ATTACHMENT 6

ANP-3000(NP)
Revision 0

ST. LUCIE NUCLEAR UNIT 1 EPU -
INFORMATION TO SUPPORT LICENSE AMENDMENT REQUEST

FLORIDA POWER AND LIGHT
ST. LUCIE PLANT UNIT 1

This coversheet plus 117 pages

ANP-3000(NP)
Revision 0

St. Lucie Unit 1 EPU – Information to Support License Amendment Request

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AREVA NP Inc.

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St. Lucie Unit 1 EPU – Information to Support License Amendment Request

Nature of Changes

Item	Page	Description and Justification
1.	All	Initial Release

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Nomenclature

Acronym	**Definition**
AOO	Anticipated Operational Occurrence
AOR	Analysis of Record
ASGPT	Asymmetric Steam Generator Pressure Trip
ASGT	Asymmetric Steam Generator Transient
ASI	Axial Shape Index
ASME	American Society of Mechanical Engineers
AST	Alternate Source Term
BE	Best Estimate
BOC	Beginning of Cycle
BOHL	Bottom of Heated Length
CE	Combustion Engineering
CE-NSSS	Combustion Engineering Nuclear Steam Supply System
CEA	Control Element Assembly
CHF	Critical Heat Flux
CL	Cold Leg
COLR	Core Operating Limit Report
CRGT	Control Rod Guide Tube
CWAP	Control Element Assembly Withdrawal Error at Power
DC-UH	Downcomer – Upper Head
DC-HL	Downcomer – Hot Leg
DNB	Departure From Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
EOC	End-of-Cycle
EPU	Extended Power Uprate
ESFAS	Engineered Safety Feature Actuation System
FCM	Fuel Centerline Melt
HA	Hot Assembly
HEM	Homogeneous Equilibrium Model
HFP	Hot Full Power
HL	Hot Leg
HPPT	High Pressurizer Pressure Trip
HPSI	High Pressure Safety Injection
HZP	Hot Zero Power

Nomenclature *(Continued)*

Acronym	Definition
LAR	License Amendment Request
LBLOCA	Large Break Loss-of-Coolant Accident
LOCA	Loss of Coolant Accident
LOEL	Loss of External Load
LPSI	Low Pressure Safety Injection
LR	Licensing Report
LS	Loop Seal
MDNBR	Minimum Departure From Nucleate Boiling Ratio
MFW	Main Feedwater
MFWP	Main Feedwater Pump
MSIV	Main Steam Isolation Valve
MSLB	Main Steam Line Break
MSSV	Main Steam Safety Valve
MTC	Moderator Temperature Coefficient
NR	Narrow Range
NRC	Nuclear Regulatory Commission
PCI	Pellet/Cladding Interaction
PCMI	Pellet/Cladding Mechanical Interaction
PCT	Peaking Cladding Temperature
PDIL	Power Dependent Insertion Limits
PORV	Power Operated Relief Valve
PSV	Pressurizer Safety Valve
PWR	Pressurized Water Reactor
RCP	Reactor Coolant Pump
RCPB	Reactor Coolant Pressure Boundary
RCS	Reactor Coolant System
RPS	Reactor Protection System
RTP	Rated Thermal Power
RV	Reactor Vessel
SAFDL	Specified Acceptable Fuel Design Limit
SBLOCA	Small Break Loss-of-Coolant Accident
SDM	Shutdown Margin
SG	Steam Generator
SIAS	Safety Injection Actuation Signal
SIT	Safety Injection Tank
SRP	Standard Review Plan

Nomenclature *(Continued)*

Acronym	**Definition**
Tavg	RCS Average Temperature
Tcold	Cold Leg Temperature
TM/LP	Thermal Margin/Low Pressure
TOHL	Top of Heated Length
TS	Technical Specification
UFSAR	Updated Final Safety Analysis Report
USNRC	United States Nuclear Regulatory Commission
VHP	Variable High Power
VHPT	Variable High Power Trip

1.0 Introduction

Through review of several recent submittals, the Nuclear Regulatory Commission (NRC) staff has identified some issues related to AREVA methodologies (References 1 through 5), some of which were employed in the development of the St. Lucie Unit 1 extended power uprate (EPU) license amendment request (LAR). These issues, and the proposed remedies, were discussed with the NRC in a meeting on March 16, 2011. The purpose of this document is to support NRC review of the St. Lucie Unit 1 EPU LAR by providing information related to methodology changes implemented as a result of the NRC's concerns.

Issues with the affected methodology documents are identified in Section 2.0 together with the respective responses. The following table provides a summary of the methodology issues addressed in this document.

Discipline	Topic
Large Break LOCA	• Refer to Reference 6
Small Break LOCA	• Break Spectrum and Loop Seal Clearing
	• Safety Injection (SI) Line Break
	• Delayed Reactor Coolant Pump (RCP) Trip
Non-LOCA Transient and Accident Analysis	• Overpressure Events
	• Locked Reactor Coolant Pump Rotor
	• Control Element Assembly (CEA) Withdrawal at Power
	• Control Element Assembly (CEA) Ejection Acceptance Criteria
	• Control Element Assembly (CEA) Ejection at Part-Power
	• Overpressure protection
	• Harsh Condition Uncertainties
	• Main Steam Line Break (MSLB) (Mode 3)
	• Asymmetric Steam Generator Transient
	• Pressurizer Level Plots for Condition II Events

The results to the identified issues contained herein are specific to the analyses supporting the St. Lucie Unit 1 EPU LAR submittal.

2.0 Issue Dispositions

2.1 *Large Break Loss of Coolant Accident Analysis*

Refer to the revised St. Lucie Nuclear Plant Unit 1 EPU Cycle Realistic Large Break LOCA Summary Report with Zr-4 Fuel Cladding (Reference 6).

2.2 *Small Break Loss of Coolant Accident*

2.2.1 Break Spectrum and Loop Seal Clearing

Issue

EMF-2328 (Reference 2) does not prescribe modeling approaches for the break spectrum. The NRC staff has observed selected break spectra based on generic geometry that does not reflect plant phenomenology. The spectrum needs to consider those break sizes that prevent safety injection tank deployment until immediately before and after the time of PCT. In the case of St. Lucie and the proposed evaluation, this would require tightening the break spectrum between 0.06 ft^2 and 0.08 ft^2.

i. This issue has been shown to result in a significant under-prediction of the peak cladding temperature.

ii. Refer to Item 1.a.ii for the applicable regulatory requirement.

iii. The staff may accept a proposal to use an augmented methodology, requiring the use of a finer break spectrum that is based on the phenomena governing the accident rather than an arbitrary prescription of the analyzed break spectrum.

Issue

The EMF-2328 evaluation model does not provide for a conservative representation of reactor coolant loop seal clearing.

i. This has been shown to result in a significant under-prediction of the peak cladding temperature.

ii. Refer to Item 1.a.ii for the applicable regulatory requirement.

iii. The staff may accept a proposal to use an augmented methodology that includes the use of a more conservative loop seal modeling approach.

Dispostion

Small Break Loss-of-Coolant Accident (SBLOCA) has been re-analyzed for the EPU with the AREVA EMF-2328(P)(A) evaluation model using a refined break spectrum. Specifically, the break spectrum has been refined in the range between [

] to determine the PCT and the limiting break size based on phenomena. The refined break spectrum addresses the phenomenology where Safety Injection Tank (SIT) flow begins just prior to or just after the increase in cladding temperature has effectively been mitigated by High Pressure Safety Injection (HPSI) flow.

[

] The re-analysis of the SBLOCA event has used [

]

[

]

Table 2.2.1-1 shows the results of the break spectrum analysis, including the time of PCT, the time of SIT flow initiation, and the number of loop seals that cleared for each break size analyzed. The [ ] break was identified as the limiting break size with respect to PCT. [

]

Table 2.2.1-2 shows the sequence of events for the [ ] break case. Figure 2.2.1-1 through Figure 2.2.1-14 show the system response for the [ ] break case. From Figure 2.2.1-14, it can be observed that the increase in cladding temperature was being mitigated by HPSI flow just prior to the cladding being quenched by SIT flow. This typifies the limiting case.

[

]

Table 2.2.1-1 Summary of Results for Break Spectrum Cases

Table 2.2.1-1 Summary of Results for Break Spectrum Cases ***(Continued)***

Table 2.2.1-2 Sequence of Events for [ ] Break Case

RV = Reactor Vessel
MFW = Main Feedwater
TM/LP = Thermal Margin/Low Pressure
SG = Steam Generator
SIAS = Safety Injection Actuation Signal

Figure 2.2.1-1 Reactor Power for [ ] Break

Figure 2.2.1-2 Pressurizer and Steam Generator Pressure for [ ] Break

Figure 2.2.1-3 Break Void Fraction for [ ] Break

Figure 2.2.1-4 Break Flow Rate for [ ] Break

Figure 2.2.1-5 Loop Seal Void Fractions for
[ ] Break

Figure 2.2.1-6 RCS Loop Flow Rate for [ ] Break

Figure 2.2.1-7 Main Feedwater Flow Rate for [ ] Break

Figure 2.2.1-8 Auxiliary Feedwater Flow Rate for [ ] Break

Figure 2.2.1-9 Steam Generator Total Mass for
[ ] Break

Figure 2.2.1-10 Total HPSI Mass Flow Rate for
[ ] Break

Figure 2.2.1-11 Total SIT Mass Flow Rate for [ ] Break

Figure 2.2.1-12 RCS and Reactor Vessel Mass Inventories for [ ] Break

Figure 2.2.1-13 Hot Assembly Collapsed Liquid Level and Mixture Level for [ ] Break

Figure 2.2.1-14 Hot Spot Cladding Temperature and Coolant Temperature for [ ] Break

2.2.2 Safety Injection Line Break

Issue

Provide the results of an analysis of the severed injection line with the degraded injection into the reactor coolant system (RCS) since one of the line spills to containment while others inject at the much higher RCS pressures.

Disposition

The following is additional information for the NRC regarding the St. Lucie Unit 1 EPU LAR Attachment 5, LR Section 2.8.5.6.3.3, Small Break LOCA.

In addition to a break spectrum analysis, an analysis of a double-ended-guillotine break in a SIT line was performed. The SIT line break area analyzed was 0.5592 ft^2 (10.126 in. diameter), which is the area of the SIT discharge line. This represents about 11.4% of the cold leg pipe area. [

] The assumed ECCS configuration bounds single failures in either one of two HPSI pumps or a single failure of one of two emergency diesel generators (i.e. failure of one train of safety injection).

Table 2.2.2-1 SIT Line Break HPSI Flow Table

[ ]

Table 2.2.2-2 shows the sequence of events for the SIT line break. Figure 2.2.2-1 through Figure 2.2.2-5 show the system and cladding temperature response. Figure 2.2.2-4 and Figure 2.2.2-5 show that the core collapsed liquid level is stabilized following SIT injection and the cladding remains quenched, respectively, with [ ]

The PCT for this case was calculated to be [ ]. The SIT line break results are non-limiting compared to the break spectrum results.

Table 2.2.2-2 Sequence of Events for SIT Line Break

CL = Cold Leg
MFWP = Main Feedwater Pump

Figure 2.2.2-1 SIT Line Break: RCS-side Break Flow Rate and Void Fraction

Figure 2.2.2-2 SIT Line Break: Pressurizer and Secondary Pressures

Figure 2.2.2-3 SIT Line Break: ECCS Injection

[ ]

Figure 2.2.2-4 SIT Line Break: Vessel Liquid Levels

CL = Cold Leg
LS = Loop Seal
BOHL = Bottom of Heated Length
TOHL = Top of Heated Length

Figure 2.2.2-5 SIT Line Break: Peak Cladding and Local Vapor Temperatures

2.2.3 Delayed Reactor Coolant Pump Trip

Issue

Perform a delayed reactor coolant pump trip analysis to demonstrate that the limiting break location for the RCP trip timing criteria has been identified.

