ArevaEPRDCDocsPEm Resource

From: WILLIFORD Dennis (AREVA) [Dennis.Williford@areva.com]
Sent: Thursday, June 27, 2013 2:10 PM
To: Snyder, Amy
Cc: Gleaves, Bill; ANDERSON Katherine (EXTERNAL AREVA); DELANO Karen (AREVA); HONMA George (EXTERNAL AREVA); LEIGHLITER John (AREVA); LEWIS Ray (EXTERNAL AREVA); ROMINE Judy (AREVA); RYAN Tom (AREVA); SHEPHERD Tracey (AREVA); VANCE Brian (AREVA); HOLM Jerald (EXTERNAL AREVA); RITCHEY Calvin (AREVA)
Subject: Advanced Response to U.S. EPR Design Certification Application FINAL RAI No. 582, FSAR Ch. 15 - NEW PHASE 4 RAI (ATWS), Question 15.08-1
Attachments: Advanced Response to RAI 582 Question 15.08-1 US EPR DC.pdf

Importance: High

Amy,

Attached is an Advanced Response to RAI 582, Question 15.08-1 prior to the final response date of August 30, 2013 .

To keep our commitment to send a final response to this question by the commitment date, we need to receive all NRC staff feedback and comments no later than **August 5, 2013**.

Please let me know if NRC staff has any questions or if the response to this question can be sent as final.

Sincerely,

Dennis Williford, P.E.
U.S. EPR Design Certification Licensing Manager
AREVA NP Inc.
7207 IBM Drive, Mail Code CLT 2B
Charlotte, NC 28262
Phone: 704-805-2223
Email: Dennis.Williford@areva.com

From: KOWALSKI David (RS/NB)
Sent: Tuesday, June 18, 2013 12:47 PM
To: Amy.Snyder@nrc.gov
Cc: Bill.Gleaves@nrc.gov; ANDERSON Katherine (External AREVA NP INC.); DELANO Karen (RS/NB); LEIGHLITER John (RS/NB); LEWIS Ray (External RS/NB); ROMINE Judy (RS/NB); RYAN Tom (RS/NB); SHEPHERD Tracey (RS/NB); VANCE Brian (RS/NB); GUCWA Len (External RS/NB); KOWALSKI David (RS/NB); WILLIFORD Dennis (RS/NB); HOLM Jerald (External RS/NB); RITCHEY Calvin (EP/PE)
Subject: Response to U.S. EPR Design Certification Application FINAL RAI No. 582, FSAR Ch. 15 - NEW PHASE 4 RAI (ATWS)
Importance: High

Amy,

Attached please find AREVA NP Inc.'s response to the subject request for additional information (RAI). The attached file, "RAI 582 Response US EPR DC," provides a schedule since a technically correct and complete response to the question is not provided.

The following table indicates the respective pages in the response document, "RAI 582 Response US EPR DC.pdf," that contain AREVA NP's response to the subject question.

Question #	Start Page	End Page
RAI 582 — 15.08-1	2	2

The schedule for a technically correct and complete final response to this question is provided below.

Question #	Response Date
RAI 582 — 15.08-1	August 30, 2013

Sincerely,

David Kowalski for

Dennis Williford, P.E.
U.S. EPR Design Certification Licensing Manager
AREVA NP Inc.
7207 IBM Drive, Mail Code CLT 2B
Charlotte, NC 28262
Phone: 704-805-2223
Email: Dennis.Williford@areva.com

From: Snyder, Amy [mailto:Amy.Snyder@nrc.gov]
Sent: Thursday, June 13, 2013 7:09 AM
To: ZZ-DL-A-USEPR-DL
Cc: Budzynski, John; Donoghue, Joseph; Gleaves, Bill; Segala, John
Subject: U.S. EPR Design Certification Application FINAL RAI No. 582, FSAR Ch. 15 - NEW PHASE 4 RAI (ATWS)

Attached please find the subject request for additional information (RAI). A draft RAI was provided to you on April 4, 2013. On May 30, 2013, there was a public meeting with AREVA on this draft RAI. On May 31, 2013, you informed us that the draft RAI does not contain proprietary information and that the draft RAI is clear and no further clarification is needed. As result, the RAI was not changed.

The schedule we have established for review of your application assumes technically correct and complete responses within 30 days of receipt of RAIs,. For any RAIs that cannot be answered **within 30 days or July 15, 2013**, it is expected that a date for receipt of this information will be provided to the staff within the 30-day period so that the staff can assess how this information will impact the published schedule.

Thank You,

Amy

Amy Snyder, U.S. EPR Design Certification Lead Project Manager
Licensing Branch 1 (LB1)
Division of New Reactor Licensing
Office of New Reactors
U.S. Nuclear Regulatory Commission
☎ Office: (301) 415-6822

Fax: (301) 415-6406
Mail Stop: T6-C20M
E-mail: Amy.Snyder@nrc.gov

Hearing Identifier: AREVA_EPR_DC_Docs_Public
Email Number: 38

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Subject: Advanced Response to U.S. EPR Design Certification Application FINAL RAI No. 582, FSAR Ch. 15 - NEW PHASE 4 RAI (ATWS), Question 15.08-1
Sent Date: 6/27/2013 2:09:50 PM
Received Date: 6/27/2013 2:10:14 PM
From: WILLIFORD Dennis (AREVA)

Created By: Dennis.Williford@areva.com

Recipients:
"Gleaves, Bill" <Bill.Gleaves@nrc.gov>
Tracking Status: None
"ANDERSON Katherine (EXTERNAL AREVA)" <katherine.anderson.ext@areva.com>
Tracking Status: None
"DELANO Karen (AREVA)" <Karen.Delano@areva.com>
Tracking Status: None
"HONMA George (EXTERNAL AREVA)" <George.Honma.ext@areva.com>
Tracking Status: None
"LEIGHLITER John (AREVA)" <John.Leighliter@areva.com>
Tracking Status: None
"LEWIS Ray (EXTERNAL AREVA)" <Ray.Lewis.ext@areva.com>
Tracking Status: None
"ROMINE Judy (AREVA)" <Judy.Romine@areva.com>
Tracking Status: None
"RYAN Tom (AREVA)" <Tom.Ryan@areva.com>
Tracking Status: None
"SHEPHERD Tracey (AREVA)" <Tracey.Shepherd@areva.com>
Tracking Status: None
"VANCE Brian (AREVA)" <Brian.Vance@areva.com>
Tracking Status: None
"HOLM Jerald (EXTERNAL AREVA)" <Jerald.Holm.ext@areva.com>
Tracking Status: None
"RITCHEY Calvin (AREVA)" <Calvin.Ritchey@areva.com>
Tracking Status: None
"Snyder, Amy" <Amy.Snyder@nrc.gov>
Tracking Status: None

Post Office: FUSLYNCMX03.fdom.ad.corp

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MESSAGE	3856	6/27/2013 2:10:14 PM	
Advanced Response to RAI 582 Question 15.08-1 US EPR DC.pdf			1067791

