FPL

Florida Power & Light Company, 6501 S. Ocean Drive, Jensen Beach, FL 34957

September 22, 2011

L-2011-389
10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Re: St. Lucie Plant Unit 1
Docket No. 50-335
Renewed Facility Operating License No. DPR-67

Response to NRC Reactor Systems Branch Request for Additional Information Regarding Extended Power Uprate License Amendment Request

References:

(1) R. L. Anderson (FPL) to U.S. Nuclear Regulatory Commission (L-2010-259), "License Amendment Request (LAR) for Extended Power Uprate," November 22, 2010, Accession No. ML103560419.

(2) Email from T. Orf (NRC) to C. Wasik (FPL), "St. Lucie 1 EPU draft RAIs – Reactor Systems (SRXB)," August 23, 2011.

By letter L-2010-259 dated November 22, 2010 [Reference 1], Florida Power & Light Company (FPL) requested to amend Renewed Facility Operating License No. DPR-67 and revise the St. Lucie Unit 1 Technical Specifications (TS). The proposed amendment will increase the unit's licensed core thermal power level from 2700 megawatts thermal (MWt) to 3020 MWt and revise the Renewed Facility Operating License and TS to support operation at this increased core thermal power level. This represents an approximate increase of 11.85% and is therefore considered an Extended Power Uprate (EPU).

By email from the NRC Project Manager dated August 23, 2011 [Reference 2], additional information related to Chemical and Volume Control System (CVCS) malfunction; and Inadvertent Operation of the Emergency Core Cooling System (ECCS) was requested by the NRC staff in the Reactor Systems Branch (SRXB) to support their review of the EPU LAR. The request for additional information (RAI) identified thirty-two (32) questions. The response to RAI numbers thirty-six (36) and thirty-seven (37) of Reference 2 is provided in Attachment 1 to this letter.

A001
NRR

In accordance with 10 CFR 50.91(b)(1), a copy of this letter is being forwarded to the designated State of Florida official.

This submittal does not alter the significant hazards consideration or environmental assessment previously submitted by FPL letter L-2010-259 [Reference 1].

This submittal contains no new commitments and no revisions to existing commitments.

Should you have any questions regarding this submittal, please contact Mr. Christopher Wasik, St. Lucie Extended Power Uprate LAR Project Manager, at 772-467-7138.

I declare under penalty of perjury that the foregoing is true and correct to the best of my knowledge.

Executed on Sept. 22, 2011.

Very truly yours,

Robert J Hughes FOR RLA

Richard L. Anderson
Site Vice President
St. Lucie Plant

Attachments (1)

cc: Mr. William Passetti, Florida Department of Health

ATTACHMENT 1

Response to NRC Reactor Systems Branch Request for Additional Information Regarding Extended Power Uprate License Amendment Request

(Cover Page Plus 17 Pages)

Controlled Document

ANP-3037
Revision 0

St. Lucie Unit 1 EPU – Information to Support NRC Review of CVCS Malfunction

September 2011

AREVA NP Inc.

Controlled Document

AREVA NP Inc.

ANP-3037
Revision 0

St. Lucie Unit 1 EPU – Information to Support NRC Review of CVCS Malfunction

Nature of Changes

Item	Page	Description and Justification
1.	All	Initial Release

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Nomenclature

AOO	Anticipated Operational Occurrence
CVCS	Chemical and Volume Control System
DNBR	Departure from Nucleate Boiling Ratio
DTC	Doppler Temperature Coefficient
ECCS	Emergency Core Cooling System
EOC	End-of-Cycle
EPU	Extended Power Uprate
ESF	Engineered Safety Feature
FSAR	Final Safety Analysis Report
HFP	Hot Full Power
HPPT	High Pressurizer Pressure Trip
LAR	Licensing Amendment Request
LCO	Limiting Condition for Operation
MTC	Moderator Temperature Coefficient
NRC	Nuclear Regulatory Commission
PHLA	Pressurizer High Level Alarm
PORV	Power-Operated Relief Valve
PSV(s)	Pressurizer Safety Valve(s)
PWR	Pressurized Water Reactor
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RPS	Reactor Protective System
SRP	Standard Review Plan
TS	Technical Specifications

1.0 Introduction

The Nuclear Regulatory Commission (NRC) staff requested additional information to support the review of the Chemical Volume Control System (CVCS) Malfunction and inadvertent operation of the Emergency Core Cooling System (ECCS) section of the St. Lucie Unit 1 extended power uprate (EPU) license amendment request (LAR). This NRC information request is included in draft Request for Additional Information (RAI) SRXB-36 and SRXB-37.

