NuScale Power™

RAIO-0717-55059

July 24, 2017

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 28 (eRAI No. 8771) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 28 (eRAI No. 8771)," dated May 26, 2017

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) response to the referenced NRC Request for Additional Information (RAI).

The Enclosure to this letter contains NuScale's response to the following RAI Question from NRC eRAI No. 8771:

- 15-1

This letter and the enclosed response make no new regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions on this response, please contact Darrell Gardner at 980-349-4829 or at dgardner@nuscalepower.com.

Sincerely,

Zackary W. Rad
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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 8771

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 8771

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 8771
Date of RAI Issue: 05/26/2017

NRC Question No.: 15-1

15.0.06 Return to Power

According to General Design Criterion (GDC) 27, reactivity control systems shall be designed to reliably control reactivity changes to ensure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained. As described in the staff's response (ML16116A083) to the NuScale Gap Analysis Summary Report, Revision 1, Gap 11, GDC27, the staff has determined the applicant's design does not meet GDC 27 and, as such, the applicant has requested an exemption to GDC 27. The exemption to GDC 27 states, in part, that the return to power assuming a stuck rod is sufficiently unlikely and that specified acceptable fuel design limits (SAFDLs) for critical heat flux would not be exceeded even if it occurred. To demonstrate that SAFDLS are not exceeded, the applicant has analyzed a return to power in FSAR Section 15.0.06. An accurate determination of the minimum critical heat flux (MCHF) is dependent upon assessing reactor stability at these low power conditions and hence GDC 12 also applies.

Based on the detail provided in Section 15.0.06, the staff is unable to reach a reasonable assurance finding that the SAFDLs will not be exceeded. The staff is requesting that Final Safety Analysis Report (FSAR) Section 15.0.06, which assumes the use of the decay heat removal system (DHRS), be modified to include details provided for a licensing basis event, and the short term transient analyses in the various subsections of Chapter 15 should reference Section 15.0.06 for the potential long term acceptability of the events. Additional information should include, but not be limited to, a description of the evaluation model(s); the critical heat flux (CHF) correlation used and basis for the correlation (or a reference there to); the assumed radial and axial power distributions and the basis for those power distributions (or a reference there to); and a justification of how reactor stability is maintained under these conditions such that the MCHF can be accurately determined. In addition to Figure 15.0-8, "Power Response on a Return to Power," the staff is requesting the FSAR be updated to include plots of reactor coolant system average temperature, pressurizer pressure, core reactivity, hot channel nuclear enthalpy rise factor (F-delta-H), and MCHF ratio verses time.

NuScale Response:

As requested, the FSAR Section 15.0.6, Return to Power, has been updated to include a more detailed description of the return to power evaluation. A conclusion section was added that includes comparing results of the analysis to typical SRP acceptance criteria including: minimum critical heat flux ratio (MCHFR), maximum reactor coolant system (RCS) and steam generator (SG) pressure and ensuring the event does not escalate to a more serious plant condition.

The event was evaluated with and without AC and DC (EDSS) power available. The progression of the event is different depending on the availability of DC power. If DC power is available, the NPM will reach an equilibrium power level based on the DHRS heat removal capability. If DC power is not available, ECCS valves will open when DHRS reduces RCS pressure below the inadvertent actuation block release pressure. In the MCHFR case, the ECCS valves are assumed to open at the time of peak power. During the depressurization, RCS inventory flashes and the power drops sharply over the next 30 seconds to less than 2 MW. The MCHFR occurs shortly after ECCS actuates but never approaches the acceptance limit. Analysis of the long term ECCS conditions conclude the power levels, although low, are sufficient to generate the core voiding necessary to suppress the critical power level. The overcooling return to power analysis of MCHFR is conservatively evaluated during the initial ECCS actuation at peak power, which is more limiting than the lower power oscillatory conditions that occur later during ECCS mode cooling.

In both event sequences, RCS and SG pressures are not challenged and the event does not progress to a more serious event. Figures of merit were added to describe the event with DC power (peak power case) and without DC power (MCHFR case).

