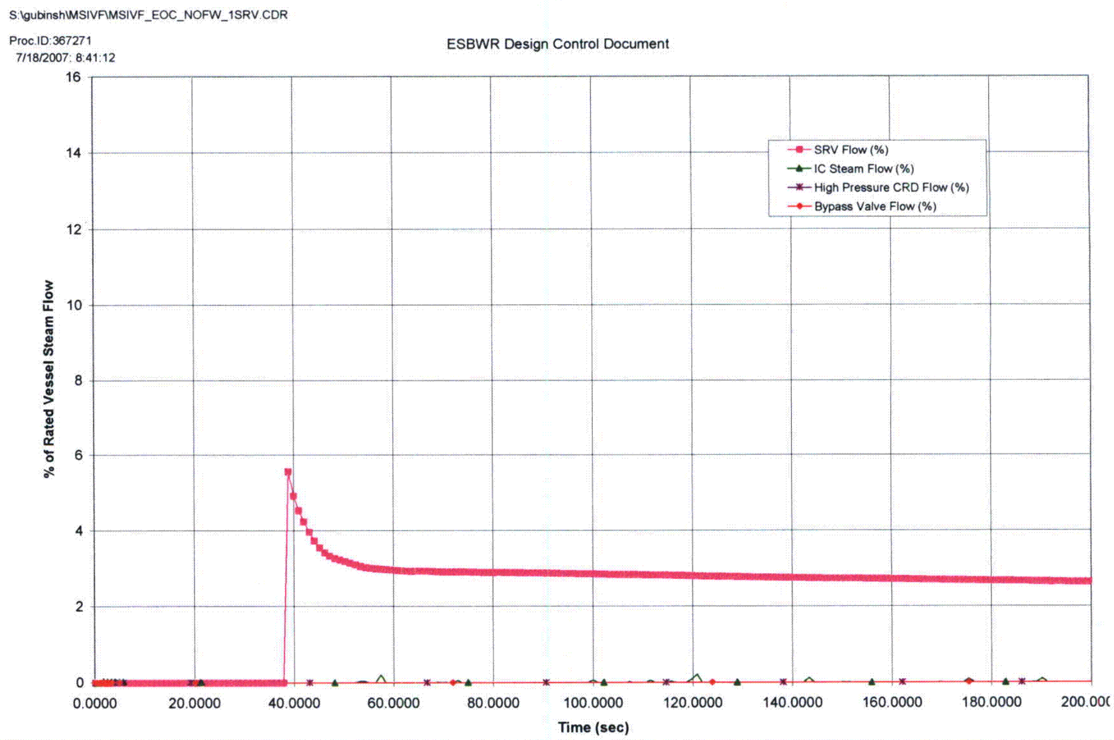


Figure 15.5-11b. MSIV Closure With Flux Scram

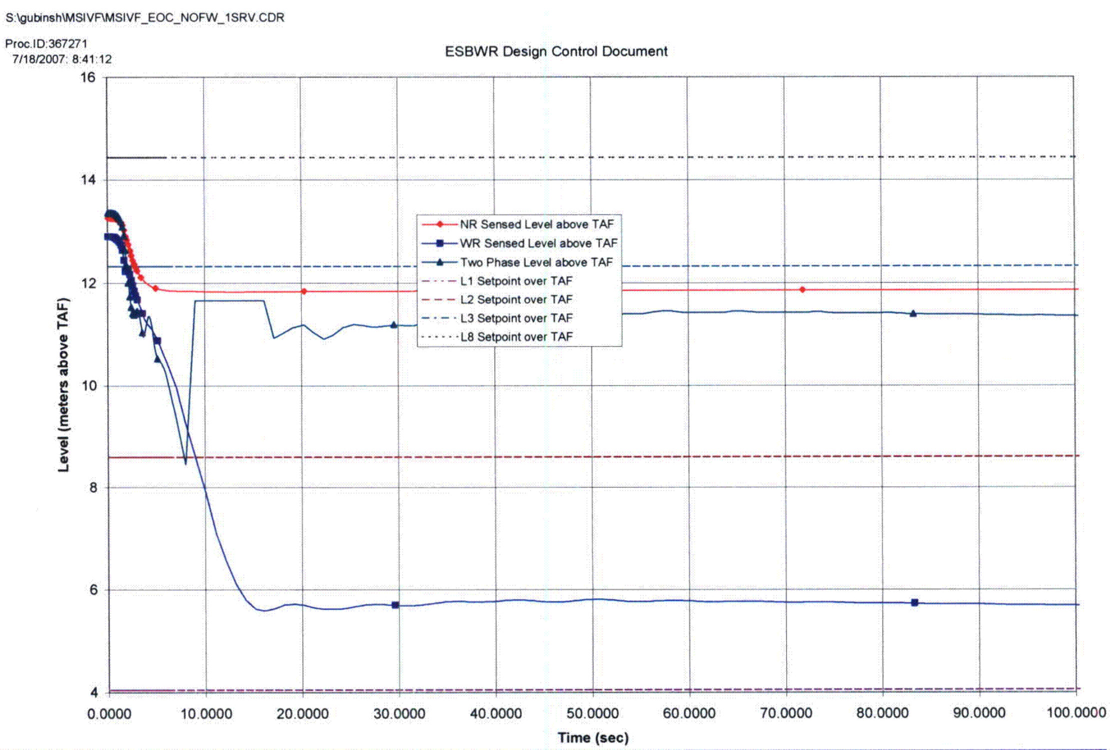