Disposition

The following is additional information for the NRC regarding the St. Lucie Unit 1 EPU LAR Attachment 5, LR Section 2.8.5.6.3.3, Small Break LOCA.

2.2.3.1 Delayed RCP Trip Analysis Using Appendix K Models

The break spectrum analysis described in Section 2.2.1 assumed RCP trip at reactor trip, coincident with loss of offsite power. An evaluation of delayed RCP trip using Appendix K models was performed since delayed RCP trip following loss of subcooling margin (or reactor coolant system pressure of 1600 psia) can potentially produce more limiting results. Continued pump operation can result in more integrated mass lost out the break. Continued pump operation also tends to maintain RCS pressure at a plateau until the RCPs are tripped. This could potentially result in a reduced HPSI flow rate early in the transient. The combined effect will be less RCS and RV mass, more core uncovery, and a higher PCT relative to the break spectrum cases.

Both cold leg and hot leg break cases with various RCP trip delay times were analyzed. Table 2.2.3-1 shows results for the cold leg break delayed RCP trip calculations. The results for the cold leg break cases indicate that [

]

Table 2.2.3-2 shows results for the hot leg break delayed RCP trip calculations. The results for the hot leg break cases were more limiting than the results for the cold leg break cases. The results for the hot leg break delayed RCP trip cases indicate that [

]

Table 2.2.3-1 Cold Leg Break Delayed RCP Trip Results Using Appendix K Models - PCT (All 4 RCPs tripped simultaneously)

[ ]

Table 2.2.3-2 Hot Leg Break Delayed RCP Trip Results Using Appendix K Models - PCT (All 4 RCPs tripped simultaneously)

[ ]

Table 2.2.3-2 Hot Leg Break Delayed RCP Trip Results Using Appendix K Models – PCT (All 4 RCPs tripped simultaneously) ***(Continued)***

[ ]

[ ]

2.2.3.2 Delayed RCP Trip Analysis Using [ ]

A delayed RCP trip analysis was also performed using [

]

Both cold leg and hot leg break cases with various RCP trip delay times were analyzed. Table 2.2.3-3 shows the results for the cold leg break cases with delayed RCP trip. The cold leg break cases indicate [

]

Table 2.2.3-4 shows the results for the hot leg break cases with delayed RCP trip. The hot leg break cases also indicate [

]

Table 2.2.3-3 Cold Leg Break Delayed RCP Trip Results Using [ ] - PCT (All 4 RCPs Tripped Simultaneously)

Table 2.2.3-4 Hot Leg Break Delayed RCP Trip Results Using [ ] - PCT (All 4 RCPs Tripped Simultaneously)

2.3 *Non-LOCA Transient and Accident Analysis*

2.3.1 Overpressure Events

Issue

Per Page 5-5 of EMF-2310(P)(A) (Reference 3): [

] The methodology does not speak to the analysis of pressurization transients.

i. As indicated in a comparison of current licensing basis loss of external load (LOEL) analysis to the proposed EPU analysis, the EPU analysis predicts a lower peak pressure for the same transient initiated at nominal initial conditions as opposed to a conservatively low pressure. The staff believes this result is non-conservative.

ii. 10 CFR 50.36 states that LCOs are limiting initial conditions applied to process variables important to safety. Analyses are inconsistent with this requirement.

iii. The staff may consider supplementation of the report with sensitivity studies identifying the limiting initial pressure, and that the reload safety analysis methodology be supplemented to reflect analyzing the transient with conservative initial conditions.

Disposition

Additional parameter sensitivities were evaluated for events that significantly challenge the overpressure criteria. Those events are the Loss of External Load, CEA Ejection, and Control Element Assembly Withdrawal Error at Power events. Results of those evaluations are presented below.

2.3.1.1 Loss of External Load Event (LR Section 2.8.5.2.1, Loss of External Electrical Load, Turbine Trip, and Loss of Condensor Vacuum)

The LOEL is discussed in St. Lucie Unit 1 EPU LAR Attachment 5, LR Section 2.8.5.2.1, Loss of External Electrical Load, Turbine Trip, and Loss of Condenser Vacuum. The LOEL event was determined to be the limiting event for both primary side and secondary side pressurization. For the LOEL event, cases were analyzed from a Hot Full Power (HFP) initial condition to assess the challenge to acceptance criteria for primary side pressure and secondary side pressure. In addition, part-power cases were analyzed to assess the impact to secondary side pressures due to varying numbers of main steam safety valves (MSSV)s being out-of-service. Limiting case results for the LOEL are summarized in Table 2.3.1-1.

Key input parameters that characterize the sensitivity calculations performed relative to the analysis documented in LR Section 2.8.5.2.1 are described below.

- Initial Conditions – For cases initiated from HFP plus measurement uncertainty, both primary and secondary side pressure cases were analyzed. [

  ]

  For the part-power cases, the secondary side peak pressure was calculated for one, two and three out-of-service MSSVs per steam line. Initial conditions were conservatively treated, [

]

High Pressurizer Pressure Trip (HPPT) Uncertainty – The HPPT uncertainty [

] the actual calculated uncertainty for HPPT of <30 psi.

Power Operated Relief Valve (PORV) Operation - Operation of PORVs was conservatively modeled for both the primary side and secondary side pressurization analyses. [

]

2.3.1.1.1 LOEL Primary Side Pressurization Results

The limiting primary side pressurization case is the case with [

] The peak RCS pressure for the limiting case is less than 110% of design (i.e., 2750 psia).

The sequence of events for the limiting primary side pressurization case is given in Table 2.3.1-2, and the results are given in Table 2.3.1-1, [

]. The transient response for the limiting primary side pressurization case is shown in Figure 2.3.1-1 through Figure 2.3.1-9. Figure 2.3.1-1 shows the reactor power as a function of time. Figure 2.3.1-2 shows the pressurizer and peak RCS pressure compared with the RCS design pressure and 110% of RCS design pressure limit. Pressurizer liquid level is shown in Figure 2.3.1-3, Pressurizer Safety Valve (PSV) flow rate is shown in Figure 2.3.1-4, Figure 2.3.1-5 shows the RCS loop temperatures, and Figure 2.3.1-6 shows the RCS cold leg mass flow rates. Figure 2.3.1-7 shows the steam line pressures compared to the MSSV opening setpoints, Figure 2.3.1-8 shows the MSSV flow rates, and Figure 2.3.1-9 shows the reactivity feedback.

Results of the primary pressurization calculations demonstrate the following changes tend to increase the maximum primary side pressure:

[

]

2.3.1.1.2 LOEL Secondary Side Pressurization Results

The limiting secondary side pressurization case for full power operation is the case with [

]

The peak secondary side pressure for the limiting case is less than 110% of design (i.e., 1100 psia). The sequence of events is given in Table 2.3.1-3, and the results providing the peak main steam system pressure (SG dome) are given in Table 2.3.1-1, [

].

The transient response for the limiting case is shown in Figure 2.3.1-10 through Figure 2.3.1-17. Figure 2.3.1-10 shows the reactor power as a function of time. Figure 2.3.1-11 through Figure 2.3.1-17 show the pressurizer pressure, the pressurizer liquid level, the RCS loop temperatures, the RCS cold leg loop mass flow rates, the main steam system (SG dome) pressures, the MSSV flow rates, and the reactivity feedback, respectively.

For the part-power cases with one, two and three MSSVs out-of-service per SG, the calculated peak main steam system pressure was calculated to be less than 110% of design (i.e., 1100 psia), as shown in Table 2.3.1-4. [

]

2.3.1.2 CEA Ejection (LR Section 2.8.5.4.6, Spectrum of Rod Ejection Accidents)

Control rod ejection accidents cause a rapid positive reactivity insertion which increases RCS pressure and could lead to overpressurization of the reactor coolant pressure boundary. The consequences of a control rod ejection accident were evaluated in St. Lucie Unit 1 EPU LAR Attachment 5, LR Section 2.8.5.4.6, Spectrum of Rod Ejection Accidents.

Detailed thermal-hydraulic analyses of the CEA ejection event were performed as described in LR Section 2.8.5.4.6, using conditions described in LR Section 2.8.5.0, Accident and Transient Analyses. The peak RCS pressure analysis demonstrated that Beginning of Cycle (BOC) HFP conditions were the most conservative with respect to peak RCS pressure. The peak pressure result from the BOC HFP case was calculated to be 2696 psia, as compared to the Loss of External Load Event value of 2708 psia in LR Section 2.8.5.2.1. [

]

For the CEA ejection event, peak primary side pressure occurs after reactor trip, which occurs very early in the event – CEA insertion begins within one second of event initiation. Therefore, conditions that tend to increase the maximum primary side pressure are those that produce the fastest increase in pressure. Thus, [

]

The maximum reactor coolant pressure boundary (RCPB) pressure for this event is limited to that which causes local yielding, which is typically taken to be 120% of design pressure or 3000 psia. The peak RCS pressure calculated has a margin of greater than 50 psi to 110% of the design pressure and significantly more margin to 120% of the design pressure. The calculated pressure is also [ ]. The impact of [ ] on CEA ejection peak pressure will be well within the margin available [ ].

2.3.1.3 Uncontrolled Control Rod Assembly Withdrawal at Power (LR Section 2.8.5.4.2, Uncontrolled Rod Cluster Control Assembly Withdrawal at Power)

The Control Element Assembly Withdrawal Error at Power (CWAP) event is described in St. Lucie Unit 1 EPU LAR Attachment 5, LR Section 2.8.5.4.2, Uncontrolled Rod Cluster Control Assembly Withdrawal at Power.

As described in LR Section 2.8.5.4.2, RCS pressurization calculations were performed to evaluate the peak RCS pressure for this event. Part-power levels were analyzed as well as full power conditions. Both BOC and End of Cycle (EOC) kinetics were analyzed for each initial power level. Key input parameters were biased conservatively in order to determine the limiting peak RCS pressure. The calculations demonstrate that maximum RCS pressures occurred at the intersection of the VHPT and HPPT. The results, given in LR Table 2.8.5.4.2-1, show that peak RCS pressure increases with increasing core power with the overall limiting initial condition being HFP with BOC reactivity feedback. The peak RCS pressure was calculated to be 2657 psia which is less than the acceptance criterion of 2750 psia. The peak RCS pressure for this event is bounded by the Loss of External Load Event (LR Section 2.8.5.2.1).

The results in LR Section 2.8.5.4.2 are supported by sensitivity calculations that were performed [

]

Thus, the CWAP event will not exceed the 110% of design pressure criterion (2750 psia), and is bounded by the LOEL event for primary side pressurization.

The analysis presented in LR Section 2.8.5.4.2 shows that the CWAP event does not challenge the pressurizer level for overfill.

