FPL Energy
Seabrook Station

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November 7, 2005

SBK-L-05226
Docket No. 50-443

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

Seabrook Station
Facility Operating License NPF-86
Completion of License Condition 2.K

References:

1. FPL Energy Seabrook, LLC letter (NYN-04016) to USNRC Document Control Desk, LAR 04-03, "Application for Stretch Power Uprate," dated March 17, 2004.
2. FPL Energy Seabrook, LLC letter (SBK-05054) to USNRC Document Control Desk, "Supplemental Response to Request for Additional Information Regarding License Amendment Request 04-03, Application for Stretch Power Uprate," dated February 25, 2005.
3. USNRC letter to FPL Energy Seabrook, LLC, "Seabrook Station, Unit No. 1 – Issuance of Amendment, RE: 5.2 Percent Power Uprate (TAC No. MIC2364), dated February 28, 2005.

By letter dated March 17, 2004 (Reference 1) and supplemented by letter dated February 25, 2005 (Reference 2), FPL Energy Seabrook, LLC (FPL Energy Seabrook) requested amendment to facility operating license NPF-86 and the Technical Specifications for Seabrook Station. This amendment request was an application for a stretch power uprate which increased the Seabrook Station licensed reactor core power by 5.2% from 3411 megawatts thermal (MWt) to 3587 MWt.

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Nuclear Regulatory Commission (NRC) letter dated February 28, 2005 (Reference 3) issued the license amendment approving the increase in licensed core power for Seabrook Station. In addition to the increase in licensed core power, the license amendment contained license condition 2.K requiring FPL Energy Seabrook to address the Inadvertent Actuation of the Emergency Core Cooling System either through re-analysis using an NRC approved methodology or by qualification of the pressurizer power-operated relief valves for water relief prior to startup from refueling outage 11.

The enclosure to this letter contains the analysis methodology, results, and conclusions for the Inadvertent Actuation of the Emergency Core Cooling System event. This submittal satisfies FPL Energy Seabrook's commitment identified in facility operating license NFP-86 license condition 2.K.

Should you have any questions concerning this LAR, please contact Mr. Stephen T. Hale, Power Uprate Project Manager, at (603) 773-7561.

Very truly yours,

FPL Energy Seabrook, LLC

Gene St. Pierre

Gene St. Pierre
Site Vice President

Enclosure

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Enclosure to Letter SBK-L-05226

Inadvertent Emergency Core Cooling System Initiation at Power Event Evaluation

INADVERTENT EMERGENCY CORE COOLING SYSTEM INITIATION AT POWER

1.0 PURPOSE

The purpose of this evaluation is to demonstrate that an inadvertent Emergency Core Cooling System initiation at power event will not result in the pressurizer becoming water-solid prior to operator action to mitigate and terminate the event. This will prevent challenging the pressurizer power-operated relief valves and the pressurizer safety valves for water relief.

2.0 BACKGROUND

As part of the Seabrook Station stretch power uprate (SPU) license amendment request [Reference 1], the accident analyses were re-analyzed at a core thermal power level of 3659 megawatts thermal (MWt) (3678 MWt Nuclear Steam Supply System power level). The revised analysis of the inadvertent Emergency Core Cooling System initiation at power concluded:

- The departure from nucleate boiling ratio is never less than the initial value. Thus, there will be no cladding damage and no release of fission products to the Reactor Coolant System.
- There will be no water flow through the pressurizer safety valves as a consequence of the event provided that the operators isolate the high head Emergency Core Cooling System injection flowpath prior to exceeding 10.1 minutes. No credit for pressure relief from the pressurizer power-operated relief valves is assumed. The analysis indicates that even though the pressurizer becomes water-solid, the pressurizer pressure at 10.1 minutes is still below the power-operated relief valve setpoint of 2400 psia.
- If an uncertainty is applied to the pressurizer power-operated relief valve setpoint, there is a potential that the power-operated relief valves would open during an inadvertent Emergency Core Cooling System initiation at power event. If the power-operated relief valves were to open and the pressurizer was water-solid, water relief through the power-operated relief valves would occur. Water relief through the power-operated relief valves was considered acceptable because if a power-operated relief valve were to stick open, the power-operated relief valve could be isolated with the power-operated relief valve block valve. Additionally, the valves, piping, and supports are qualified for water.

The Nuclear Regulatory Commission staff identified a concern associated with instrument uncertainties for the Reactor Coolant System pressure and pressurizer level instrumentation. Using the uncertainties, the potential exists for the pressurizer to become water-solid prior to the operator terminating the event. A water-solid pressurizer could result in water relief through the power-operated relief valves and the potential for this Condition II event becoming a Condition III event.