Options
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Return Notification: No
Reply Requested: No
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Recipients Received:

Advanced Response to

Request for Additional Information No.582, Question 15.08-1

4/4/2013
U.S. EPR Standard Design Certification
AREVA NP Inc.
Docket No. 52-020
SRP Section: 15.08 - Anticipated Transients Without Scram
Application Section: SRP 15.8
SRSB Branch

Question 15.08-1:

AREVA's technical report ANP-10304, "USEPR Diversity and Defense - in-Depth Assessment" lists the increase in steam flow event as an anticipated operational occurrence. Consistent with paragraph (b) of Part 50.62, (Anticipated Transients Without Scram) ATWS events are anticipated operational occurrences as defined by Appendix A of Part 50 followed by the failure of the reactor trip portion of the protection system specified in General Design Criterion (GDC) 20. The current AREVA ATWS analysis does not provide an analysis of the increase in steam flow event followed by the postulated failure of the reactor protection system to perform the necessary safety functions. Therefore, the staff cannot conclude that the EPR design meets the requirements for ATWS events set forth in paragraph (c) of 10 CFR 50.62. Provide the analysis that demonstrates compliance with paragraph (c) for the subject increase in steam flow Anticipated Operational Occurrence (AOO) or provide a basis with the appropriate exemption request that supports a conclusion that compliance with paragraph (c) is not necessary.

Response to Question 15.08-1:

The regulation 10 CFR 50.62 (c) (1) states:

"(c) *Requirements*. (1) Each pressurized water reactor must have equipment from sensor output to final actuation device, that is diverse from the reactor trip system, to automatically initiate the auxiliary (or emergency) feedwater system and initiate a turbine trip under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner and be independent (from sensor output to the final actuation device) from the existing reactor trip system."

The U.S EPR plant contains a diverse actuation system (DAS) to automatically initiate the emergency feedwater (EFW) system, and to initiate a turbine trip under conditions indicative of an anticipated transient without scram (ATWS). The DAS is designed to perform its function in a reliable manner and is independent from the existing reactor trip system. The DAS initiates EFW when a condition indicative of an ATWS is sensed and then initiates a reactor trip and then a turbine trip. The condition indicative of an ATWS is defined for the U.S. EPR plant as the steam generator wide range low level setpoint. The purpose of the requirement in the 10 CFR 50.62 rule to initiate EFW and turbine trip, for "conditions indicative of an ATWS" versus for all ATWS events, is to assure that the actions are only taken when they serve to mitigate the event consequences, and not when the actions might increase the event consequences. U.S. FSAR Tier 2 Sections 15.8.1.2 and 15.8.1.3 will be revised to clarify the behavior of the DAS for an ATWS.

The analysis of the increase in steam flow event, an anticipated operational occurrence (AOO), in conjunction with an ATWS has been performed with a number of best estimate input assumptions. This is consistent with the statement in NUREG-0800, Section 15.8, Revision 2, March 2007:

"The probability of an AOO, in coincidence with multiple failures or a common mode failure, is much lower than the probability of any of the other events that are evaluated under SRP Chapter 15. An ATWS event cannot therefore be classified as either an AOO or a design-basis accident."

The criteria for an ATWS are defined the NUREG-0800, Section 15.8, Revision 2, March 2007 as:

1. Coolable geometry for the reactor core
2. Maintain reactor coolant pressure boundary integrity
3. Maintain containment integrity

Analysis Description

An ATWS analysis of the increase in steam flow AOO has been performed by utilizing appropriate methodology. ATWS events are considered beyond the design basis and are defined in NUREG-0800, Section 15.8 as an AOO followed by the failure of the reactor trip portion of the protection system. The ATWS increase in steam flow analysis has been performed utilizing best estimate methods:

- All safety and non-safety systems (except reactor trip portion of the protection system) are assumed to function normally;
- No single failure assumption;
- No stuck rod assumption;
- No equipment maintenance considerations;
- Nominal system setpoints without uncertainties applied;
- Nominal reactor power level without uncertainty;
- Best estimate system flow rates (RCS, SIS, and EFW);
- Best estimate reactivity feedback parameters (no biasing).

The U.S. EPR FSAR Tier 2, Chapter 15.1.2 Increase in Steam Flow limiting case (which is initiated by the inadvertent opening of five turbine bypass valves at end-of-cycle (EOC), full power conditions, with control rod positions being automatically controlled) has been re-analyzed using the best estimate method outlined above. The DAS trip functions are available as the backup to the failed reactor trip portion of the protection system. The event has been analyzed using the best-estimate, thermal hydraulic system performance code S-RELAP5.

Analysis Results

A sequence of events for this analysis is presented in Table 15.8-1–1. The transient is initiated at EOC full-power conditions by the inadvertent opening of five turbine bypass valves (Figure 15.8-1–1). The main feedwater flow controllers attempt to follow the increased steam flow (Figure 15.8-1–2), until the MFW control valves are wide open. With the main feedwater flow being less than the steam flow, the steam generator levels (Figure 15.8-1–3) begin to decrease.

The opening of the turbine bypass valves causes the main steam line pressures (Figure 15.8-1–4) to decrease. At 9.3 seconds after transient initiation, the protection system's high rate of main steam line pressure decrease setpoint for actuating reactor trip and main steam isolation is reached. The reactor trip portion is assumed to fail, but at 9.8 seconds the MSIVs begin to close, and by 14.8 seconds they are fully closed.

The decrease in steam generator pressures (Figure 15.8-1–5) and temperatures that starts at transient initiation causes the RCS to cool down (Figure 15.8-1–6). This increases the moderator and Doppler reactivities (Figure 15.8-1–7) and initiates an increase in reactor power (Figure 15.8-1–8), in conjunction with the EOC conditions.

When the MSIVs close, the steam generator depressurization, RCS cooldown, and reactor power increase are all arrested. At 17.6 seconds, the reactor power reaches its maximum value (122 percent of RTP) and begins to decrease. At 18.4 seconds, the average fuel rod heat flux reaches its maximum value (117.8 percent of its value at RTP) and begins to decrease.

During the initial RCS cooldown, the ex-core neutron flux detectors that the diverse actuation system monitors for its high neutron flux reactor trip channel indicate lower-than-actual reactor power levels (because of the denser-than-calibrated fluid between the detectors and the core). The ex-core detector decalibration and trip signal processing time delays insertion of scram rods by the diverse actuation system (on a high neutron flux signal) until 18.9 seconds — which is after the reactor power has already started to decrease.

Post-scram decay heat removal is provided by the main steam relief trains (MSRTs) and the main feedwater system.

As indicated in Table 15.8-1–2, this scenario does not challenge any of the SRP Section 15.8 acceptance criteria for system overpressure (primary or secondary) or for the specified acceptable fuel design limits (SAFDLs). The minimum departure from nucleate boiling ratio (MDNBR) and the peak linear heat generation rate (LHGR) results are calculated using the LYNXT computer code utilizing the system response calculated by S-RELAP5. Containment integrity is never challenged because this event has no containment release.