Table 2.3.1-1 Summary of Results for the Limiting HFP LOEL Primary and Secondary Side Pressure Cases

Table 2.3.1-2 Sequence of Events for Limiting HFP LOEL Primary Side Pressure Case

Event	Time (sec)
Event initiation (Turbine Trip)	[ ]
High Pressurizer Pressure trip setpoint reached	[ ]
Reactor trip occurred on High Pressurizer Pressure (including trip response delay)	[ ]
CEA insertion begins	[ ]
Peak reactor power occurred	[ ]
Pressurizer safety valves opened	[ ]
Peak primary pressure occurred	[ ]
Peak core-average RCS temperature occurred	[ ]
Steam generator Bank 1 MSSVs opened (both SGs)	[ ]
Peak pressurizer level occurred	[ ]
Peak main steam system pressure (SG dome) occurred	[ ]

Table 2.3.1-3 Sequence of Events for Limiting HFP LOEL Secondary Side Pressure Case

Event	Time (Sec)
Event initiation (Turbine Trip)	[ ]
Pressurizer spray begins	[ ]
Steam generator Bank 1 MSSVs opened (both SGs)	[ ]
Steam generator Bank 2 MSSVs opened (both SGs)	[ ]
High Pressurizer Pressure trip setpoint reached	[ ]
Reactor trip occurred on High Pressurizer Pressure (including trip response delay)	[ ]
Peak reactor power occurred	[ ]
CEA insertion begins	[ ]
Pressurizer safety valves opened	[ ]
Peak main steam system pressure (SG dome) occurred	[ ]

Table 2.3.1-4 Summary of Results for Inoperable MSSV Part-Power Secondary Side Pressure Cases

[ ]

Figure 2.3.1-1 Loss of External Load (Primary Side Pressure) - Reactor Power

Figure 2.3.1-2 Loss of External Load (Primary Side Pressure) - Pressurizer and Peak RCS Pressure

Figure 2.3.1-3 Loss of External Load (Primary Side Pressure) - Pressurizer Liquid Level

Figure 2.3.1-4 Loss of External Load (Primary Side Pressure) - Pressurizer Safety Valve Flow

Figure 2.3.1-5 Loss of External Load (Primary Side Pressure) - RCS Loop Temperatures

[ ]

Figure 2.3.1-6 Loss of External Load (Primary Side Pressure) - RCS Cold Leg Loop Flow Rates

Figure 2.3.1-7 Loss of External Load (Primary Side Pressure) - Steam Line Pressures

Figure 2.3.1-8 Loss of External Load (Primary Side Pressure) - MSSV Flow Rates

Figure 2.3.1-9 Loss of External Load (Primary Side Pressure) - Reactivity Feedback

Figure 2.3.1-10 Loss of External Load (Secondary Side Pressure) - Reactor Power

Figure 2.3.1-11 Loss of External Load (Secondary Side Pressure) - Pressurizer Pressure

Figure 2.3.1-12 Loss of External Load (Secondary Side Pressure) - Pressurizer Liquid Level

Figure 2.3.1-13 Loss of External Load (Secondary Side Pressure) - RCS Loop Temperatures

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Figure 2.3.1-14 Loss of External Load (Secondary Side Pressure) - RCS Cold Leg Loop Flow Rate

Figure 2.3.1-15 Loss of External Load (Secondary Side Pressure) - Main Steam System (SG Dome) Pressures

Figure 2.3.1-16 Loss of External Load (Secondary Side Pressure) - MSSV Flow Rates

Figure 2.3.1-17 Loss of External Load (Secondary Side Pressure) - Reactivity Feedback

2.3.2 Locked Reactor Coolant Pump Rotor

Issue

For asymmetric transients, S-RELAP5 assumes [ ], but does allow [ ]. [ ], however, the methodology is not clear as to whether these assumptions result in an overall conservative analytic approach. In a recent application of the EMF-2310 method reviewed by the staff, comparison to more detailed thermal-hydraulic analyses indicated that the assumptions relied upon in EMF-2310 may not have had the appropriate technical basis.

i. This results in a potentially non-conservative DNBR evaluation.

ii. Standard Review Plan (SRP) 15.3.3/15.3.4 states that system parameters to be reviewed include the core flow and flow distribution. The staff does not believe that the core flow distribution is conservatively modeled.

iii. The staff may consider sensitivity studies using more realistic flow modeling, and supplementation of the reload safety analysis method to reflect the use of appropriately conservative modeling techniques, if necessary.

Disposition

The following is additional information for the NRC regarding St. Lucie Unit 1 EPU LAR Attachment 5, LR Section 2.8.5.3.2, Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break.

The non-LOCA analyses provided in the St. Lucie Unit 1 EPU submittal were performed using the AREVA methodology from EMF-2310(P)(A). The NRC has questioned the application of EMF-2310(P)(A) to certain analyses with respect to obtaining conservative Departure from Nucleate Boiling (DNB) results. For the RCP rotor seizure event, the assumption of cross flow into the affected quadrant in the lower plenum has been questioned. An additional DNB analysis has been performed for St. Lucie Unit 1 to address NRC concerns related to inlet flow asymmetry. For the additional analysis, [

] to therefore produce more conservative DNB results. The revised DNB calculational method will become the analysis of record (AOR) for St. Lucie Unit 1. Details and results are provided below.

Scoping analyses performed for a 2x4 loop Combustion Engineering-Nuclear Steam Supply System (CE-NSSS) plant that is similar to St. Lucie Unit 1 justified an inlet flow asymmetry corresponding to a [ ] as being conservative. This change in the flow and the corresponding DNB modeling has a small adverse impact on the calculated MDNBR because [

]. The scoping study also showed that [

]

A map of the core configuration, showing the impacted region for the flow gradient case, is provided as Figure 2.3.2-1. [

]

The results in Table 2.3.2-1 show that at the time of MDNBR the [

]. The minimum DNBR remains above the limit, resulting in no DNB fuel failures. The radiological dose consequences documented in St. Lucie Unit 1 EPU LAR Attachment 5, LR Section 2.9.2, Radiological Consequences Analyses Using Alternative Source Terms (AST), thus remain bounding for this event.

Table 2.3.2-1 Reactor Coolant Pump Rotor Seizure Results and Comparison to Previous Results

Figure 2.3.2-1 Reactor Coolant Pump Rotor Seizure Inlet Flow Distribution

2.3.3 Control Element Assembly Withdrawal at Power

Issue

Some reactivity and power distribution anomalies can be more severe at lower power levels because the allowable power shape operating space is less restrictive, and potentially more severe transient variations in power distribution can occur at lower power levels. EMF-2310, however, relies on analysis at zero- and full-power levels only, and uses only an array of steady-state power shapes for analysis.

i. This issue may result in a non-conservative DNBR evaluation, the generation of a non-conservative set of core operating limits, and disregard of a potentially limiting primary system pressurization transient.

ii. SRP 15.4.2, "Uncontrolled Control Rod Assembly Withdrawal at Power," Section III, "Review Procedures," Item 1, states: "The review considers the entire power range from low to full power, and the allowed extreme range of reactor conditions during the operating fuel cycle."

iii. Full- and part-power analyses have been provided demonstrating that, for the chosen set of core operating limits, the part-power transients are less severe than the full-power analysis. The staff may consider a proposal to augment the methodology to include consideration of transient power redistribution, and a generic basis for full-power only analysis, or that the methodology be revised to reflect the analysis of intermediate power levels.

Disposition

Part-power analyses, documented in St. Lucie Unit 1 EPU LAR Attachment 5, LR Section 2.8.5.4.2, Uncontrolled Control Rod Assembly Withdrawal at Power, evaluate the challenge to the Specified Acceptable Fuel Design Limits (SAFDL)s as well as the RCS overpressure limit. These analyses conclude that the acceptance criteria are met for events initiated from part-power conditions.

2.3.4 Control Element Assembly Ejection Acceptance Criteria

Issue

The AREVA Control Element Assembly (CEA) ejection analytic method uses an acceptance criterion for fuel cladding mechanical integrity that does not reflect more recently obtained (1994) experimental data.

i. Adherence to the 280 cal/g acceptance criterion may result in a significant underprediction of the radiological consequences of this event.

ii. Information Notice 94-64 discusses data indicating that higher-burnup fuel may fail at significantly lower burnups than the acceptance criterion of 280 cal/g; Appendix B to SRP 4.2 discusses more restrictive interim acceptance criteria; Appendix H.1 to RG 1.183 describes acceptable ways to calculate radiological consequences for fuel failures due to fuel melt and due to cladding failure resulting from departure from nucleate boiling.

iii. The staff may consider a proposal to adhere to more restrictive acceptance criteria and augment the methodology to distinguish between fuel failures due to centerline melt and due to cladding mechanical failure, and treat the radiological consequences appropriately.

Disposition

The CEA Ejection event is discussed in the St. Lucie Unit 1 Updated Final Safety Analysis Report (UFSAR) Chapter 15.4.5. St. Lucie Unit 1 EPU LAR Attachment 5, LR Section 2.8.5.4.6 Spectrum of Rod Ejection Accidents, provided the EPU analysis for the CEA Ejection event and this same acceptance criterion of 280 cal/gm. More recent experimental data shows that 280 cal/gm acceptance criterion for high burned fuel may be non-conservative from fuel coolability considerations and may result in underprediction of fuel failures and the subsequent radiological consequences.

Appendix B to SRP 4.2 discusses more restrictive interim acceptance criteria for reactivity initiated accidents and Appendix H.1 of RG 1.183 provides guidance for calculating radiological consequences for CEA ejection accidents due to fuel melt and fuel cladding failures. Compliance to these criteria for the St. Lucie Unit 1 EPU CEA ejection accident is discussed below. In addition to the results presented in LR Section 2.8.5.4.6, analyses were performed at part power conditions and results are provided in Section 2.3.5.

2.3.4.1 Acceptance Criteria for Fuel Coolability

Per Appendix B to SRP 4.2, the acceptance criterion for coolability is 200 cal/gm. The Hot Zero Power (HZP) and HFP total deposited enthalpy results are provided in LR Tables 2.8.5.6.6-2 and 2.8.5.4.6-3. The total deposited enthalpy results for the part power cases are provided in Table 2.3.5-1. For the events (HZP, part power and HFP) analyzed for the EPU, the total deposited enthalpy is calculated to be less than 170 cal/gm, which is less than the criterion of 200 cal/gm. This criterion is therefore met for St. Lucie Unit 1 EPU.

2.3.4.2 Acceptance Criterion for Cladding Failures

For HZP, the restrictive acceptance criterion for cladding failures, per Appendix B to SRP 4.2, is 150 cal/gm peak radial average fuel enthalpy. As shown in LR Table 2.8.5.4.6-2, the maximum calculated total deposited enthalpy for St. Lucie Unit 1 EPU HZP event is much less than 100 cal/gm, which meets the acceptance criterion of 150 cal/gm.

For at power events, the acceptance criterion for fuel cladding failure, per Appendix B to SRP 4.2, is the local heat flux not exceeding thermal design limit (DNBR). The HFP MDNBR result is provided in LR Table 2.8.5.4.6-3. The MDNBR results for the part power cases are provided in Table 2.3.5-1. For St. Lucie Unit 1 EPU analyses, the MDNBR is calculated to be greater than the DNBR limit for all analyzed power levels, thus meeting the acceptance criteria for cladding failures.

Although no specific limit currently exists for pellet/cladding interaction (PCI) and pellet/cladding mechanical interaction (PCMI) failures, the EPU analyses performed at all power levels show that the enthalpy rise for the peak rods, is below 100 cal/gm, which meets the 150 cal/gm limit depicted in Figure B-1 of Appendix B to SRP 4.2 for lower burned fuel

2.3.4.3 Fuel Centerline Melt

The HZP and HFP fuel centerline temperature results are provided in LR Tables 2.8.5.4.6-2 and 2.8.5.4.6-3. The fuel centerline temperature results for the part power cases are provided in Table 2.3.5-1. For the CEA ejection accident analyses performed for St. Lucie Unit 1 EPU at

HZP, part powers and HFP, the fuel centerline temperature is calculated to be below the centerline melt temperature. Thus there are no fuel melt failures for the EPU CEA ejection accident.

2.3.4.4 Radiological Consequences

The radiological consequences analysis for the CEA ejection accident, described in LR Section 2.9.2, Radiological Consequences Analyses Using Alternative Source Term (AST), is performed consistent with Appendix H.1 to RG 1.183, Alternative Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors. The fuel failures used in this analysis are a total of 10%, which included 9.5% DNB failures and 0.5% fuel melt failures. Since the actual fuel failures calculated for this event are zero, the radiological consequences analysis remains bounding.

2.3.5 Control Element Assembly Ejection at Part-Power

Issue

The AREVA CEA ejection analytic method, by considering only hot full power and hot zero power cases at beginning and end of cycle conditions only, may not cover the full range of extreme conditions permissible throughout the cycle.

i. This issue may result in an underprediction of the radiological consequences of this accident.

ii. SRP Chapter 15.4.8, "Spectrum of Rod Ejection Accidents (PWR)," Section III, "Review Procedures," Items 1.A-D describe the spectrum of possible initial conditions that should be considered in the accident, including zero, intermediate, and full power, possible control rod patterns, reactivity coefficients, and reactivity feedback weighting.

iii. The staff may consider a proposal to augment the methodology to consider more extreme permissible operating conditions than would be covered by the four statepoints currently considered.

