FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302 /LICENSE NUMBER DPR-72

ENCLOSURE 2

AREVA 86-9167251-001 - SUMMARY OF CR3 EPU INADVERTENT PRESSURIZER RELIEF VALVE OPENING

AREVA

CALCULATION SUMMARY SHEET (CSS)

Document No. 86 - 9167251 - 001 Safety Related: ☒ Yes ☐ No

Title Summary of CR3 EPU Inadvertent Pressurizer Relief Valve Opening

PURPOSE AND SUMMARY OF RESULTS:

This document summarizes the inputs and results of an analysis of the Inadvertent Pressurizer Relief Valve Opening (IPRVO) accident performed for CR-3 at an Extended Power Uprate (EPU) rated thermal power level of 3014 MWt (plus 0.4% heat balance uncertainty). The IPRVO analysis is summarized in Reference [1]. Reference [2] is the analysis input summary document for the IPRVO analysis.

All acceptance criteria for the IPRVO accident are met. The total power increase throughout the transient is <3%. Thermal power remains less than the 112% of full power for the duration of the event; therefore, fuel cladding integrity is maintained. The peak RCS pressure is not challenged during this event, which is a depressurization event.

THE FOLLOWING COMPUTER CODES HAVE BEEN USED IN THIS DOCUMENT:

CODE/VERSION/REV	CODE/VERSION/REV
N/A	

THE DOCUMENT CONTAINS ASSUMPTIONS THAT SHALL BE VERIFIED PRIOR TO USE

☐ YES
☒ NO

Summary of CR3 EPU Inadvertent Pressurizer Relief Valve Opening

Review Method: ☒ Design Review (Detailed Check)
☐ Alternate Calculation

Signature Block

Name and Title (printed or typed)	Signature	P/R/A and LP/LR	Date	Pages/Sections Prepared/Reviewed/Approved
Douglas Brownson Supervisor, B&W Safety Analysis	Douglas Brownson	P	9/13/11	All
Mohsin Reza Engineer IV	Mohsin Reza	R	9-13-2011	All
Keith Higar Manager	K. S. Higar	A	9-13-11	All – Reviewer Independence

Note: P/R/A designates Preparer (P), Reviewer (R), Approver (A);
LP/LR designates Lead Preparer (LP), Lead Reviewer (LR)

Project Manager Approval of Customer References (N/A if not applicable)

Name (printed or typed)	Title (printed or typed)	Signature	Date
N/A			

Mentoring Information (not required per 0402-01)

Name (printed or typed)	Title (printed or typed)	Mentor to: (P/R)	Signature	Date
N/A				

Record of Revision

Revision No.	Pages/Sections/Paragraphs Changed	Brief Description / Change Authorization
000	All	Initial Release
001	Section 2.0	Removed Figures 2-1 through 2-4 (proprietary issue)
	Section 3.1	Removed sentence comparing IPRVO event to an SBLOCA event
	Section 3.2	Removed sentence regarding availability of HPI. Removed discussion regarding core uncovery aspects of event being bounded by the SBLOCA event.

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1.0 ASSUMPTIONS

1.1 Unverified Assumptions

There are no unverified assumptions associated with the analysis.

1.2 Justified Assumptions

There are no justified assumptions associated with the analysis.

1.3 Modeling Simplifications

There are no modeling simplifications associated with the analysis.

2.0 DESIGN INPUTS

Reference [2] provides a detailed description of the inputs to the IPRVO analysis. The input is summarized in Table 2-1 of this document.

Table 2-1: Key Input Parameters

Parameter	Value
RCS Conditions	
Core Power, MWt	3026.1
Decay Heat	1.0*ANS71+B&W actinides
RCP Power, MWt per pump	4.1
Tave, °F	582
RCS Pressure, psia	2170 (Note 1)
Total RCS Flow Rate, gpm	374,880
Core Bypass Flow Fraction, %	7.5
Pressurizer	
Initial Indicated PZR Level, in	220
Number of Pressurizer Code Safety Valves (PCSVs)	N/A
PCSV Nominal Setpoint, psig	N/A
PCSV Setpoint Tolerance, %	N/A
PCSV Blowdown, %	N/A
PCSV Capacity, lbm/hr/valve	317,973 @ 2750 psig
PORV Capacity, lbm/hr/valve	The analysis conservatively uses the capacity of a single PCSV in place of the PORV capacity.
Pressurizer Heaters	Not Modeled
Pressurizer Spray	N/A
MFW and Steam Generator Tube Plugging	
MFW Temperature, °F	460 (Note 2)
MFW Flow, lbm/s per SG	(Note 2)
SG Tube Plugging – average %	0
SG Level – operate range, %	~70
EFW	
Flow, gpm per SG	N/A
Temperature, °F	N/A

Parameter	Value
Turbine & Main Steam System	
Turbine Header Pressure, psia	930 (Note 2)
Turbine Trip Delay Time, seconds	0.0 (at time of reactor trip)
TSV Stroke Time, sec	0.2
MSSV Nominal Setpoint per SG, psig	2 @ 1050 psig (one assumed out of service) 2 @ 1070 psig 2 @ 1090 psig 2 @ 1100 psig (Note 3)
MSSV Capacity per SG, lbm/hr/valve	7 @ 845,759 lbm/hr at 1169.7 psia 1 @ 583,574 lbm/hr at 1169.7 psia (Note 3)
MSSV Setpoint Tolerance, %	+3.0
MSSV Accumulation, %	+3.0
MSSV Blowdown, %	-5.0
Reactivity Control	
Moderator Temperature Coefficient, Δk/k/°F	+0.75E-04
Doppler Reactivity (Δk/k/°F)	-1.30E-05
Prompt Neutron Generation Time (μs)	24.8
Effective Delayed Neutron Fraction	0.0070
Shutdown Margin, %Δk/k	1.000
Scram Curve	Table 2-2
Reactor Protection System	
Low RCS Pressure Setpoint (psia)	1894 (Note 1)
Low RCS Pressure Response Time (s)	0.610
Miscellaneous	
Offsite Power	Available
Single Failure	None
Operator Actions	None Credited

Notes:

1. As measured at the hot leg pressure tap.
2. The MFW temperature and flow as well as the turbine header pressure were varied to achieve the desired RCS and SG conditions for the heat load of the event.
3. The 1 MSSV per SG that has a capacity of 583,574 lbm/hr at 1169.7 psia is at a nominal setpoint of 1100 psig.

Table 2-2: Scram Curve > 15%FP

Time After Reactor Trip sec	Reactivity Insertion %
0.0	0.00
0.2	0.58
0.3	0.99
0.4	1.83
0.6	5.29
0.8	12.33
1.0	21.41
1.2	33.09
1.4	50.75
1.6	72.96
1.8	91.30
2.0	99.26
2.2	99.99
2.3	100.00

3.0 ANALYSIS SUMMARY

3.1 Event Description

The inadvertent opening of a primary system pressure relief valve results in a decrease in reactor coolant inventory and Reactor Coolant System (RCS) pressure, since the loss of inventory will be in excess of the capability of the makeup system. The event is initiated by a spurious electrical signal or operator error. This type of initiating event could only occur with the Pilot Operated Relief Valve (PORV). However, the analysis will conservatively model a break flow consistent with a single PCSV, since the PCSV has a larger capacity. The decrease in pressure affects neutron flux via moderator density feedback. For this event, feedback is based on a positive moderator temperature coefficient such that the result of the feedback is an increase in core power. The RCS continues to depressurize until the low RCS pressure RPS setpoint is reached. The resulting insertion of scram reactivity ends the power excursion and negates further challenge to the core Departure from Nucleate Boiling (DNB) limits. For the IPRVO event, it is anticipated that the operator would have ample time to isolate the relief valve prior to any concerns with core uncovery.

3.2 Applicable Acceptance Criteria

The IPRVO analyses will demonstrate compliance with the acceptance criteria for the evaluation of this accident. Per Section 15.6.1 of the Standard Review Plan (SRP) [5], the applicable acceptance criteria for this event are:

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

2) Fuel cladding integrity is maintained if the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit.

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

The first acceptance criterion will not be challenged, since the RCS is depressurizing and the event does not challenge the secondary plant response.

The second criterion will be shown to be met by demonstrating that thermal power remains less than 112% of full power for the duration of the event.

As explained in Section 6.0 of Reference [2], the equivalent break size of a fully opened PCSV is at the lowest end of the spectrum for small break loss-of-coolant accidents (SBLOCA). For the IPRVO event, there is ample time for operator action to isolate the break long before core uncovery is a concern. Therefore, the event will not evolve into a different classification.

