FPL Energy
Seabrook Station

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February 25, 2005

Docket No. 50-443
SBK-L-05054

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

Seabrook Station
Facility Operating License NPF-86
Supplemental Response to Request for Additional Information Regarding
License Amendment Request 04-03, Application for Stretch Power Uprate

References:

1. FPL Energy Seabrook, LLC letter NYN-04016, LAR 04-03, "Application for Stretch Power Uprate," dated March 17, 2004.

2. NRC letter to FPL Energy Seabrook, LLC, "Seabrook Station Unit 1 – Request for Additional Information for Proposed Amendment Request Regarding the Application for Stretch Power Uprate (TAC MC2364)," dated August 18, 2004.

3. FPL Energy Seabrook, LLC letter SBK-L-04072, "Response to Request for Additional Information Regarding License Amendment Request 04-03, Application for Stretch Power Uprate," dated October 12, 2004.

By letter dated March 17, 2004 (Reference 1), 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 (LAR) is an application for a stretch power uprate which will increase the Seabrook Station licensed reactor core power by 5.2% from 3411 megawatts thermal (MWt) to 3587 MWt.

In Reference 2, the NRC requested additional information to support its review of Seabrook Station LAR 04-03. By letter dated October 12, 2004 (Reference 3), FPL Energy Seabrook provided its responses to the requests for additional information (RAIs) provided in your correspondence.

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Based on requests from the NRC staff during teleconferences on December 7, 2004, January 5, and 11, February 3, 16, and 24, 2005, FPL Energy Seabrook is providing additional information related to the inadvertent Emergency Core Cooling System (ECCS) initiation at power event and the Loss of External Load event.

This supplemental information provides an additional evaluation (Enclosure 1) of the inadvertent ECCS event at power for the SPU. Enclosure 2 contains a FPL Energy Seabrook commitment for a License Condition regarding the inadvertent ECCS initiation at power event with implementation prior to startup following the Fall 2006 (OR11) refueling outage. Enclosure 3 provides graphs of plant parameters associated with the interim evaluation for the inadvertent ECCS initiation at power event. Enclosure 4 provides the results of analysis for the Loss of External Load event crediting the second safety grade reactor trip signal.

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

Mark E. Warner
Site Vice President

Enclosures (4)

cc: S. J. Collins, NRC Region I Administrator
V. Nerses, NRC Project Manager, Project Directorate I-2
G. T. Dentel, NRC Resident Inspector

Mr. Bruce Cheney, Director
New Hampshire Bureau of Emergency Management
State Office Park South
107 Pleasant Street
Concord, NH 03301-3809

Oath and Affirmation

I, Mark E. Warner, Site Vice President of FPL Energy Seabrook, LLC hereby affirm that the information and statements contained within the supplemental information regarding License Amendment Request 04-03 are based on facts and circumstances which are true and accurate to the best of my knowledge and belief.

Sworn and Subscribed
Before me this

25 day of February, 2005

Michael D. O'Keefe
Notary Public

Mark E. Warner
Site Vice President

MICHAEL D. O'KEEFE
MY COMMISSION EXPIRES FEB. 5, 2008
NOTARY PUBLIC
NEW HAMPSHIRE

Enclosure 1 to Letter SBK-L-05054

**Supplemental Information to NRC
Regarding Seabrook Station LAR 04-03
Application for Stretch Power Uprate**

INADVERTENT EMERGENCY CORE COOLING SYSTEM INITIATION AT POWER EVENT EVALUATION

1.0 PURPOSE AND SCOPE

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 if nominal operating conditions are assumed.

This evaluation is being developed utilizing the guidance provided in NRC Generic Letter 91-18, Revision 1, "Information to Licensees Regarding NRC Inspection Manual Section on Resolution of Degraded and Nonconforming Conditions," dated October 8, 1997.

2.0 BACKGROUND

The Seabrook Station stretch power uprate (SPU) analysis indicates that the pressurizer would become water solid in 8.9 minutes. The analysis demonstrates that operator action within 10.1 minutes would prevent opening of a pressurizer safety valve, with no credit taken for pressure relief from the pressurizer power-operated relief valves.