**Figure 15.5-11c. MSIV Closure With Flux Scram**

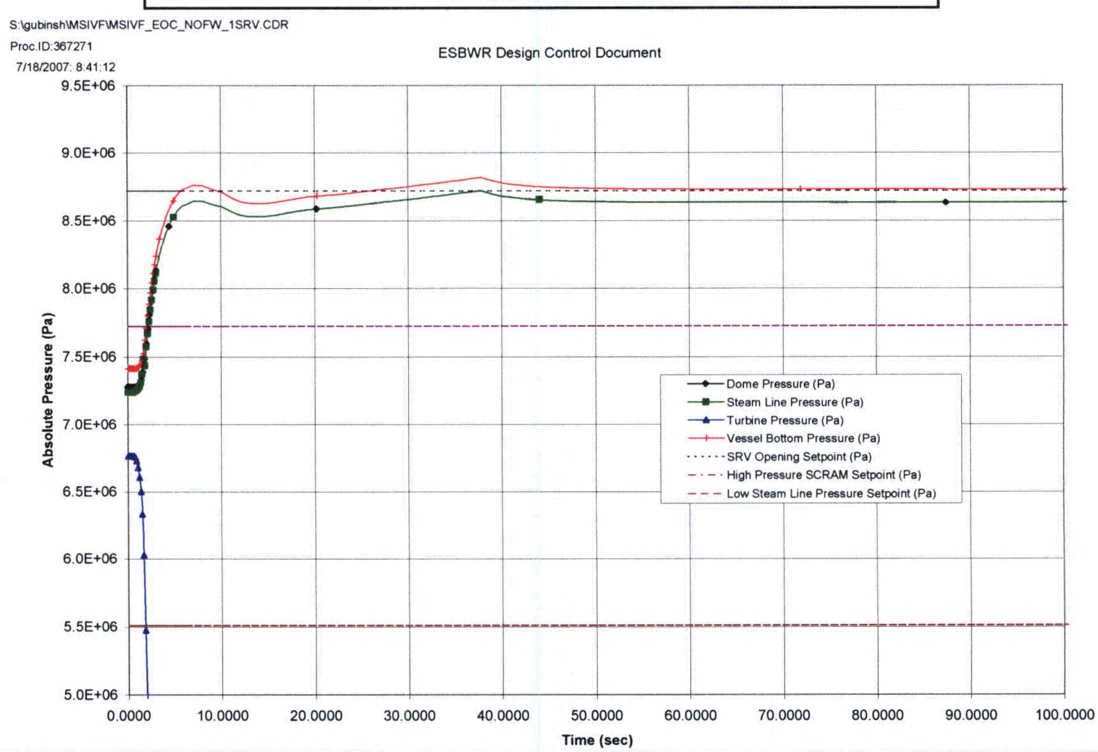


**Figure 15.5-11d. MSIV Closure With Flux Scram**

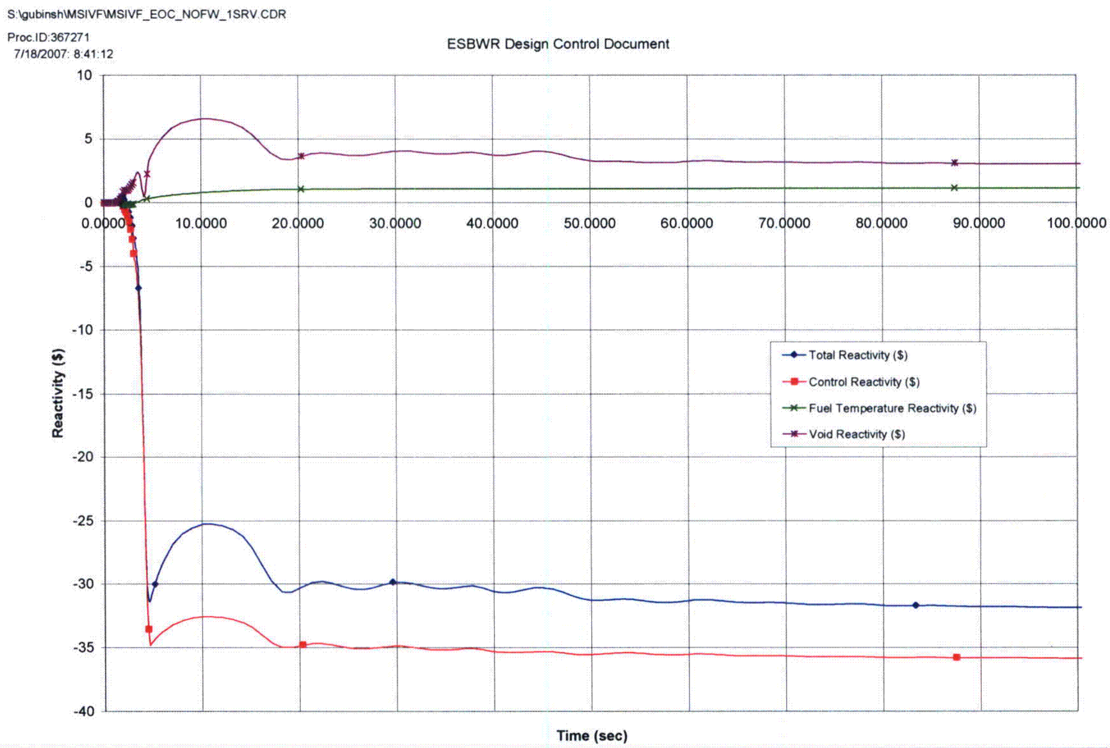