3.3 Methodology

The IPRVO analysis uses the approved methodology in Reference [3]. Where possible, this methodology utilizes the plant design bases to establish acceptance criteria and input boundary conditions. The approved methodology includes the manner for determining the responses of the primary system, the secondary system, and the core to postulated accidents. In addition, the methodology requires the use of conservative setpoints, valve and pump capacities, and reactivity coefficients to demonstrate adequate margin to the applicable limits.

The analysis is performed using the RELAP5/MOD2-B&W computer code. This code has been approved by the NRC for use in non-LOCA safety analyses [3] [4]. The code simulates RCS and secondary system operation. The reactor core model is based on a point kinetics solution with reactivity feedback for control rod assembly insertion, fuel temperature changes, and moderator temperature changes. The RCS model provides for heat transfer from the core, transport of the coolant to the SGs, and heat transfer to the SGs. The secondary model includes a detailed depiction of the main steam system, including steam relief to the atmosphere through the MSSVs, TBVs, and simulation of the TSVs. The secondary model also includes the delivery of feedwater, both main and emergency, to the SGs.

3.4 Steady-State Initialization

The analysis in Reference [1] began by creating a steady-state input model that targeted the conditions in Table 2-1. Table 3-1 demonstrates the comparison between the targeted conditions and the conditions achieved by the steady-state input model.

Table 3-1: Steady-State Verification Results

Parameter	Target	Steady-State Value
Core Power, MWt	3026.1	3026.1
RCS T_{ave}, °F	582	581.94
RCS Pressure, psia	2170	2169.7
RCS Flow Rate, gpm	374,880	374,866
Core Bypass Flow, %	7.5	7.5030
RCP Heat, MWt	16.4	16.522
PZR Level, in	220	220.22
SG Level - operate range, % span	70	72.567 (SG A) 72.754 (SG B)
Turbine Header Pressure, psia	930	883.30

With the exception of the turbine header pressure, the parameters are reasonably close to the targeted conditions. As noted in Section 7.4.1 of Reference [2], the turbine header pressure was adjusted as needed to achieve the targeted Tave, and is therefore acceptable.

3.5 Transient Analysis

The plots associated with the transient are provided in Figures 3-1 through 3-14.

The IPRVO event is initiated by assuming that one PCSV opens suddenly during power operation and remains open during the event. The PCSV Flow Rate reaches 63.2 lbm/s at t=1s, and decreases slowly to reach 54 lbm/s before reactor trip due to RCS depressurization. The valve opening causes loss of reactor coolant system (RCS) inventory, rapid RCS depressurization, and decrease in fluid density. Pressurizer heaters are assumed to be off for this event and the RCS pressure continues to decrease, leading to a rapid change in density. RCS flow is maintained by the running reactor coolant pumps (RCPs). The reduction in moderator density causes positive reactivity to be inserted as a result of the positive moderator temperature coefficient. Consequently, the core power increases. As the depressurization continues, the Reactor Protection System (RPS) automatically functions at 45 seconds, causing a reactor trip on Low RCS Pressure (P<1894 psia). The reactor trip automatically trips the turbine. The resulting insertion of scram reactivity ends the power excursion and negates further challenge to the core DNB limits. Hence, the event is considered terminated.

The total neutron power increases from 3026.10 to 3100.92 MWt at 45 s, and as can be seen in Figure 3-2. This is an increase of 2.5%, and is 2.9% higher than the rated thermal power of 3014 MWt. The total neutron power increase bounds the thermal power; therefore, thermal power remains less than 112% of full power for the duration of the event. Since the thermal power remains below 112% of the rated thermal power, the fuel cladding integrity is maintained.

Figure 3-1 shows that the RCS pressure is constantly decreasing until reactor trip, and is always less than the initial value. Therefore, the peak RCS pressure acceptance criteria is not challenged during the IPRVO event.

Based on the analysis results, all acceptance criteria are met.