The information contained herein is specific to the St. Lucie Unit 1 EPU LAR submittal.

2.0 NRC Information Request

SRXB-36: [2.8.5.5.a]: The St. Lucie FSAR §15.1.1, "Classification of Accidents", states that the Chapter 15 accident analyses are based on RG 1.70 and NUREG-0800. Both documents include the Chemical and Volume Control System Malfunction. The latter reference contains guidelines for review of analyses of the Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory in all operating modes. Please supply an analysis of the Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory.

SRXB-37: [2.8.5.5.b]: The application notes that high-head safety injection pumps are not capable of injecting water into the reactor coolant system at normal operating pressure. However, the application also proposes a change in the Technical Specifications (TS 3/4.5.2, EMERGENCY CORE COOLING SYSTEMS (ECCS) – OPERATING) that would add LCO 3.5.2.d, which accounts for the incorporation of the charging pumps into the ECCS. Since the charging pumps are capable of injecting water into the reactor coolant system at normal operating pressure, the Inadvertent Operation of the ECCS is a credible event and ought to be analyzed. Provide an analysis of the Inadvertent Operation of the ECCS consistent with the SRP guidelines and Regulatory Issue Summary (RIS) 2005-029.

3.0 Response to Information Request

3.1 *Identification of Causes and Accident Description*

The CVCS Malfunction event is initiated by the failure of the pressurizer level transmitter which results in an erroneous low-low level signal for pressurizer level. The generated signal is transmitted to the controller, which responds by starting the two stand-by charging pumps and closing the letdown flow control valve to its minimum flow position. With the mismatch between letdown and charging flow, the pressurizer mixture level and pressure increase. The pressurizer sprays limits the pressure increase. The operators are alerted to the event either by a high pressurizer pressure trip (HPPT) or by the pressurizer high level alarm (PHLA) and mitigate the event by reducing charging flow and/or restoring letdown flow.

For St. Lucie Unit 1, the consequences of this event are also applicable to an inadvertent operation of the ECCS where inadvertent operation of the charging pumps results in an increase in pressurizer level. The boundary conditions for the analysis performed were chosen to bound either scenario; Inadvertent CVCS or ECCS Actuation.

3.2 *Description of Analyses and Evaluations*

The purpose of this analysis was to evaluate the CVCS Malfunction event / inadvertent operation of ECCS event. Detailed analysis was performed using the S-RELAP5 code (Reference 1). The S-RELAP5 code was used to model the key primary and secondary system components, Reactor Protective System (RPS) and Emergency Safety Features (ESF) actuation trips and core kinetics. The calculation was performed to determine the operator action time necessary to mitigate the CVCS Malfunction event.

3.3 *Input Parameters and Assumptions*

A single limiting case was analyzed. Parameter biasing, assumptions, and an assumed single-failure were designed to ensure a conservatively high CVCS charging flow rate, maximize initial pressurizer level, and provide maximum reactivity feedback. Assumptions regarding operator actions and mitigating systems and functions, along with a limiting single-failure, produce the most challenging scenario regarding pressurizer fill.

The input parameters and biasing for the analysis of this event are shown in Table 1.

- Initial Conditions – The event was initiated from rated power plus uncertainty conditions with a maximum core inlet temperature and minimum Technical Specification (TS) Reactor Coolant System (RCS) flow.

- Reactivity Feedback – End-of-cycle (EOC) Doppler and moderator feedback were assumed for this event. Minimum scram worth with the most reactive rod stuck out of the core was assumed.

- Reactor Protective System Trips and Delay – Reactor protection trip setpoints and delay times were biased to conservatively estimate the operator action time. The high pressurizer pressure trip was based on the nominal value plus uncertainty.

- Pressurizer Conditions - A nominal pressurizer pressure and nominal pressurizer level plus uncertainty were assumed. The pressurizer safety valve (PSV) setpoint was based on the nominal value minus tolerance while the pressurizer high level alarm was based on the nominal high level alarm value plus uncertainty. The biasing of pressurizer parameters ensures the calculation of a minimum time to fill the pressurizer.

- CVCS Charging – Maximum CVCS charging flow and a conservative charging temperature were assumed to ensure the most limiting conditions for the CVCS event.

- Steam Generator Tube Plugging – Maximum steam generator tube plugging was assumed (10% average).

- Single-Failure – The assumed single-failure is the complete closure of letdown flow control valve that occurs concurrently with the start of the second and the third charging pumps.

- Charging Boron Concentration – The charging flow boron concentration is assumed to be equal to the initial RCS boron concentration.