Disposition

St. Lucie Unit 1 EPU LAR Attachment 5, LR Section 2.8.5.4.6 Spectrum of Rod Ejection Accidents, covered cases which were initiated from HFP or HZP conditions. A question was raised about whether the radiological dose consequences may be under-predicted by analyzing only HFP and HZP initial conditions. In response to this question, additional analyses to evaluate the potential for fuel failure due to DNB and/or fuel centerline melt (FCM) were conducted. This response gives the results of the analyses performed for potential events which are initiated when a single CEA is ejected from the core during operation at part power conditions.

The part power analysis was performed using the approved EMF-2310(P)(A) methodology for the plant system and core response (including MDNBR and peak fuel centerline temperature), and the approved XN-NF-78-44(NP)(A) methodology for deposited fuel enthalpy. The values used for key input parameters were chosen consistent with these methodologies. The key inputs and assumptions that characterize the analysis of part-power initial conditions relative to the analyses documented in LR Section 2.8.5.4.6 are:

- Initial Conditions - The analysis was performed at initial conditions corresponding to 20% RTP and 70% RTP, with corresponding bounding initial fuel rod hot spot temperatures and maximum core inlet fluid temperatures. Power measurement uncertainties were applied consistent with the initial power level. These power levels were selected based on the COLR power dependent insertion limit (PDIL) breakpoints.

- Core Power Distributions - Initial core hot spot power peaking factors were [ ]. The hot spot power peaking during the event was determined from detailed core neutronic calculations of both pre-ejection and post-ejection conditions.

- Reactivity Feedback - Reactivity feedbacks were modeled that conservatively bounded conditions at both BOC and EOC for each initial condition. [ ]

- Reactor Protection System Trips and Delays - The event is primarily protected by the VHPT. The VHPT setpoints were set to values consistent with the initial power levels, including the trip uncertainty.

- Ejected CEA Worth – [ ]

Four cases were analyzed: (1) 70% RTP initial conditions at BOC, (2) 70% RTP initial conditions at EOC, (3) 20% RTP initial conditions at BOC and (4) 20% RTP initial conditions at EOC. Results are given in Table 2.3.5-1. The peak hot spot centerline temperatures were calculated to be less than the fuel melt temperature limit; thus, no fuel failure is predicted to occur as a result of fuel centerline melting. MDNBRs were calculated to be above the 95/95 critical heat flux (CHF) correlation limit; thus, no fuel failure is predicted to occur as a result of DNB. The total deposited fuel enthalpies were less than the deposited fuel enthalpy limit; thus, no fuel failure is predicted to occur as a result of deposited fuel enthalpy. The results of the part-power cases are bounded by the results of the limiting case (BOC HFP) discussed in LR Section 2.8.5.4.6. Because no fuel failures were predicted to occur, there is no impact on the radiological consequences analysis performed for this event.

Table 2.3.5-1 CEA Ejection at Part Power Key Inputs and Results

2.3.6 Overpressure Protection

Issue

Provide a discussion regarding SRP Section 5.2.2 and crediting the second safety grade trip for overpressure protection.

Disposition

The following information is provided to assist the NRC in the review of the St. Lucie Unit 1 EPU LAR related to overpressure protection documented in St. Lucie Unit 1 EPU LAR Attachment 5, LR Section 2.8.4.2, Overpressure Protection During Power Operation.

Specific review criteria for overpressure protection are contained in SRP Section 5.2.2 and Matrix 8 of RS-001, United States Nuclear Regulatory Commission (USNRC) Review Standard for Extended Power Uprates. St. Lucie Unit 1 was licensed before the SRPs were issued, such that adequate overpressure protection is demonstrated by the UFSAR safety analyses. The specific overpressure protection requirements for St. Lucie Unit 1 are stated in UFSAR, Appendix 5A, Nuclear Steam Supply System Overpressure Protection Report for Florida Power & Light Company St. Lucie Unit No. 1. For primary and secondary overpressure protection, this report concludes, "The steam generators and reactor coolant system are protected from overpressurization in accordance with the guidelines set forth in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III. Peak reactor coolant system and main steam system pressures are limited to 110% of design pressures during worst case loss of turbine-generator load. Overpressure protection is afforded by pressurizer safety valves, main steam safety valves and the reactor protective system".

The overpressure protection analyses credit the high pressurizer pressure safety-grade reactor trip signal and do not credit non-safety components, instrumentation, or controls to mitigate the event. The analyses also do not credit the highly reliable but non-safety grade reactor trip on turbine trip signal, which is the first trip actuated in these analyses. This overall approach of crediting this second trip on high pressurizer pressure, which is a safety-grade trip, is consistent with the current St. Lucie Unit 1 design basis.

St. Lucie Unit 1 EPU LAR Attachment 5, LR Section 2.8.5.2.1, Loss of External Electrical Load, Turbine Trip, and Loss of Condenser Vacuum, and the results in Section 2.3.1 of this document discussed the results of the analyses which produce the limiting peak primary and peak main steam system pressure conditions. The limiting overpressure event is the LOEL.

The LOEL event analyses demonstrate that the plant will continue to have sufficient pressure relief capacity to ensure that primary and main steam system pressure limits will not be exceeded at the EPU conditions. The analyses assume that the reactor is operating at the EPU power level, and that key system and core parameters are biased within their normal operating range to produce the highest anticipated pressure. The analysis credits the safety-grade high pressurizer pressure signal for Reactor Protection System (RPS) trip; however, it does not credit the highly-reliable, non-safety grade reactor trip on turbine trip. In addition, no credit is taken for the steam dump bypass system, the pressurizer sprays, or the pressurizer power-operated relief valves to mitigate the overpressure challenge. Peak RCS pressure was found to be below 110% of design pressure or 2750 psia at the limiting RCS location. Peak main steam system pressure was found to be below 110% of design pressure or 1100 psia in the steam generator dome location.

Therefore, the analysis of the limiting LOEL overpressure event, under EPU conditions, demonstrates that the pressurizer safety valves, main steam safety valves and the reactor protective system provide the requisite overpressure protection during power operation in accordance to the St. Lucie Unit 1 licensing basis

2.3.7 Harsh Condition Uncertainties

Issue

Discuss the application of harsh environment uncertainties to the potentially affected RPS and ESAFS setpoints.

Disposition

The following is additional information for the NRC to assist in the review of the St. Lucie Unit 1 EPU LAR related to the treatment of harsh environment uncertainties applied to RPS and Engineered Safety Feature Actuation System (ESFAS) trip setpoints assumed in the analyses documented in St. Lucie Unit 1 EPU LAR Attachment 5, LR Section 2.8.5.0, Accident and Transient Analyses, for the events that have the potential for developing a harsh containment environment.

Harsh environment uncertainties were applied to the RPS and ESFAS trip setpoints and the uncertainties were credited for events that generate a harsh containment environment. These events include inside containment MSLB, SBLOCA and Large Break Loss-of-Coolant Accident (LBLOCA). A summary of the setpoints and uncertainties applied in the analyses for the events that generate a harsh environment is provided in Table 2.3.7-1 and Table 2.3.7-2. The setpoints and uncertainties modeled in the transient analyses were conservatively applied to provide bounding simulations of the plant response. To the extent that the RPS and ESFAS are credited in the accident analyses, the setpoints have been verified to adequately protect the plant for EPU operation.

Table 2.3.7-3 and Table 2.3.7-4 provide data to supplement LR Section 2.8.5.0.

Table 2.3.7-1 RPS Harsh Condition Setpoints and Events

RPS Trip	Nominal Setpoint	Harsh Condition Uncertainty	Analytical Setpoint	Event(s)
Steam Generator Pressure - Low	≥ 600 psia	200 psi	[ ]	MSLB
Pressurizer Pressure - Low *(TM/LP)*	Min floor = 1,887 psia	80 psi (low pressure)	[ ]	MSLB SBLOCA
Containment Pressure - High	≤ 3.3 psig	1.3 psi	[ ]	MSLB

Table 2.3.7-2 ESFAS Harsh Condition Setpoints and Events

ESFAS Trip	Nominal Setpoint	Harsh Condition Uncertainty	Analytical Setpoint	Event(s)
Main Steam Isolation • Steam Generator Pressure -Low	≥ 600 psia	200 psi	[ ]	MSLB
Auxiliary Feedwater Actuation • Steam Generator Level – Low	≥ 19% NR	14%	[ ]	SBLOCA
Safety Injection • Pressurizer Pressure Low	≥ 1,600 psia	80 psi	[ ]	MSLB SBLOCA LBLOCA
Safety Injection • Containment Pressure - High	≤ 5.0 psig	1.3 psi	[ ]	LBLOCA

NR = Narrow Range

Table 2.3.7-3 RPS Trip Setpoints Summary

Trip	Nominal Trip Setpoint	Normal Uncertainty	Harsh Condition Uncertainty
Power Level – High • Four Reactor Coolant Pumps Operating	≤ 9.61% above thermal power with a minimum setpoint of 15% RTP and a maximum of ≤ 107.0% RTP	3%	No events that generated harsh conditions actuated this trip
Thermal Margin/Low Pressure (TM/LP)	$P_{VAR} = f(T_{IN}$, Power, ASI) Min. floor = 1,887 psia	± 40 psi (Low Pressure) ± 155 psi (P_{VAR})	± 80 psi (Low Pressure)
Reactor Coolant Flow – Low	≥ 95% of four pump design reactor coolant flow	± 4%	No events that generated harsh conditions actuated this trip
Pressurizer Pressure – High	≤ 2,400 psia	± 30 psi[a]	± 80 psi No events that generated harsh conditions actuated this trip
Steam Generator Pressure – Low	≥ 600 psia	± 40 psi (normal) ± 80 psi (high normal)	± 200 psi
Steam Generator Water Level – Low	≥ 20.5% NR (each steam generator)	± 5%	± 14% No events that generated harsh conditions actuated this trip
Steam Generator Pressure Difference – High	≤ 135 psid	± 64 psi (normal) ± 80 psi (high normal)	No events that generated harsh conditions actuated this trip
Containment Pressure – High	≤ 3.3 psig	± 0.55 psi (meas. uncert) ± 1.30 psi (trip uncert.)	N/A ± 1.30 psi

[a] Except for LOEL Main Steam System pressurization events, all other events used a value equal to or higher than 35 psi. These values bound the actual calculated uncertainty which is <30 psi.

Table 2.3.7-4 ESFAS Trip Setpoints Summary

Actuation	Nominal Actuation Setpoint	Normal Uncertainty	Harsh Condition Uncertainty
Main Steam Isolation • Steam Generator Pressure – Low	≥ 600 psia	± 40 psi (normal) ± 80 psi (high normal)	± 200 psi
Auxiliary Feedwater Actuation • Steam Generator Level – Low	≥ 19.0% NR	± 5%	± 14%
Safety Injection • Pressurizer Pressure – Low	≥ 1,600 psia	± 40 psi	± 80 psi
Safety Injection • Containment Pressure - High	≤ 5.0 psig	± 0.55 psi (meas. uncert) ± 1.30 psi (trip uncert.)	N/A ± 1.30 psi

2.3.8 Main Steam Line Break (Mode 3)

Issue

In lower modes, certain trip functions and ESFAS equipment important in the mitigation of the event may be unavailable. Discuss the availability of safety related equipment and demonstrate that the HZP case bounds scenarios initiated from lower modes.

Disposition

The following is additional information for the NRC regarding St. Lucie Unit 1 EPU LAR Attachment 5, LR Section 2.8.5.1.2, Steam System Piping Failures Inside and Outside Containment.

