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CR-3 EPU – Feedwater Line Break Analysis Sensitivity Studies

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Nomenclature

Acronym	Definition
β_{eff}	Delayed Neutron Fraction
BOC	Beginning of Cycle
CR-3	Crystal River Unit 3
DTC	Doppler Temperature Coefficient
EFIC	Emergency Feedwater Initiation and Control
EFW	Emergency Feedwater
EOC	End of Cycle
EPU	Extended Power Uprate
FOGG	Feed Only Good Generator
FWLB	Feedwater Line Break
HZP	Hot Zero Power
ICS	Integrated Control System
LAR	Licensing Amendment Request
LOCA	Loss of Coolant Accident
LOFW	Loss of Feedwater
LOOP	Loss of Offsite Power
MFW	Main Feedwater
MSSV	Main Steam Safety Valves
MTC	Moderator Temperature Coefficient
NRC	Nuclear Regulatory Commission
PORV	Pilot Operated Relief Valve
PSV	Pressurizer Safety Valve
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RPS	Reactor Protection System
SDM	Shutdown Margin
SG	Steam Generator
SRP	Standard Review Plan
T_{ave}	Average RCS Temperature
TSV	Turbine Stop Valves

1.0 INTRODUCTION

Section 2.8.5.2.4 of the Crystal River Unit 3 (CR-3) extended power uprate (EPU) licensing amendment request (LAR) describes the feedwater line break (FWLB) evaluations that were performed to support the power uprate. The evaluations that were performed are consistent with approved methodology [2] and the current licensing basis of the plant, except that the licensing basis is being revised for EPU to use the Standard Review Plan (SRP) Section 15.2.8 reactor coolant system (RCS) pressure limit of 120% of the design pressure, or 3000 psig. Since the CR-3 licensing basis is being revised to use the SRP pressure limit, additional sensitivity studies are performed to address requirements of SRP 15.2.8. The sensitivity studies documented in this report are for the following parameters:

- Initial RCS Pressure
- Initial RCS Flow
- Initial Power Level
- Time in Cycle
- Initial RCS Average Temperature (Tave)
- Initial steam generator (SG) Pressure
- Initial SG Level
- Initial Pressurizer Level
- Loss of offsite power (LOOP)
- Break Size
- Break Location

2.0 ANALYTICAL METHODOLOGY

The thermal-hydraulic analysis of the FWLB at the CR-3 EPU power level is performed using the RELAP5/MOD2-B&W computer program (Reference [1]). The code simulates RCS and secondary system operation. The reactor core model is based on a point kinetics solution with reactivity feedback for control rod assembly insertion, fuel temperature changes, moderator temperature changes, and changes in boron concentration. The RCS model provides for heat transfer from the core, transport of the coolant to the steam generators, and heat transfer to the steam generators. The secondary model includes a detailed depiction of the main steam system, including steam relief to the atmosphere through the main steam safety valves (MSSVs) and simulation of the turbine stop valves (TSVs). The secondary model also includes the delivery of feedwater, both main and emergency, to the steam generators.

The RELAP5/MOD2-B&W code has been approved by the Nuclear Regulatory Commission (NRC) for use in non Loss of Coolant (LOCA) safety analyses (Reference [2]). The FWLB analysis documented in Section 2.8.5.2.4 of the CR-3 EPU LAR is consistent with Reference [2]. The methodology in Reference [2] dictates the input boundary conditions for many plant parameters, such as the RCS flow, Tave, and the RCS initial pressure. The intent of the evaluations in this report is to determine the effect of changing initial conditions on the FWLB transient to support compliance with SRP Section 15.2.8. Therefore, the analyses in this document include input conditions that are not consistent with Reference [2]. However, the conditions specified in Reference [2] are included or bounded by the range of conditions considered in the sensitivity studies.

3.0 ANALYSIS INPUTS

Many of the input parameters in the FWLB sensitivity studies are consistent with the FWLB analysis without pressurizer spray performed to support Section 2.8.5.2.4 of the CR-3 EPU LAR. The input parameters that are consistent are listed below:

- The maximum power level considered is the targeted EPU power level of 3026.1 MWt (100.4% of 3014 MWt). A conservative reactor coolant pump (RCP) heat of 16.4 MWt is included when the RCPs are operating.
- The nominal average RCS temperature (Tave) is 582 °F, consistent with the increase in Tave that is planned in conjunction with the EPU.
- The nominal hot leg pressure is 2170 psia.
- The minimum RCS flow rate considered is 374,880 gpm.
- The nominal indicated pressurizer level plus uncertainty is 240 inches.
- Two pressurizer safety valves (PSVs) were modeled with a nominal lift setpoint of 2500 psig, plus 3% lift tolerance, and 0% accumulation. A blowdown of 4% was also considered.
- Pressurizer heaters and pressurizer spray were not modeled.
- The Pilot Operated Relief Valve (PORV) was not modeled for the FWLB analysis.
- Reactor trip was modeled to occur based on the reactor protection system (RPS) high RCS pressure trip function. The high RCS pressure trip setpoint includes the effects of elevated pressure that may exist inside containment post-FWLB.
- The control rod worth modeled in the analysis is the minimum rod worth that will meet the shutdown margin (SDM) requirement.
- After reactor trip, the core heat generation rate was conservatively based on 1.0 times the ANS 1971 decay heat standard for fission plus heavy actinides.
- Beginning-of-cycle (BOC) typical reactivity coefficients (Doppler temperature coefficient (DTC) = -1.30 × 10^{-5} Δk/k/°F and moderator temperature coefficient (MTC) = 0.0 × 10^{-4} Δk/k/°F) were evaluated.
- The MSSVs were modeled to lift at the nominal setpoints plus 3% lift tolerance and 3% accumulation. A nominal blowdown of 5% was used.
- A single failure of one train of the Emergency Feedwater Initiation and Control (EFIC) system was assumed as the worse case single failure. Therefore, only one of the two emergency feedwater (EFW) pumps was assumed to be available to provide flow to the SGs. The EFW temperature was modeled as 120 °F. The EFIC system contains Feed Only Good Generator (FOGG) logic, thus all EFW was provided to the unaffected SG.
- Steam generator tube plugging of 5% was modeled.

- No operator actions were credited.
- No Integrated Control System (ICS) actions were credited.