3.4 *Acceptance Criteria*

This event is classified as an AOO. The acceptance criteria for this event are:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values,

2. Fuel cladding integrity should be maintained by ensuring that the minimum departure from nucleate boiling ratio (MDNBR) remains above the 95/95 DNBR limit for pressurized water reactors (PWRs), and

3. An AOO should not generate a more serious plant condition without other faults occurring independently.

The principally challenged acceptance criterion for the CVCS Malfunction is to demonstrate that the event does not generate a more serious plant condition. The analysis objective is to show that the pressurizer does not become water-solid before the operator can terminate the transient, within 10 minutes after the event begins. This ensures that no solid-water is relieved through the pressurizer power operated relief valves (PORVs).

3.5 *Results*

A single limiting case was analyzed for the CVCS Malfunction event at hot full power (HFP) conditions. The sequence of events is shown in Table 2. The system response is presented in Figure 1 to Figure 4. The analysis showed that the operators will have more than 13 minutes from the event initiation to terminate the event. The analysis also showed that the operators have 651.125 seconds (more than 10 minutes) from the time of receipt of the pressurizer high level alarm to terminate the event before the pressurizer reaches a water-solid condition.

Table 1 CVCS Malfunction: Initial Conditions and Biasing

Parameter	Value
Initial Reactor Power	3029.06 MWt
Initial Core Inlet Temperature	551 °F
Initial Reactor Coolant Flow Rate	375,000 gpm
Initial Pressurizer Pressure	2250 psia
Initial Pressurizer Liquid Level	68.6%
Moderator Temperature Coefficient (MTC)	- 32 pcm/°F
Doppler Temperature Coefficient (DTC)	- 1.75 pcm/°F
Scram Reactivity[1]	6017.22 pcm
High Pressurizer Pressure Trip[2]	2435 psia
Pressurizer High Level Alarm	73.6%
CVCS Charging Flow (total)	147 gpm
CVCS Charging Temperature	104 °F
RCP Bleedoff	4 gpm
Steam Generator Tube Plugging	10%
Pressurizer Spray	On Auto
Pressurizer PORVs	Not Credited
PSVs	Open on pressure higher than 2437.5 psia / Close on pressure lower than 2413.1 psia
Reactor Coolant Pumps	Operating
Main Feedwater	Automatic
Letdown Flow	Isolated
Pressurizer Proportional Heaters	On Auto
Pressurizer Backup Heaters	Conservatively turned On when the pressurizer level is greater than 10% above nominal level

Note 1 and Note 2 – No reactor trip occured in the analysis.

Table 2 CVCS Malfunction: Sequence of Events

Event	Time[1] (s)
Event initiation – Erroneous low-low level pressurizer level control system signal; Second and third charging pumps start; Letdown flow is isolated	10.0
Pressurizer High Level Alarm reached	181.325
Maximum pressurizer pressure occurred	786.0
Maximum pressurizer volume occurred	832.45

Note 1 – These values include the 10 second steady-state time.