The Main Steam Line Break (MSLB) event is analyzed for return-to-power behavior because it could result in fuel failure due to DNB or FCM. Analyses were performed for St. Lucie Unit 1 at the conditions associated with a proposed EPU. For the EPU, HFP and HZP conditions were analyzed. MSLB in Mode 3 is considered bounded by the HZP cases, with respect to potential fuel failures due to exceeding DNB and FCM limits, as described herein. The scope of this disposition is limited to the fuel response due to a MSLB event occurring from a Mode 3 initial condition.

The main difference between Mode 3 and HZP conditions, with respect to MSLB, is the availability of HPSI system for providing borated water to offset the positive reactivity due to the system cooldown, and consequently decrease the transient power if a return to criticality and power were to occur. In Mode 3 (hot standby), the limiting condition is at a pressurizer pressure just under 1725 psia, with SIAS on low pressurizer pressure bypassed resulting in no HPSI systems available (St. Lucie Unit 1 TS Table 3.3-3), and two boron injection paths available (St. Lucie Unit 1 TS 3.1.2.2). For pressurizer pressures ≥ 1725 psia the availability of HPSI in Mode 3 is the same as HZP conditions and thus the MSLB event at these pressures is no worse, with respect to DNB and FCM, than the analyzed HZP cases.

Two HZP cases were analyzed for EPU conditions (LR Section 2.8.5.1.2):

- Offsite Power is assumed to be available
- Offsite power is assumed to be lost.

Both cases resulted in a return to power and borated flow from one HPSI pump was credited to decrease the power level reached.

The case with offsite power available returned to power and achieved essentially a new "steady-state" condition (reactivity feedback and power are balanced) with core power reaching a plateau (see LR Table 2.8.5.1.2-3 and LR Figure 2.8.5.1.2-20) prior to the HPSI injection of borated water. Thus in Mode 3, if HPSI flow was not available, the peak core power level would not be more adverse than that in the HZP analysis with offsite power available.

For the HZP case with a loss of offsite power, the peak core power is reached prior to injection of any borated HPSI flow (see LR Table 2.8.5.1.2-3 and LR Figure 2.8.5.1.2-30). Thus, the peak power would not be affected if HPSI were not available.

The aforementioned reactivity balance assumed no boron injection and did not take credit for favorable Mode 3 conditions. Per TS requirements in the assumed Mode 3 scenario at least two boron injection paths would be available to provide negative reactivity to decrease the power level reached during the event and to stabilize the plant. Crediting these injection paths for borating the RCS would provide the same effect as that of HPSI. The overall impact on the DNB or FCM due to the assumption of no HPSI flow in Mode 3 is therefore no worse than the HZP cases.

Additionally, the following Mode 3 (Pressurizer Pressure < 1725 psia) conditions are favorable for MSLB in comparison to the HZP cases:

Moderator Temperature Coefficient (MTC)

The HZP analysis used the COLR negative MTC limit. Since MTC becomes most negative at HFP conditions due to the higher operating temperatures and lower coolant densities relative to HZP, the COLR negative MTC limit is conservative for HZP. Compared to the COLR MTC limit, the MTC will be less negative at Mode 3 conditions. This is significant because moderator density feedback is the primary means of reactivity insertion resulting in an erosion of shutdown margin (SDM) and a potential return to power during a MSLB event.

Stuck CEA Assumption and Available Shutdown Margin

- No Stuck CEA

By having all of the CEAs fully inserted and verified, the full SDM is available at the transient initiation without the localized peaking effects of a stuck rod. With no severe power peaking, the conditions for minimum DNB and FCM are much less severe than the analyzed HZP cases.

- 1 Stuck CEA

If there is a stuck CEA, the RCS is borated in excess of the minimum SDM to offset the condition corresponding to the stuck CEA. This means that there is additional negative shutdown reactivity at the beginning of the event (compared to the HZP case) which will also tend to offset the effects of the RCS cooldown and minimize the potential for return to power.

The Mode 3 MSLB event, therefore, remains bounded by the analyzed HZP MSLB cases with respect to potential fuel failures due to exceeding DNB and FCM limits.

2.3.9 Asymmetric Steam Generator Transient

Issue

Provide an analysis of the Asymmetric Steam Generator Transient (ASGT) (i.e., Loss of Load to One Steam Generator) using a justified asymmetric core inlet temperature distribution and consequent core radial power distribution to capture the unique aspects of the ASGT.

Disposition

The following is additional information for the NRC regarding St. Lucie Unit 1 EPU LAR Attachment 5, LR Section 2.8.5.2.5, Asymmetric Steam Generator Transient.

The ASGT or Loss of Load to One SG event was reanalyzed to address a concern dealing with the unique asymmetric characteristics of this event relative to the potential for augmented radial peaking due to the asymmetric core inlet coolant temperatures. Relative to LR Section 2.8.5.2.5, the following key modeling changes were made:

- Core and Reactor Vessel Model - Due to the similarities of this event with the pre-scram phase of a MSLB, the pre-scram MSLB model described in the approved methodology, EMF-2310(P)(A) Revision 1, was used for this analysis. Consistent with the approved methodology, [

  ]

- Core and Reactor Vessel Mixing - In the plant, mixing between the parallel affected and unaffected sectors within the reactor pressure vessel will tend to occur in the lower plenum, the core, and the upper plenum—due to lateral momentum imbalances, turbulence or eddy mixing, and the relative angular positions of the cold legs to the hot legs. Some mixing may also occur in the downcomer. Mixing and/or crossflow acts to reduce the positive reactivity feedback effects—due to a reduced rate and magnitude of cooldown of the unaffected loop. [

  ]

- Reactivity Weighting - Power fractions [

] producing a conservative overall core power response.

- Main Steam Isolation Valve (MSIV) Closure Times - Two cases were analyzed: one case with a nearly instantaneous MSIV closure time and a second case with a maximum MSIV closure time of 6.9 seconds.

- Radial Peaking Augmentation – The asymmetric core inlet temperatures cause a slightly asymmetric core power distribution. A bounding radial peaking augmentation factor, [ ] was applied to the peak rod power for the DNB calculations.

SG-1 is defined as the SG with the closed MSIV and SG-2 is defined as the SG without the closed MSIV. Table 2.3.9-1 provides the sequence of events for both cases. Figure 2.3.9-1 to Figure 2.3.9-10 show the transient responses of key parameters for the case with an instantaneous MSIV closure time and Figure 2.3.9-11 to Figure 2.3.9-20 show the response for the case with a maximum MSIV closure time of 6.9 seconds. [ ] Figure 2.3.9-3 and Figure 2.3.9-13 show the diverging SG pressures. With pressure increasing in SG-1, limited by the opening of the MSSVs, and decreasing in SG-2, the diverging SG pressures produced an asymmetric steam generator pressure trip (ASGPT) signal (Figure 2.3.9-4 and Figure 2.3.9-14). Figure 2.3.9-5 and Figure 2.3.9-15 show the asymmetric core inlet temperatures. The core inlet temperature asymmetry is less than about 3°F at the time of scram for both cases. The asymmetry increases to about 8°F by the time the clad surface heat flux drops to about 90% RTP after the scram. The case with instantaneous MSIV closure had an earlier trip time, but the asymmetry evolved more quickly. The case with a 6.9 second MSIV closure time had a later trip time, but the asymmetry evolved more slowly. [ ]

PRISM calculations based on [ ]. The MDNBR was calculated to be [ ] which is above the 95/95 CHF correlation limit. Due to the relatively benign power excursion and inlet temperature asymmetry to the time of reactor scram and peak rod surface heat flux, the limiting MDNBR is primarily a function of the pressure transient response. Pressure is predicted to increase through the event to the time of reactor scram; thus the minimum pressurizer (and core exit) pressure for the MDNBR analysis occurs at event initiation. The MDNBR was conservatively calculated based on the initial conditions at the event initiation with the bounding augmented radial peaking factor.

Table 2.3.9-1 ASGT: Sequences of Events

Event	MSIV Closure	
	Instantaneous	6.9 sec
	Time (sec)	Time (sec)
Initiation of event (initiation of closure of MSIV on SG-1)	0.0	[ ]
MDNBR occurred (radial peaking augmentation factor conservatively applied to initial conditions)	0.0	[ ]
SG-1 MSIV fully closed	0.01	[ ]
MSSV flow begins for SG-1	1.7	[ ]
ASGPT setpoint reached	3.1	[ ]
Peak core average heat flux occurs	3.5	[ ]
ASGPT occurs (after 0.9 sec. delay)	4.0	[ ]
CEA insertion begins (after 0.5 sec. delay)	4.5	[ ]

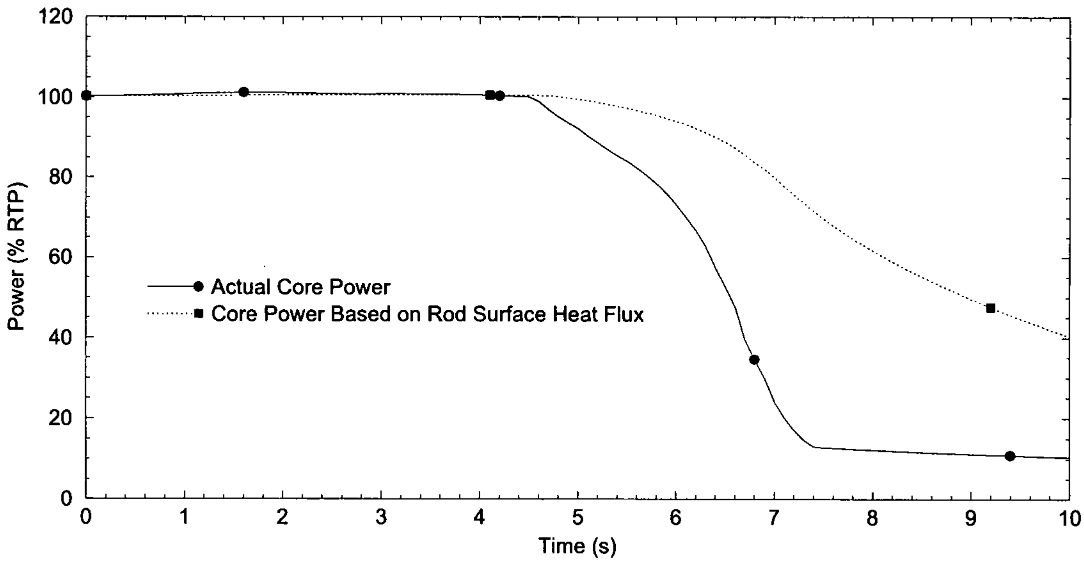

Figure 2.3.9-1 ASGT: Reactor Power (Instantaneous MSIV Closure)

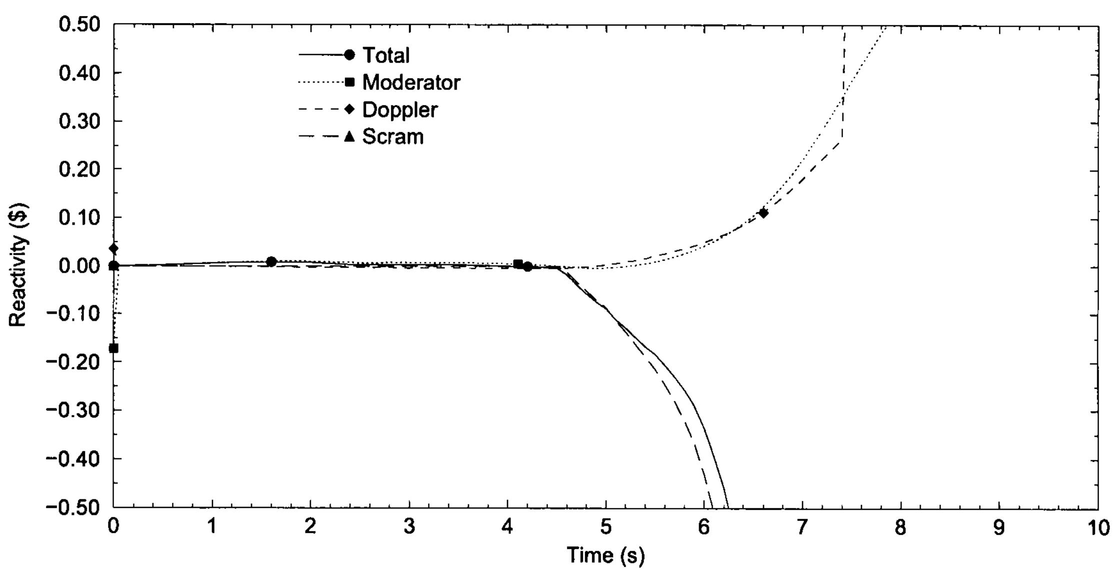

Figure 2.3.9-2 ASGT: Reactivity Feedback (Instantaneous MSIV Closure)