The analysis of the increase in steam flow event in conjunction with an ATWS shows that the event does not experience conditions indicative of an ATWS as defined for the U.S. EPR plant. EFW is therefore not initiated for this event. As discussed above, the analysis does indicate that the three criteria for an AOO + ATWS are met.

FSAR Impact:

U.S. EPR FSAR Tier 2, Section 15.8.1.2 and 15.8.1.3 will be revised as described in the response and indicated on the enclosed markup.

Table 15.8-1–1: Sequence of Events

Sequence of Event	Time (seconds)
With reactor operating at EOC full-power conditions (and control rod positions being automatically controlled), five turbine bypass valves inadvertently open (and remain open).	0.0
Indicated Main Steam Line 2 pressure reaches protection system's high rate of main steam line pressure decrease reactor trip and main steam isolation signal setpoint (but reactor-trip-portion of protection system is assumed to fail).	9.3
Main steam isolation signal is issued; MSIVs begin to close.	9.8
MSIVs are fully closed.	14.8
Reactor neutron power reaches maximum (122.0% RTP) and begins to decrease; Pressurizer spray is initiated.	17.6
Indicated reactor neutron power (decalibrated by excess heat removal) reaches diverse actuation system's high neutron flux reactor trip setpoint (115.0% RTP).	17.9
Reactor trip signal is issued.	18.5
Scram rods begin to drop into core, and reactor power begins to rapidly decrease.	18.9
Turbine trips, and admission valves begin to close (but MSIVs are already closed).	19.5
Peak RCS pressure (2457.2 psia) occurs at the bottom of the reactor vessel	21.0
RCS average temperature reaches maximum (~1°F above initial value) and begins to decrease.	~21.5
Pressurizer dome pressure reaches maximum (2373.1 psia) and begins to decrease.	~21.8
Indicated main steam line pressures reach MSRT setpoint.	22.9
Pressurizer spray is terminated.	~23.7
MSRTs begin to open.	~24.9
SG secondary-side pressures reach maximum (1449.7 psia), at bottom, and begin to decrease.	~25.0
Relief through MSRTs ends.	~63.5
Simulation is terminated.	90.0

Table 15.8-1-2: Results Comparison to Acceptance Criteria

Acceptance Criteria	Calculated Value	Applicable Limit	Time of Limiting Value (seconds)
Primary System Pressure	2457.2 psia	2803.2 psia	21.0
Secondary System Pressure	1449.7 psia	1593.2 psia	~25
DNBR	2.57	1.25	18.9
Linear Heat Generation Rate			
Fuel Centerline Melt	10.00 kW/ft	20.45 kW/ft	17.6
Clad Strain		17.2 kW/ft	