As a result of the concern identified above, FPL Energy Seabrook submitted an interim analysis [Reference 2] based on nominal values. Westinghouse performed a revised analysis of the Seabrook Station inadvertent Emergency Core Cooling System initiation at power event using nominal values for pressurizer pressure, pressurizer level, and heat removal using the steam dumps. The conclusions of the revised analysis using nominal values were:

- The time for the pressurizer to become water-solid is 14.9 minutes which is longer than the 10.1 minutes required for the operator to isolate the high head Emergency Core Cooling System injection flowpath.
- As a result, neither the pressurizer power-operated relief valves nor the pressurizer safety valves are challenged to relieve water within 10.1 minutes following an inadvertent Emergency Core Cooling System initiation at power event.

The NRC accepted the interim analysis for use during one core operating cycle. License Amendment 101 [Reference 3] was issued to increase the core rated thermal power to 3587 MWt. This amendment also included license condition 2.K which stated:

> Prior to startup from refueling outage 11, FPL Energy Seabrook commits to either upgrade the controls for the pressurizer power operated relief valves to safety-grade status and confirm the safety-grade status and water-qualified capability of the pressurizer power operated relief valves, pressurizer power operated relief valve block valves and associated piping or to provide a reanalysis of the inadvertent safety injection event, using NRC-approved methodology, that concludes that the pressurizer does not become water-solid within the minimum allowable and verifiable time for operators to terminate the event.

Section 3.0 below contains the revised analysis for the inadvertent Emergency Core Cooling System initiation at power event necessary to satisfy license condition 2.K.

3.0 INADVERTENT EMERGENCY CORE COOLING SYSTEM INITIATION AT POWER EVENT REVISED ANALYSIS

3.1 ACCIDENT DESCRIPTION

An inadvertent or spurious actuation of the Emergency Core Cooling System at power event results in an increase in the Reactor Coolant System inventory leading to the potential filling of the pressurizer. The inadvertent Emergency Core Cooling System initiation at power event could be caused by operator error or a false electrical actuating signal. Spurious actuation in plants may be caused by any of the following signals:

- High containment pressure
- Low pressurizer pressure
- Low steamline pressure
- Manual actuation

Following actuation of one or more of the above signals, the Emergency Core Cooling System is actuated which results in borated water being pumped from the refueling water storage tank. The centrifugal charging pumps then inject borated water into the cold leg of each Reactor Coolant System loop. The safety injection pumps start automatically, but provide no flow when the Reactor Coolant System is at normal pressure.

Normally, a safety injection signal results in an immediate automatic reactor trip, which in turn generates a turbine trip. However, it cannot be assumed that a single fault that actuated the Emergency Core Cooling System will also produce a reactor trip. Even without an immediate reactor trip, the reactor will experience a negative reactivity excursion as a result of the injected borated water. This negative reactivity excursion results in a decrease in reactor power.

In manual rod control, the power mismatch causes a drop in T_{avg} and a shrinkage of the reactor coolant. Prior to a reactor trip, this results in a decrease in pressurizer pressure and water level and the turbine load will decrease due to the effect of reduced steam pressure on load. The decrease in Reactor Coolant System pressure results in an increase in safety injection flow associated with the Emergency Core Cooling System pump performance characteristics.

In automatic rod control, the above effects may be compensated for by rod cluster control assembly withdrawal as the control system responds to maintain programmed T_{avg}. Once the rods have been fully withdrawn, the event continues as described for operation in manual rod control. The transient is eventually terminated by the reactor protection system low pressurizer pressure trip or by manual trip.

This event is not significant with respect to departure from nucleate boiling, since the conditions resulting from injecting borated water into the Reactor Coolant System are beneficial with respect to departure from nucleate boiling. Depending on the control systems in operation, core power and Reactor Coolant System temperatures either remain near the initial nominal conditions or decrease during this event. The Reactor Coolant System flow remains constant throughout the event. A decrease in Reactor Coolant System pressure is the only condition that may occur which would adversely affect departure from nucleate boiling. However, for the decrease in Reactor Coolant System pressure that may occur, the effects are more than offset by beneficial changes in power and temperature. The net effect is a departure from nucleate boiling ratio that remains near the initial departure from nucleate boiling ratio or increases throughout the event.