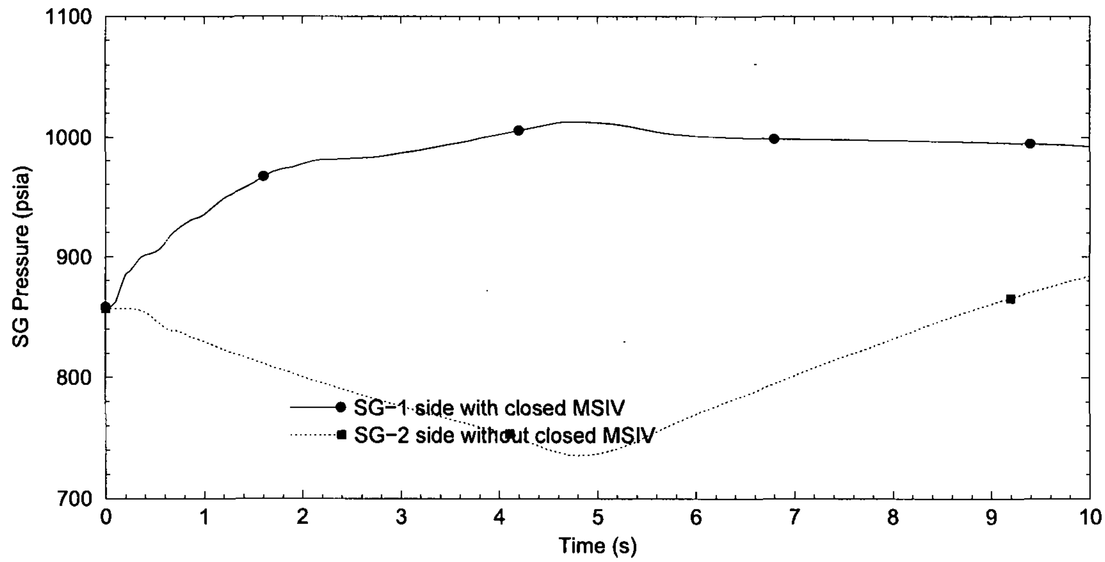

Figure 2.3.9-3 ASGT: SG Pressures (Instantaneous MSIV Closure)

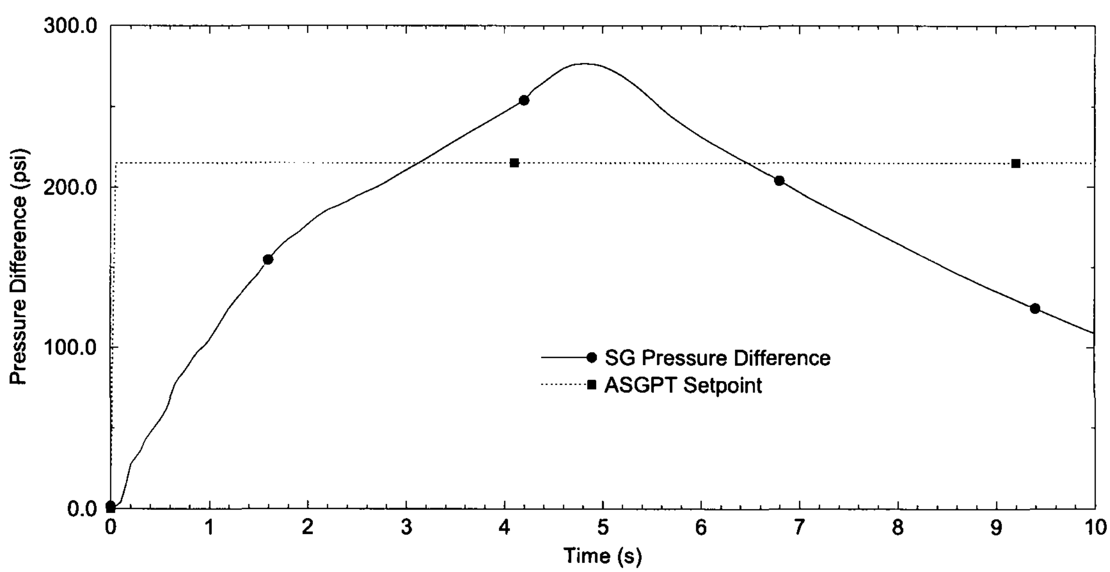

Figure 2.3.9-4 ASGT: SG Pressure Difference vs. ASGPT Setpoint (Instantaneous MSIV Closure)

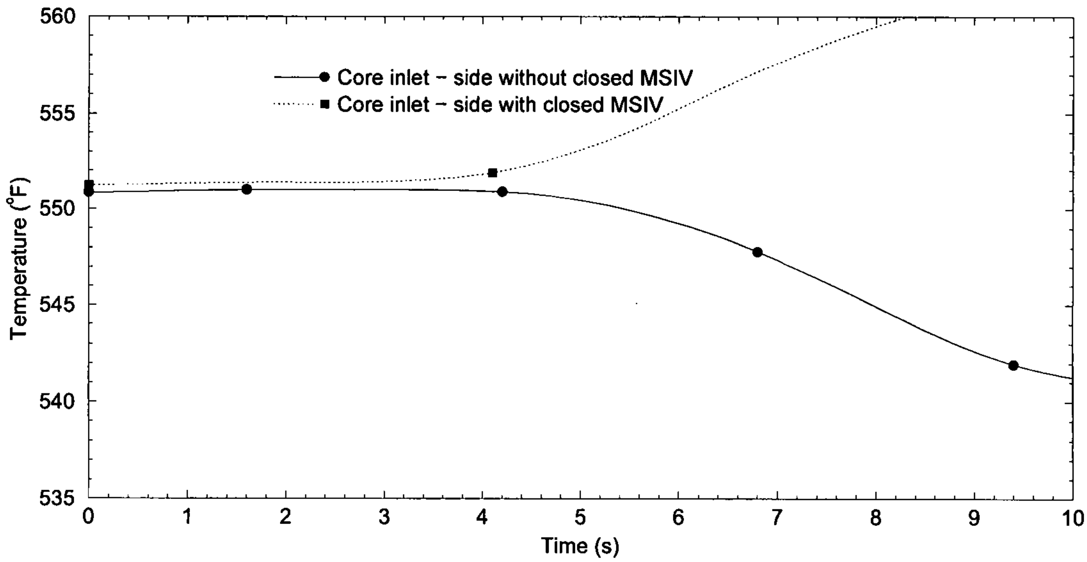

Figure 2.3.9-5 ASGT: Core Inlet Temperatures (Instantaneous MSIV Closure)

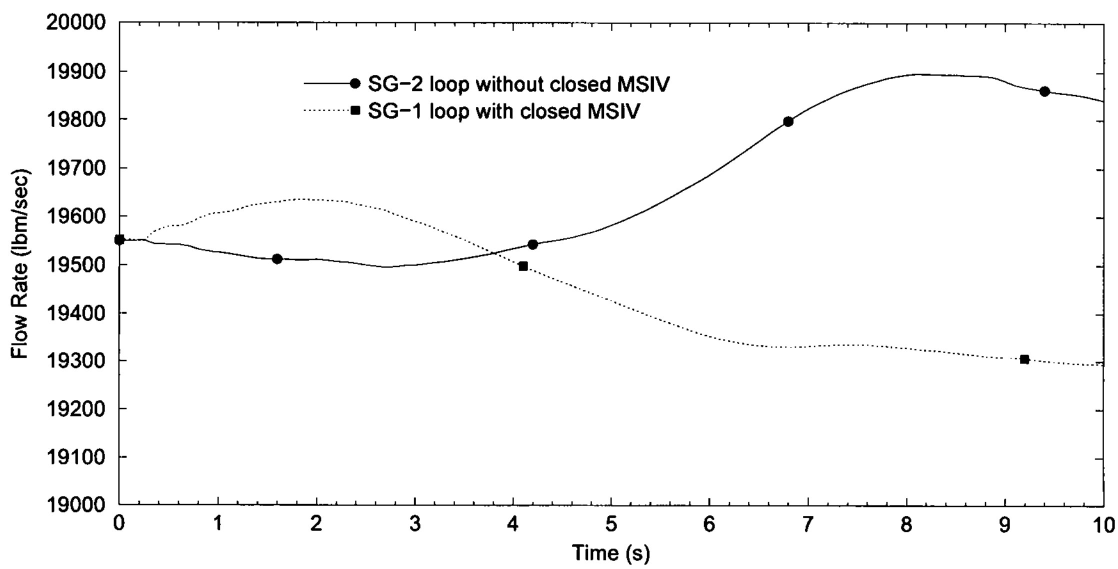

Figure 2.3.9-6 ASGT: RCS Loop Flow Rates (Instantaneous MSIV Closure)

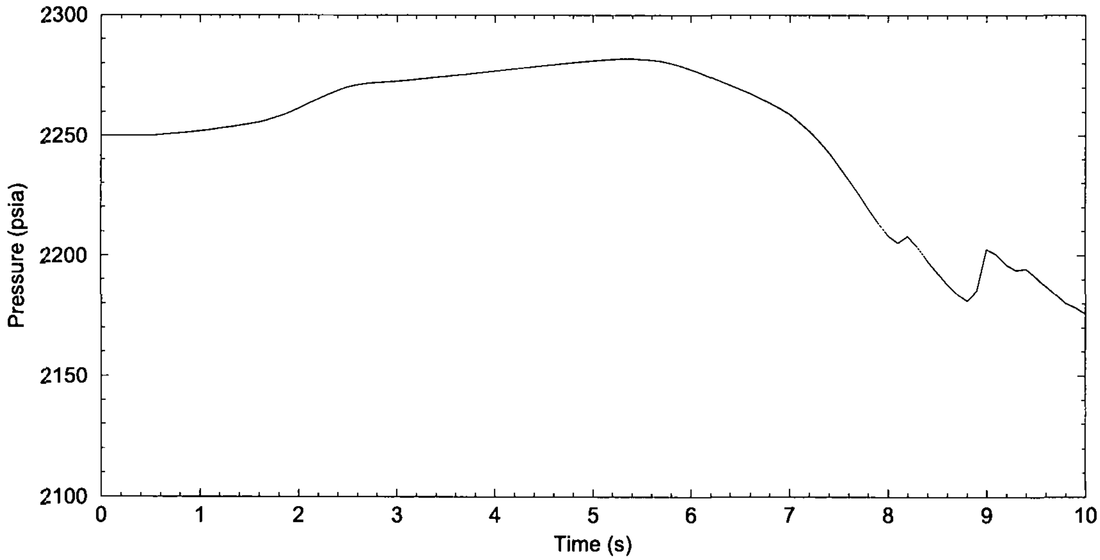

Figure 2.3.9-7 ASGT: Pressurizer Pressure (Instantaneous MSIV Closure)