Figure 15.8-1–1: Main Steam Valve Flow Rates

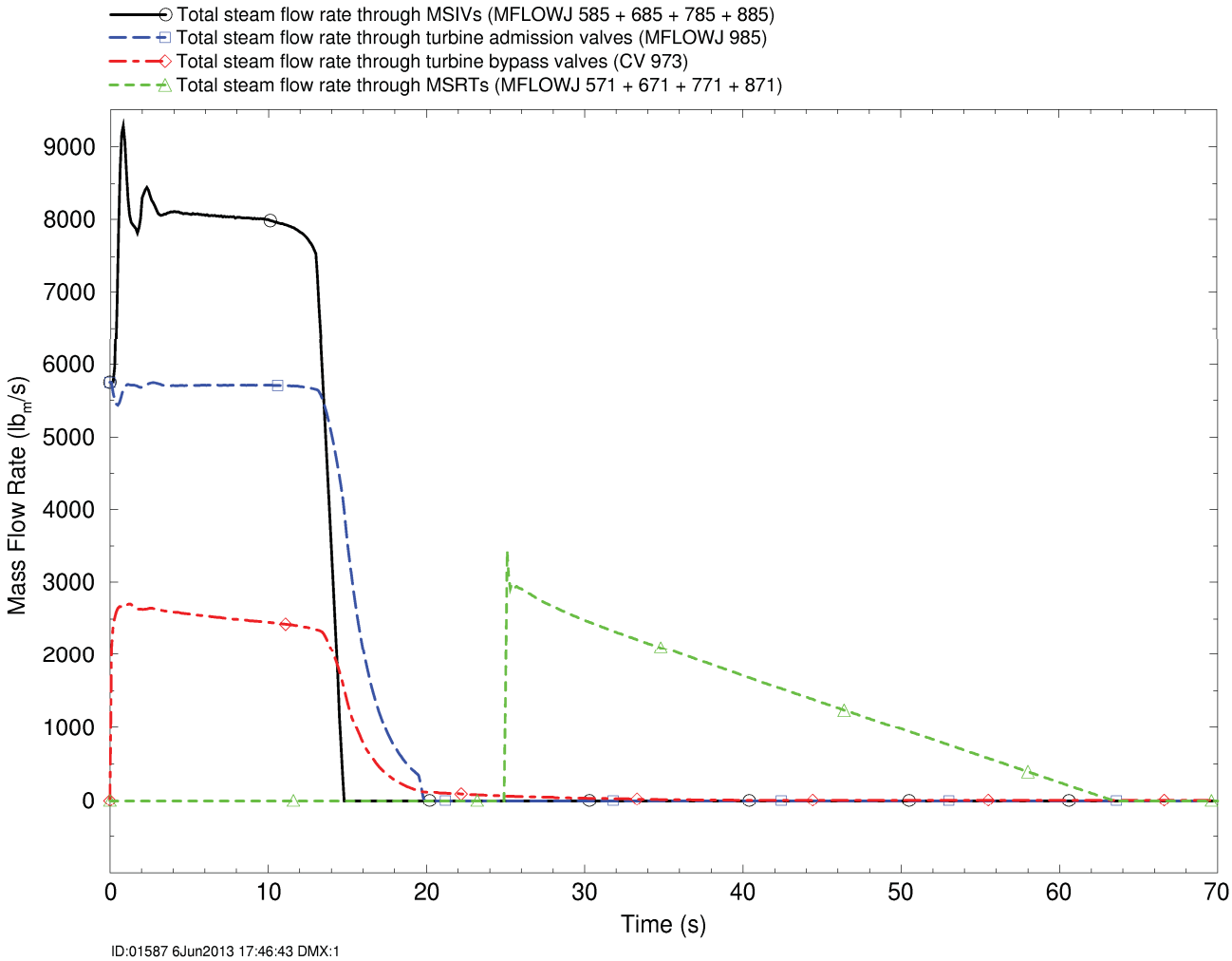


Figure 15.8-1–2: SG Steam, MFW, and Blowdown Flow Rates

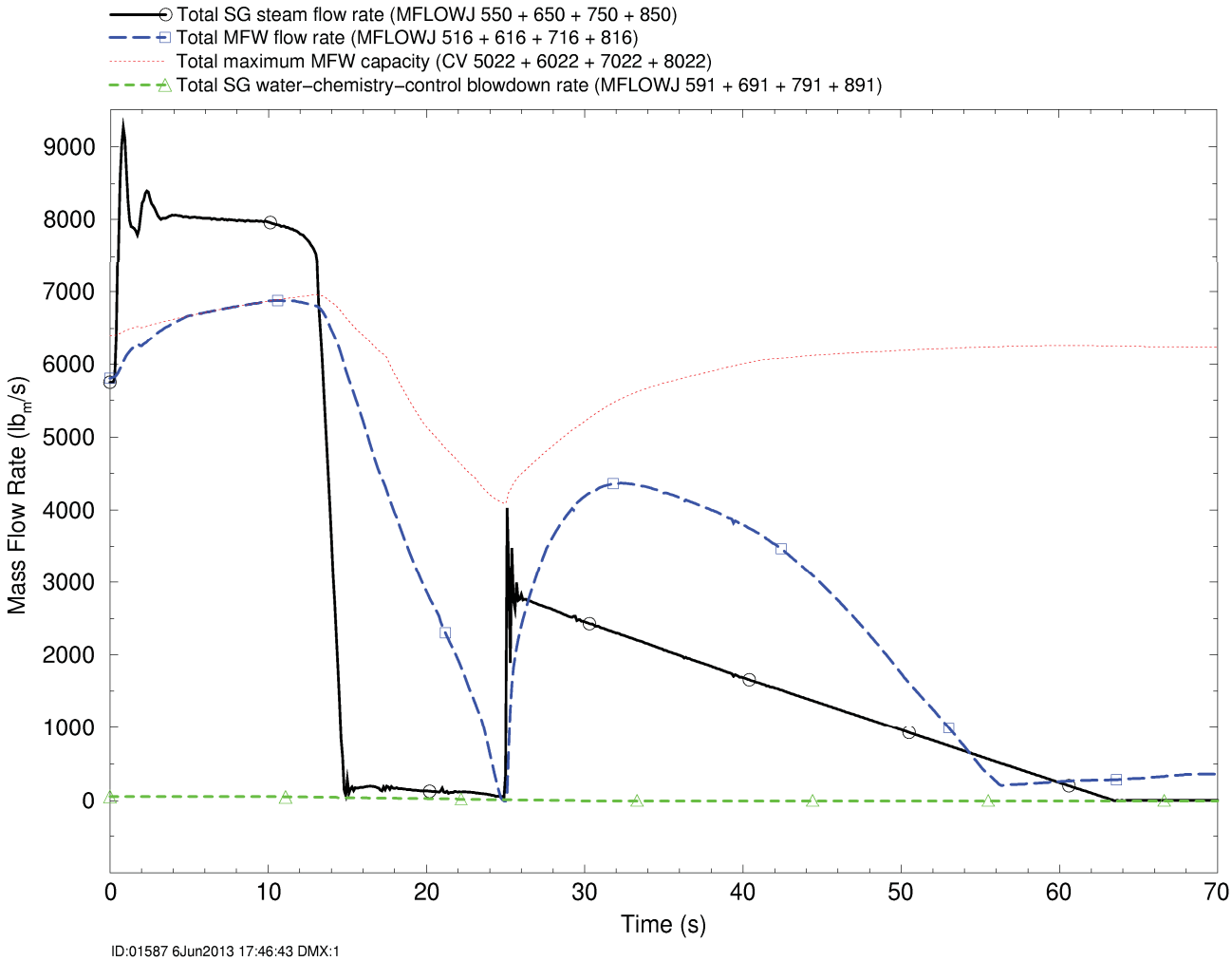