The major concern from an inadvertent Emergency Core Cooling System initiation at power event is the potential to violate the Condition II acceptance criterion where *an incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently*. The pressurizer water volume increases for this event as a result of the continuous safety injection flow. Operator action is required to terminate safety injection flow prior to filling the pressurizer or mitigate the consequences of this event by demonstrating that pressurizer valve integrity is maintained with water relief such that the ability to isolate the Reactor Coolant System is preserved, thus demonstrating that this Condition II event does not propagate to a more serious plant

condition. Therefore, to preclude pressurizer filling (and actuation of the pressurizer power-operated relief and safety valves), the event is analyzed to demonstrate that sufficient time is available for the appropriate operator actions to be taken to preclude a pressurizer water-solid condition.

3.2 METHOD OF ANALYSIS

The inadvertent Emergency Core Cooling System initiation at power event is analyzed with a Westinghouse version of the RETRAN-02 computer code [Reference 4]. The RETRAN-02 computer code is a digital computer code, developed to simulate transient behavior in light water reactor systems. The main features of the program include a point kinetics and one-dimensional kinetics model, one-dimensional homogeneous equilibrium mixture thermal-hydraulic model, control system models, two-phase natural convection heat transfer correlations, a non-equilibrium pressurizer model, etc. The code computes pertinent plant variables including temperatures, pressures and power level.

For maximizing the potential for pressurizer filling (and the possibility for subsequent water relief through the pressurizer power-operated relief valves and pressurizer safety valves), the most limiting case is the maximum reactivity feedback condition with an immediate reactor and turbine trip on safety injection.

The major assumptions in the revised analysis are the same as those in the original SPU analysis except as noted in Table 1 below.

To ensure that the pressurizer does not become water-solid for this event, operator actions are modeled. These operator actions include terminating the flow from all but one centrifugal charging pump at 9 minutes after the beginning of the event, and termination of all charging flow at 13 minutes into the event based on analyzed maximum fill rate. The timing for all operator actions assumed in the analysis has been confirmed through time studies on the plant simulator in September and October 2005. The approach to terminating this event is consistent with Westinghouse NSAL-93-013, "Inadvertent ECCS Actuation at Power" [Reference 5], Option II.

3.3 RESULTS

Table 1 below compares the assumptions in the analysis described above with the original SPU LAR analysis and the subsequent interim analysis. Table 2 below presents the sequence of events for the inadvertent Emergency Core Cooling System initiation at power event as described above. Figures 1 through 4 below show the evolution of the key parameters during the transient.

Reactor trip occurs at event initiation followed by a rapid initial cooldown of the Reactor Coolant System. Coolant contraction results in a short-term reduction in pressurizer pressure and water level. The combination of the Reactor Coolant System heatup, due to residual Reactor Coolant System heat generation, and Emergency Core Cooling System injected flow causes the pressure and level transient to rapidly turn around. Pressurizer water level then increases throughout the transient. Per Emergency Operating Procedures, Reactor Coolant System temperature (T_{avg}) is maintained at 557°F through heat removal from the steam generators using the steam generator atmospheric steam

dump valves, and flow from all but one centrifugal charging pump is terminated early in the event. These steps, which are already part of the emergency operating procedure, were not credited in the analysis submitted with Seabrook Station LAR 04-03, "Application for Stretch Power Uprate" [Reference 1], which assumed the pressurizer became water-solid shortly before 9 minutes into the transient.

The revised analysis credits heat removal through the steam generators using the atmospheric steam dump valves, stopping of all but one centrifugal charging pump at 9 minutes after the beginning of the event, and termination of all charging flow at 13 minutes into the event based on analyzed maximum fill rate. The results of the revised analysis indicate that at no time does the pressurizer become water-solid.

3.4 CONCLUSIONS

Results of the analysis of the inadvertent Emergency Core Cooling System initiation at power event demonstrates that there is no hazard to the integrity of the Reactor Coolant System. This analysis was performed using Nuclear Regulatory Commission approved methodologies and is consistent with the requirements of NUREG-0800. This approach to terminating this event is also consistent with Option II of Westinghouse NSAL 93-013 [Reference 5].

The departure from nucleate boiling ratio is never less than the initial value. Thus, there will be no cladding damage and no release of fission products to the Reactor Coolant System.

Operator action terminating safety injection flow is sufficient to preclude a pressurizer water-solid condition and prevent actuation of the pressurizer power-operated relief and safety valves. By demonstrating that sufficient time is available for the appropriate operator actions to preclude a pressurizer water-solid conditions, the pressurizer valve integrity can be maintained for the inadvertent Emergency Core Cooling System initiation at power event. No credit for operation of the pressurizer power-operated relief valves was assumed. Therefore, the ability to isolate the Reactor Coolant System and maintain the integrity of the Reactor Coolant System pressure boundary confirms that this event does not lead to a more serious plant condition, hence demonstrating acceptability of the Condition II acceptance criteria.