Figure 3-1: CR-3 Full Power at 3026.1 MWt

Inadvertent Pressurizer Relief Valve Opening

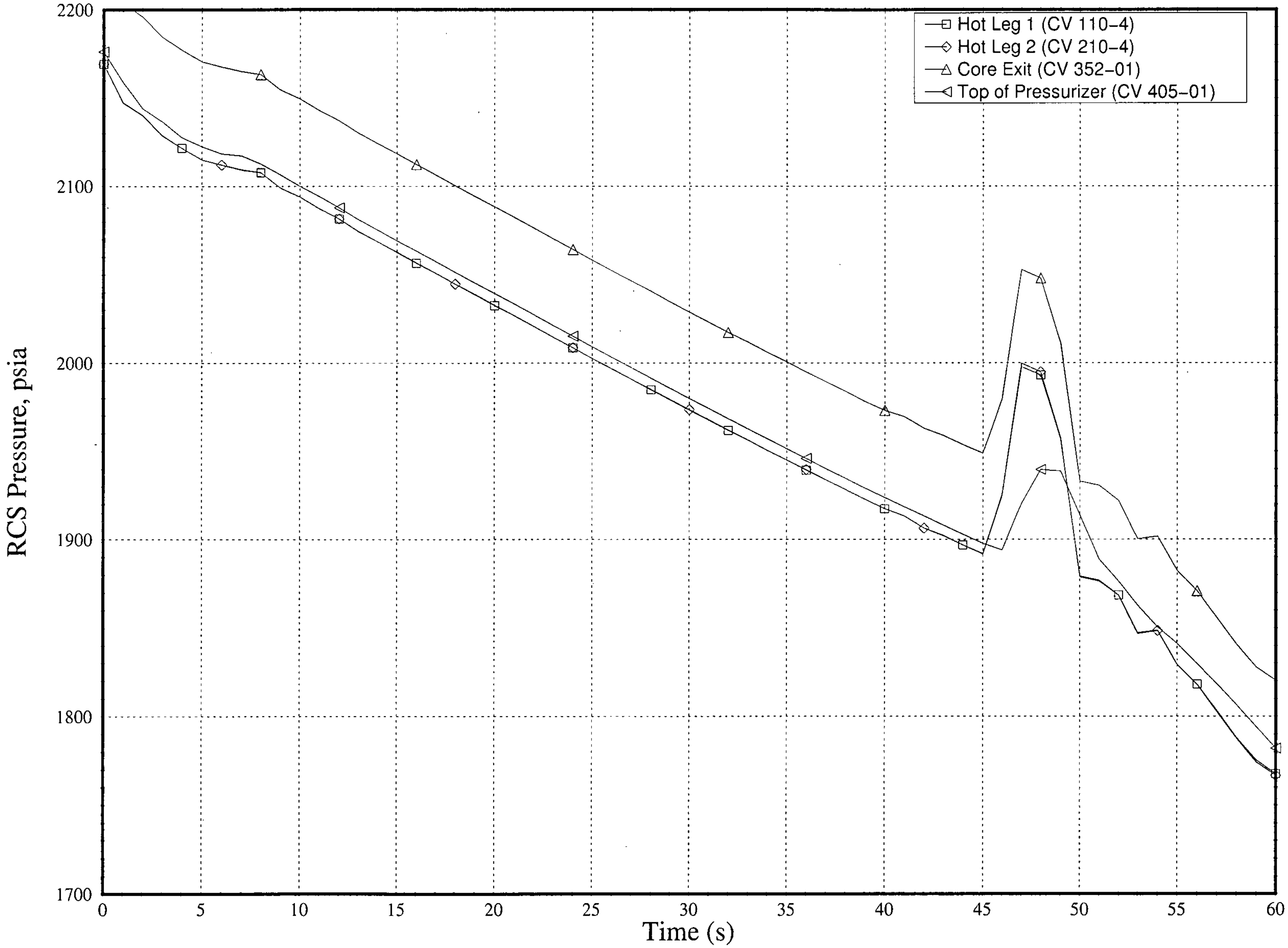

Figure 3–2: CR–3 Full Power at 3026.1 MWt

Inadvertent Pressurizer Relief Valve Opening

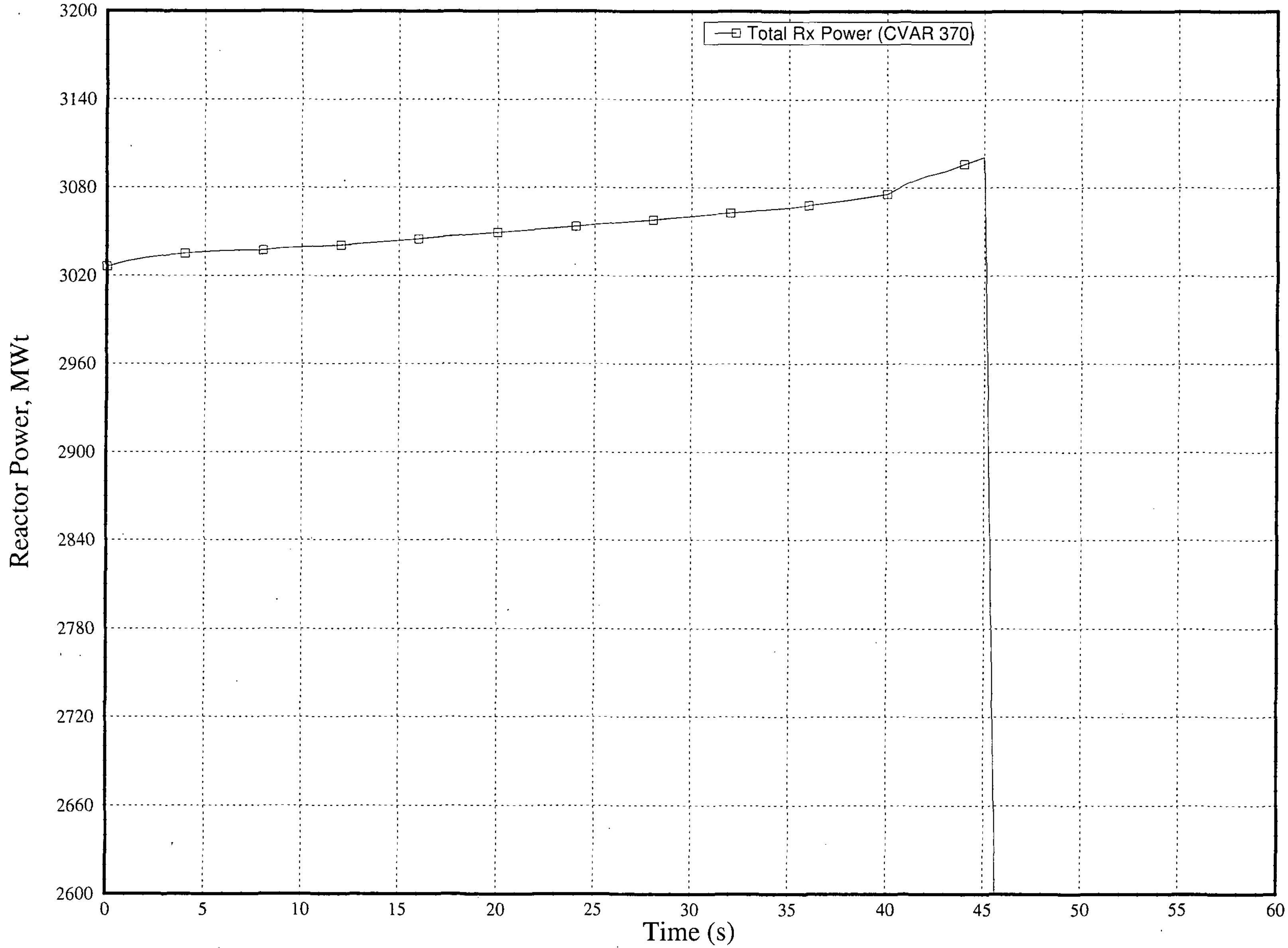

Figure 3–3: CR–3 Full Power at 3026.1 MWt

Inadvertent Pressurizer Relief Valve Opening

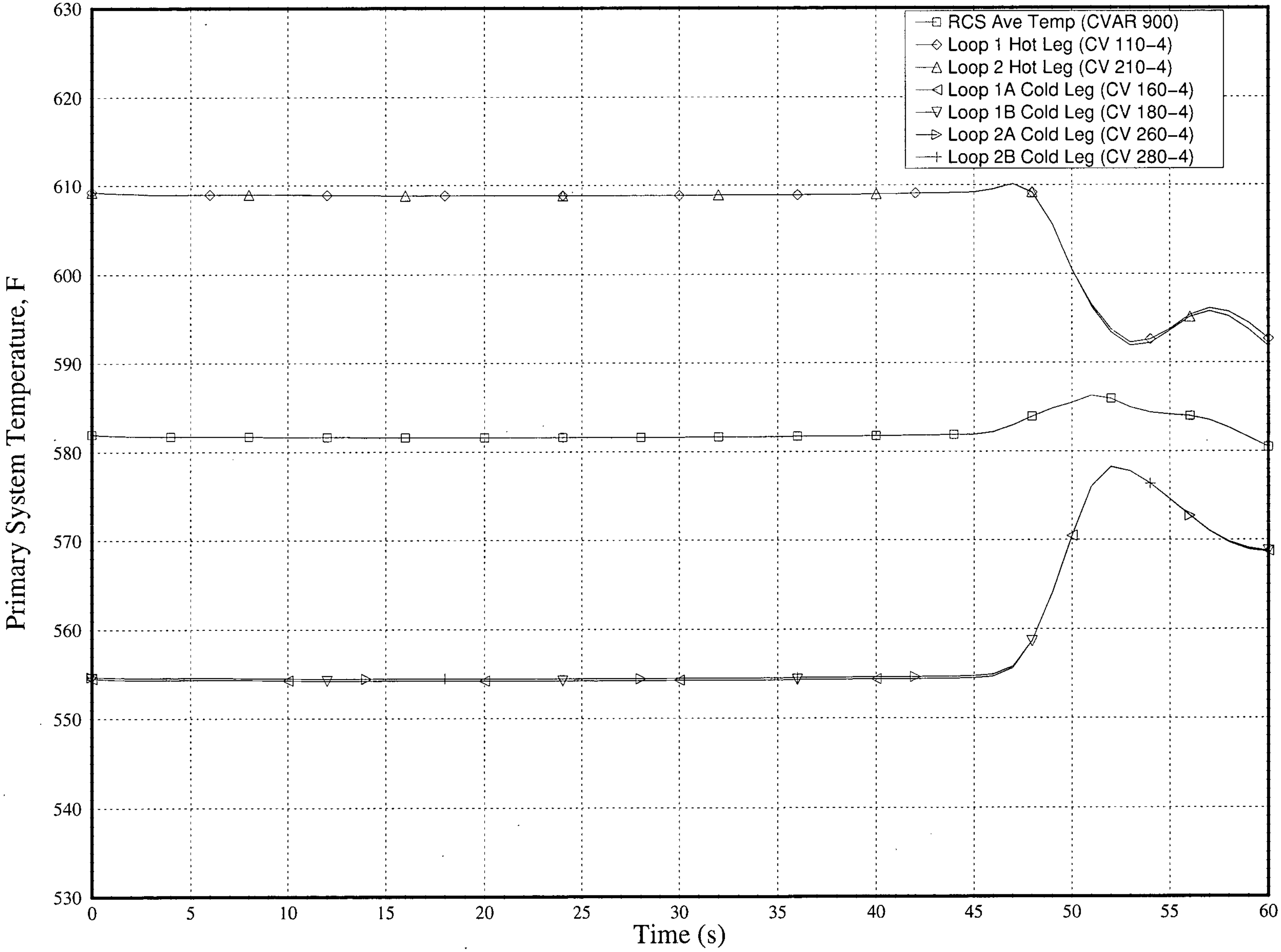

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Figure 3–4: CR–3 Full Power at 3026.1 MWt