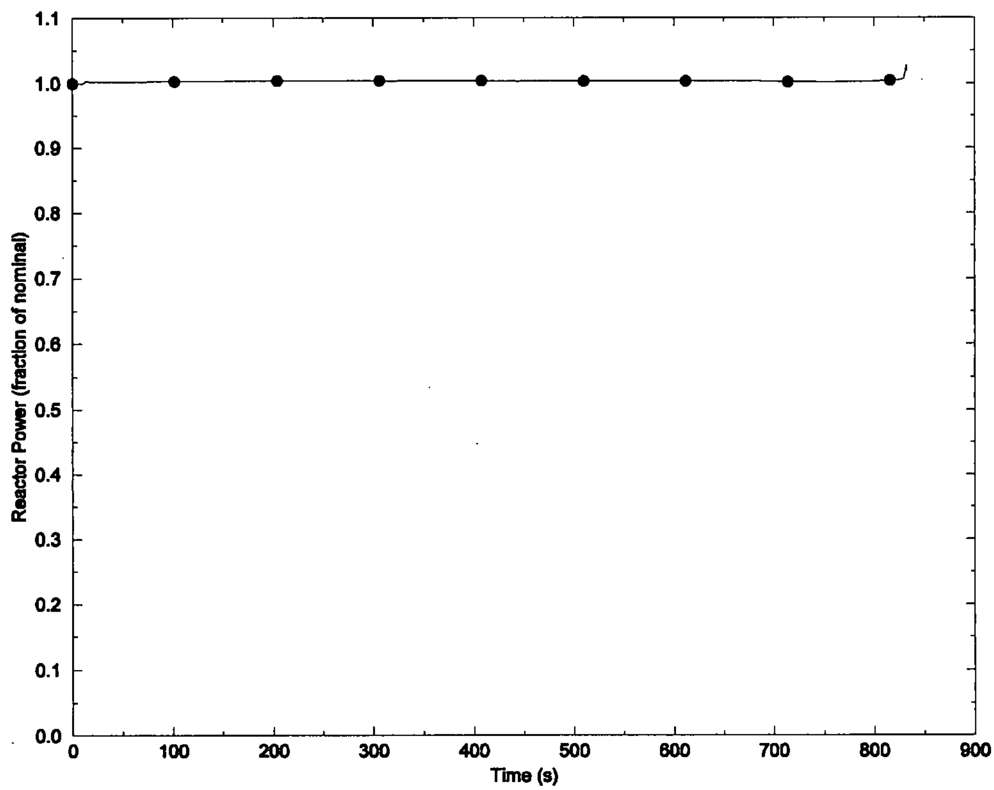

Figure 1 CVCS Malfunction – Reactor Power

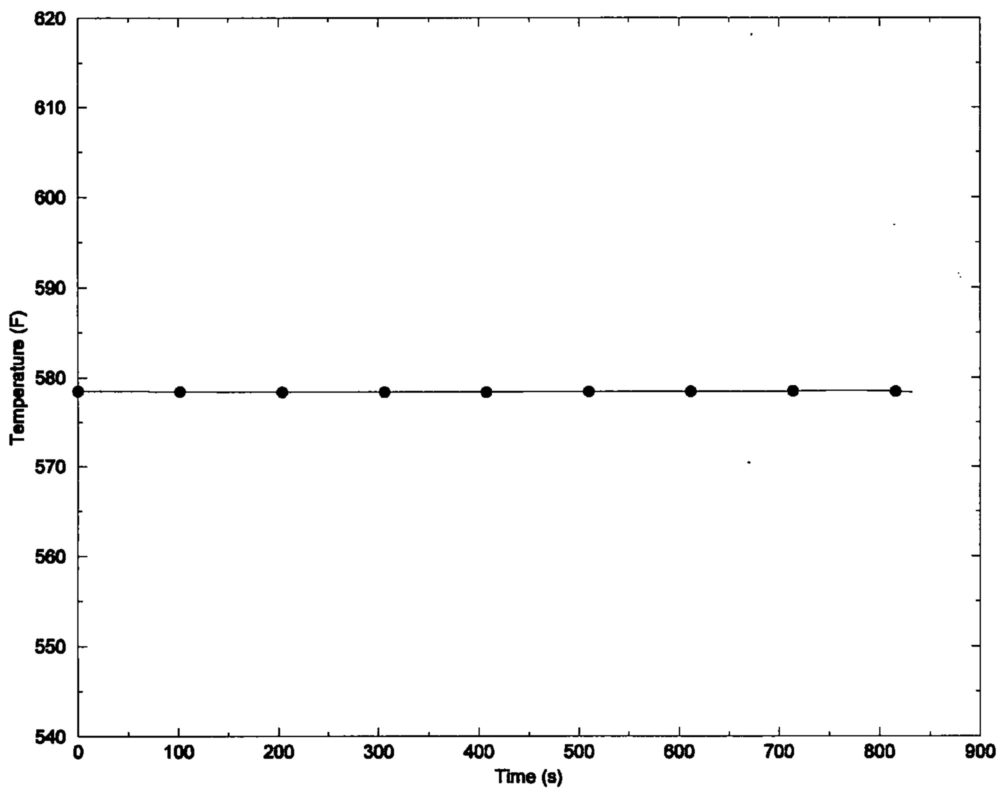

Figure 2 CVCS Malfunction – RCS Average Temperature

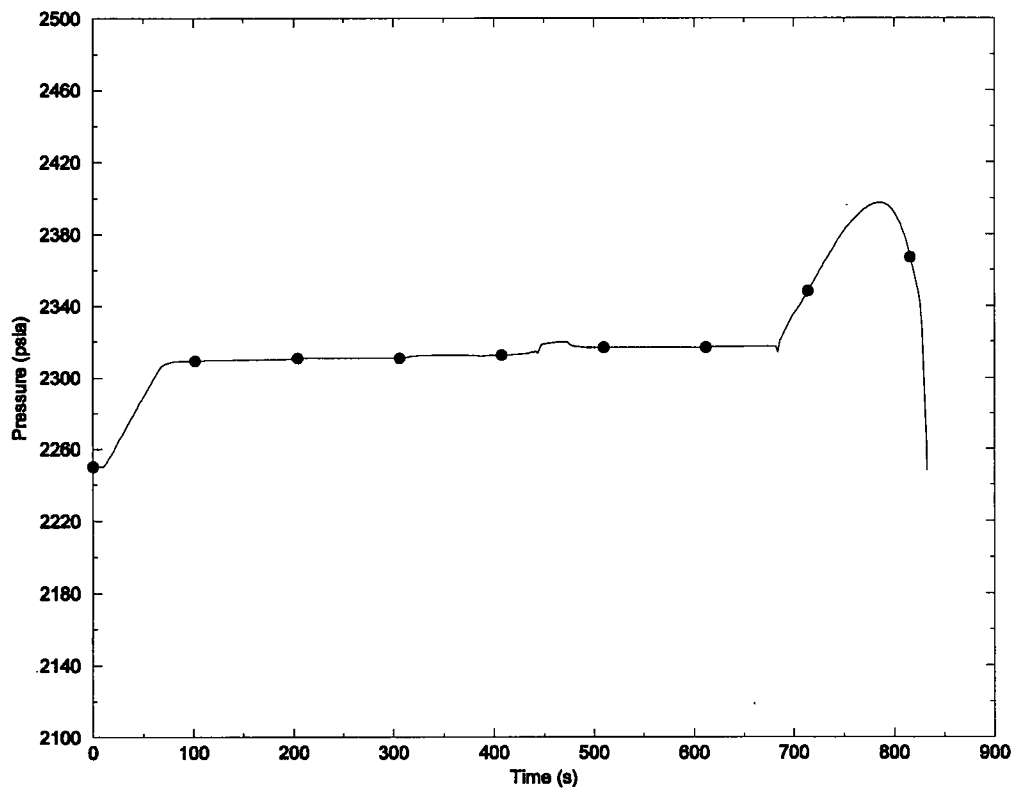