Section 15.0.4 describes the organization of the section 15 of the FSAR where the traditional short term plant transients are evaluated until a safe stabilized condition is reached. The two possible long term scenarios are then presented generically in sections 15.0.5 and 15.0.6, which are each evaluated to the design basis coping period of 72 hours. This section provides the generic cross reference from the safe stable condition, described as the end state of the short term transient description, to the two long term event scenarios. Since the long term scenarios are presented generically, specific references in individual events is not needed.

Impact on DCA:

The text in FSAR Section 15.0.6 was revised and Table 15.0-16 and Table 15.0-17 as well as Figure 15.0-10 through Figure 15.0-17 were added. These FSAR Sections have been revised as described in the response above and as shown in the markup provided in this response.

the core power very low. If decay heat exceeds 100 kW, the reactor will be maintained with k_{eff}<1 even with a worst rod stuck out because of negative density reactivity. Therefore, the heat produced after a return to power with the RVVs and RRVs is bounded by maximum decay heat. Consequently, the maximum decay heat curve, with the reactor at k_{eff}<1, is used to evaluate maintaining fuel integrity and ECCS performance. Maintaining fuel integrity is addressed in Sections 15.6.5 and 15.6.6 and ECCS performance is addressed in Section 6.3.

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15.0.6.1 Identification of Causes and Accident Description

Design basis events are analyzed with an assumed highest worth control rod stuck fully withdrawn in order to evaluate the immediate shutdown capability of the negative reactivity insertion due to a reactor trip with the control rods inserting into the core, consistent with Section 15.0.0.3. In the event of an extended DHRS cooldown, when the RCS is at low boron concentrations and the CVCS is unavailable to add boron, it may be possible for DHRS to cool the core to the point of reestablishing some level of critical neutron power if the most reactive control rod stuck out is assumed. This potential overcooling could cause a unique reactivity event similar to a steam line break for traditional multi-loop PWRs. Therefore, this event is specifically evaluated for specified acceptable fuel design limits (SAFDLs).

RAI 15-1

The purpose of this analysis is to evaluate the thermal hydraulic and core neutronic response of the NPM for an extended DHRS overcooling return to power. This analysis is intended to provide a generic bounding evaluation of the extended DHRS cooling that could result following any DBE, therefore AOO acceptance criteria and conservative analysis assumptions are applied. This event is analyzed specifically for the parameters that generate the most severe overcooling return to power core power overshoot so that it can be concluded that the postulated overcooling return to power following a DBE would not be more limiting than the event presented in this analysis. Additionally this event is analyzed for the postulated transition from DHRS cooling to ECCS cooling in order to demonstrate that transition to ECCS cooling can be safely accomplished.

RAI 15-1

15.0.6.2 Sequence of Events and Systems Operation

Consistent with the AOO acceptance criteria, the parameters of interest for the return to power event are reactor vessel pressure, secondary pressure and minimum critical heat flux ratio (MCHFR). To appropriately characterize the event, two cases are presented: return to power with and without transition from DHRS cooling to ECCS cooling. The sequence of events for an overcooling return to power event with DHRS cooling is provided in Table 15.0-16 and with the transition to ECCS cooling is provided in Table 15.0-17.

RAI 15-1

For the overcooling return to power event, it is assumed that a reactor trip occurs at end of cycle (EOC) with the most reactive control rod stuck out of the core. The subsequent DHRS cooldown is left unmitigated and boron addition does not occur. While there are simple operational means for mitigating the DHRS extended cooldown and thereby eliminating the need for boron addition, operator action is not credited for either mitigating the cooldown or adding boron, consistent with Section 15.0.0.6.4.

RAI 15-1

The overcooling return to power event assumes a reactor trip coincident with the loss of normal AC power as the initiating event. This analysis concerns the post-reactor trip return to power; therefore, the MPS is not specifically credited.

RAI 15-1

In the event that the highly reliable DC power (EDSS) is assumed to be unavailable concurrent with the initiating event, ECCS would be actuated while RCS pressure is above the IAB release pressure, and the ECCS valves would not initially open. During an extended DHRS cooling event, RCS pressure decreases due to reactor pressure vessel (RPV) heat loss and reactor coolant system (RCS) shrinkage causing an expansion of the pressurizer vapor space. Although unlikely, if the initial pressurizer pressure and level were sufficient, it is possible to postulate an IAB release concurrent with the overcooling return to power peak. This scenario generates the most challenging CHF conditions and is presented as the transition to ECCS cooling scenario.