Key input parameters modeled in the sensitivity studies that are different from the FWLB analyses in Section 2.8.5.2.4 of the CR-3 EPU LAR include:

- The sensitivity studies considered power levels ranging from 60% to 100.4% of the EPU rated thermal power level (3014 MWt). The sensitivity demonstrates that the peak RCS pressure is much less at lower power levels, so evaluations below 60% were not required.
- The sensitivity studies considered a Tave range of 582 +/- 3 °F, allowing for an uncertainty of up to 3 °F.
- The sensitivity studies considered hot leg pressure ranging from the minimum allowed value of 2078.7 psia to a maximum value of 2244.7 psia, which is 25 psi above the setpoint for opening the pressurizer spray valves to allow for uncertainty.
- The sensitivity studies considered RCS flow ranging from the minimum DNB flow of 374,880 gpm to a maximum flow of 398,850 gpm. The maximum flow value is based on no tube plugging, a 2.5% uncertainty, and the conservative assumption that all of the SG tube walls are at the minimum tube wall thickness.
- The sensitivity studies considered initial pressurizer level ranging from 240 inches to 290 inches, which is the maximum pressurizer level currently allowed for CR-3. A higher pressurizer level is more limiting, so levels below 240 inches were not evaluated.
- The sensitivity studies evaluated the effect of time in cycle on the FWLB transient including end of cycle (EOC) typical reactivity coefficients (DTC = -2.00×10^{-5} Δk/k/°F and MTC = -4.0×10^{-4} Δk/k/°F).
- The sensitivity studies evaluated initial SG levels ranging from 50 %OR to 95 %OR.
- The sensitivity studies evaluated initial turbine header pressures in the range of 930 +/- 50 psia.
- The sensitivity studies evaluated the FWLB transient with and without a LOOP. When modeled, the LOOP occurs at the time of turbine trip because a LOOP at the start of the event is less limiting.
- The FWLB analyses in Section 2.8.5.2.4 of the CR-3 EPU LAR considered a high RCS pressure reactor trip setpoint of 2445.45 psia. This trip setpoint is based on the conservative assumption that the containment pressure is at the maximum allowed pressure (55 psig) by the time the trip setpoint is reached following a FWLB. A more realistic evaluation of containment pressure following a secondary side pipe break demonstrated that the containment pressure increases by less than 29.1 psi by the time the high RCS pressure trip setpoint is reached in a FWLB transient. Therefore, the sensitivity studies are based on a high RCS pressure setpoint of 2420 psia.
- The FWLB analyses in Section 2.8.5.2.4 of the CR-3 EPU LAR conservatively modeled a 1.0 %Δk/k SDM requirement at hot zero power (HZP). The Modes 1 and 2 minimum SDM requirement is being increased to 1.3 %Δk/k for the CR-3 EPU. Therefore, the sensitivity studies are based on the EPU SDM requirement of 1.3 %Δk/k at HZP.

- The FWLB analyses in Section 2.8.5.2.4 of the CR-3 EPU LAR conservatively modeled the EFW flow as 550 gpm (total) occurring with a 60-second delay after the low SG level initiation setpoint. The CR-3 EPU loss of feedwater (LOFW) analysis in Section 2.8.5.2.3 of the CR-3 EPU LAR requires a minimum EFW flow of 660 gpm and a 40-second delay. Therefore, the sensitivity studies are based on a minimum EFW flow of 660 gpm and a 40-second delay after the low SG level initiation setpoint for consistency with the CR-3 EPU LOFW analyses.

4.0 RESULTS

4.1 Base Case for Sensitivity Studies

As identified in Section 3.0 of this report, several of the key input assumptions in the FWLB analysis in Section 2.8.5.2.4 of the CR-3 EPU LAR contained excess conservatism. These assumptions, as well as some details in the model, were modified prior to performing the sensitivity studies. In particular, the following input assumptions were changed:

1. The minimum EFW flow of 550 gpm with a 60-second delay time after the low SG level initiation setpoint is reached was changed to a minimum EFW flow of 660 gpm with a 40-second minimum delay time. This change makes the FWLB analysis consistent with the requirements of the LOFW transient. Note that the minimum EFW flow is based on the limiting single failure assumption that only one of the two EFW pumps is available to provide flow to the SGs.

2. The SDM requirement was increased from 1.0 %Δk/k to 1.3 %Δk/k at HZP with the maximum worth control rod removed from the core. The Modes 1 and 2 minimum shutdown margin requirement is being increased to 1.3 %Δk/k for the EPU.

3. The RPS high RCS pressure trip setpoint was reduced from 2445.45 psia to 2420 psia. Evaluations determined that a setpoint of 2420 psia continues to conservatively account for elevated pressure that may exist inside containment post-FWLB at the time that the reactor trip setpoint is reached.

4. More detailed calculations of the form loss in the pressurizer surge line as a function of surge line flow rate were incorporated into the FWLB model.

5. A more detailed calculation of the reverse form loss through the MFW nozzles was performed. The results of the calculation were incorporated into the FWLB model.

6. A preliminary sensitivity study of initial SG inventory determined that the peak RCS pressure is highest for an initial SG level of 80 %OR. Therefore, the initial SG level was changed to 80 %OR.

The peak RCS pressure for the FWLB analysis in Section 2.8.5.2.4 of the CR-3 EPU LAR is 2896.20 psia. The FWLB transient described in Section 2.8.5.2.4 of the CR-3 EPU LAR was re-evaluated with the changes described above. The resulting peak RCS pressure was 2829.42 psia. Unless otherwise stated, the model resulting in a peak RCS pressure of 2829.42 psia is used as the base case for sensitivity studies described in this report. The base case targets the following initial conditions:

- 2170 psia hot leg pressure
- Minimum RCS flow of 374,880 gpm
- 100.4% of 3014 MWt
- Beginning of cycle typical parameters (MTC = 0.0×10^{-4} Δk/k/°F, DTC = -1.30×10^{-5} Δk/k/°F, βeff = 0.0070, Fuel Temperature ~ 1400 °F)
- Tave of 582 °F
- Pressurizer indicated level of 240 inches
- SG level of 80 %OR
- No LOOP
- Double-ended guillotine break in the main feedwater line shortly before the line splits into two smaller branches that feed the SG (see Figure 4-3).

4.2 RCS Pressure Sensitivity

This section presents the results of the RCS pressure sensitivity study. The RCS pressure sensitivity starts from the base case (nominal RCS pressure) and changes the initial hot leg pressure. The peak RCS pressure and reactor trip times from the RCS pressure sensitivity study are presented in Table 4-1.

The RCS pressure sensitivity study demonstrates that the time of reactor trip during a FWLB transient is directly affected by the initial RCS pressure. For cases evaluated near the nominal hot leg pressure (i.e. 2170 psia +/- 30 psia), the change in the time of reactor trip does not significantly affect the increase in RCS pressure that occurs after reactor trip. Consequently, the peak RCS pressure for each case analyzed near the nominal hot leg pressure is within 5 psi of the other cases.

In cases that are initialized further away from the nominal hot leg pressure (i.e. 2078.7 psia and 2244.7 psia), the change in the time of reactor trip is more significant. The transient progression during the time leading up to reactor trip for these cases leads to a lower peak RCS pressure than the case evaluated at the nominal hot leg pressure.

Based on the RCS pressure sensitivity study, setting the initial RCS pressure to the nominal value is appropriate for evaluating the FWLB transient.

Table 4-1: RCS Pressure Sensitivity Results

4.3 RCS Flow Sensitivity

This section presents the results of the RCS flow sensitivity study. The RCS flow sensitivity starts from the base case (minimum RCS flow) and changes the initial RCS flow to the maximum value. Establishing steady-state conditions at higher RCS flow will result in a change in Tave, the SG pressure, the SG inventory, or a combination of these parameters. The RCS flow sensitivity maintains a consistent SG inventory for all cases and evaluates one case where only Tave changes and one case where only the SG pressure changes. The peak RCS pressure and reactor trip times from the RCS flow sensitivity study are presented in Table 4-2.

The RCS flow sensitivity study shows that the RCS flow does not significantly change the hot leg pressure during the transient, and therefore does not affect the time of reactor trip. However, since the hot leg pressure is the same and the RCS flow is higher, the ΔP from the hot leg to the location of the peak RCS pressure (i.e. at the bottom of the reactor vessel) increases. Consequently, the peak RCS pressure is [ ] higher for cases evaluated at the maximum RCS flow.