2.1 STANDARD REVIEW PLAN REQUIREMENTS

The NRC Standard Review Plan (NUREG-0800) acceptance criteria for an inadvertent Emergency Core Cooling System initiation at power is based on 10 CFR 50, Appendix A, General Design Criteria. For this event, general design criteria 10, 15, 17, and 26 are applicable.

Criterion 10 – Reactor Design The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Criterion 15 – Reactor Coolant System Design The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

Criterion 17 – Electric Power Systems An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.

Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

<u>Criterion 26 – Reactivity Control System Redundancy And Capability</u> Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

The specific criteria necessary to meet the requirements of General Design Criteria 10, 15, and 26 for incidents of moderate frequency are:

- Pressure in the reactor coolant and main steam systems shall be maintained below 110% of the design values.
- Fuel cladding integrity shall be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 departure from nucleate boiling ratio limit for pressurized water reactors based on acceptable correlations.
- An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.
- An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered. For such accidents, fuel failure

must be assumed for all rods for which the departure from nucleate boiling ratio falls below the value cited above for cladding integrity unless it can be shown, based on acceptable fuel damage model that fewer failures occur. There shall be no loss of function of any fission barrier other than fuel cladding.

- To meet the requirements of General Design Criteria 10, 15, and 26, the guidelines of Regulatory Guide 1.105, "Instrument Setpoints for Safety-Related Systems," are used with regard to their impact on the plant response to the inadvertent Emergency Core Cooling System initiation at power event.
- The most limiting plant systems single failure, as defined in the "Definitions and Explanations" of Appendix A of 10 CFR Part 50, shall be identified and assumed in the analysis and shall satisfy the guidelines stated in Regulatory Guide 1.53, "Application of the Single Failure Criterion to Nuclear Power Plant Protection Systems."

2.2 SEABROOK INITIAL SAFETY EVALUATION REPORT

The original Seabrook Station safety evaluation report (NUREG-0896) was issued in March 1983. The analysis of the inadvertent Emergency Core Cooling System initiation at power event concluded that the minimum departure from nucleate boiling ratio is 1.3. Thus, no fuel failure is predicted to occur, core geometry and control rod insertability are maintained with no loss of core cooling capability, and the maximum Reactor Coolant System pressure remains below 110% of design pressure. The NRC staff found the results of these analyses in conformance with the acceptance criteria of the Standard Review Plan Sections 15.5.1 through 15.5.2, and therefore acceptable.

2.3 WESTINGHOUSE NUCLEAR SAFETY ADVISORY LETTER 93-013

Westinghouse Nuclear Safety Advisory Letter NSAL-93-013, "Inadvertent ECCS Actuation at Power," was issued on June 30, 1993 and NSAL-93-013, Supplement 1 was issued on October 28, 1994.

NSAL-93-013 S1 addressed the means of satisfying the acceptance criteria for the inadvertent Emergency Core Cooling System initiation at power event identified in the NRC Standard Review Plan (NUREG-0800). The specific criterion of concern for NSAL-93-013 S1 is "An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently." The Standard Review Plan requirements for the inadvertent Emergency Core Cooling System initiation at power event are provided in Section 2.1 above.

NSAL-93-013 S1 concludes that to meet the Standard Review Plan criterion discussed above, it must be demonstrated:

- The pressurizer does not become water solid within the minimum allowable operator action time,
- The pressurizer safety valves do not relieve water or the pressurizer safety valves are capable of successfully closing following sub-cooled water relief,
- The downstream piping is capable of handling the water discharge flow, and

- The radiological consequences of breaking the pressurizer relief tank rupture disk do not violate the applicable offsite dose limits.

NSAL-93-013 S1 also states that water relief through the pressurizer power-operated relief valves is not a concern because the power-operated relief valve block valves would be available to isolate the power-operated relief valves should one open and fail to close.

The Westinghouse NSAL included three recommendations:

1. Reduce the maximum Emergency Core Cooling System flow used in the safety analysis.
2. Use a less restrictive operator response time. Per ANSI/ANS 58.8-1992, "Time Response Design Criteria For Safety-Related Operator Actions," credit can be taken in the analysis for the operator to stop one pump at 7 minutes, a second pump at 8 minutes, and depending on the plant-specific design, the third at 9 minutes.
3. Credit the use of one or more pressurizer power-operated relief valves to help mitigate the accident.