RAI 15-1

15.0.6.3 Thermal Hydraulic and Critical Heat Flux Analyses

15.0.6.3.1 Evaluation Models

The transient evaluations are performed in separate parts. First, the peak power portion of the analysis, where EDSS is available, is analyzed using the non-LOCA NRELAP5 model discussed in Section 15.0.2. The purpose of the peak power analysis is to demonstrate the limited magnitude of the return to power, to characterize the event should DHRS cooling be sustained and to examine the various sensitivities that influence the moderator temperature-driven power response to inform the CHF modeling of the appropriate case to simulate.

RAI 15-1

The MCHFR portion of the analysis, where EDSS is unavailable, uses the LOCA NRELAP5 model. The CHF correlation applied in the LOCA evaluation model discussed in Reference 15.0-3 is evaluated against the 95/95 CHFR acceptance criterion of an AOO, as described in Section 15.6.6.

RAI 15-1

15.0.6.3.2 Input Parameters and Initial Conditions

As stated above, this event is analyzed specifically for the parameters that generate the most severe overcooling return to power core power event. A bounding DHRS cooldown following a DBE is evaluated with conservative assumptions to maximize the rate of reactivity insertion during a return to power to maximize the peak power. The following assumptions, for the case with EDSS available, ensure that the results have sufficient conservatism.

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- The reactor is at hot zero power (HZP) for the initial condition. The core power response is due to the moderator temperature-driven reactivity insertion that creates a bounding power overshoot that is several times larger than the eventual steady state power level. From a HZP initialization, the RCS shrinkage does not impede cooldown rate due to the much higher initial RCS density. Additionally, the HZP initial condition will tend to have lower decay heat levels and lower initial RCS temperature than higher power initializations resulting in a faster cooldown.

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- The most negative HZP moderator temperature coefficient (-15 pcm/°F) is used, as it will produce a bounding rate of increase in moderator reactivity worth during cooldown.

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- The least negative Doppler coefficient (-1.40 pcm/°F) is used because it results in the least strong negative reactivity feedback during the return to power, bounding the maximum peak power for the transient.

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- Uniform radial and axial moderator and Doppler reactivity feedback weighting is applied to ensure the power response is not suppressed due to the local heating effects.

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- The reactor is shut down with an assumed minimum required shutdown margin of two percent at 420 degrees Fahrenheit. A minimum shutdown margin allows for a return to power early in the cooldown transient while the RCS cools down at a higher rate.

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- The DHRS heat transfer is increased by 30 percent to ensure the consequences of the cooldown are maximized after DHRS actuation.

RAI 15-1

- A reactor pool temperature of 40 degrees Fahrenheit is used leading to a conservatively high cooldown rate, which adds the maximum positive reactivity.

RAI 15-1

No single failure is assumed. Failure of the main steam or feedwater isolation valves to close could result in a reduction of DHRS cooling, which would be non-conservative for the overcooling return to power event. Full ECCS actuation will be more limiting for CHF, therefore, an ECCS valve failure to open is not considered.

RAI 15-1

For the limiting MCHFR portion of the analysis, a loss of highly reliable DC power (EDSS) is assumed at the time of DHRS initiation, resulting in ECCS actuation. The timing of the ECCS valve opening is near the power peak as determined by a timing sensitivity analysis. The following conservatisms are applied to the MCHFR portion of the analysis:

RAI 15-1

- IAB release timing is sequenced with the timing of the power peak in order to evaluate the most limiting ECCS transition event sequence.

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- The maximum radial peaking ($F_{\Delta h}$) due to the stuck control rod is 6.5. The return to power is driven by the lack of necessary negative reactivity insertion due to the postulated most reactive control rod stuck in a fully withdrawn position. The critical power will be localized in this location generating higher than normal radial peaking.

RAI 15-1

- The bottom shaped axial power distribution is applied consistent with the LOCA EM.