Based on the RCS flow sensitivity study, the peak RCS pressure for a FWLB transient should be evaluated assuming maximum RCS flow. This would not be conservative for evaluating departure from nucleate boiling (DNB). However, as explained in Section 2.8.5.2.4.2 of the CR-3 EPU LAR, since the reactor coolant pumps (RCPs) remain operating, the RCS fluid remains subcooled, and the core power remains less than 112% throughout the FWLB analysis, it is concluded that the minimum DNB ratio would remain above the applicable correlation limit without performing an explicit analysis for FWLB. Therefore, the RCS flow for the FWLB transient should be biased to challenge the peak RCS pressure acceptance criteria.

Table 4-2: RCS Flow Sensitivity Results

[ ]

4.4 Power Level Sensitivity

This section presents the results of the power level sensitivity study. The base case is evaluated at 100.4% of the proposed EPU power level. The power level sensitivity study compares the base case to evaluations performed at 95.4 %FP, 80 %FP, and 60 %FP. Table 4-3 documents the steady-state conditions achieved for each power level. The peak RCS pressure and reactor trip times from the FWLB power level sensitivity study are presented in Table 4-4.

The FWLB results show that as the power level decreases, the lower initial SG inventory results in an earlier reactor trip. The earlier reactor trip in conjunction with a lower initial core power level causes the peak RCS pressure to be significantly lower as the power level decreases. The peak RCS pressure is less than 110% of the RCS design pressure (< 2764.7 psia) for initial power levels of 80 %FP and below. Given the significant reduction in peak RCS pressure observed at 80 and 60 %FP, it was concluded that power levels <60 %FP did not need to be evaluated.

CR-3 is allowed to operate with a positive MTC at lower power levels. As discussed in Section 2.8.5.4.1.2 of the CR-3 EPU LAR, the most positive MTC that will be allowed for EPU core designs is $+7.5 \times 10^{-5}$ $\Delta k/k/°F$. The FWLB power level sensitivity study included evaluations at 95.4 %FP, 80 %FP, and 60 %FP with an MTC of $+7.5 \times 10^{-5}$ $\Delta k/k/°F$. A positive MTC resulted in a higher peak RCS pressure for each power level analyzed; however, the base case (100.4 %FP, 0.0×10^{-5} $\Delta k/k/°F$) remained the limiting case.

Based on the power level sensitivity study, the FWLB transient should be evaluated at 100.4 %FP.

Table 4-3: Steady-State Results for Power Level Sensitivity

[ ]

Table 4-4: Power Level Sensitivity Results

[ ]

4.5 Time in Cycle Sensitivity

This section presents the results of the time in cycle sensitivity study. The base case is evaluated at conservative BOC conditions. The MTC (a constant 0.0×10^{-5} Δk/k/°F) is the most positive MTC allowed at full power. The DTC is a value (-1.3×10^{-5} Δk/k/°F) that is more positive than expected for EPU fuel cycle designs (see Table 2.8.2-2 of the CR-3 EPU LAR). The fuel temperature (~1400 °F) is representative of BOC conditions at the EPU power level. Finally, the effective delayed neutron fraction (βeff) is a BOC typical value (0.0070). The rod worth used in the base case is determined to be the minimum required rod worth needed to meet the SDM definition for the modeled MTC, DTC, and fuel temperature. The time in cycle sensitivity study evaluates the effects of changing MTC, DTC, fuel temperature, and βeff on the FWLB transient. Table 4-5 presents the key results from the FWLB time in cycle sensitivity study.

The base case utilizes an MTC that does not change with moderator temperature or density. Therefore, the time in cycle sensitivity analysis compared the base case to cases using constant MTC values of -10.0×10^{-5} Δk/k/°F and -20.0×10^{-5} Δk/k/°F. When a constant MTC is used, the thermal power level of the core does not change significantly prior to reactor trip. Consequently, the time of reactor trip is approximately the same for all three cases evaluated with a constant MTC. Although the pre-trip transient behavior is approximately the same in all three cases, the results show that a more negative MTC resulted in a lower peak RCS pressure. This occurs because a more negative MTC requires more rod worth in order for the SDM requirement to be met. The additional rod worth results in a faster reduction in the core thermal power shortly after trip, which in turn reduces the peak RCS pressure. The trend versus MTC is clear, so evaluations were not performed for constant MTCs that are more negative than -20.0×10^{-5} Δk/k/°F.

The time in cycle sensitivity study also includes cases that allow the moderator reactivity feedback to change with moderator density. The sensitivity study considered five reactivity versus moderator density curves. The MTCs from the curves at initial hot full power conditions are 0.0×10^{-5} Δk/k/°F, -10.0×10^{-5} Δk/k/°F, -20.0×10^{-5} Δk/k/°F, -30.0×10^{-5} Δk/k/°F, and -40.0×10^{-5} Δk/k/°F. Table 2.8.2-2 of the CR-3 EPU LAR lists the expected EPU most negative MTC limit as -37.5×10^{-5} Δk/k/°F, so a -40×10^{-5} Δk/k/°F MTC bounds typical EPU cycles. The sensitivity study performed using different reactivity versus moderator density curves demonstrates that as the MTC becomes more negative, the peak thermal power increases and the time to reactor trip decreases. However, in all of the cases the peak thermal power remained well below 112 %FP. The increase in thermal power that occurs prior to reactor trip is more than offset by the increased rod worth that is needed to meet the SDM definition at more negative MTCs. Therefore, as the MTC becomes more negative the peak RCS pressure decreases.

The time in cycle sensitivity study included a case to evaluate the effect of DTC. The case is evaluated with an MTC that is representative of EOC (i.e. -40×10^{-5} Δk/k/°F). The DTC sensitivity study considers values of -1.30×10^{-5} Δk/k/°F and -2.00×10^{-5} Δk/k/°F, which bound the typical EPU values in Table 2.8.2.2 of the CR-3 EPU LAR. The DTC comparison shows that a more negative DTC causes the maximum thermal power to decrease because of the additional negative reactivity feedback. Furthermore, the more negative DTC requires more rod worth to meet the minimum SDM requirement. These two factors result in a lower peak RCS pressure for the case evaluated with a more negative DTC.

The effect of fuel temperature is evaluated with reactivity coefficients that are representative of EOC. The fuel temperature sensitivity reduces the BOC typical fuel temperature (~1400 °F) to a fuel temperature that is representative of EOC conditions (~1080 °F). The fuel temperature is changed by increasing the fuel gap thermal conductance. The fuel temperature comparison demonstrates that the increased fuel gap thermal conductance associated with EOC fuel temperatures results in a slightly higher peak thermal

power. In addition, a lower initial fuel temperature means that less rod worth is needed to meet the minimum SDM requirement because the initial fuel temperature is closer to the HZP temperature. However, the lower initial fuel temperature also results in less heat stored in the fuel that must be removed following reactor trip. Consequently, the peak RCS pressure is significantly less for cases evaluated at EOC typical fuel temperatures.