As a result of the above, the Seabrook Station UFSAR was revised to address the inadvertent ECCS actuation at power event as an overpressure event, and operator actions were established to ensure safety injection was terminated to prevent water relief through the pressurizer safety valves. Water relief through the pressurizer power-operated relief valves was considered acceptable based on the NSAL-93-013 S1 guidance regarding the ability to isolate a stuck open valve with the power-operated relief valve block valves, and because the valves, piping, and supports are qualified for water relief.

2.4 CURRENT UFSAR

Seabrook Station UFSAR Section 15.5.1 contains the evaluation of the inadvertent Emergency Core Cooling System initiation event. The event was evaluated using the RETRAN computer code.

The assumptions are as follows:

<u>Initial Operating Conditions</u>

Initial reactor power is assumed to be at the maximum value, initial reactor coolant temperature is assumed to be at the minimum value, initial pressurizer pressure is assumed to be at its minimum value, and the initial pressurizer water level is assumed to be at its maximum value, consistent with steady-state full power operation including allowances for calibration and instrument errors.

<u>Moderator and Doppler Coefficients of Reactivity</u>

A least-negative moderator temperature coefficient was used. A low (absolute value) Doppler power coefficient was assumed.

<u>Reactor Control</u>

The reactor was assumed to be in manual control.

Pressurizer Control

Pressurizer heaters and spray are assumed to be operable to increase the rate of pressurizer filling. The pressurizer sprays act to reduce the Reactor Coolant System pressure, thus increasing Emergency Core Cooling System injection. The pressurizer heaters act to add energy to the pressurizer fluid, thus increasing the pressurizer fluid volume through thermal expansion.

Boron Injection

At time zero, two charging pumps conservatively inject 2,900 ppm borated water into the cold leg of each loop.

Reactor Trip

The reactor and turbine are assumed to trip upon receipt of the safety injection signal. Assuming reactor and turbine trip on safety injection minimizes the heat removal capability of the Reactor Coolant System, thereby maximizing the Reactor Coolant System inventory increase through safety injection flow and thermal expansion of the Reactor Coolant System fluid.

Results of the analysis show that spurious safety injection presents no hazard to the integrity of the Reactor Coolant System. No radioactivity releases are anticipated as a direct result of this malfunction. Consequently, no radiological consequences are predicted.

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 there will be no water flow through the pressurizer safety valves as a consequence of inadvertent operation of Emergency Core Cooling System during power operation provided a minimum of 10.1 minutes is available for operator action to terminate Emergency Core Cooling System, and thereby prevent pressurizer safety valve challenges without credit for pressure relief through the pressurizer power-operated relief valves.

If the reactor does not trip immediately, the low pressure reactor trip will be actuated. This trips the turbine and prevents excess cooldown, thereby expediting recovery from the incident.

3.0 LAR 04-03 SEABROOK STATION SPU

As part of the Seabrook Station SPU, the accident analyses were re-analyzed at a core thermal power level of 3659 MWt (3678 MWt Nuclear Steam Supply System power level). The review 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 ECCS 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.

4.0 INTERIM EVALUATION

4.1 SPU ANALYSIS BASED ON NOMINAL CONDITIONS

Westinghouse has performed a re-analysis of the Seabrook Station inadvertent ECCS initiation at power event using nominal values for pressurizer pressure, pressurizer level, and heat removal using the steam dumps. The nominal values represent normal operating conditions and the steams dumps are the likely means for heat removal. It is more likely that these conditions would be present should an inadvertent ECCS initiation at power event occur than the adverse values assumed in the SPU analysis. The transient was analyzed using the RETRAN computer code that was also used in the SPU analysis. The model and assumptions used in the re-analysis were those used in the SPU analysis except for the nominal values as provided below:

PARAMETER	SPU ANALYSIS	INTERIM ANALYSIS
RCS pressure (psia)	2200 psia (2250 psia – 50 psia uncertainty)	2250 psia
Pressurizer level (%)	65% (60% + 5% uncertainty)	60%
Secondary heat removal	Main steam safety valves	Steam dump valves to the condenser
Time to water solid pressurizer	8.9 minutes	14.9 minutes

The conclusions of the re-analysis using nominal values of pressurizer pressure, pressurizer level and heat removal using the steam dumps are:

- 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 ECCS 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 ECCS initiation at power event.