Inadvertent Pressurizer Relief Valve Opening

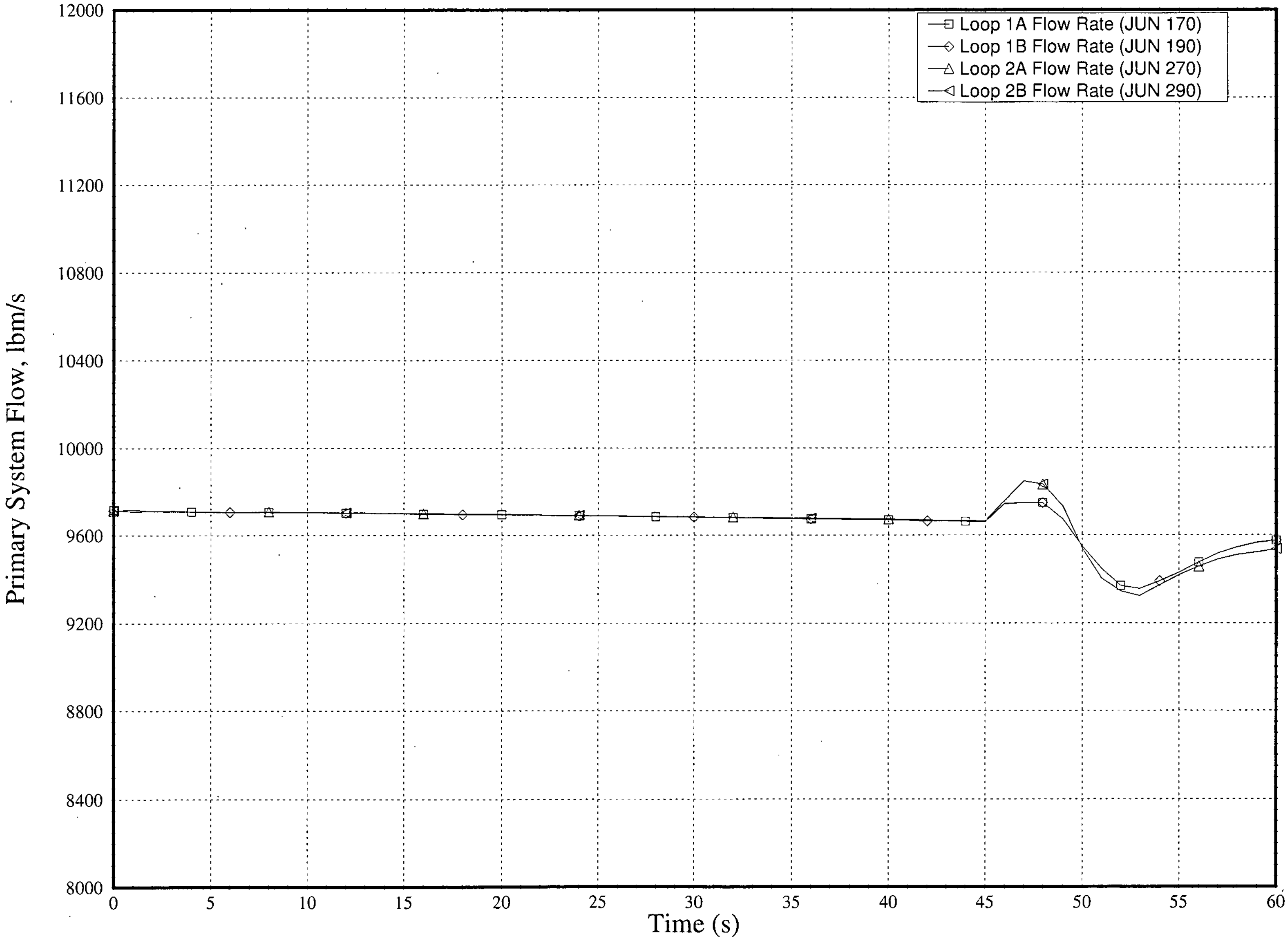

Figure 3–5: CR–3 Full Power at 3026.1 MWt

Inadvertent Pressurizer Relief Valve Opening

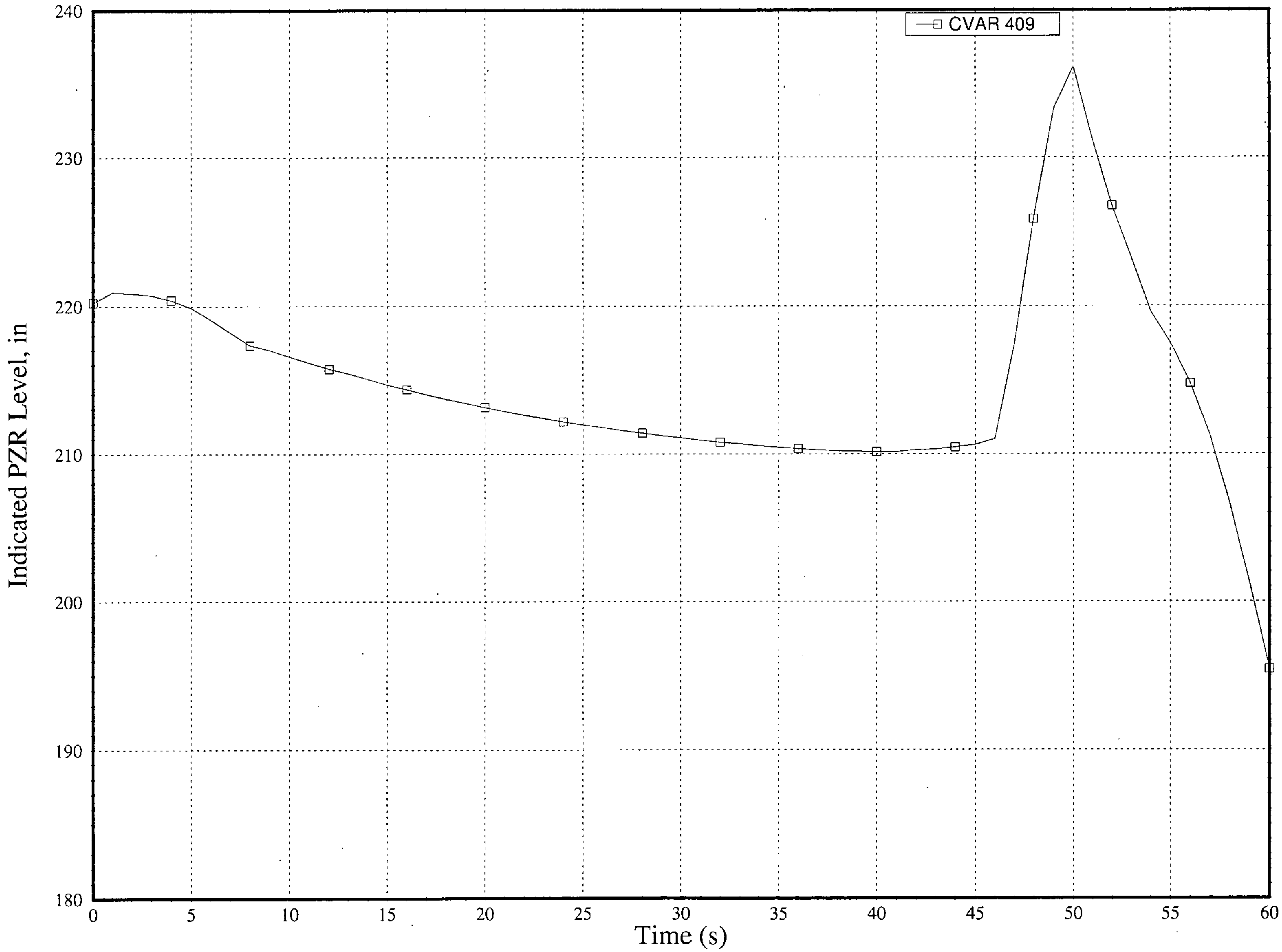

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Figure 3–6: CR–3 Full Power at 3026.1 MWt