**Figure 15.5-11e. MSIV Closure With Flux Scram**

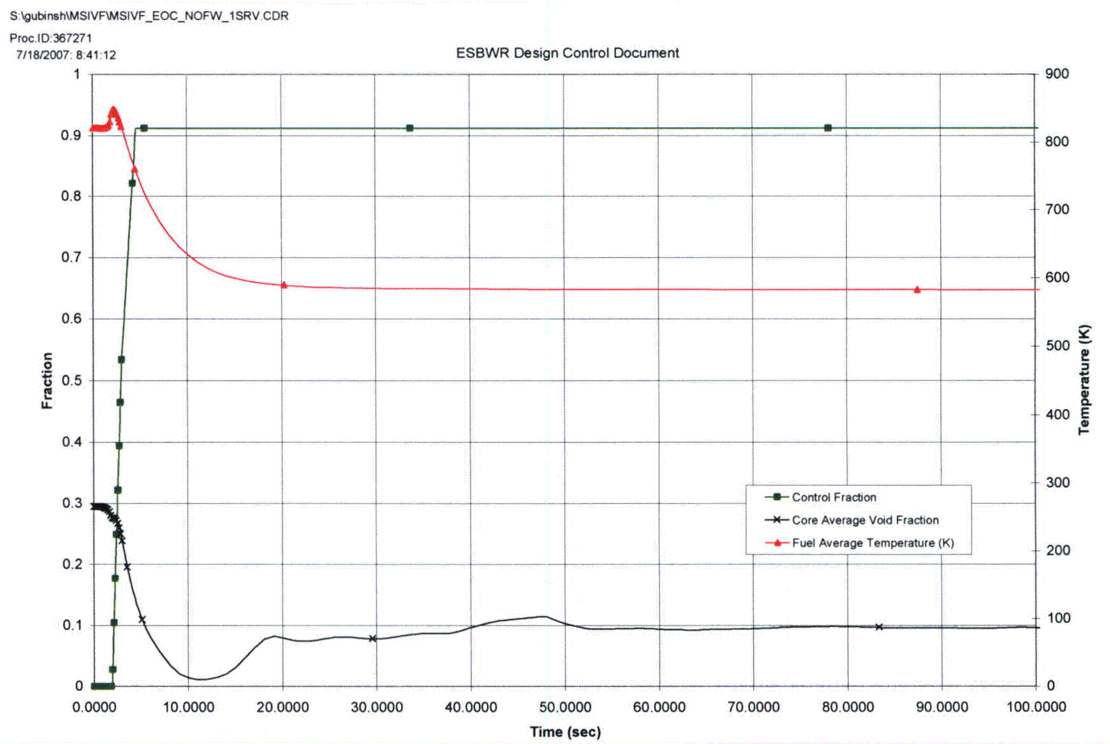


**Figure 15.5-11f. MSIV Closure With Flux Scram**