Figure 3 CVCS Malfunction – Pressurizer Pressure

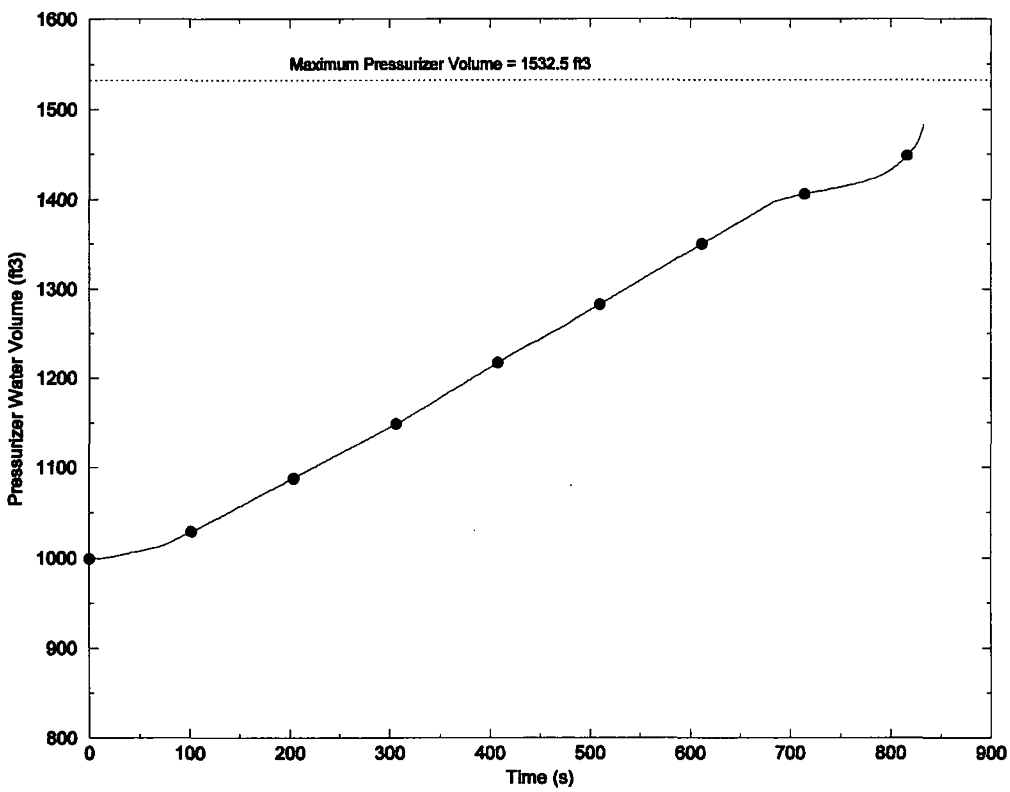

Figure 4 CVCS Malfunction – Pressurizer Volume

4.0 References

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