Inadvertent Pressurizer Relief Valve Opening

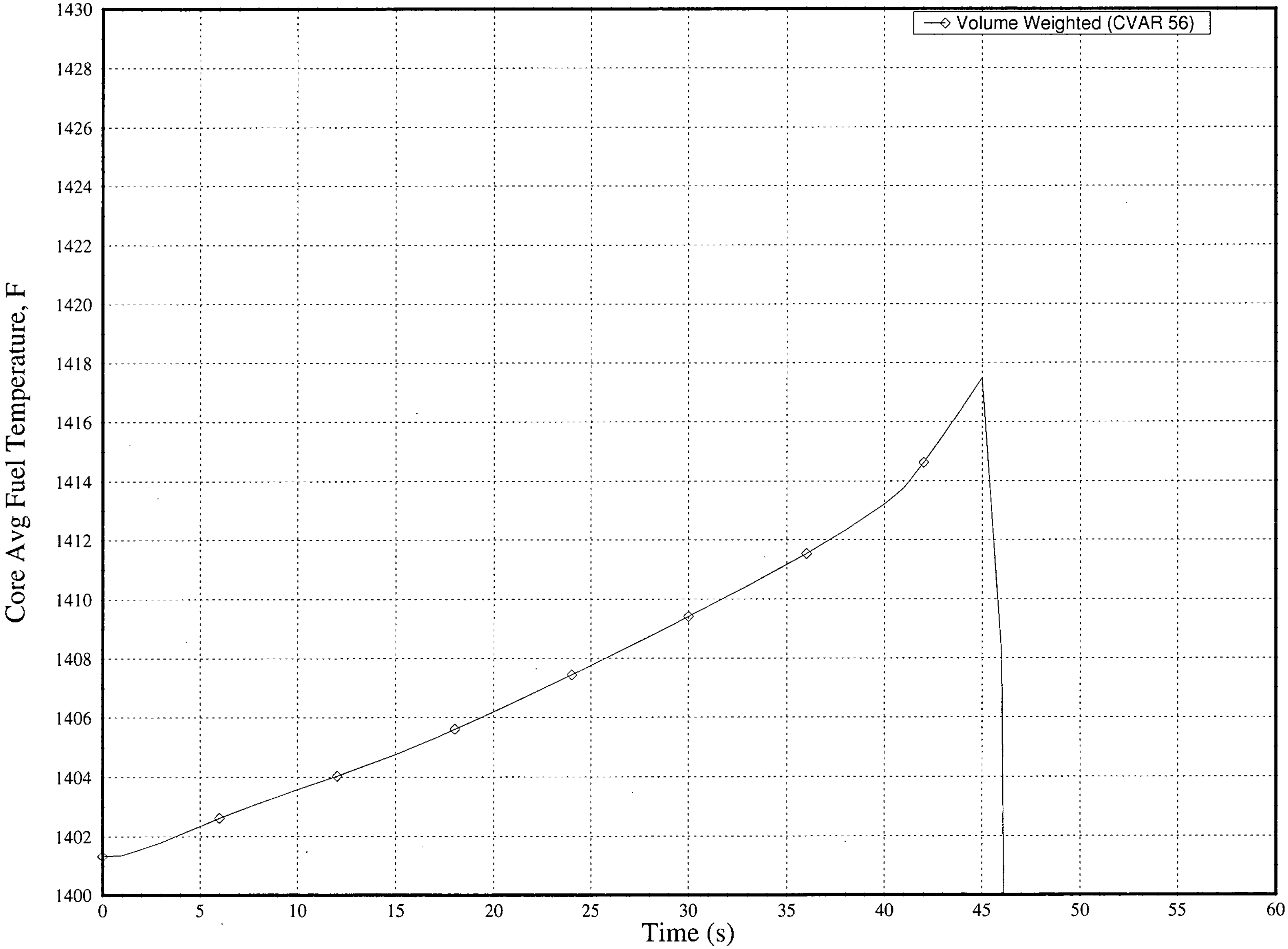

Figure 3-7: CR-3 Full Power at 3026.1 MWt

Inadvertent Pressurizer Relief Valve Opening

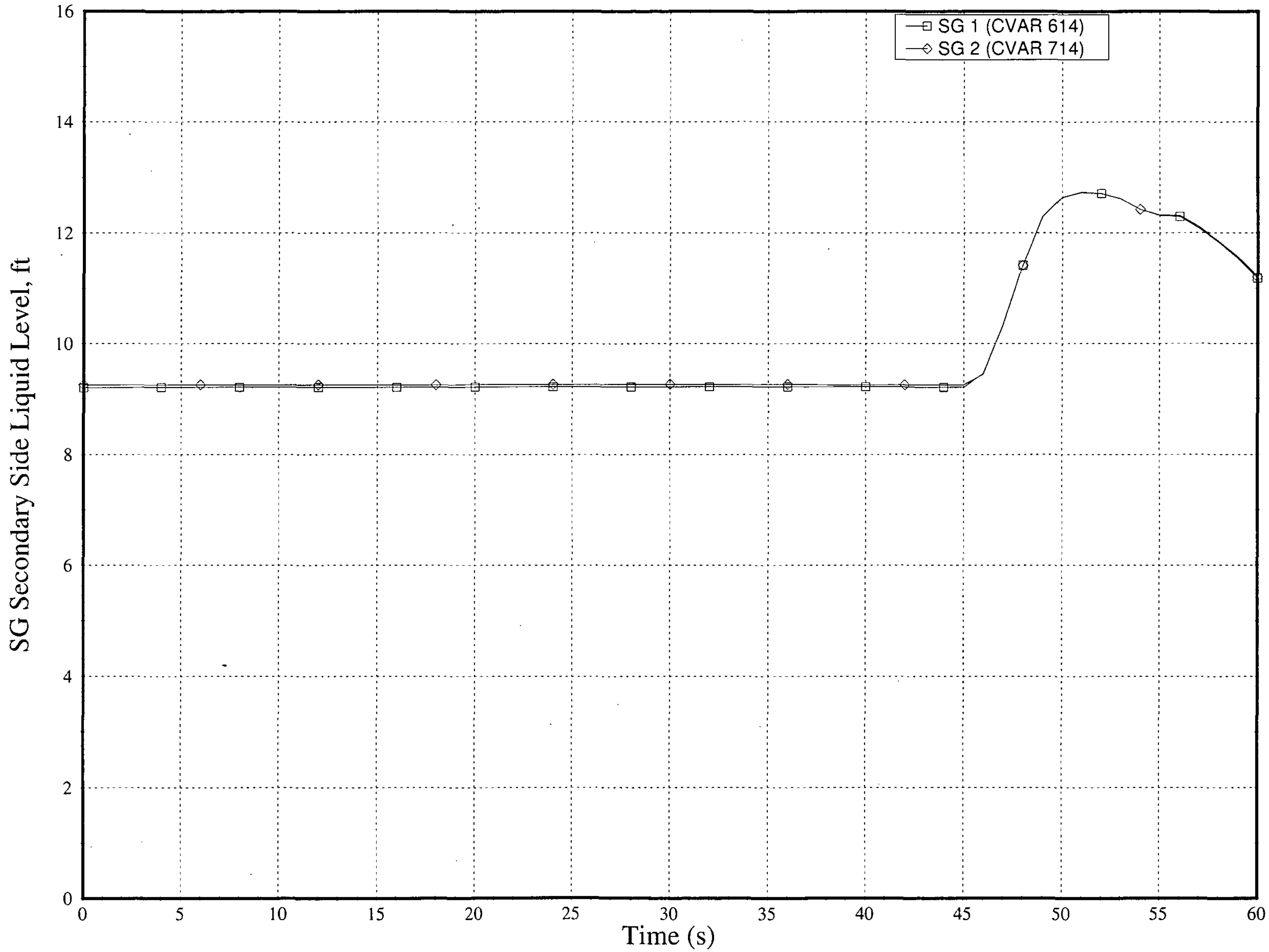

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Figure 3–8: CR–3 Full Power at 3026.1 MWt