RAI 15-1

- The ECCS valve characteristics are conservatively set to maximize the depressurization effect on MCHFR.

RAI 15-1

- Uniform radial and axial density reactivity feedback is used to conservatively bound the localized reactivity suppression due to the localized power generated around the stuck rod location.

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15.0.6.3.3 Results

The sequence of events for the DHRS overcooling event is provided in Table 15.0-16. Figure 15.0-8 provides the power response on the return to power. Figure 15.0-10 through Figure 15.0-13 show the transient behavior of key parameters.

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The overcooling return to power event begins with an initial negative reactivity insertion that is gradually removed as the transient progresses until a return to power occurs (Figure 15.0-8 and Figure 15.0-10). The biased initial conditions, with increased heat transfer, and low pool temperature results in a slightly larger return to power. A sensitivity analysis on background decay heat shows a minor sensitivity, where higher decay heat results in a slightly slower event progression and marginally decreased peak power. Initial negative reactivity insertion also has little impact on the calculated peak power. Therefore, it is concluded that protection of shutdown margin is insignificant for this event.

RAI 15-1

The return to power (Figure 15.0-8 and Figure 15.0-10) occurs more than two hours from the start of the transient, meaning the initiating event has little impact on the event progression and results. These cases are initiated from HZP conditions, which subsumes initiation events from hot full power (HFP) conditions due to the nature of the reactivity balance. The time of return to power would be greatly delayed with realistic decay heat levels and an event initiating from HFP conditions.

RAI 15-1

With the initial negative reactivity insertion, the average RCS temperature (Figure 15.0-11), average fuel temperature (Figure 15.0-12), and RPV pressure (Figure 15.0-13) decrease until the return to power occurs. At the time of the return to power, the temperature and pressure increase slightly and then either stabilize at a low value or continue to decline.

RAI 15-1

For the limiting MCHFR event, which is a loss of highly reliable DC power (EDSS) resulting in a transition to ECCS cooling, the CHFR is presented in Figure 15.0-14. This analysis considers the short term transition period to demonstrate that even at elevated local power distributions, a transition to ECCS cooling is not a safety concern. Further analysis demonstrates that once ECCS equilibrium conditions are established, the density reactivity feedback is sufficient that even very low heat levels will suppress the critical power response. Therefore, it is concluded that the limiting condition for MCHFR is at the time of the ECCS transition when the core power levels are much higher than the equilibrium ECCS cooling power levels. Reactor power, RCS flow rate, and hot channel heat flux are provided in Figure 15.0-15, Figure 15.0-16, and Figure 15.0-17, respectively.

RAI 15-1

15.0.6.3.4 Conclusions

The AOO acceptance criteria outlined in Table 15.0-2 are used as the basis for the overcooling return to power event. The acceptance criteria, followed by how the NuScale design meets them, are listed below:

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1) Potential core damage is evaluated on the basis that it is acceptable if the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit. Minimum critical heat flux ratio is used instead of minimum DNBR, as described in Section 4.4.2.

   Fuel integrity is not challenged by an overcooling return to power event. The MCHFR is 2.25 and is shown in Figure 15.0-14. The MCHFR for evaluated cases occurs in the Hench-Levy correlation flow range, therefore the 95/95 design limit is 1.122. The CHF analysis confirms that the DHRS overcooling return to power event can safely transition to ECCS cooling without challenging MCHFR limits.

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2) RCS pressure should be maintained below 110 percent of the design value.

   Due to the nature of the overcooling return to power event, primary pressure is not challenged and is non-limiting for this event.

RAI 15-1

3) The main steam pressure should be maintained below 110 percent of the design value.

Due to the nature of the overcooling return to power event, main steam pressure is not challenged and is non-limiting for this event.

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4) The event should not generate a more serious plant condition without other faults occurring independently.

The overcooling return to power analysis demonstrates that DBEs, where a most reactive control rod is assumed stuck out upon reactor trip, can be safely cooled by DHRS, or DHRS transitioning to ECCS cooling, without challenging MCHFR limits. Additionally, return to power scenarios with extended ECCS core cooling are limited by the density reactivity feedback as generated by the boiling in the core such that these scenarios are well bounded by the DHRS transition event due to the relative power levels in the core.