Figure 15.8-1–3: SG Wide-Range Levels

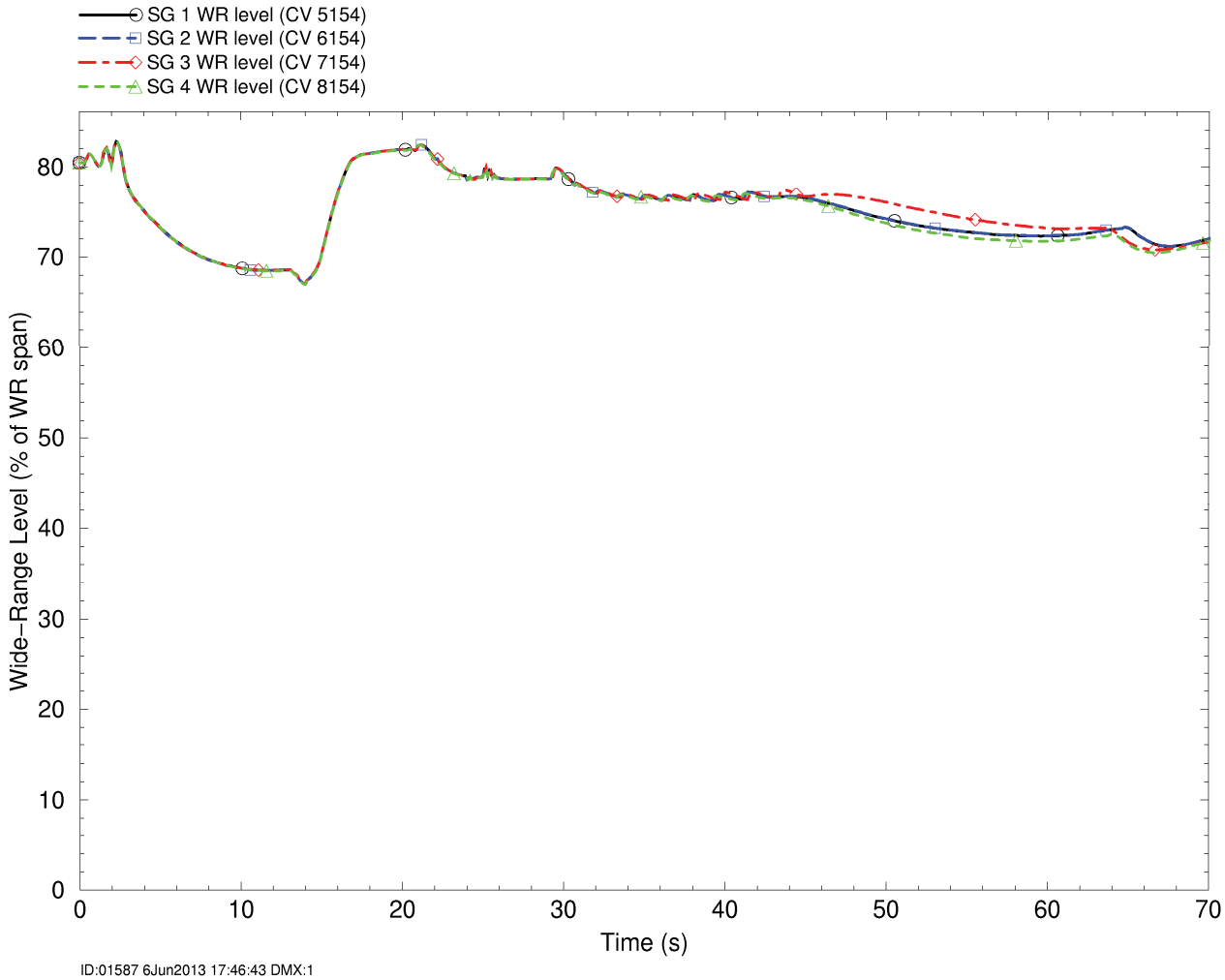


Figure 15.8-1–4: Main Steam Line Pressures

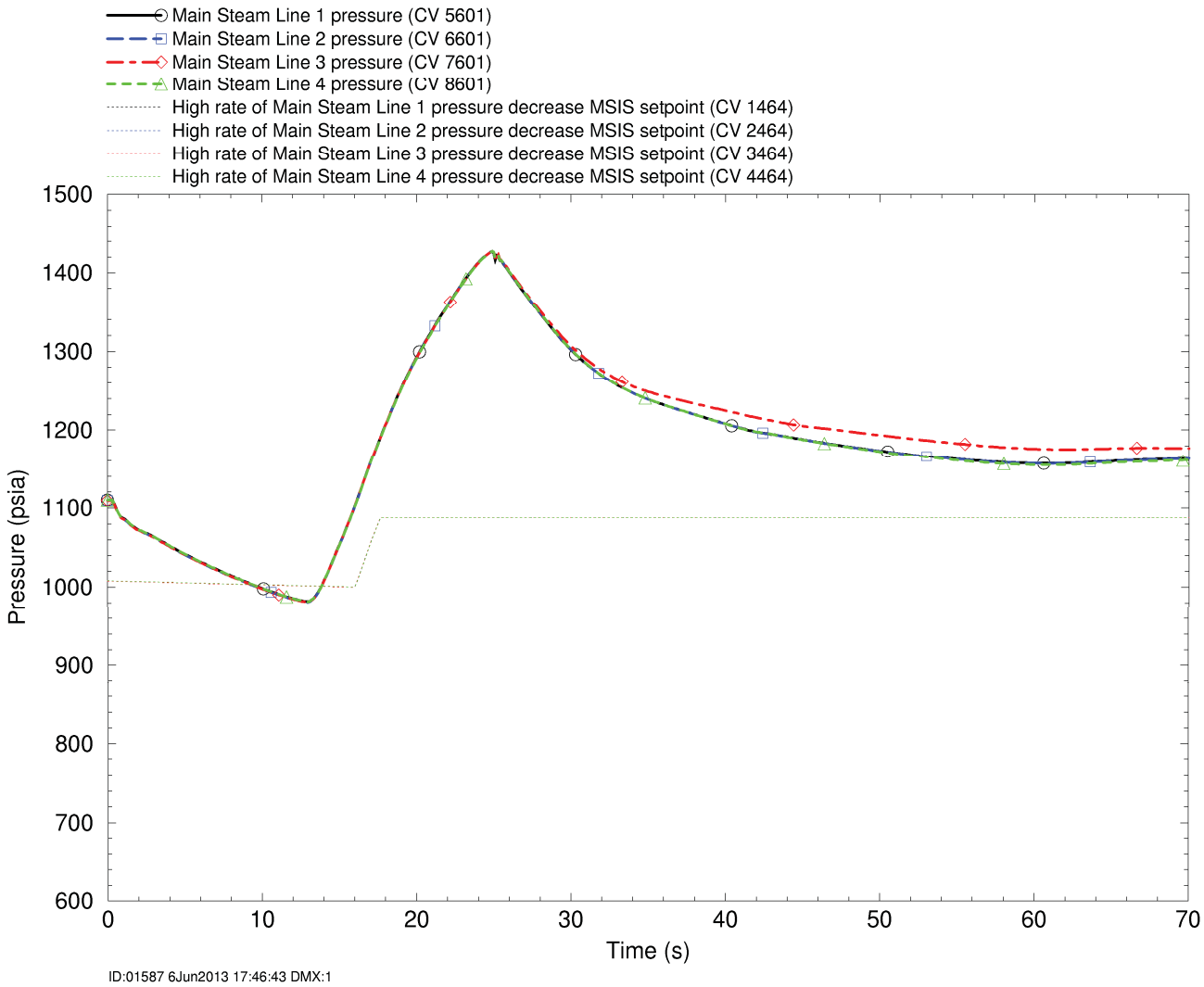