The effect of βeff is evaluated with an MTC, DTC, and fuel temperature that are representative of EOC. The βeff sensitivity study considers values of 0.0045 and 0.0070, which bound the typical EPU values in Table 2.8.2.2 of the CR-3 EPU LAR. The βeff comparison shows that smaller values of βeff result in a faster positive reactivity response prior to trip, which increases the maximum thermal power. However, smaller values of βeff also result in a faster negative reactivity response as the control rods are inserted. The effect of βeff on the post-trip portion of the transient has more effect on the peak RCS pressure, which occurs a few seconds after reactor trip. Consequently, the peak RCS pressure is significantly less for cases evaluated at EOC typical values of βeff.

The sensitivity studies on time in cycle demonstrated that the peak RCS pressure in a FWLB transient is higher when BOC typical values are modeled for MTC, DTC, fuel temperature, and βeff.

Table 4-5: Time in Cycle Sensitivity Results

[ ]

4.6 Steam Generator Inventory (Tave, SG Pressure, and SG Level) Sensitivity

This section presents the results of the SG inventory sensitivity study, which varies RCS Tave, SG pressure, and SG level to achieve different initial SG inventories. The SG inventory sensitivity study starts from the base case. The RCS flow is set to a maximum value because the evaluations summarized in Section 4.3 determined that a maximum RCS flow results in a higher peak RCS pressure than a case at a minimum DNB flow. The initial RCS pressure, initial power level, and time in cycle evaluated in the SG inventory sensitivity study match the base case because the values used in the base case were shown to be the limiting conditions. The RCS Tave, SG pressure, and SG level are varied within the following spectrum of initial conditions:

- RCS Tave in the range of 582 °F +/- 3 °F
- Turbine header pressure in the range of 930 psia +/- 50 psi
- SG level ranging from 50 %OR to 95 %OR

Although the above ranges are considered, every possible combination cannot realistically be achieved. For a given Tave, the minimum level that can be achieved in the SG is affected by the orifice plate in the SG downcomer and the initial SG pressure. Therefore, the minimum SG levels considered for a given RCS Tave and SG pressure combination is either 50 %OR or the minimum level when the orifice plate is fully opened. Similarly, the maximum SG pressure that can be modeled for a given RCS Tave is the pressure that results in a SG level of 95 %OR when the orifice plate is fully open.

Table 4-6 summarizes the steady-state SG inventory and the key FWLB transient results for each of the 34 cases that were evaluated to consider the effects of RCS Tave, SG pressure, and SG level on the FWLB transient. The results clearly show that as the SG inventory increases, the time to reactor trip increases. This is demonstrated by Figure 4-1. The longer time to reactor trip means that cases starting with more initial SG inventory also discharge more inventory out of the break by the time reactor trip is reached. This behavior leads to relatively small differences in the affected SG inventory when reactor trip occurs. Since each case has approximately the same inventory in the affected SG at the time of reactor trip, the RCS pressure increase shortly after reactor trip is close to the same for each case considered. Furthermore, since the reactor trip is initiated by a high RCS pressure trip function, each case is at the same hot leg pressure when the reactor trip occurs. Consequently, the peak RCS pressure in a FWLB transient is not sensitive to RCS Tave, SG pressure, SG level, or SG inventory. This is demonstrated by the similarity in the peak RCS pressure results in Table 4-6 and Figure 4-2. The highest peak RCS pressure for any of the cases considered is 2842.15 psia at an RCS Tave of 585 °F, a turbine header pressure of 940 psia, and a SG level of 80 %OR.

Although the peak RCS pressure is not sensitive to changes in RCS Tave, SG pressure, SG level, or SG inventory, these parameters will affect the mass and energy released to containment. Cases with higher initial SG inventory would release more mass from the SG side of the break and would delay reactor trip allowing more mass from the MFW side of the break to reach containment. Conversely, minimizing the initial SG inventory will minimize the mass and energy released to containment.

Table 4-6: SG Inventory Sensitivity Results

Figure 4-1: Time to Reactor Trip versus Initial Steam Generator Inventory

Figure 4-2: Peak RCS Pressure versus Initial Steam Generator Inventory

4.7 Pressurizer Level Sensitivity

This section presents the results of the pressurizer level sensitivity study. The evaluations summarized in Sections 4.1 through 4.6 are based on an indicated pressurizer level of 240 inches, which is the nominal pressurizer level of 220 inches plus 20 inches for uncertainty. The pressurizer level sensitivity study evaluates pressurizer levels as high as 290 inches, which is the maximum pressurizer level currently allowed for CR-3.

Six of the cases evaluated in Section 4.6 were selected for evaluation in the pressurizer level sensitivity study. The cases chosen were selected to span the ranges of conditions evaluated in Section 4.6. The cases chosen also include the cases in Table 4-6 that had the highest peak RCS pressure. The peak RCS pressure and reactor trip times from the pressurizer level sensitivity study are presented in Table 4-7.

The pressurizer level sensitivity study demonstrated that for each case, a higher initial pressurizer level causes the RCS pressure to respond more quickly to changes in the RCS temperatures because of the smaller available steam space. Consequently, the reactor trip occurs sooner and the peak RCS pressure is higher when the initial pressurizer level is higher. The highest peak RCS pressure for any of the cases considered with a pressurizer level of 290 inches is 2861.20 psia.

In addition to evaluating the peak RCS pressure, the pressurizer level sensitivity study also evaluated the temperature of liquid in the pressurizer for those cases that predicted liquid relief out of the PSVs. The pressurizer sensitivity study verified that if liquid is passed by the PSVs, the liquid remains above 600 °F.

The sensitivity study on pressurizer level demonstrated that a higher initial pressurizer level is more limiting for the FWLB transient.

Table 4-7: Pressurizer Level Sensitivity Results

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4.8 Loss of Offsite Power

This section presents the results of the LOOP sensitivity study. The evaluations in Sections 4.1 through 4.7 are based on offsite power remaining available throughout the FWLB transient. The LOOP evaluation determines if the FWLB transient results are more limiting if offsite power is lost during the transient.

If the LOOP occurs at the time that the break opens, the loss of offsite power will result in a reactor scram, which would reduce the energy added to the RCS by the core. Therefore, a LOOP at event initiation is less limiting than if a LOOP does not occur. On the other hand, if the LOOP is modeled to occur at the time of turbine trip, which is modeled to occur coincident with reactor trip, the LOOP will not affect the energy added to the RCS by the core. Furthermore, a LOOP at turbine trip causes the RCS flow to decrease just as the RCS pressure is approaching the maximum value. Therefore, a LOOP at turbine trip is evaluated as the limiting time for a LOOP to occur.

The LOOP evaluation considers the six cases from Section 4.7, but with a LOOP modeled at turbine trip. The results with and without LOOP for each of the six cases are summarized in Table 4-8.

The LOOP evaluation demonstrates that in each case, the peak RCS pressure is higher when a LOOP occurs at turbine trip. The highest peak RCS pressure (2878.38 psia) continues to occur at the same conditions as the case with no LOOP (Tave of 585 °F, a turbine header pressure of 940 psia, and a SG level of 80 %OR). The LOOP evaluation also confirmed that liquid passed by the PSVs remains above 600 °F.

Based on the LOOP evaluation, the FWLB transient should be evaluated with a LOOP occurring at the time of turbine trip.

Table 4-8: LOOP Evaluation Results

[ ]

4.9 Break Size Sensitivity

This section presents the results of the break size sensitivity study. The evaluations in Sections 4.1 through 4.8 are based on a double-ended guillotine break of the largest main feedwater (MFW) pipe near the junction where the largest pipe splits into two smaller pipes that continue to the SG (see Figure 4-3). The break size sensitivity study evaluates the effect of smaller break sizes at the same location. The break size sensitivity study did not change the area of the MFW side of the break. Therefore, all MFW is lost to the affected SG. Only the area on the SG side of the break was reduced to determine the break size sensitivity.