TABLE 1 ASSUMPTIONS			
Parameter	**SPU Analysis**	**Interim Analysis**	**Revised Analysis**
RCS pressure	2200 psia (2250 psia – 50 psia uncertainty)	2250 psia	2200 psia (2250 psia – 50 psia uncertainty)
Pressurizer level	65%	60%	65%
Secondary heat removal	Main steam safety valves	Steam dump valves to the condenser	Atmospheric steam dump valves
Charging pump operation	Full charging flow	Full charging flow	All but one centrifugal charging pump stopped at 9 minutes All charging flow stopped at 13 minutes (Note 1)
Event termination	Safety injection cross-tie valves isolated in 10.1 minutes	Safety injection cross-tie valves isolated in 10.1 minutes	All charging flow stopped in 13 minutes (Note 1)
Acceptance criteria	Do not lift pressurizer safety valves	Do not go water-solid in the pressurizer	Do not go water-solid in the pressurizer
Time to water-solid pressurzier	8.9 minutes	14.9 minutes (Assumed normal charging flow after termination)	>14 minutes (Note 1) (Assumed single charging pump flow after 9 minutes without stopping at 13 minutes)
Note 1 – At analyzed maximum fill rate.			

TABLE 2 SEQUENCE OF EVENTS	
Event	**Time (seconds)**
Inadvertent Emergency Core Cooling System initiation	0.0
Reactor trip	0.0
Turbine trip	0.0
Terminate flow from all but one centrifugal charging pump	540.0
Terminate all charging flow	780.0
Peak pressurizer water volume occurs (6.0 cu. ft. below pressurizer fill)	1712.1