Inadvertent Pressurizer Relief Valve Opening

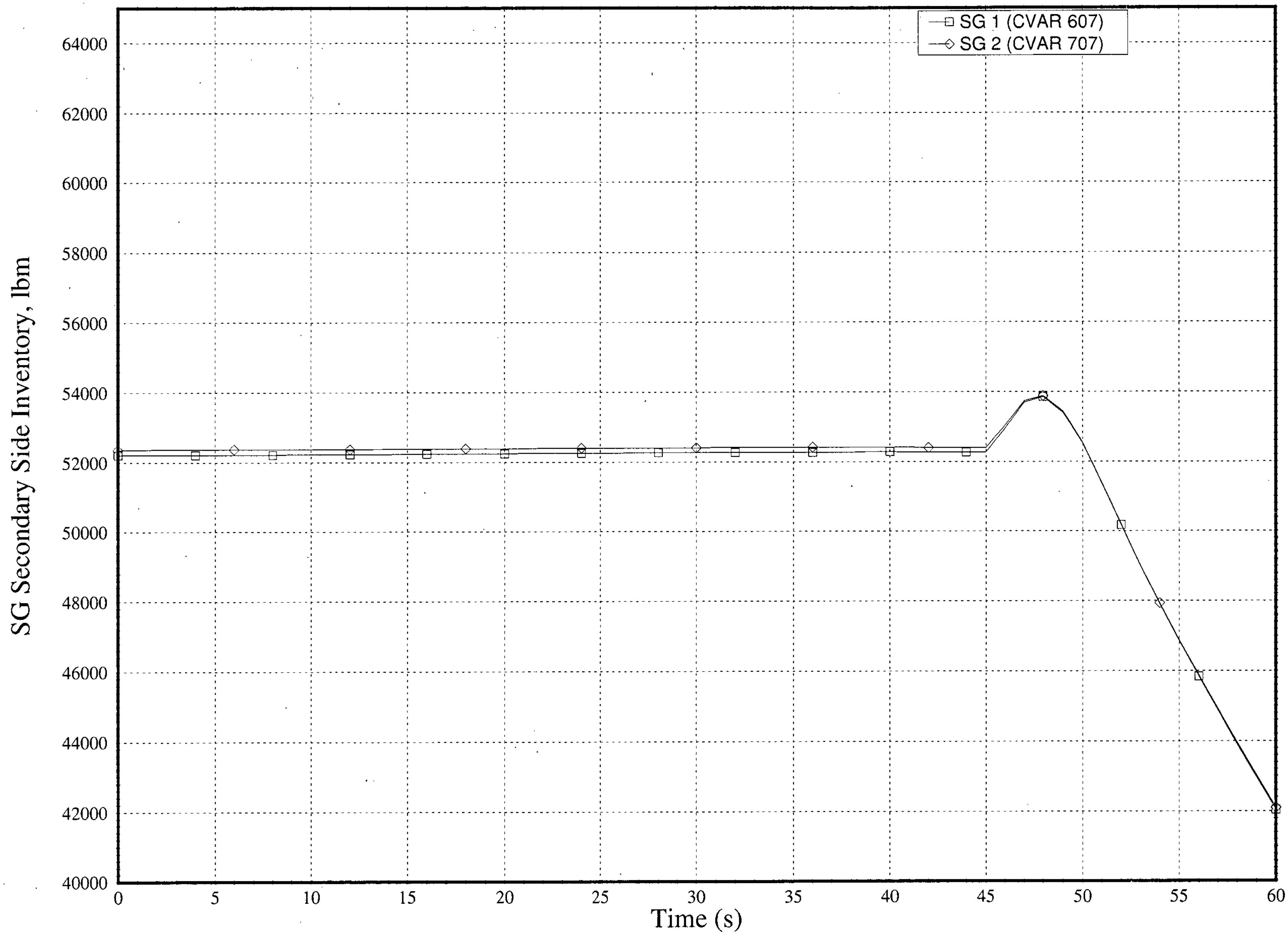

Figure 3–9: CR–3 Full Power at 3026.1 MWt

Inadvertent Pressurizer Relief Valve Opening

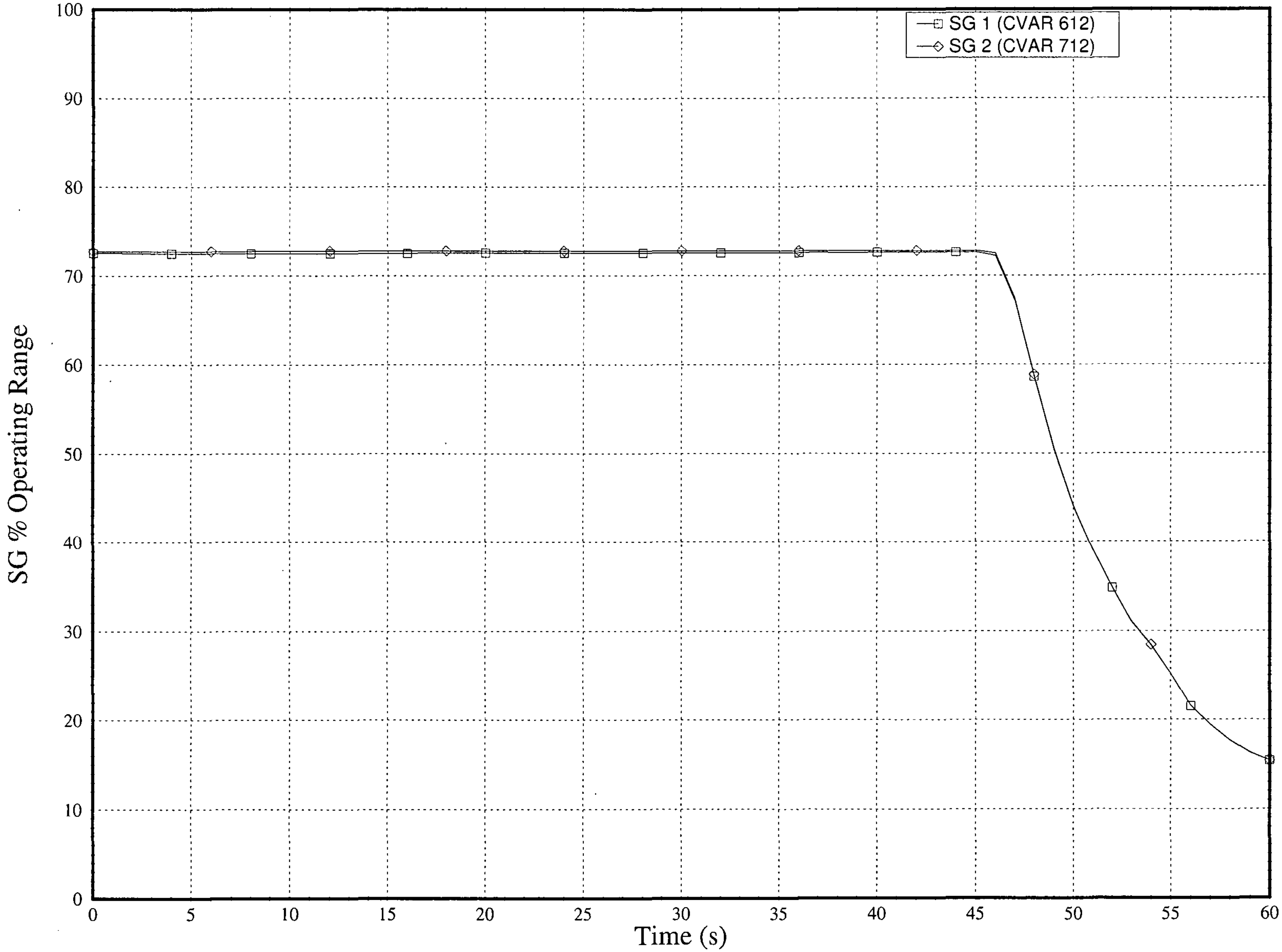

Figure 3-10: CR-3 Full Power at 3026.1 MWt

Inadvertent Pressurizer Relief Valve Opening

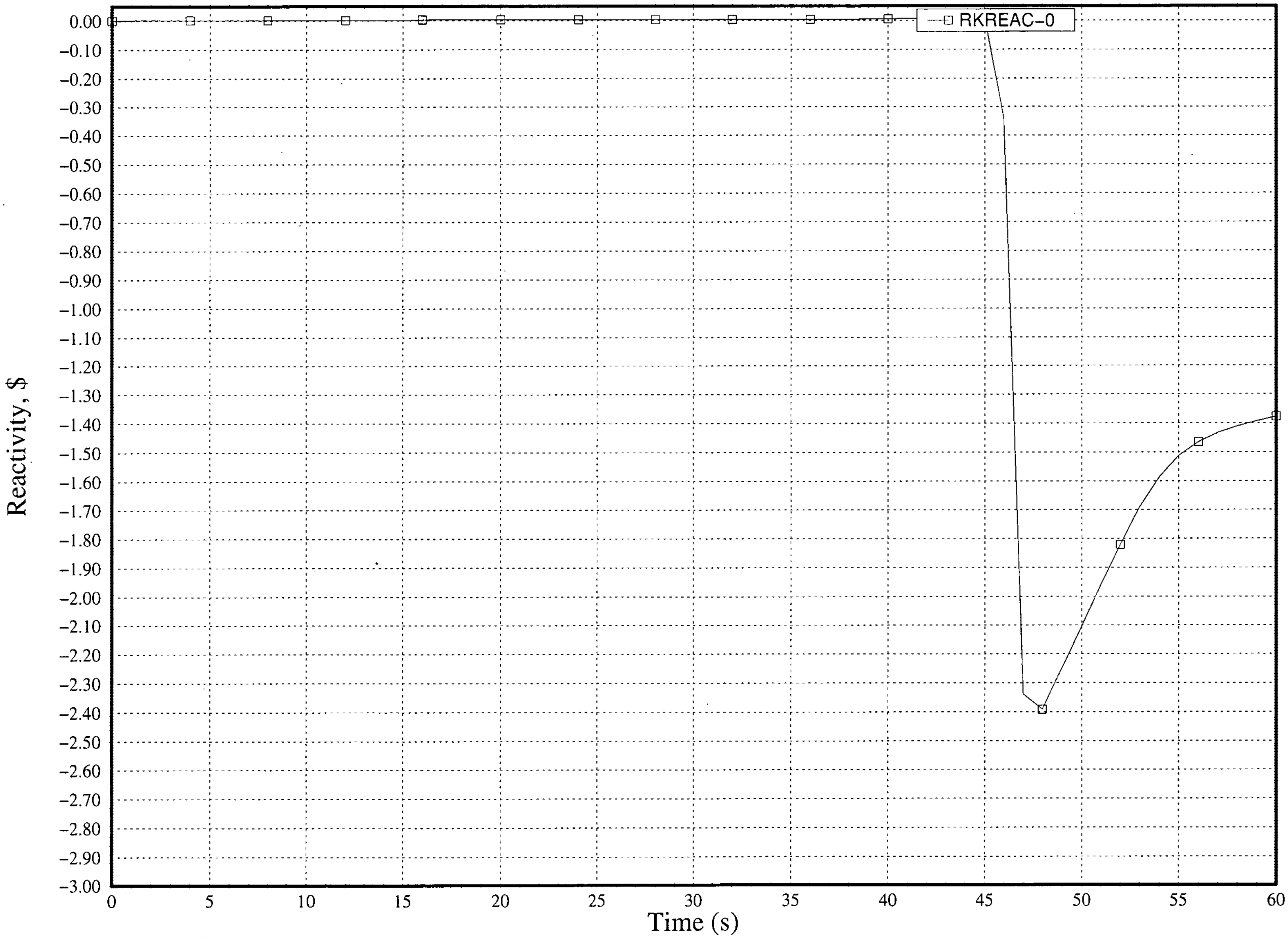

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Figure 3-11: CR-3 Full Power at 3026.1 MWt