The break size sensitivity study was performed at the same conditions as the limiting case identified in Section 4.8, namely:

- Nominal RCS Pressure
- Maximum RCS Flow
- 100.4 %FP
- BOC
- 585 °F RCS Tave
- 940 psia Turbine Header Pressure
- 80 %OR SG Level
- 290 Inch Indicated Pressurizer Level
- LOOP occurring at Turbine Trip

The model was updated to include a detailed MFW system model from the booster pumps to the SGs. The MFW system model was set to isolate MFW to the unaffected steam generator at the start of the transient for consistency with previous FWLB evaluations. Overall, the model with a simplified MFW system and the model with a detailed MFW system produced equivalent FWLB transient results for the same break size. Incorporating the detailed MFW system resulted in a slightly different peak RCS pressure (2875.31 psia versus 2878.38 psia).

The CR-3 MFW system has three pipe sizes between the isolation check valve and the SGs. The largest main feedwater pipe is 18" SCH 80 (1.418 ft^2). The largest main feedwater pipe branches into two 14" SCH 80 (0.8522 ft^2) pipes. The two 14" SCH 80 pipes then feed into a total of 32 riser pipes, each of which are 3" SCH 80 (0.04587 ft^2). These three pipe areas are considered as the area of the SG side of the break along with arbitrarily selected pipe areas to fully capture the behavior of peak RCS pressure versus break size. The results of the break size sensitivity study are shown in Table 4-9.

The break size sensitivity study demonstrates that the peak RCS pressure is effectively the same for break sizes ranging from 0.8522 ft^2 (the area of the 14" SCH 80 pipes in the MFW system) to 1.418 ft^2 (the area of the 18" SCH 80 pipe in the MFW system). For break sizes less than 0.8522 ft^2, the peak RCS pressure steadily decreases. Note that the highest peak RCS pressure was calculated for a 1.100 ft^2 break, but the difference in peak RCS pressure between a 1.418 ft^2 break and a 1.100 ft^2 break is only 2.6 psi. Furthermore, the CR-3 MFW system does not actually have a pipe with a 1.100 ft^2 area between the isolation check valves and the SG. Therefore, evaluations based on the maximum pipe area of 1.418 ft^2 represent the limiting break area.

Based on the break size sensitivity study, the limiting break area for evaluating peak RCS pressure is the full area of the largest pipe in the CR-3 MFW system between the isolation check valves and the SGs (1.418 ft^2).

Table 4-9: Break Size Sensitivity Results

4.10 Break Location Sensitivity Study

This section presents the results of the break location sensitivity study. This break location sensitivity study evaluates the effects of moving the double ended guillotine break to different pipe locations. The following locations were considered (see Figure 4-3):

- The limiting break from Section 4.9, which is a break in the main feedwater pipe in SG B. The break is located shortly before the tee connecting the 18" main feedwater line to the two 14" side branches.
- A break in the same location as the limiting case from Section 4.9, but occurring on SG A instead of SG B.
- A break in the same pipe as the limiting case from Section 4.9, but occurring immediately after the isolation check valve.
- A break in one of the side branches of SG B at the tee that connects the main feedwater line to the side branches. The break is modeled as a double-ended guillotine break of a pipe with a 1.418 ft^2 area.
- A break in one of the side branches of SG B near the entrance to the SGs. The break is modeled as a double-ended guillotine break of a pipe with a 0.8522 ft^2 area.

The peak RCS pressure and reactor trip times for each of the identified break locations are summarized in Table 4-10.

The results in Table 4-10 show that a break in SG B is clearly more limiting than a break in SG A.

The peak RCS pressure for the four break locations in SG B all result in effectively the same peak RCS pressure, but different times for reactor trip. The reactor trip time indicates how quickly the affected SG discharges inventory out of the break. The earliest reactor trip time occurs when the break is in the branch side of the tee connecting the main feedwater line to the side braches. This location allows the SG inventory to reach both sides of the 1.418 ft^2 double-ended guillotine break. Since the total effective break area is larger, the reduction in inventory occurs more quickly. However, as mentioned above, the peak RCS pressure is effectively the same as the peak RCS pressure when the break is located in the main feedwater line before the tee. Furthermore, for both break locations all of the affected SG inventory will be discharged to containment by 60 seconds and the RCS conditions at that time are approximately the same. Therefore, the FWLB transient results are effectively the same for both break locations.

A break in one of the side branches near the SG also allows the SG inventory to reach both sides of the double-ended guillotine break, but the break area in both directions is only 0.8522 ft^2. The net effect is that the reactor trip time is slightly (0.188 sec) longer than the case with a break in the main feed line before the tee. The calculated peak RCS pressure in both cases differs by only 0.10 psi, which is an insignificant difference.

A break in the main feed line just downstream of the isolation check valves requires the longest time to reach the reactor trip setpoint. The peak RCS pressure is slightly less than the case with the break in the main feedwater line right before the tee, but the difference is negligible (2.12 psi).

The break location sensitivity study confirms that the limiting break location for evaluating peak RCS pressure is a double-ended break in the main feedwater pipe in SG B, located shortly before the tee connecting the 18" main feedwater line to the two 14" side branches.

Figure 4-3: Feedwater Line Break Locations

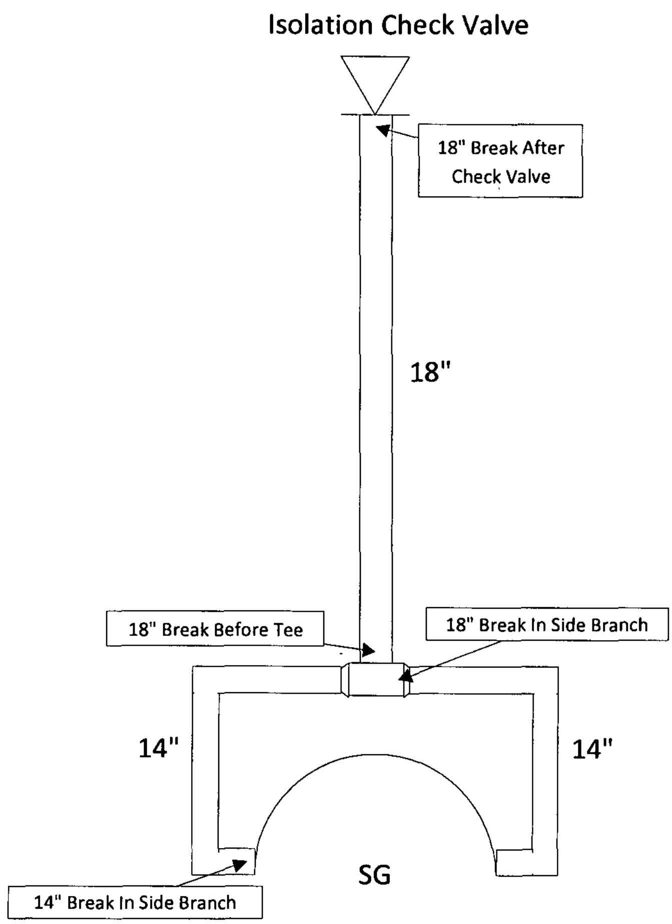

Table 4-10: Break Location Sensitivity Study

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4.11 Evaluation of Continued Main Feedwater to Unaffected SG

The CR-3 EPU FWLB evaluations conservatively model all feedwater to the unaffected SG as being lost at the start of the event. This conservative approach simplifies the analysis because the effect of control systems on the main feedwater pump speed during the transient can be ignored. This section presents an evaluation that demonstrates the conservatism added to the analysis by not crediting feedwater.