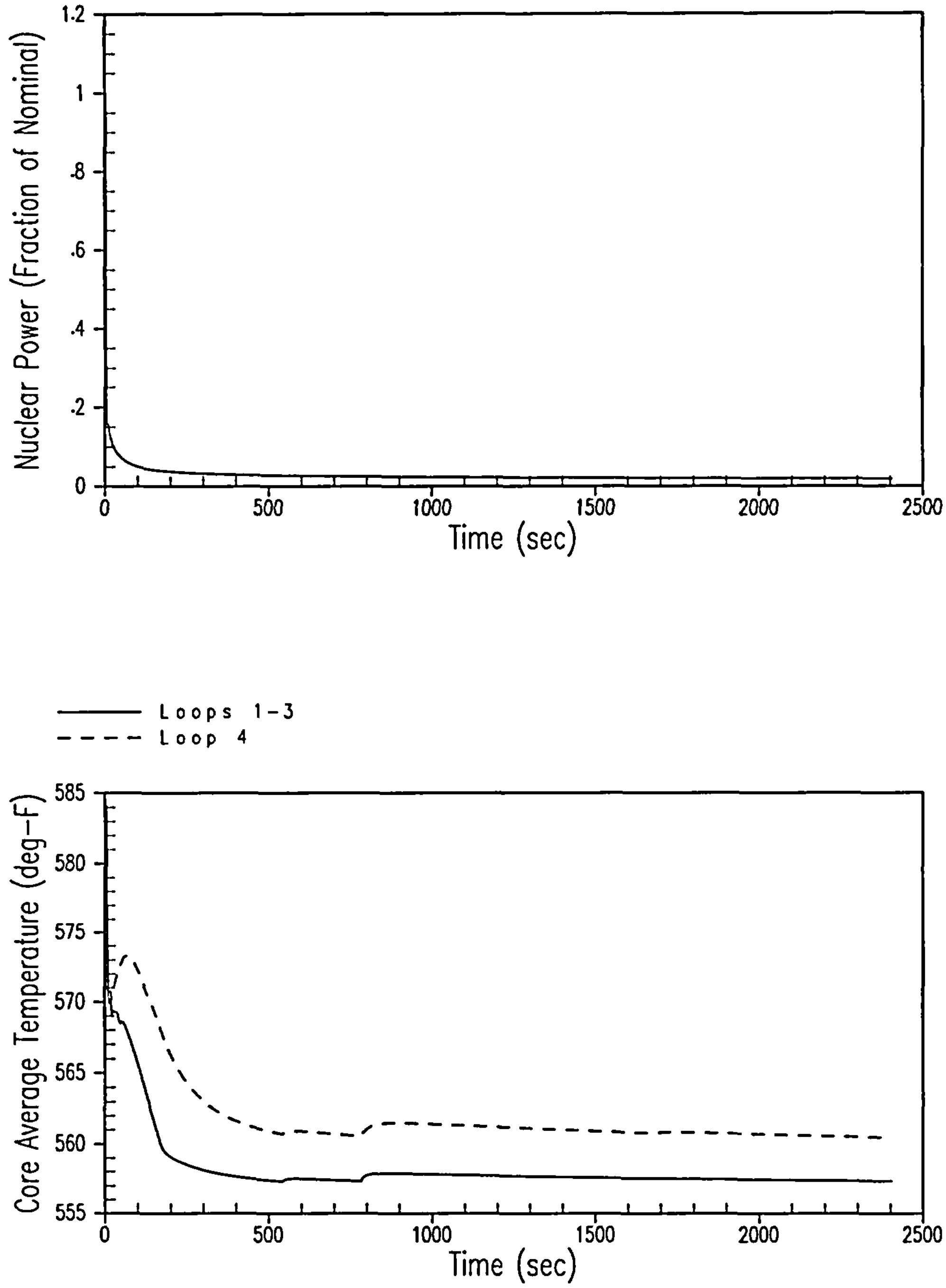


Figure 1
Nuclear Power and Vessel Average Temperature versus time

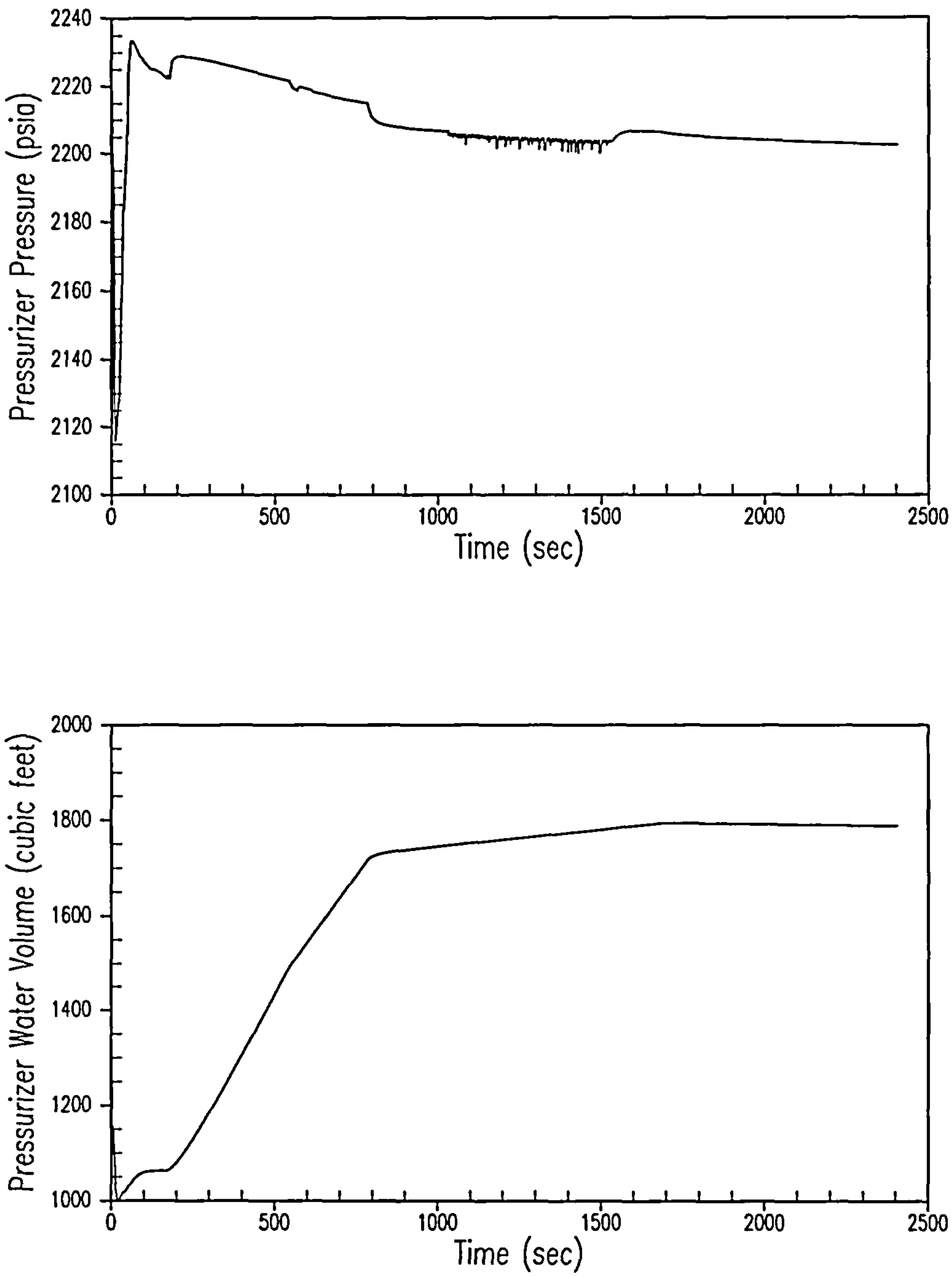
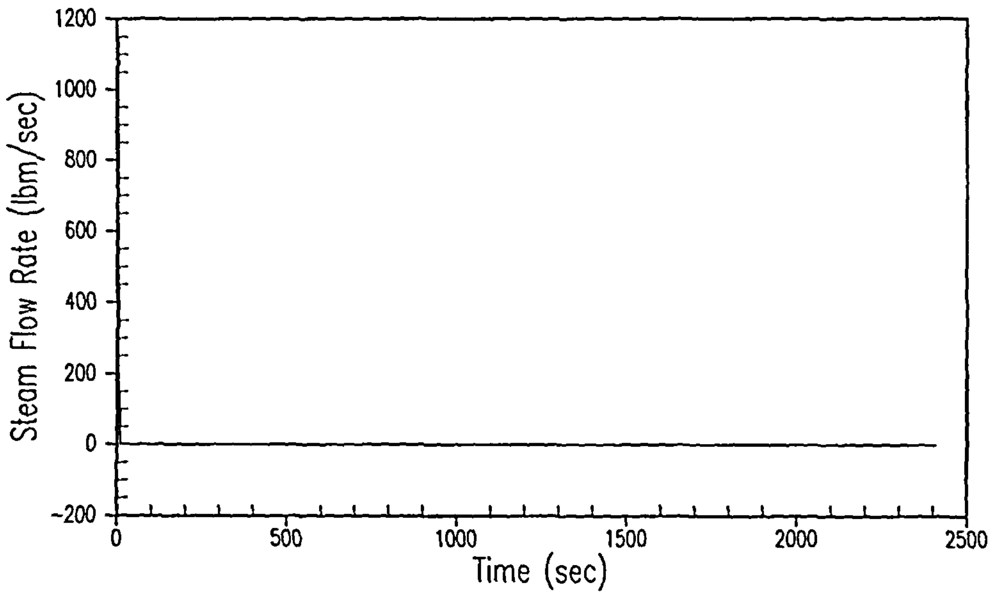