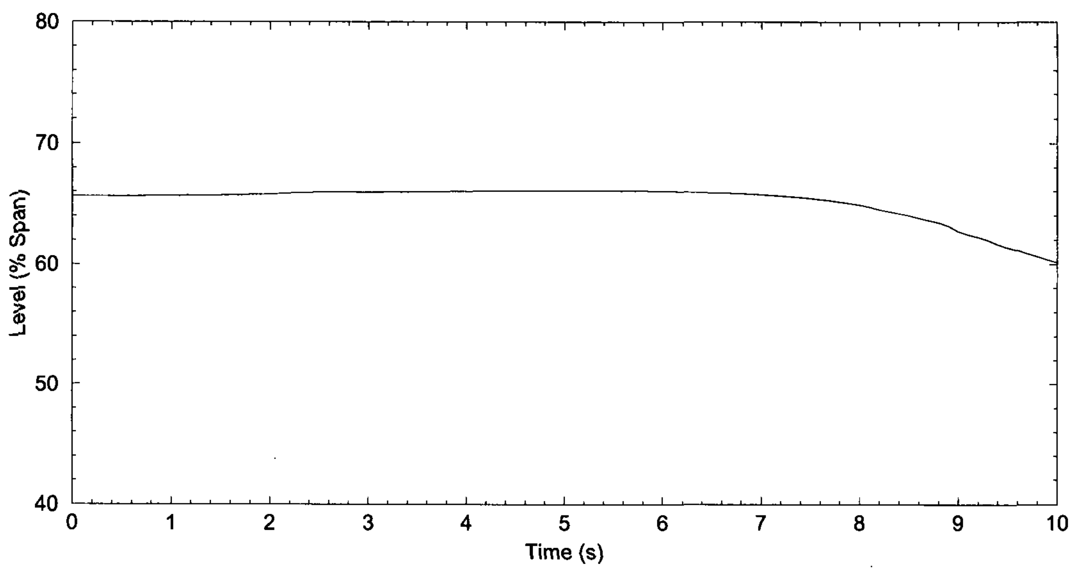

Figure 2.3.9-8 ASGT: Pressurizer Level (Instantaneous MSIV Closure)

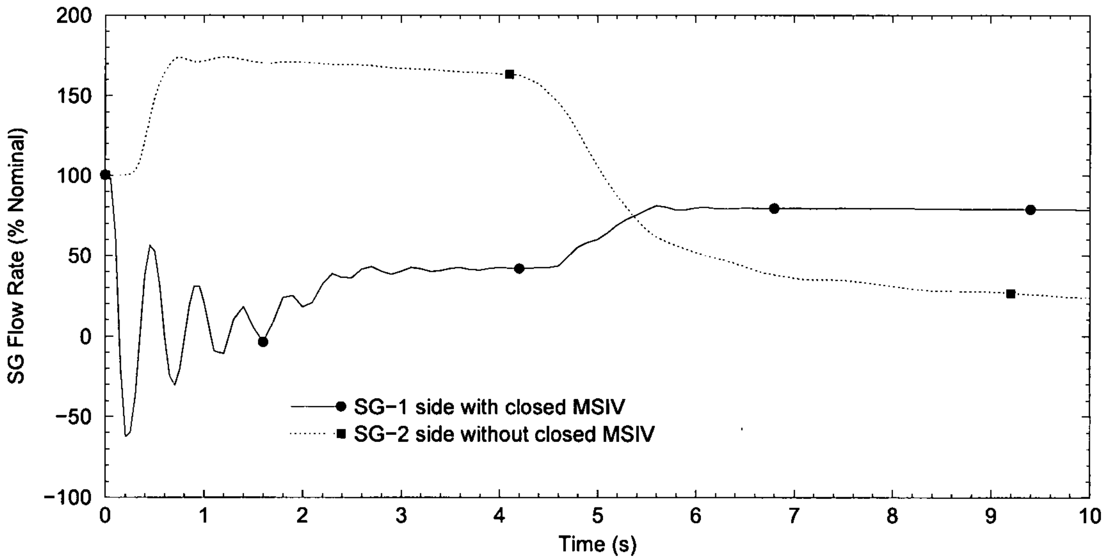

Figure 2.3.9-9 ASGT: Steam Flow Rates (Instantaneous MSIV Closure)

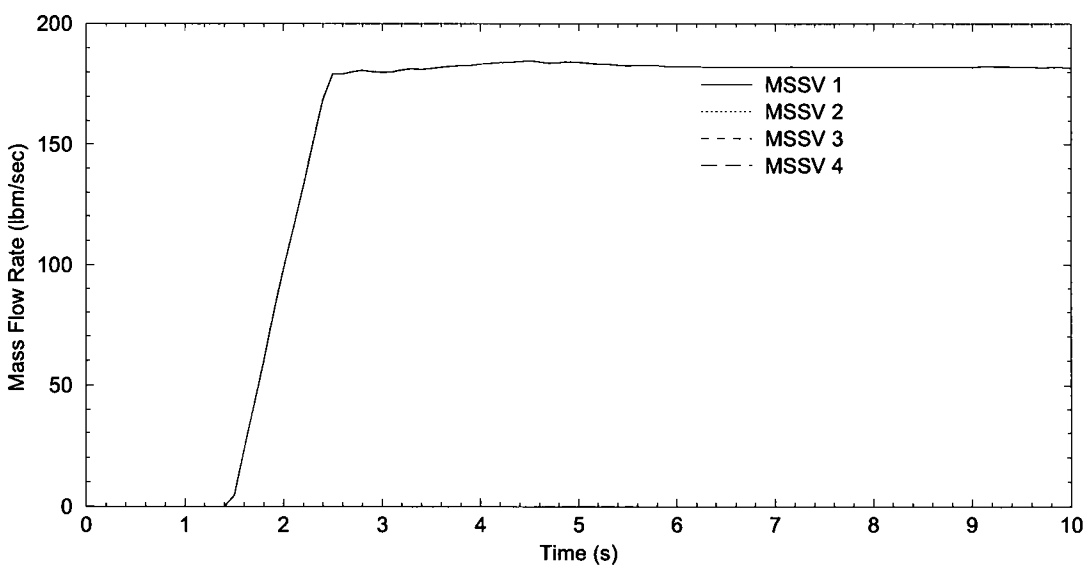

Figure 2.3.9-10 ASGT: MSSV Flows (Instantaneous MSIV Closure)

Figure 2.3.9-11 ASGT: Reactor Power (6.9 sec. MSIV Closure)

Figure 2.3.9-12 ASGT: Reactivity Feedback (6.9 sec. MSIV Closure)

Figure 2.3.9-13 ASGT: SG Pressures (6.9 sec. MSIV Closure)

Figure 2.3.9-14 ASGT: SG Pressure Difference vs. ASGPT Setpoint (6.9 sec. MSIV Closure)

Figure 2.3.9-15 ASGT: Core Inlet Temperatures (6.9 sec. MSIV Closure)

Figure 2.3.9-16 ASGT: RCS Loop Flow Rates (6.9 sec. MSIV Closure)

Figure 2.3.9-17 ASGT: Pressurizer Pressure (6.9 sec. MSIV Closure)

Figure 2.3.9-18 ASGT: Pressurizer Level (6.9 sec. MSIV Closure)

Figure 2.3.9-19 ASGT: Steam Flow Rates (6.9 sec. MSIV Closure)

Figure 2.3.9-20 ASGT: MSSV Flows (6.9 sec. MSIV Closure)

2.3.10 Pressurizer Level Plots for Condition II Events

Issue

Provide pressurizer level plots for all Condition II events to demonstrate that the pressurizer does not overfill.

Disposition

The following is additional information for the NRC to assist in the review of the St. Lucie Unit 1 EPU LAR Attachment 5, LR Section 2.8.5.0, Accident and Transient Analyses, related to the pressurizer level response for Anticipated Operational Occurrences (AOO)s.

The AOOs analyzed for the EPU submittal are:

- Increase in Steam Flow
- Inadvertent Opening of a Steam Generator Relief or Safety Valve
- Loss of External Load
- Loss of Load to One Steam Generator
- Loss of Normal Feedwater Flow
- Loss of Forced Reactor Coolant Flow
- Uncontrolled Control Rod Withdrawal from a Subcritical or Low Power Startup Condition
- Uncontrolled Control Rod Assembly Withdrawal at Power
- Inadvertent Opening of a Pressurized Water Reactor (PWR) Pressurizer Pressure Relief Valve
- CVCS Malfunction event that results in a decrease in boron concentration in the RCS (Boron Dilution)

Table 2.3.10-1 lists the pressurizer level plots for the event analyses presented in LR Section 2.8.5.0. Pressurizer level plots are included in this response for Increase in Steam Flow, Loss of Forced Reactor Coolant Flow and Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low-Power Startup Condition. The CVCS malfunction event that results in a decrease in boron concentration in the RCS is a reactivity addition event which is analyzed with the mass of the RCS and the corresponding pressurizer level remaining essentially unchanged during the event.

Table 2.3.10-1 Pressurizer Level Plots

Event Description	Figure
Increase in Steam Flow • HZP • HFP	 Figure 2.3.10-1 Figure 2.3.10-2
Inadvertent Opening of a Steam Generator Relief or Safety Valve	LR Figure 2.8.5.1.1-26
Loss of External Load • Primary Overpressure • Secondary Overpressure • SAFDL	 LR Figure 2.8.5.2.1-3 & Section 2.3.1, Figure 2.3.1-3 LR Figure 2.8.5.2.1-13 & Section 2.3.1, Figure 2.3.1-12 LR Figure 2.8.5.2.1-23
Loss of Normal Feedwater	LR Figure 2.8.5.2.3-3
Loss of Load to One Steam Generator	LR Figure 2.8.5.2.5-4
Loss of Forced Reactor Coolant Flow	Figure 2.3.10-3
Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low-Power Startup Condition • BOC • EOC	 Figure 2.3.10-4 Figure 2.3.10-5
Uncontrolled CEA Withdrawal at Power • BOC, HFP, SAFDL • EOC, HFP, SAFDL • BOC, HFP, RCS Overpressure	 LR Figure 2.8.5.4.2-4 LR Figure 2.8.5.4.2-11 LR Figure 2.8.5.4.2-17
CEA Drop	LR Figure 2.8.5.4.3-4
Inadvertent Opening of a Pressurizer Pressure Relief Valve	LR Figure 2.8.5.6.1-8