Graphs of plant parameters associated with this evaluation are provided in Enclosure 3.

4.2 OPERATOR TRAINING AND PERFORMANCE

Operators respond to the inadvertent ECCS 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 14 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, stop all but one centrifugal charging pump, establish normal charging and isolate the high head ECCS injection flowpath. A review of operator performance on the simulator in response to the inadvertent ECCS initiation at power event has verified that operators isolate the high head ECCS injection flowpath prior to exceeding 10.1 minutes. In fact, time studies done during the fourth quarter 2004 demonstrated that operators on average, isolated this flowpath in approximately 8 minutes. In addition, prior to terminating the event by isolating the high head ECCS injection flowpath, operators take action to reduce ECCS flow by shutting down all but one centrifugal charging pump. This further increases the time that operators have to respond to the event.

Operators maintain proficiency with terminating an inadvertent ECCS initiation at power event through periodic simulator training scenarios. In 2005, Licensed Operators will receive simulator training on this event during two separate training periods beginning in the second quarter 2005.

4.3 CORE DAMAGE FREQUENCY

The initiating event frequency for inadvertent ECCS initiation at power event is 2.8E-2/year. The resultant core damage frequency (CDF) is 1.67E-7/year, which is less than 1% of the total core damage frequency.

The core damage frequency accounts for possible core damage scenarios that initiate from an inadvertent ECCS 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.

4.4 REASONABLE ASSURANCE OF SAFETY

FPL Energy Seabrook has determined that there is a reasonable assurance of safety following an inadvertent ECCS initiation at power event based on the following:

- Analyses using nominal values of pressurizer pressure and pressurizer level, and heat removal through the steam dumps, resulted in increasing the time that the pressurizer becomes water solid from 8.9 minutes to 14 minutes which is greater than 10.1 minutes credited for operator action to terminate safety injection.
- If a loss of offsite power is assumed and credit is taken for heat removal from the atmospheric steam dump valves, the time for the pressurizer to become water solid is 11.8 minutes which is still greater than 10.1 minutes.
- If only heat removal using the main steam safety valves is assumed, but nominal values for pressurizer pressure and level, the time for the pressurizer to become water solid is 10.4 minutes which is still greater than 10.1 minutes.
- The SPU safety analysis does not result in the opening of the power-operated relief valves until an uncertainty is applied to the pressurizer power-operated relief valve setpoint. Based on the design and qualification of the valves, piping, and supports for water relief, there is reasonable certainty that a power-operated relief valve could be isolated if it were to stick open.
- Operating procedures to terminate an inadvertent ECCS initiation at power event require that operators stop one centrifugal charging pump when Safety Injection termination criteria are met, thus, decreasing flow and increasing the time before the pressurizer becomes water solid. This is not assumed in the SPU safety analysis or the interim evaluation and would provide additional time for operators to terminate the event.

- Licensed Operators will complete simulator training on terminating an inadvertent ECCS initiation at power event two separate times beginning in the second quarter 2005.
- The initiating event frequency for inadvertent ECCS 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.

4.5 PUBLIC HEALTH AND SAFETY

The inadvertent ECCS 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 ECCS initiation at power event at Seabrook Station.

5.0 SUMMARY AND CONCLUSION

FPL Energy Seabrook has evaluated the NRC concern that for the inadvertent ECCS initiation at power event, the pressurizer will become water solid prior to 10.1 minutes which is the time required for the operator to isolate the high head ECCS injection flowpath.

To address the NRC staff's concern, FPL Energy Seabrook performed the following:

- A review of the Standard Review Plan sections addressing inadvertent Emergency Core Cooling System initiation.
- A review of the current licensing and design bases for Seabrook Station.
- A review of Westinghouse Nuclear Safety Advisory Letter NSAL 93-013 and NSAL 93-13 Supplement 1 addressing the inadvertent ECCS initiation at power event and Seabrook-specific analyses performed in response to the letters.
- A review of operator performance on the simulator in response to this event.
- An interim evaluation using nominal values for pressurizer pressure, pressurizer level, and crediting normal heat removal through the steam generators was performed.