Figure 15.8-1–5: SG Dome Pressures

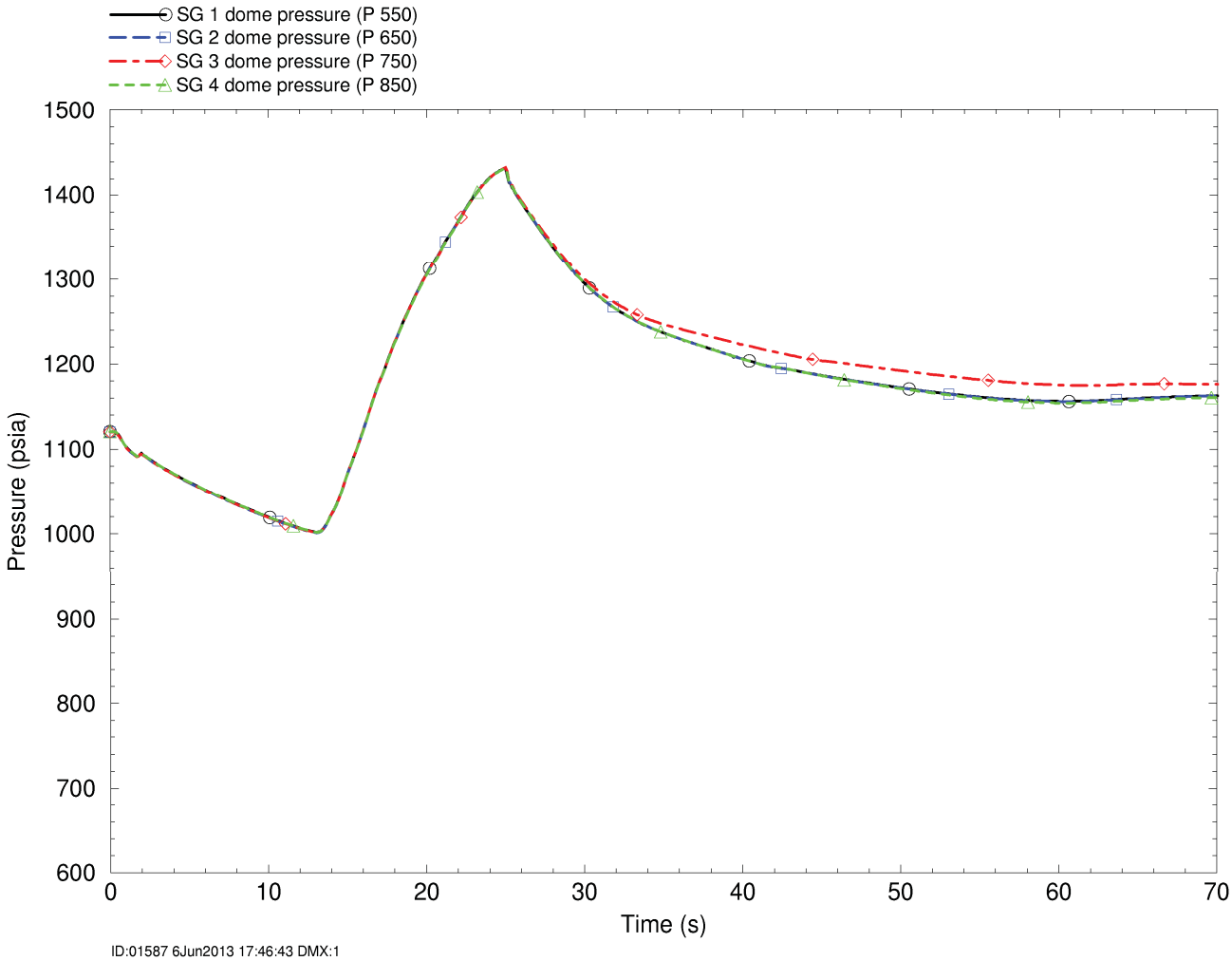


Figure 15.8-1–6: Primary Coolant Temperatures

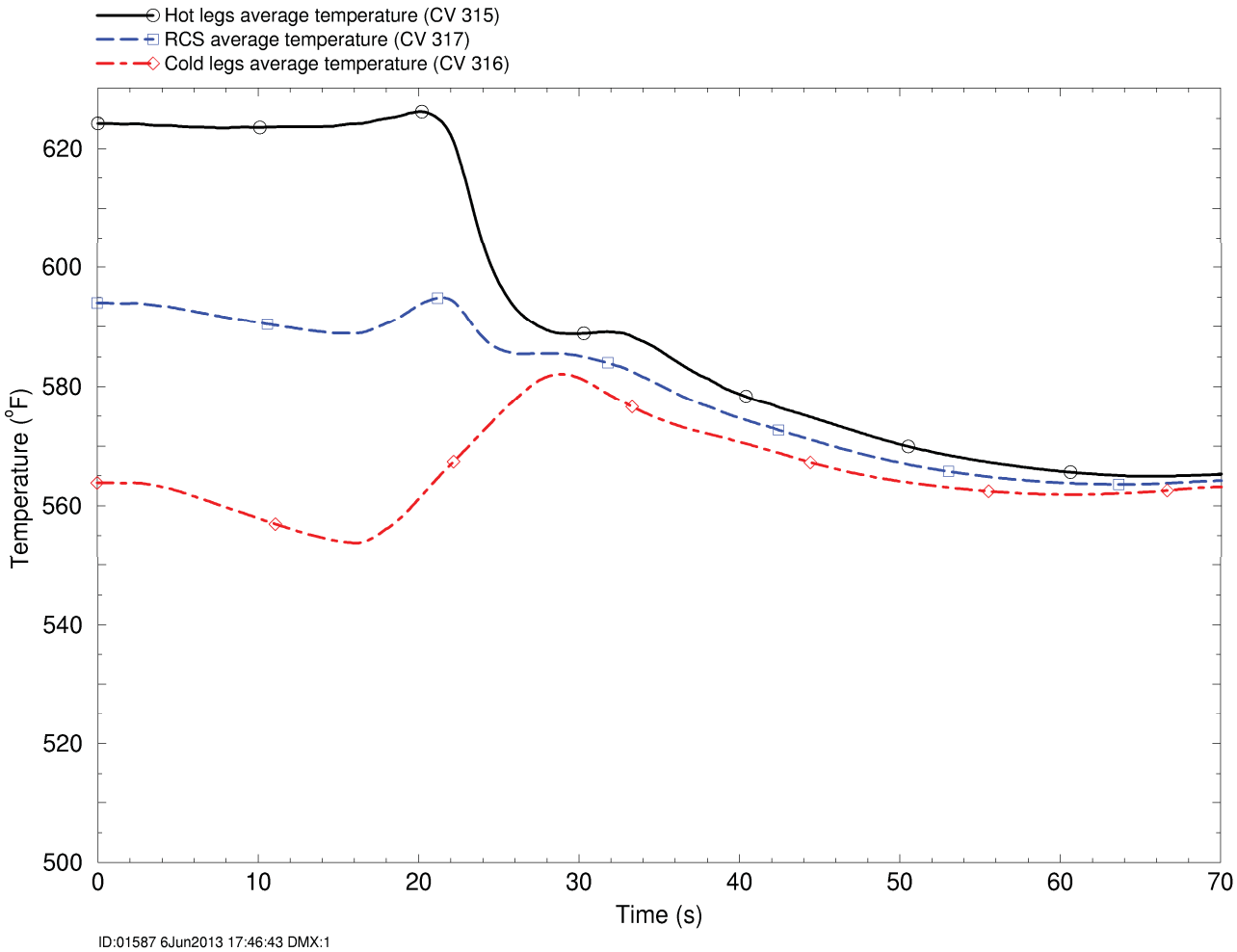


Figure 15.8-1–7: Reactivities