The evaluations with continued feedwater assume that the main feedwater pump speed is maintained at the initial condition value. EFIC actuation isolates feedwater or main feedwater is runback following a reactor trip. Once main feedwater is isolated or runback is started, the analysis conservatively assumes that all main feedwater is stopped to the unaffected SG in 3.2 seconds for consistency with the LOFW analysis (Section 2.8.5.2.3.2 of the CR-3 EPU LAR).

The detailed feedwater model includes the cross connect piping that exists between the booster pumps and the main feedwater pumps. Therefore, the evaluations with continued feedwater allow flow from the booster pump in loop A to feed the main feedwater pump on the broken loop if the transient predicts this behavior. During normal operation, the CR-3 main feedwater system does not have a cross connect between the loops downstream of the main feedwater pumps.

The FWLB evaluations with continued feedwater were performed for several break sizes. The peak RCS pressure and time of reactor trip are summarized in Table 4-11. Table 4-11 also contains the equivalent results when no credit is taken for continued MFW to the unaffected SG.

Table 4-12: Effect of Continued MFW to Unaffected SG

[ ]

The results show that for each break size, the time to reactor trip is longer and the peak RCS pressure is lower when MFW is modeled to reach the unaffected SG. For the limiting break size, the peak RCS pressure is 2875.31 – 2828.24 = 47.07 psi lower when MFW is allowed to reach the unaffected SG. The effect is less at smaller break sizes, but the reduction in peak RCS pressure is still significant.

The above evaluations demonstrate that significant conservatism is included in the FWLB evaluations because of the assumption that all feedwater is lost to the unaffected SG at the start of the event. The margins calculated above do not consider the effect of control systems changing the main feedwater pump speed during the transient. Nevertheless, any flow reaching the unaffected SG would be a benefit.

4.12 Limiting Feedwater Line Break Conditions and Results

The following targeted conditions were found to result in the highest peak RCS pressure during a FWLB transient.

- Nominal RCS pressure (2170 psia hot leg pressure)
- Maximum RCS flow (398,850 gpm)
- 100.4 %FP
- BOC (0.0×10^{-5} Δk/k/°F MTC, -1.3×10^{-5} Δk/k/°F DTC, ~1400 °F, 0.0070 βeff)
- 585 °F RCS Tave
- 940 psia turbine header pressure
- 80 %OR SG level
- 290 inch indicated pressurizer level
- LOOP occurring at turbine trip
- Double ended guillotine break located in the main feedwater line of SG B right before the tee connecting to the two side branches.

The sequence of events from this case is presented in Table 4-13. The results for the limiting case are presented in Table 4-14. Figures 4-4 through 4-8 plot key transient results for the limiting case.

Table 4-13: Sequence of Events for Limiting FWLB Transient

Event	Time (sec)
Transient initiated	0.0
MFW to both SGs interrupted	1.0E-6
Pressurizer spray begins	N/A
Peak thermal power occurs	7.9
RPS high RCS pressure trip actuated	10.0
Control rods begin to insert Turbine stop valves (TSVs) begin to close LOOP	10.6
EFIC actuated on low SG-B level	11.5
Initial PSV lift occurs	~12.5
Peak RCS pressure occurs	14.7
EFW flow begins	51.5
Affected SG depressurization complete	~55.0
Final PSV closure occurs	~200.0
Peak Tave occurs	~390.0
Transient terminated	600.0

Table 4-14: Results for Limiting FWLB Transient

Parameter	Value
Peak RCS Pressure (psia)	2878.38
Peak thermal power (%RTP)	100.50
Peak Tave (°F)	616.27