FPL Energy Seabrook has determined that based on an analysis performed using nominal values and crediting realistic heat removal through the steam generators, that the inadvertent Emergency Core Cooling System initiation at power event will not result in the pressurizer becoming water solid prior to operator action terminating the event at 10.1 minutes after initiation.

In addition, Seabrook Station procedures in response to this event require that the operator stop one centrifugal charging pump at the appropriate procedure step. This will extend the time required for the pressurizer to become water solid beyond the analyzed time that assumes two centrifugal charging pumps at maximum flow for 10.1 minutes.

Refer to Enclosure 2 for FPL Energy Seabrook's commitment relative to the NRC inadvertent Emergency Core Cooling System initiation at power event concerns.

Enclosure 2 to Letter SBK-L-05054

License Condition

License Condition

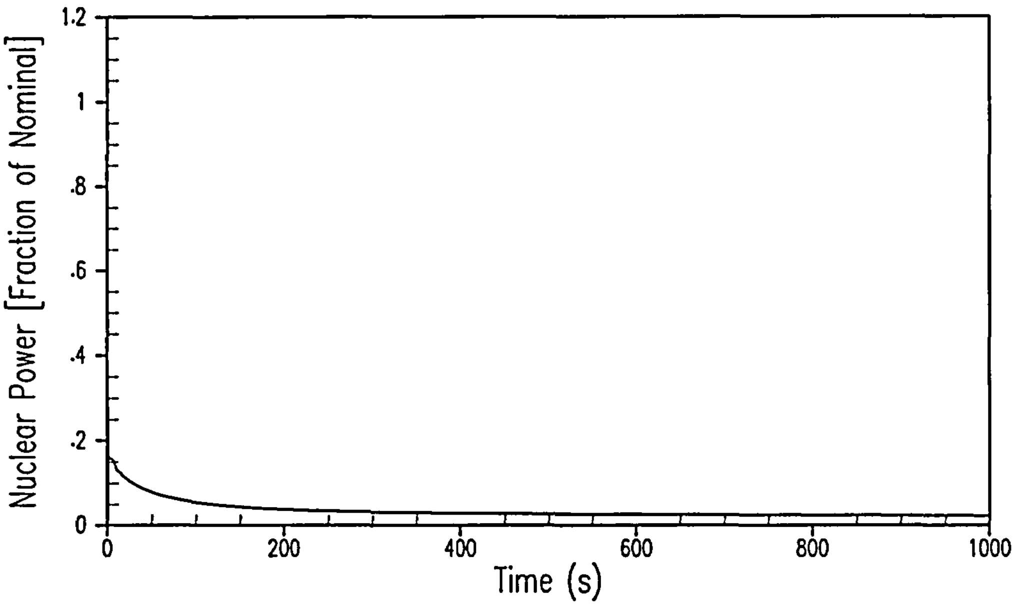
Prior to startup from refueling outage 11, FPL Energy Seabrook commits to either upgrade the controls for the pressurizer power operated relief valves (PORV) to safety-grade status and confirm the safety-grade status and water-qualified capability of the PORVs, PORV block valves and associated piping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