Figure 2
Pressurizer Pressure and Pressurizer Water Volume versus time



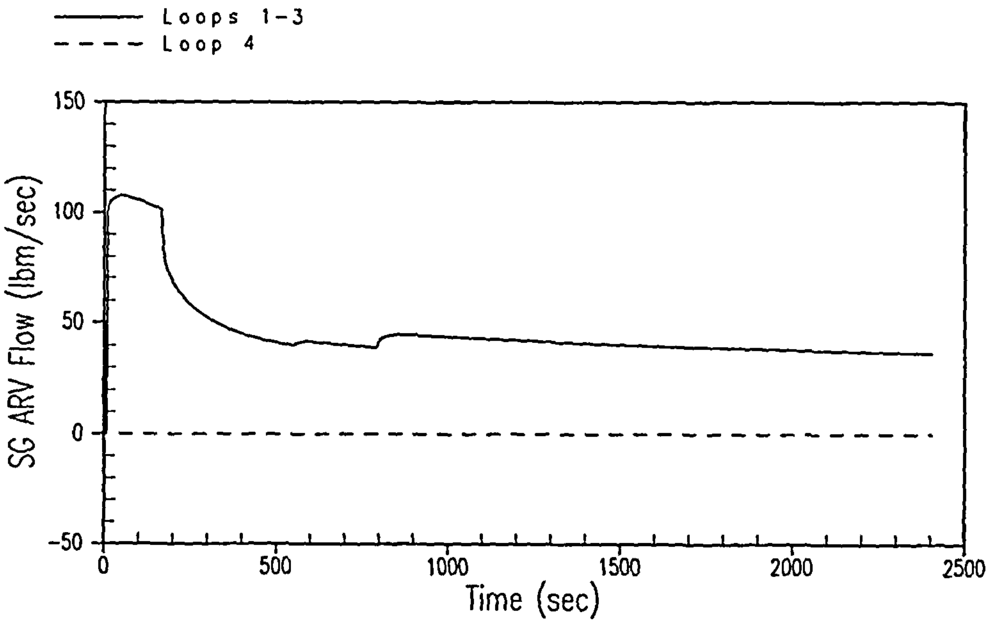
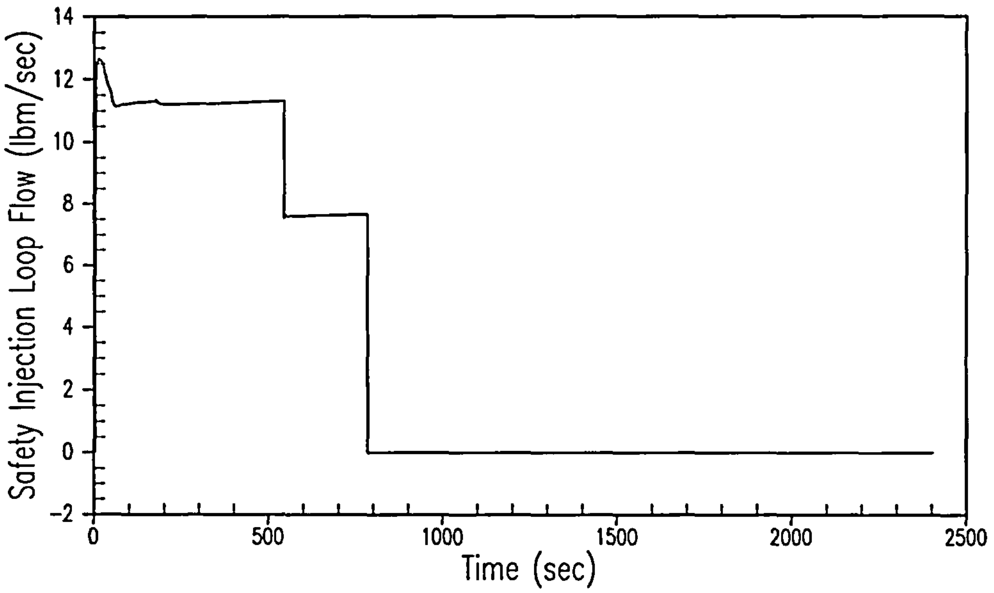