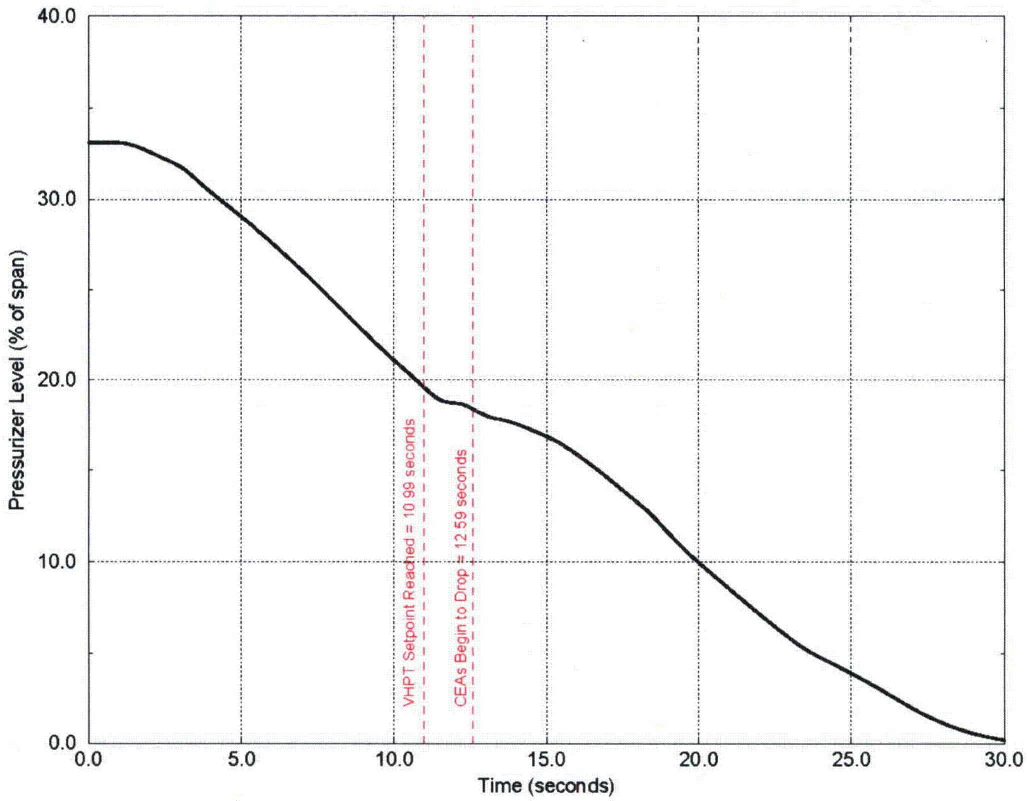

Figure 2.3.10-1 Increase in Steam Flow (HZP): Pressurizer Liquid Level

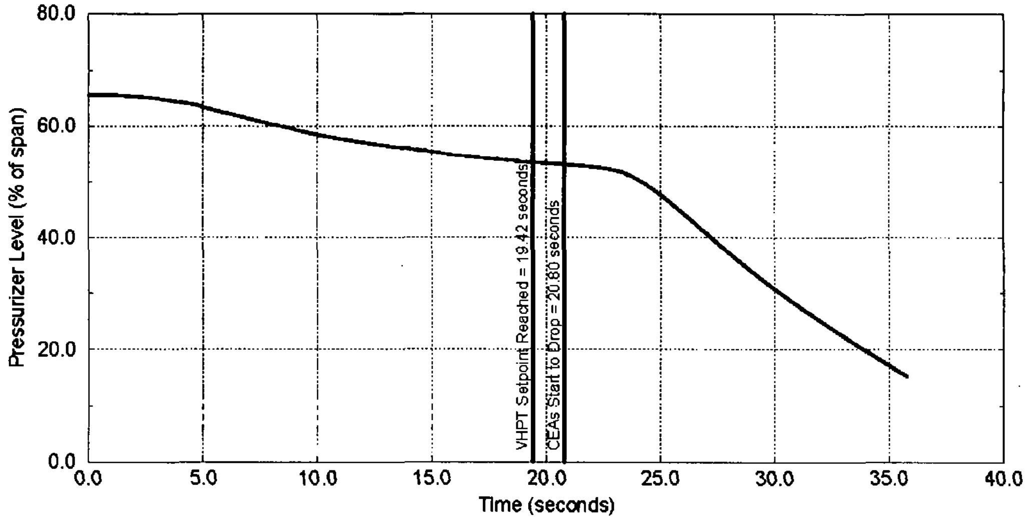

Figure 2.3.10-2 Increase in Steam Flow (HFP): Pressurizer Liquid Level

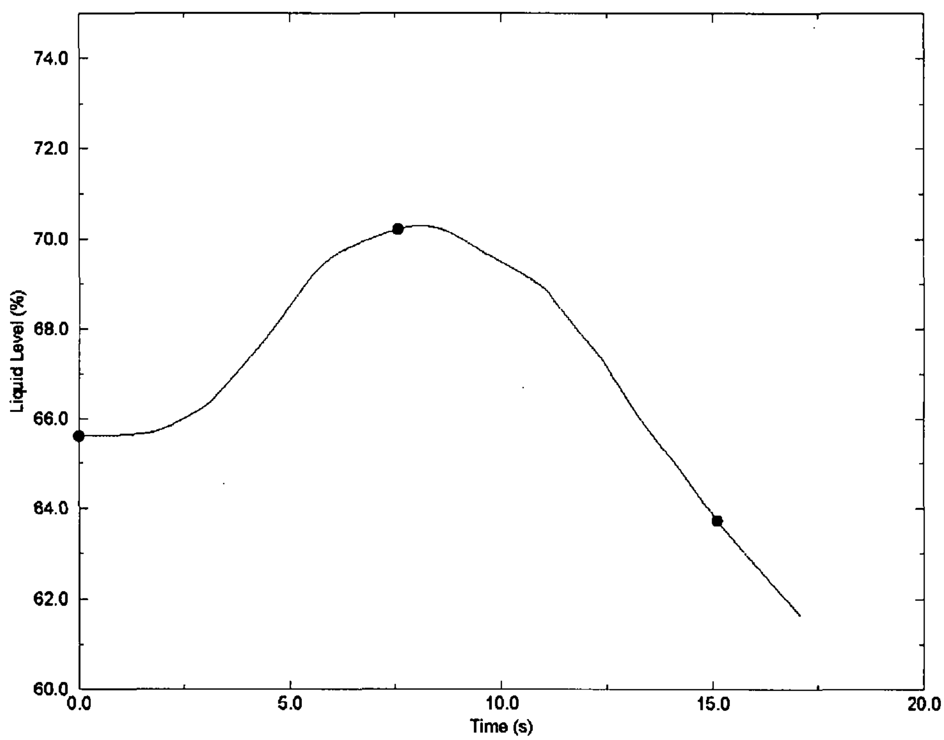

Figure 2.3.10-3 Loss of Forced Reactor Coolant Flow: Pressurizer Liquid Level

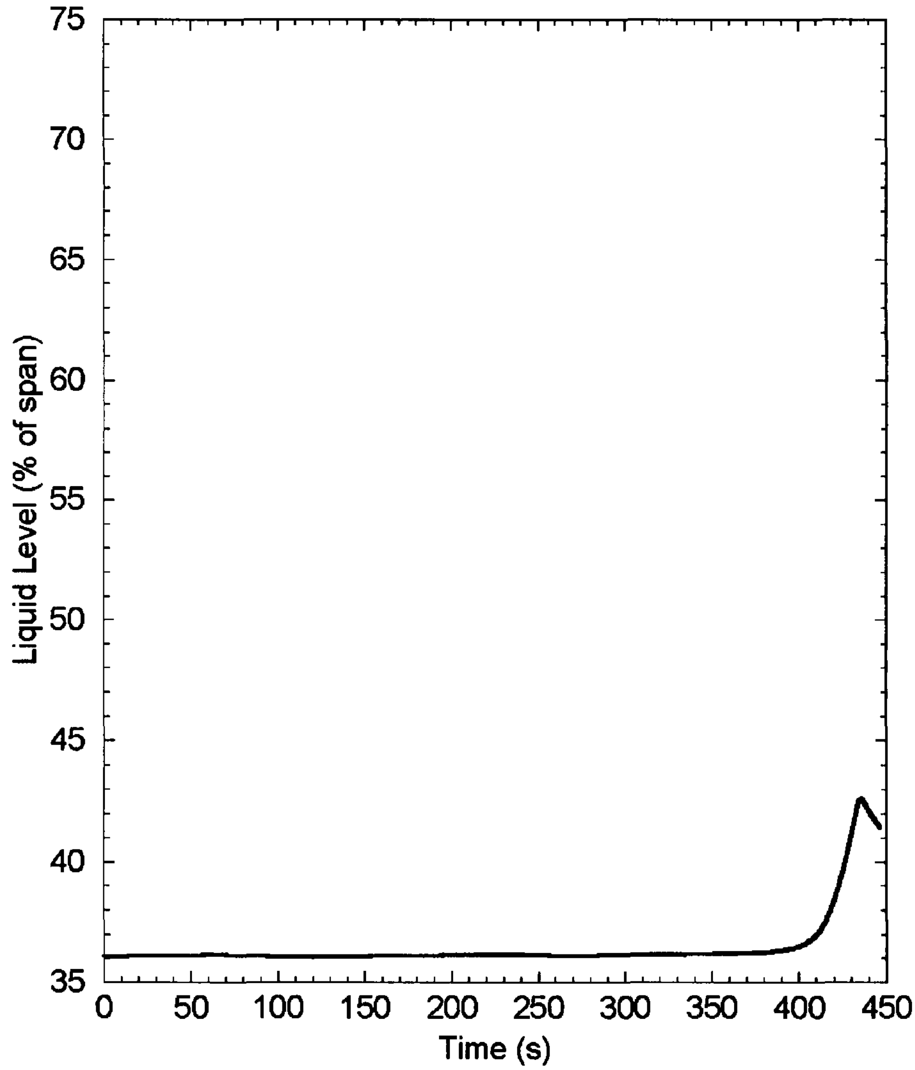

Figure 2.3.10-4 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low-Power Startup Condition: Pressurizer Level (BOC RCS Overpressure)

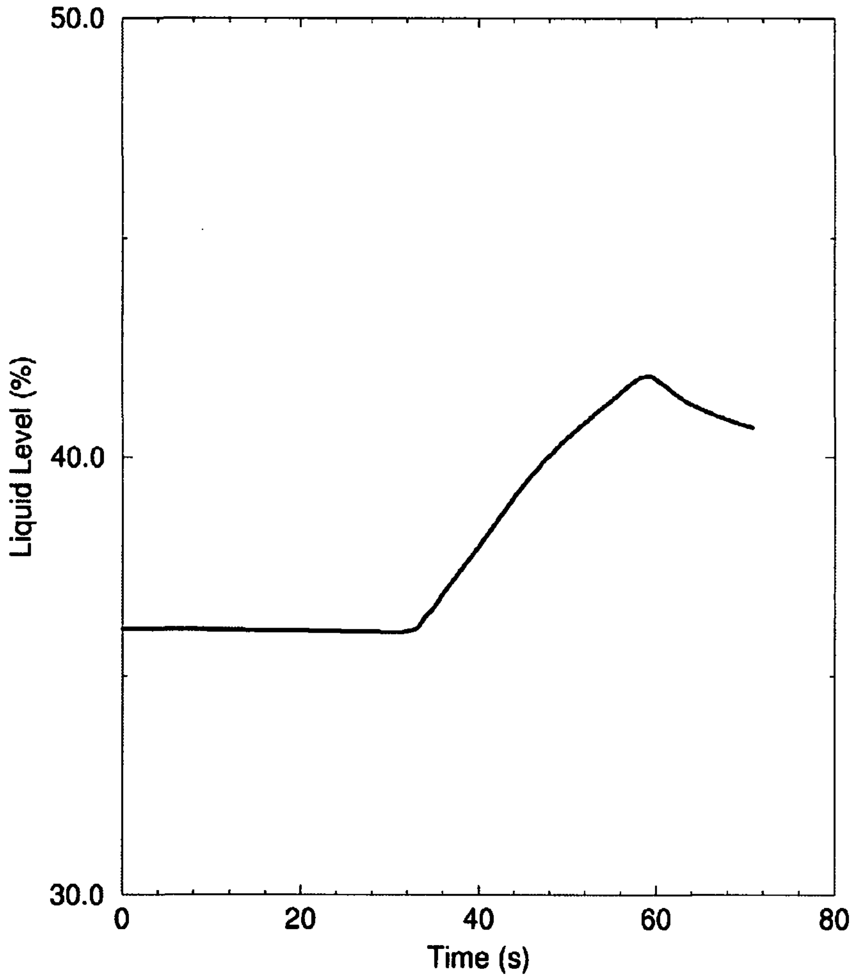

Figure 2.3.10-5 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low-Power Startup Condition: Pressurizer Level (EOC RCS Overpressure)

3.0 References

1. EMF-2103(P)(A) Revision 0, *Realistic Large Break LOCA Methodology for Pressurized Water Reactors*, April 2003

2. EMF-2328(P)(A) Revision 0, *PWR Small Break LOCA Evaluation Model, S-RELAP5 Based*, Framatome ANP, March 2001.

3. EMF-2310(P)(A) Revision 1, *SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors*, Framatome ANP, May 2004.

4. XN-NF-81-58(P)(A) Revision 2 and Supplements 1 and 2, *RODEX2 Fuel Rod Thermal Mechanical Response Evaluation Model*, March 1984.

5. XN-NF-81-58(P)(A) Revision 2 and Supplements 3 and 4, *RODEX2 Fuel Rod Thermal Mechanical Response Evaluation Model*, April 1990.

6. ANP-2903(P) Revision 1, *St Lucie Nuclear Plant Unit 1 EPU Cycle Realistic Large Break LOCA Summary Report with Zr-4 Fuel Cladding*, May 2011.