Figure 4-4: FWLB RCS Peak Pressure versus Time

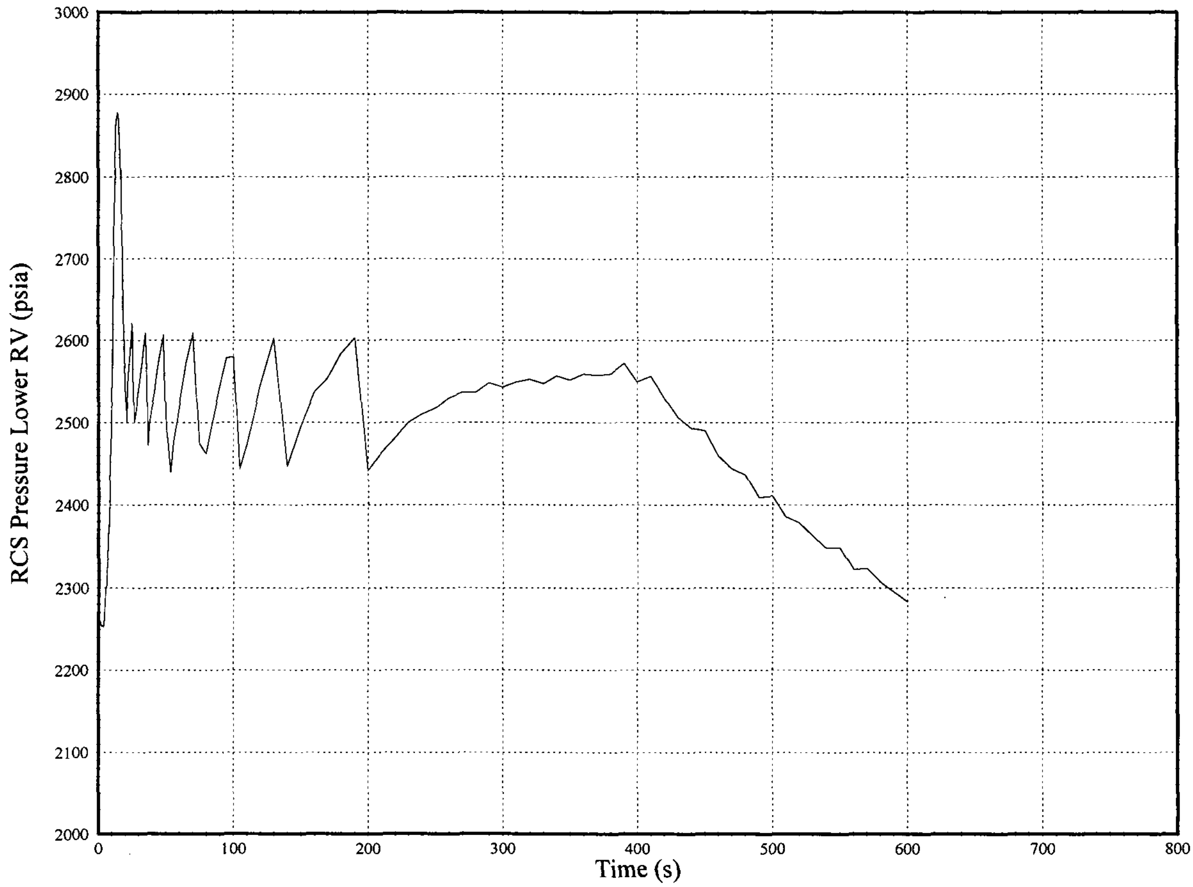


Figure 4-5: FWLB RCS Temperatures versus Time

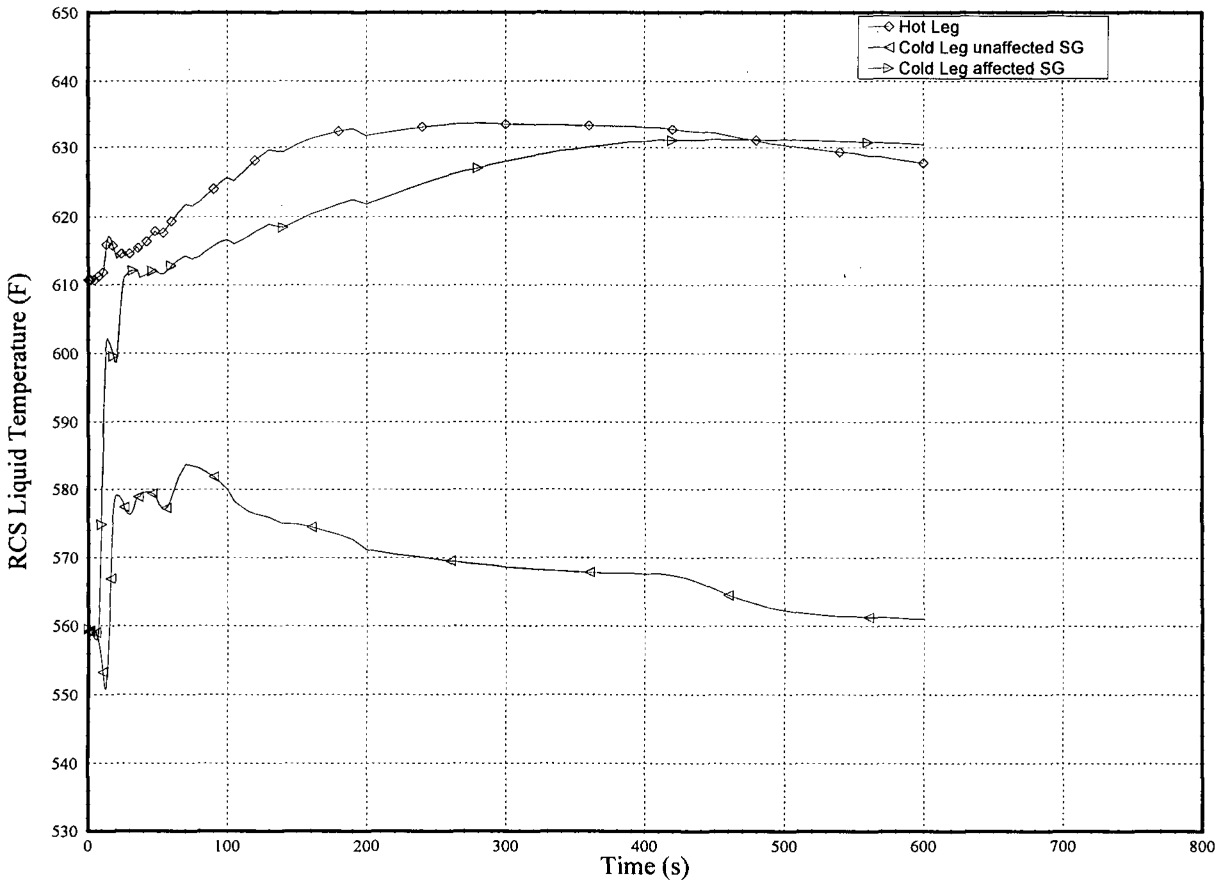


Figure 4-6: FWLB Pressurizer Level versus Time

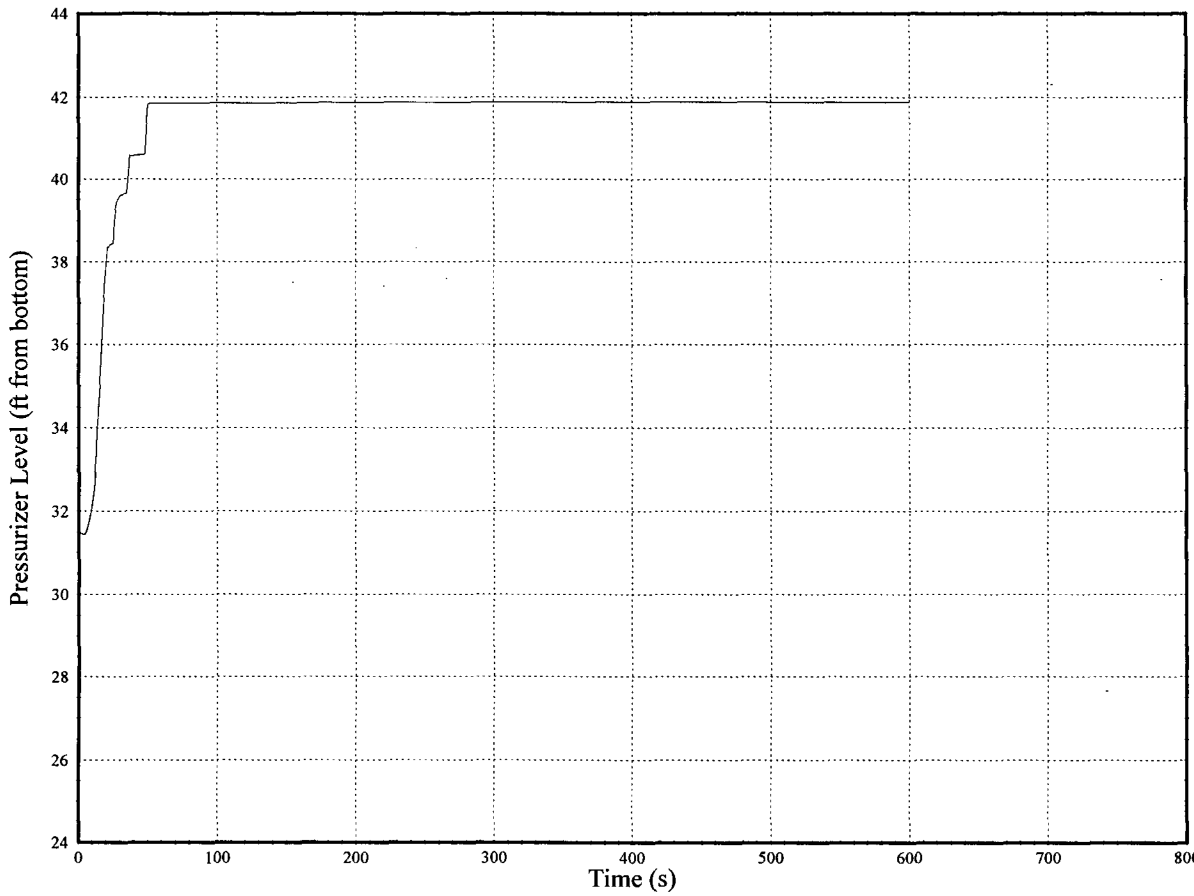


Figure 4-7: FWLB SG Pressures versus Time