Figure 3
Main Steam Flow and SG Atmospheric Relief Valve Flow versus time



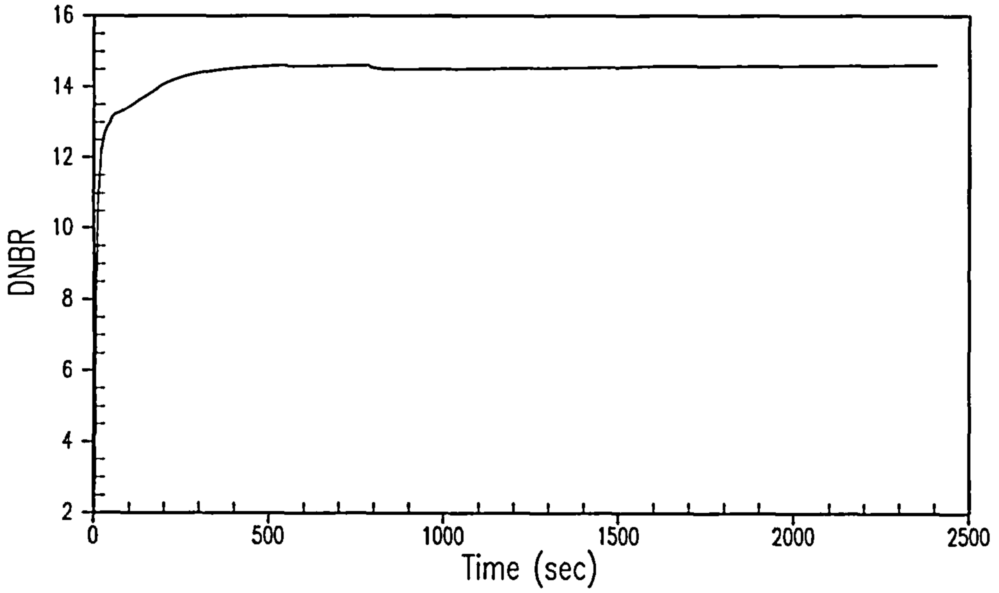


Figure 4
Safety Injection Flow and DNBR versus time

4.0 ADDITIONAL INFORMATION

4.1 OPERATOR TRAINING AND PERFORMANCE

Operators respond to the inadvertent Emergency Core Cooling System initiation at power event in accordance with approved operating procedures. Operations Department procedure E-0, "Reactor Trip or Safety Injection" contains the steps that operators would take to terminate an inadvertent Emergency Core Cooling System initiation at power event. Step 12 of E-0 checks if safety injection is required. If plant conditions indicate that safety injection is not required, then operators are directed to reset the safety injection signal, maintain Reactor Coolant System subcooling greater than 40°F, and stop all but one centrifugal charging pump. In addition, procedure E-0 is being revised to add new Steps 13 through 15 that require the operator to: 13) terminate all charging pump flow to eliminate all mass addition into the Reactor Coolant System, 14) establish normal charging flow, and 15) establish Reactor Coolant System letdown path. A review of operator performance on the simulator in response to the inadvertent Emergency Core Cooling System initiation at power event has verified that operators can secure all but one centrifugal charging pump in 9 minutes and terminate all charging pump flow well before the pressurizer becomes water-solid (>14 minutes based on analyzed maximum fill rate).