Inadvertent Pressurizer Relief Valve Opening

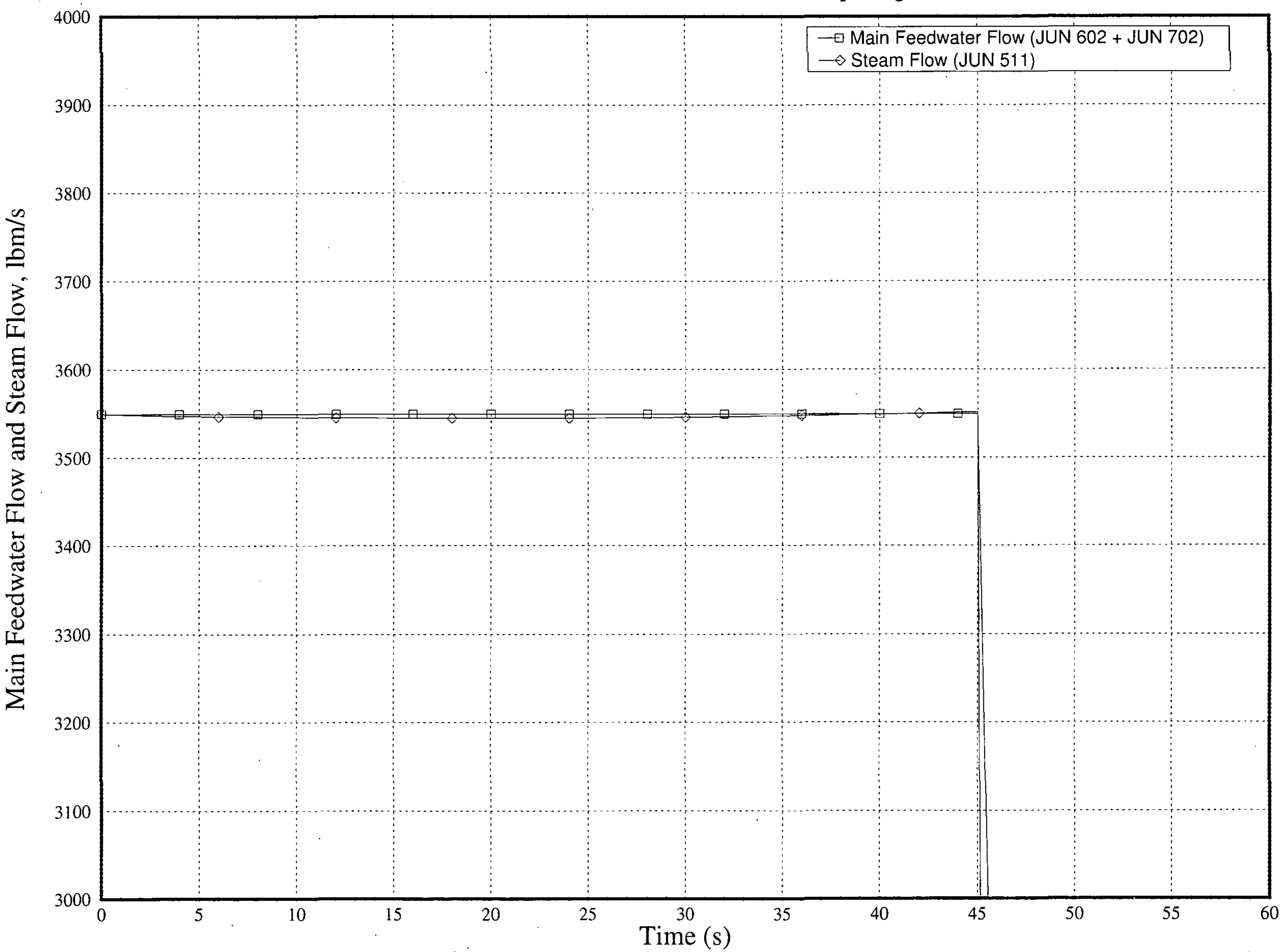

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Figure 3–12: CR–3 Full Power at 3026.1 MWt