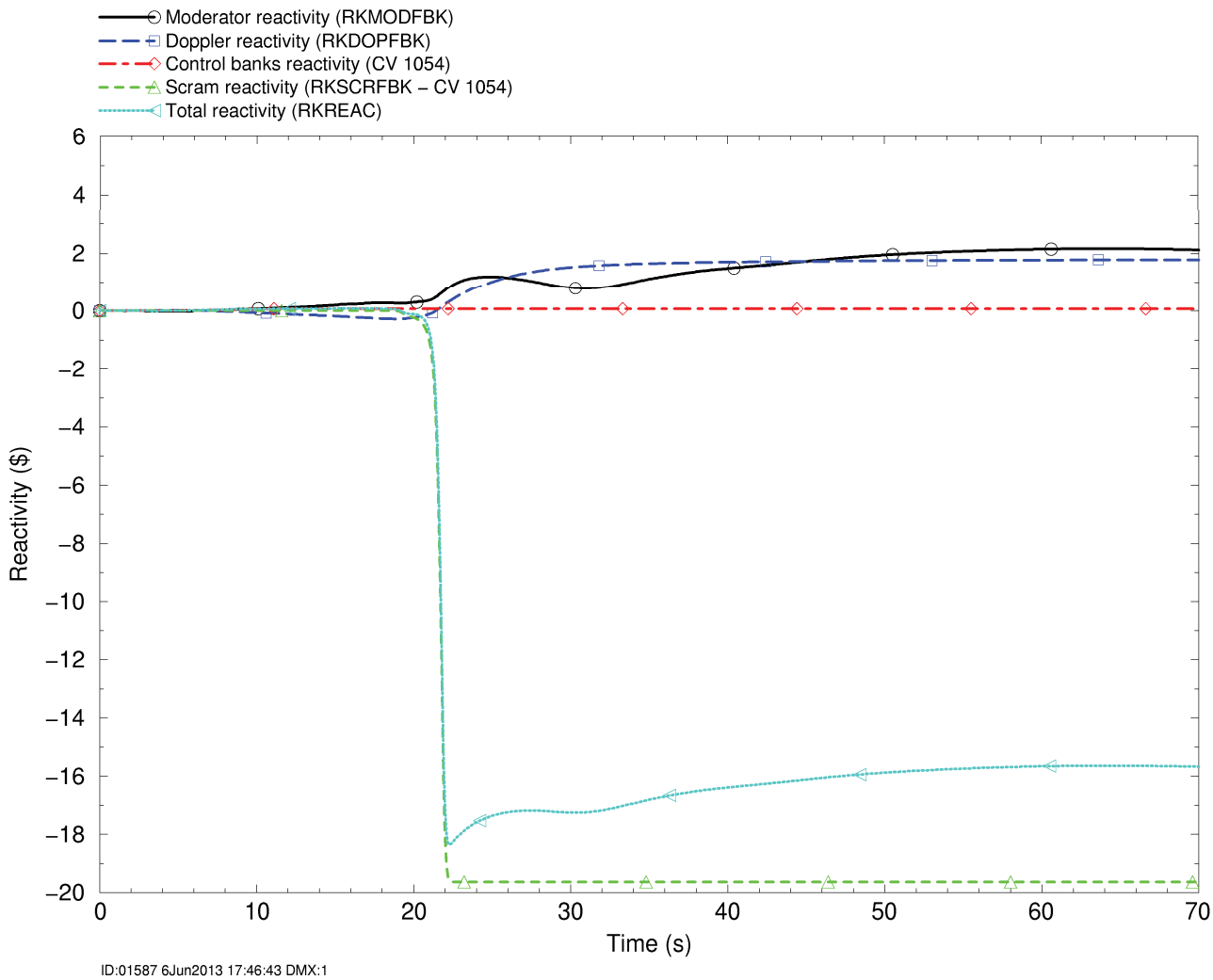


Figure 15.8-1–8: Reactor Power and Average Fuel Rod Heat Flux

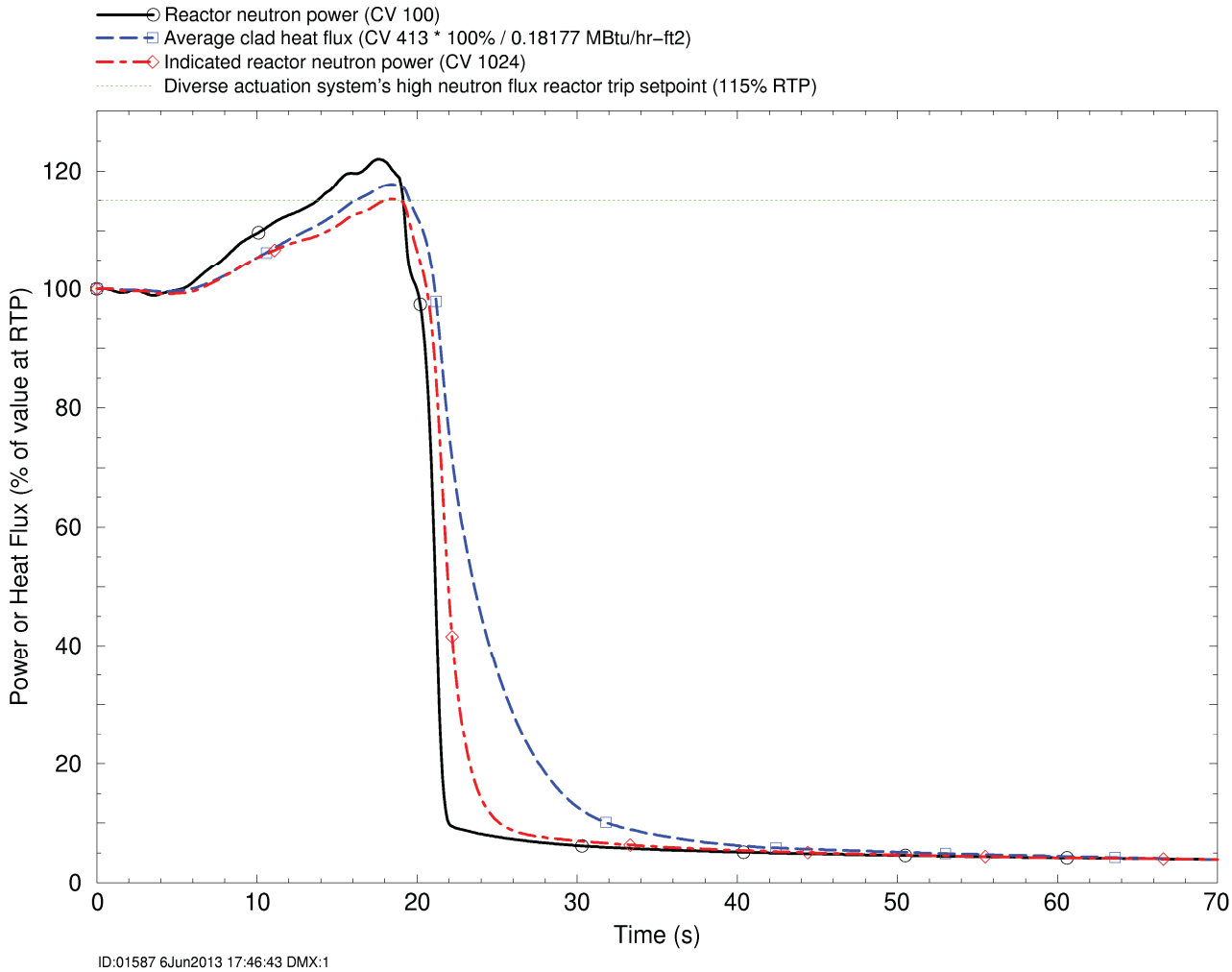
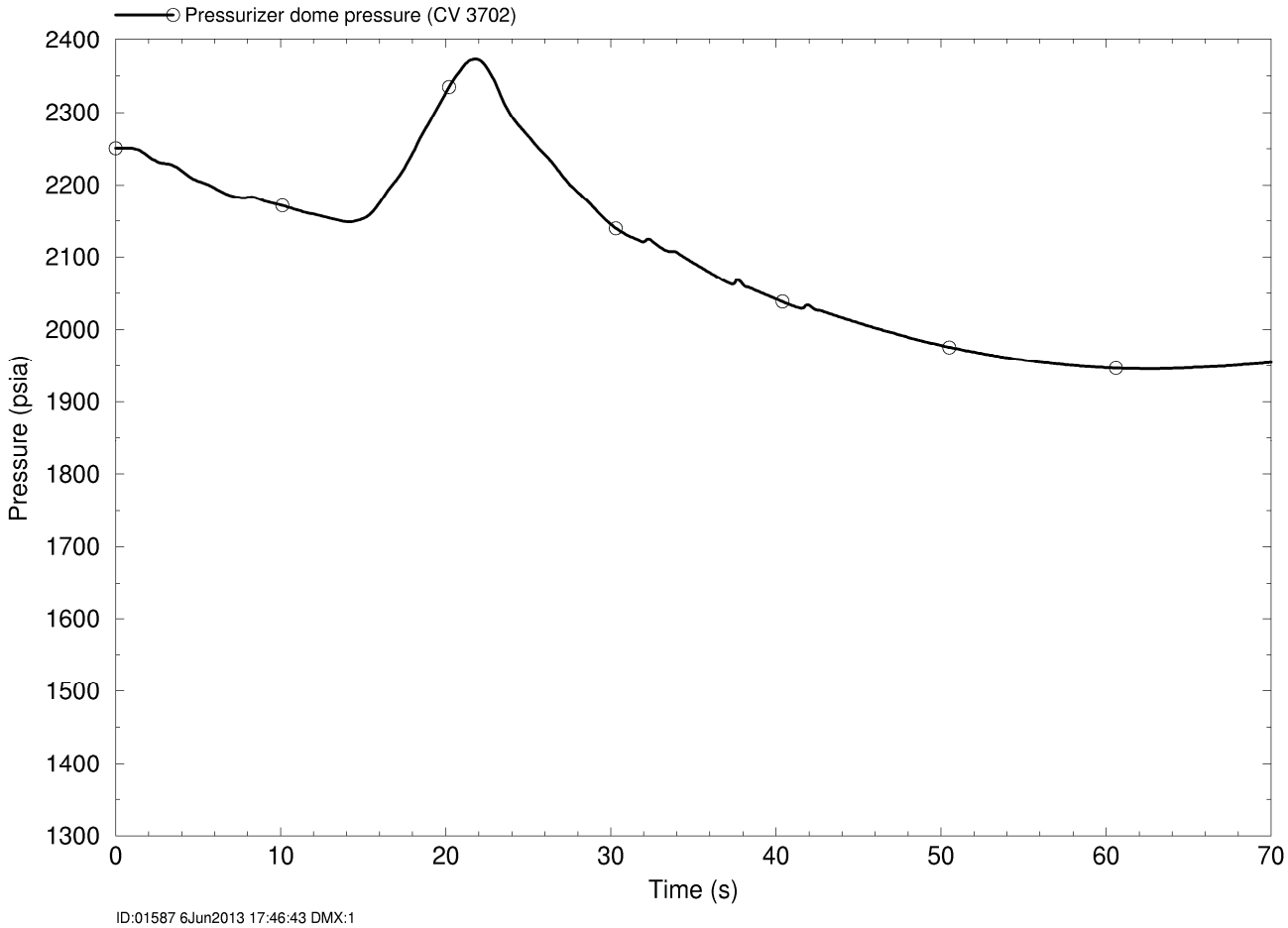