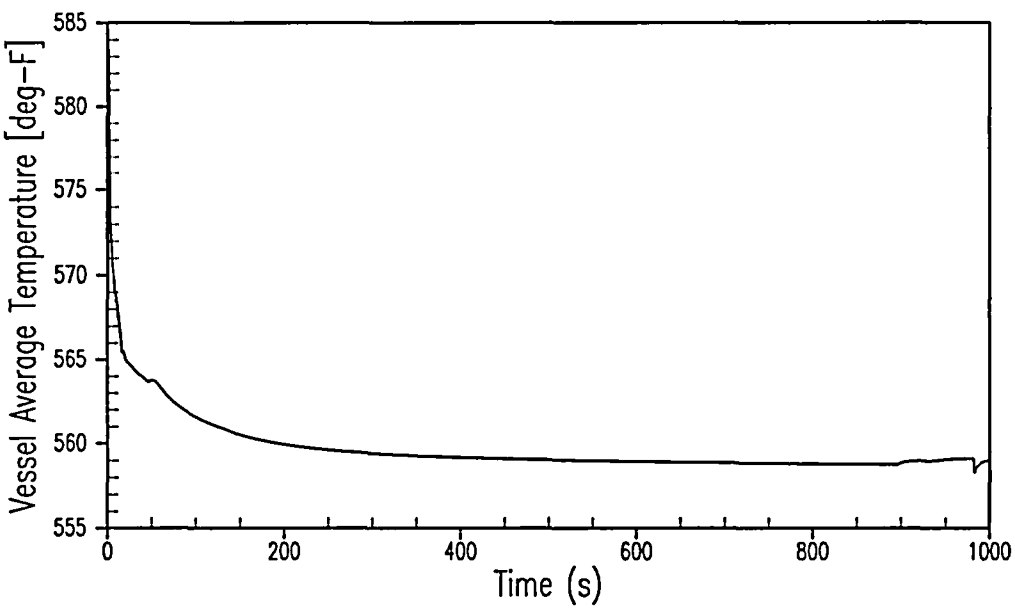
Enclosure 3 to Letter SBK-L-05054

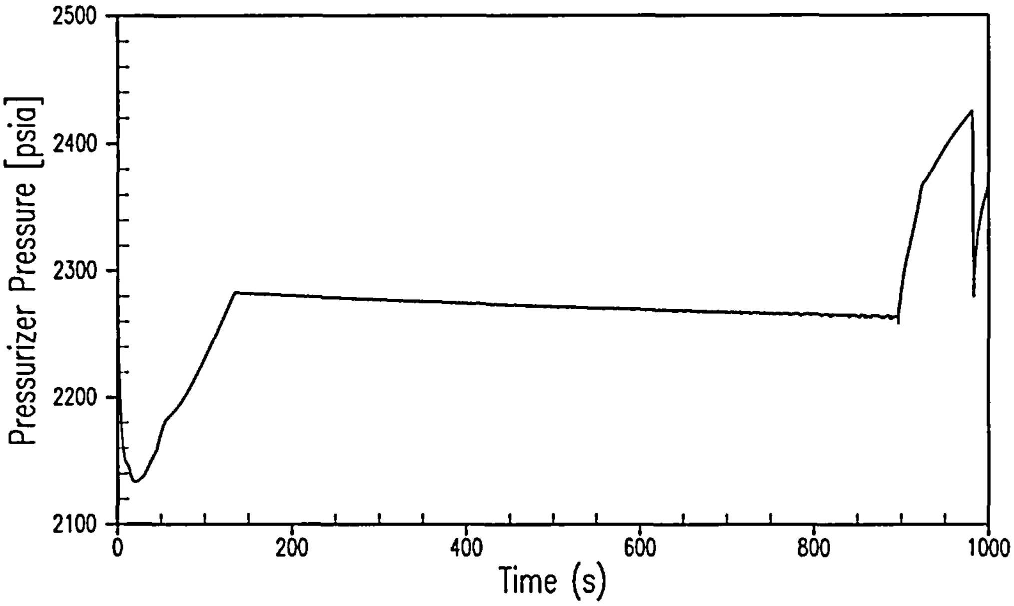
Interim Evaluation

Inadvertent ECCS Initiation at Power Graphs

U. S. Nuclear Regulatory Commission
SBK-L-05054



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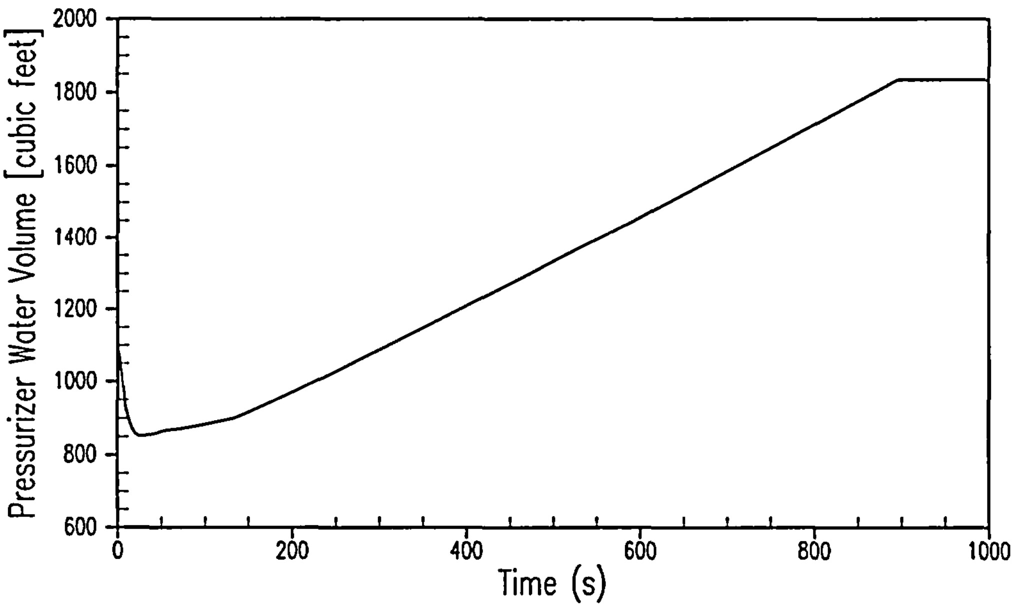
600
800
1000
Time (s)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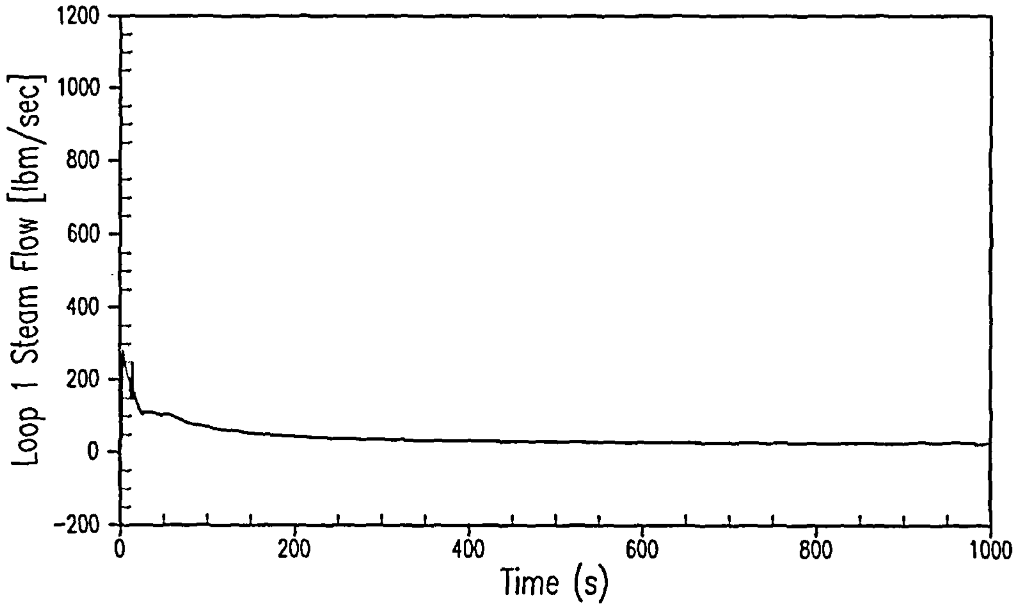
800
1000
Time (s)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800
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400
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0
-200
Loop 1 Steam Flow [lbm/sec]
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200
400
600
800
1000
Time (s)

Enclosure 4 to Letter SBK-L-05054

Additional Analysis Results

At the request of the NRC staff, an additional Loss of External Load reactor coolant system overpressure evaluation was performed crediting the second safety grade reactor trip signal (Overtemperature ΔT).

The analysis result and corresponding acceptance criteria is provided below for both the original SPU analysis (crediting the first safety grade reactor trip signal) and the evaluation crediting the second safety grade reactor trip signal.

Credited Safety Grade Reactor Trip Signal	**Maximum RCS Pressure**	**Acceptance Criteria**
High Pressurizer Pressure Trip – Original SPU Analysis	2681.9 psia	2748.5 psia
Overtemperature ΔT Trip	2684.39 psia	2748.5 psia