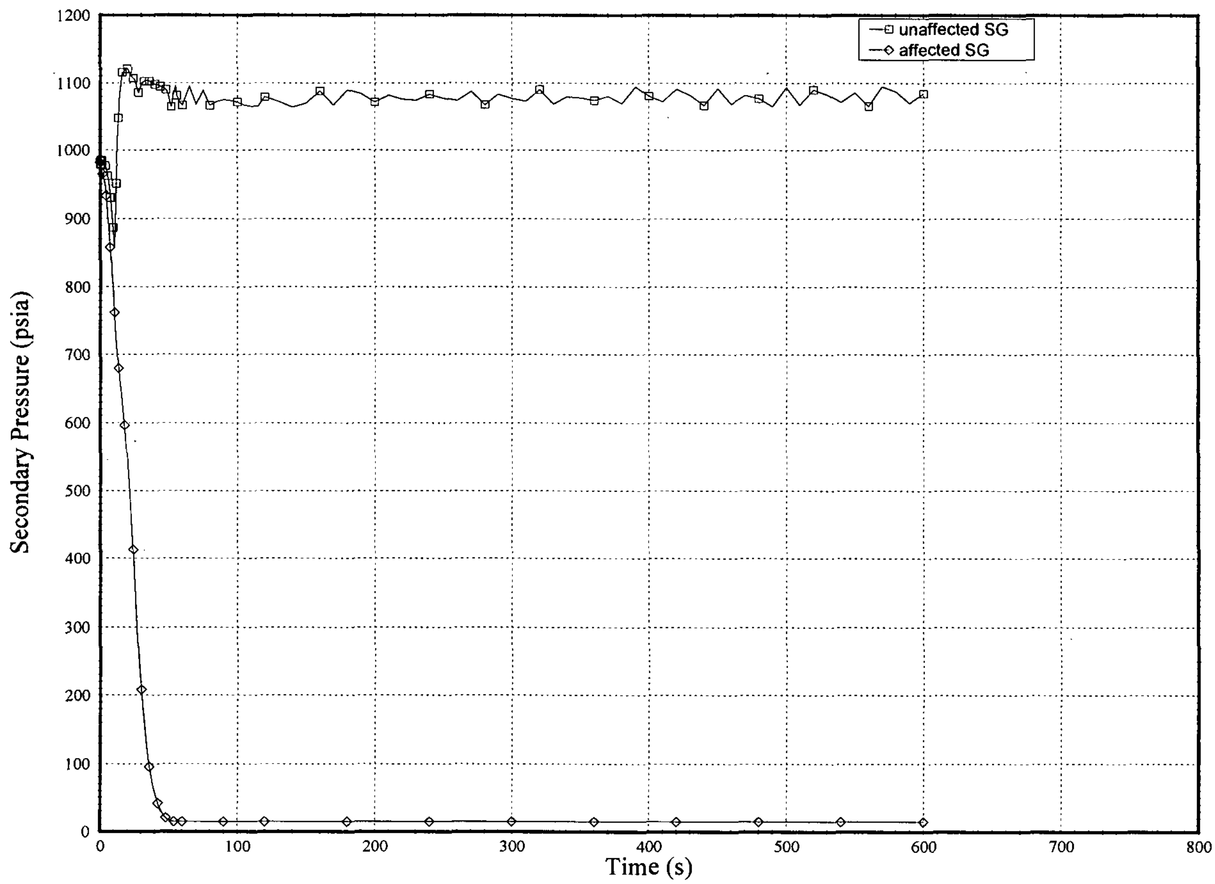
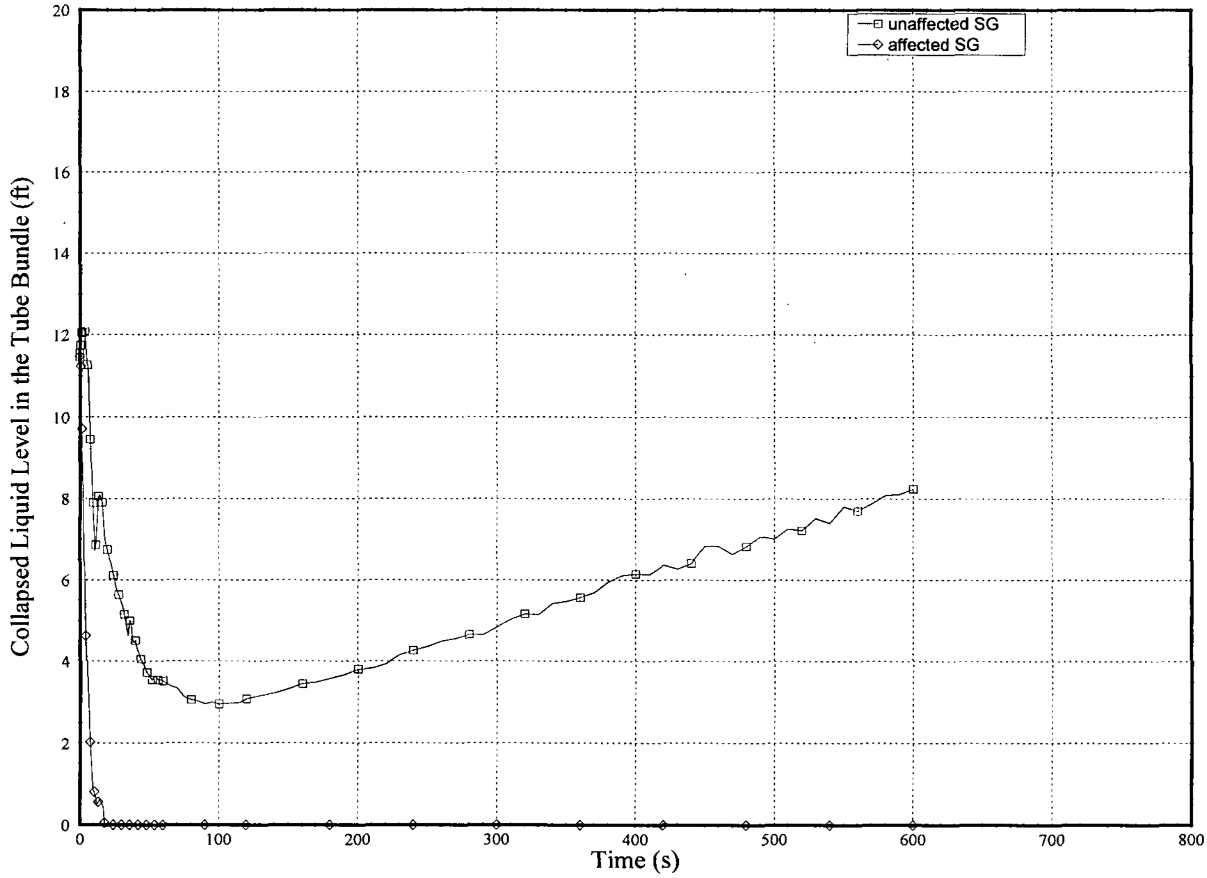


Figure 4-8: FWLB SG Levels versus Time



5.0 REFERENCES

1. BAW-10164PA-06, "RELAP5/MOD2-B&W – An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis."

2. BAW-10193PA-00, "RELAP5/MOD2-B&W for Safety Analysis of B&W-Designed Pressurizer Water Reactors."