TABLE 3
OPERATOR RESPONSE TIME STUDIES

Operator Action	Analysis Requirement	Average Operating Crew Response Time
One charging pump stopped	<9 min.	8 min. 6 sec.
All charging pump stopped	<13 min	10 min. 2 sec.

Operators maintain proficiency with terminating an inadvertent Emergency Core Cooling System initiation at power event through periodic simulator training scenarios. Licensed operators receive simulator training on this and other events during periodic training periods.

4.2 CORE DAMAGE FREQUENCY

The initiating event frequency for inadvertent Emergency Core Cooling System initiation at power event is 2.8E-2/year. The resultant core damage frequency is 1.67E-7/year, which is less than 1% of the total core damage frequency. These values include external event contributions.

The core damage frequency accounts for possible core damage scenarios that initiate from an inadvertent Emergency Core Cooling System initiation at power event. This includes such possibilities as pressurizer power-operated relief valve opening with failure to reseat and failure of the block valve, failure of the pressurizer power-operated relief

valve to open with subsequent failure of the safety valves to reseat, failure of the operators to terminate the event prior to pressurizer going solid, etc. It also includes random equipment failures that are unrelated to the initiator.

4.3 PUBLIC HEALTH AND SAFETY

The inadvertent Emergency Core Cooling System initiation at power event does not result in the release of any radioactive materials or radioactivity. There are no radiological consequences as a result of this event. Therefore, there is no reduction in the level of protection provided to the public health and safety from an inadvertent Emergency Core Cooling System initiation at power event at Seabrook Station.

5.0 SUMMARY AND CONCLUSION

Results of the revised analysis of the inadvertent Emergency Core Cooling System initiation at power event demonstrates that there is no hazard to the integrity of the Reactor Coolant System. This analysis was performed using Nuclear Regulatory Commission approved methodologies and is consistent with the requirements of NUREG-0800. This approach to terminating this event is also consistent with Option II of Westinghouse NSAL 93-013 [Reference 5]

The revised analysis credits heat removal through the steam generators using the atmospheric steam dump valves and the stopping all but of one centrifugal charging pump at 9 minutes into the transient. Prior to the pressurizer becoming water-solid (>14 minutes based on analyzed maximum fill rate), operator actions are assumed to terminate all charging flow. Thus, at no time does the pressurizer become water-solid, and the analysis demonstrates that there is no hazard to the integrity of the Reactor Coolant System.

A review of operator performance on the simulator in response to the inadvertent Emergency Core Cooling System initiation at power event has verified that operators can secure all but one centrifugal charging pump in less than 9 minutes and terminate all charging pump flow well before the pressurizer becoming water-solid (>14 minutes based on analyzed maximum fill rate).

The departure from nucleate boiling ratio is never less than the initial value. Thus, there will be no cladding damage and no release of fission products to the Reactor Coolant System.

Analytical results show that the pressurizer will not become water-solid as a consequence of inadvertent Emergency Core Cooling System initiation at power operation provided that operators perform actions to terminate charging flow within the required time frame. No credit for operation of the pressurizer power operated relief valves was assumed.

6.0 REFERENCES

1. FPL Energy Seabrook, LLC letter (NYN-04016) to USNRC Document Control Desk, LAR 04-03, "Application for Stretch Power Uprate," dated March 17, 2004.
2. FPL Energy Seabrook, LLC letter (SBK-05054) to USNRC Document Control Desk, "Supplemental Response to Request for Additional Information Regarding License Amendment Request 04-03, Application for Stretch Power Uprate," dated February 25, 2005.
3. NRC letter to FPL Energy Seabrook, LLC, "Seabrook Station, Unit No. 1 – Issuance of Amendment, RE: 5.2 Percent Power Uprate (TAC No. MIC2364), dated February 28, 2005.
4. WCAP-15234-A (Non-proprietary), RETRAN-02 "Modeling and Qualification for Westinghouse Pressurized Water Reactor Non-LOCA Safety Analyses," D. S. Huegel, et. al., May 1999.
5. Westinghouse Nuclear Safety Advisory Letter NSAL-93-013, "Inadvertent ECCS Actuation at Power," dated June 30, 1993.