Figure 15.8-1-9: Pressurizer Dome Pressure



U.S. EPR Final Safety Analysis Report Markups

- Implement interlocks and preventive actions that cope with deviations from normal operation to prevent an event resulting in the actuation of a safety system.
- Mitigate events and bring the plant to a controlled state by the protection system that controls the RT actuation system and the engineered safety features.
- Cope with the complete failure of engineered safety features and protective systems by means of risk reduction functions.
- Preserve containment integrity in case of accidents resulting in mass and energy releases into the containment.

The following sections address U.S. EPR design features that function to mitigate or reduce the risk of ATWS events.

15.8.1.2 Protection System

The PS provides the primary means for protecting fission product barriers by tripping the reactor. A description of the PS is provided in Section 7.2.

An ATWS event occurs when the control rods fail to insert following the generation of an RT signal. The specific failure mechanism is not specified, but could be the result of mechanical blockage of the control rods or the result of electrical or mechanical failures within the PS. If the ATWS is the result of a mechanical blockage of the control rods, failures within the PS are not postulated. Details of mechanical blockage are provided in Section 15.8.1.6.

If the ATWS is the result of a failure within the PS, an independent and diverse RT signal within the diverse actuation system (DAS) is provided if necessary to mitigate the event and provide compliance with the ATWS criteria.~~an independent and diverse RT signal within the diverse actuation system (DAS) bypasses the PS and initiates an RT.~~ The RT is initiated by opening the breakers mounted at the output of the motor-generator sets that energize the control rod drive mechanisms. The diverse system also processes other critical signals and initiates essential actions to trip the turbine and start other safeguards systems as needed to address the ATWS event. Details of these signals are addressed in Section 15.8.1.3.

15.8.1.3 Diverse Actuation System

The DAS includes logic that fulfills the ATWS requirements of 10 CFR 50.62. The DAS logic is independent from sensor output to the final actuation device from the PS design features, and provides a diverse means to trip the reactor, trip the turbine, and initiate emergency feedwater (EFW) on conditions indicative of an ATWS. EFW is initiated only on conditions that would require EFW to maintain the primary heat sink, namely low SG Level. These diverse functions provided by the DAS provide reasonable assurance that ~~a pressure increase does not exceed the ASME Service Level~~

~~C limit of 3200 psig (Reference 1) or does not exceed containment safety parameters.~~

- A coolable geometry for the reactor core is maintained.
- Reactor coolant pressure boundary is maintained.
- Containment integrity is maintained.

The diverse trip functions and capabilities are incorporated within DAS and used for ATWS mitigation are described in Section 7.8.1.2.

15.8.1.4 Emergency Feedwater System

The U.S. EPR provides automatic actuation of the EFW system on conditions indicative of an ATWS (see Section 15.8.1.3). The U.S. EPR is designed so that flow from the EFW system is not required for the first 30 minutes following an ATWS.

15.8.1.5 Extra Borating System

The extra borating system (EBS) is not required for ATWS mitigation. However, the system is available via manual actuation should additional negative reactivity be desired to bring the reactor into a subcritical state. A description of the EBS is provided in Section 6.8.

15.8.1.6 Mechanical Blockage of Rod Cluster Control Assemblies

ATWS events resulting from mechanical blockage of control rods are not postulated for the U.S. EPR design. The probability of an ATWS resulting from mechanical blockage of the control rods is an insignificant contributor to the overall probability of an ATWS.

As noted in the "Conclusions" section of NUREG-1780 (Reference 2), during ATWS rulemaking the NRC staff set a goal that the probability of an ATWS should be no more than 1.0E-05 per reactor year. The probability of an ATWS is defined by NUREG-1780 as "the annual frequency of an ATWS leading to plant conditions that exceed certain design parameters that can result in core melt, containment failure, and the release of radioactivity and can be viewed as the expected CDF of an unmitigated ATWS." NUREG-1780 updated the original generic ATWS regulatory analysis using operating data since the ATWS rule was implemented, and the updated results indicated that the four reactor types achieved the ATWS rule risk goal: General Electric, Westinghouse, Babcock and Wilcox (B&W), and Combustion Engineering (CE). Specifically, Table 3, Summary of ATWS Rule Risk Expectations and Outcomes, of NUREG-1780 shows that the pressurized water reactor vendors (Westinghouse, B&W, and CE) each achieved a probability of an ATWS that is at least a factor of 20 better than the goal (i.e., ≤5.0E-07 per reactor year).