Inadvertent Pressurizer Relief Valve Opening

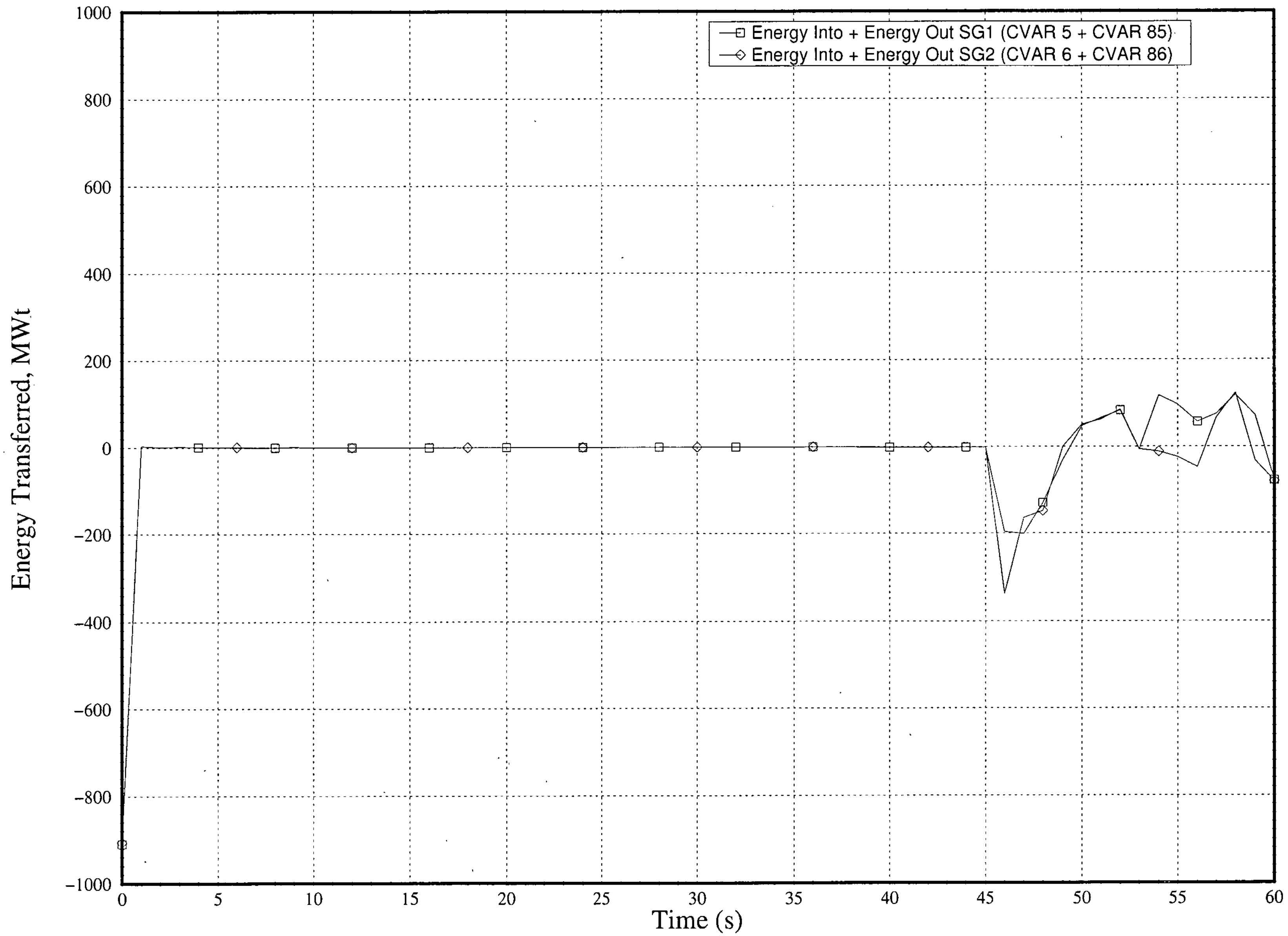


Figure 3–13: CR–3 Full Power at 3026.1 MWt

Inadvertent Pressurizer Relief Valve Opening

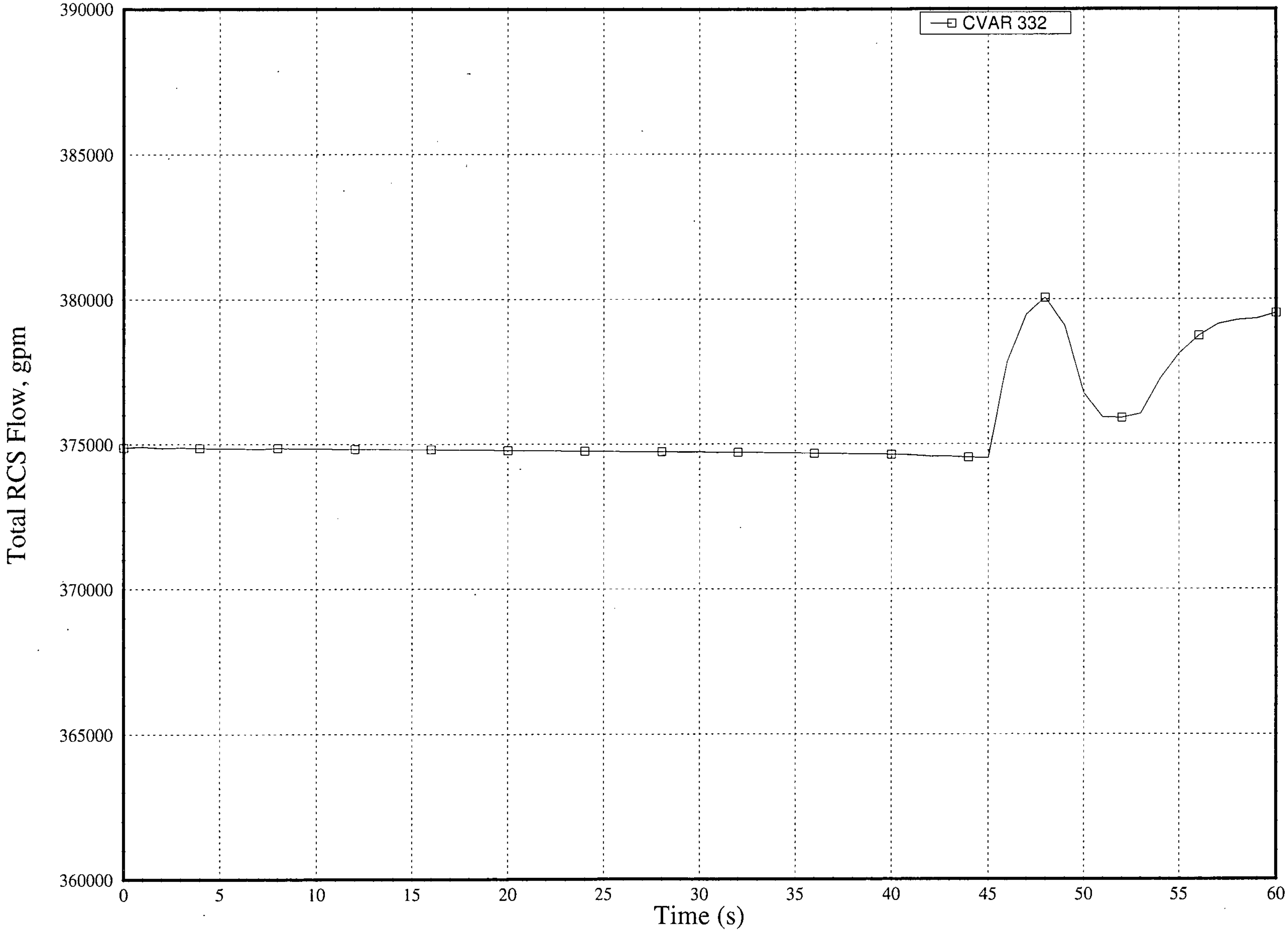
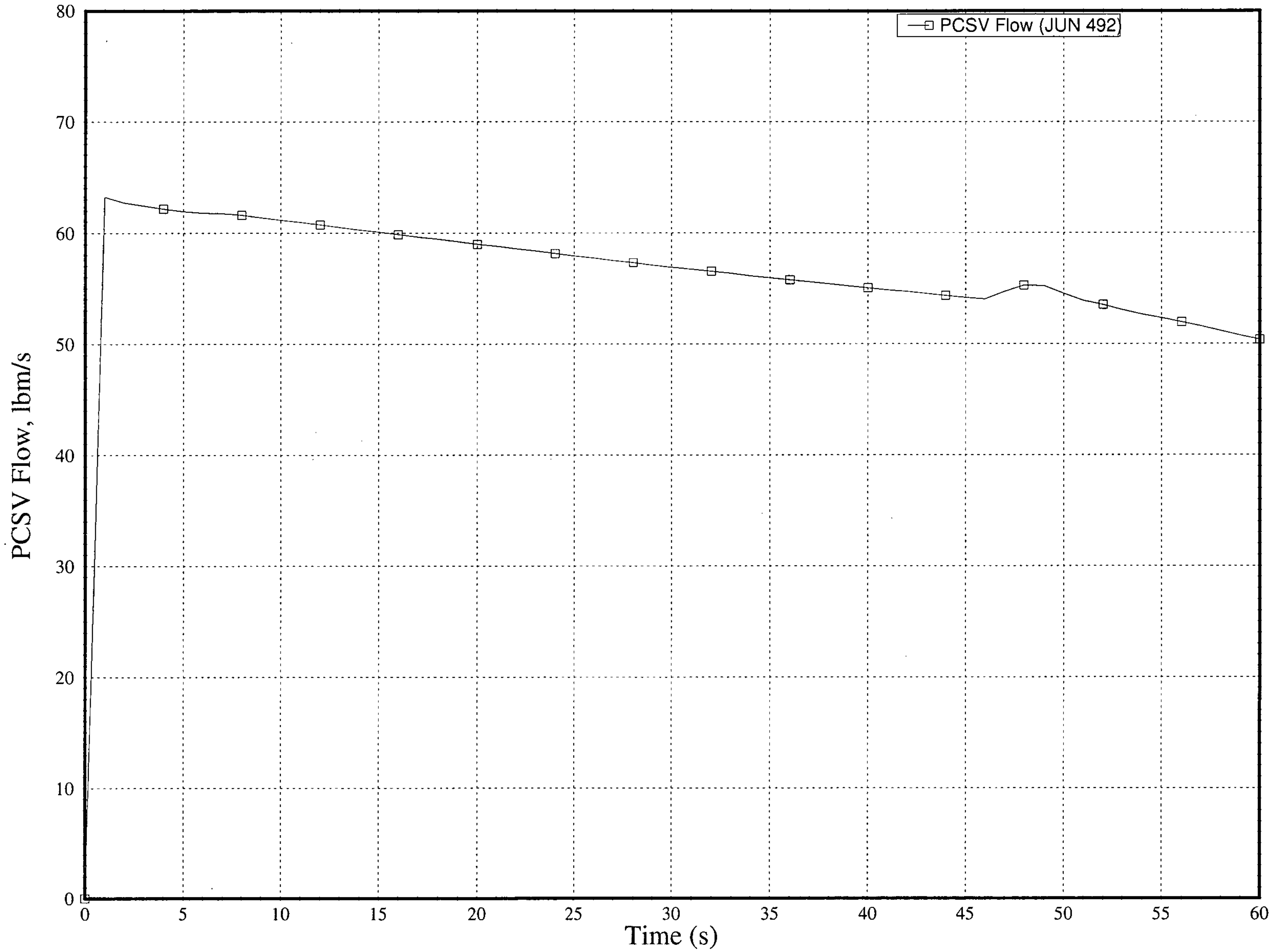


Figure 3-14: CR-3 Full Power at 3026.1 MWt

Inadvertent Pressurizer Relief Valve Opening



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4.0 REFERENCES

1) AREVA Document 32-9086732-000, "CR-3 EPU Inadvertent Pressurizer Relief Valve Opening."
2) AREVA Document 51-9083076-000, "CR-3 EPU Inadvertent Pressurizer Relief Valve Opening AIS."
3) AREVA Document 43-10193PA-00, "RELAP5/MOD2-B&W for Safety Analysis of B&W-Designed Pressurized Water Reactors."
4) AREVA Document 43-10164PA-06, "RELAP5/MOD2-B&W – An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis."
5) NUREG 0800, Section 15.6.1, "Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve," US NRC, Rev. 2, 03/2007.