RAI 15-1

The evaluation of an overcooling return to power event demonstrates that design limits are not exceeded and the overcooling return to power event is non-limiting with respect to DBEs.

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~~For those events that rely on heat removal by the DHRS, heat produced after a return to power with a stuck control rod will be limited by negative moderator temperature feedback. The time to a return to power and the power level attained are based on conservative assumptions for the purpose of demonstrating fuel protection and are not indications of plant shutdown capability.~~

~~The RCS cooldown ensues after the initial transient response to a DBE for events that rely on the DHRS for heat removal. A bounding DHRS cooldown is evaluated with conservative assumptions to demonstrate the fuel integrity is not challenged during a return to power. Conservative assumptions to maximize the rate of reactivity insertion during a return to power include:~~

- ~~The reactor is shut down with an assumed minimum required shutdown margin of 2 percent at 420 degrees Fahrenheit. A minimum shutdown margin allows for a return to power early in the cooldown transient while the RCS cools down at a higher rate. In contrast, with a nominal value for shutdown margin with the worst control rod stuck out, a return to power does not occur for at least 24 hours.~~
- ~~A reactor pool temperature of 40 degrees Fahrenheit is used which leads to a conservatively high cooldown rate, which adds the maximum positive reactivity.~~
- ~~The most negative moderator temperature coefficient is used, as it will produce a bounding rate of increase in moderator reactivity worth during cooldown.~~
- ~~The least negative Doppler coefficient is used because it results in the least strong negative reactivity feedback during the return to power, bounding the maximum peak power for the transient.~~
- ~~The DHRS heat transfer is increased by 30 percent to ensure the consequences of the cooldown are maximized after DHRS actuation.~~

~~Figure 15.0-8 shows the power response on a return to power. Minimum critical heat flux ratio is evaluated and the analysis confirms that design limits are not exceeded and the DHRS cooldown evaluation is non-bounding with respect to other events.~~

15.0.7 References

15.0-1 NuScale Power, LLC, Topical Report, "Subchannel Analysis Methodology," TR-0915-17564, Rev. 0.

15.0-2 NuScale Power, LLC, Topical Report, "NuScale Power Critical Heat Flux Correlation NSP2," TR-0116-21012, Rev. 0.

15.0-3 NuScale Power, LLC, Topical Report, "LOCA Evaluation Model," TR-0516-49422, Rev. 0.

15.0-4 NuScale Power, LLC, Topical Report, "Accident Source Term Methodology," TR-0915-17565, Rev. 1.

15.0-5 NuScale Power, LLC, Topical Report, "Non-LOCA Transient Analysis Methodology," TR-0516-49416, Rev. 0.

15.0-6 Nuclear Energy Institute, Position Paper, "Small Modular Reactor Source Terms," December 27, 2012, Washington, DC.

15.0-7 NuScale Power, LLC, Technical Report, "Long-Term Cooling Methodology," TR-0916-51299, Rev. 0.

15.0-8 K.F. Eckerman et al., "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," Federal Guidance Report 11, EPA-520/1-88-020, Environmental Protection Agency, 1988.

15.0-9 K.F. Eckerman and J.C. Ryman, "External Exposure to Radionuclides in Air, Water, and Soil," Federal Guidance Report 12, EPA-402-R-93-081, Environmental Protection Agency, 1993.

15.0-10 NuScale Power, LLC, Topical Report, "Evaluation Methodology for Stability Analysis of NuScale Power Module," TR-0516-49417, Rev.0.

15.0-11 NuScale Power, LLC, Topical Report, "NuScale Rod Ejection Accident Methodology," TR-0716-50350, Rev. 0.

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Table 15.0-16: Sequence of Events for Overcooling Return to Power Event EDSS Available (Peak Power Case)

Event	Time [s]*
Start of Transient	0.0
DHRS Actuation	2.1
Time of recriticality	6750
Maximum Return to Power (14 MW)	7850
Equilibrium power reached (3 MW)	12000

Note:

*Time is rounded.

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Table 15.0-17: Sequence of Events for Overcooling Return to Power Event EDSS unavailable (MCHFR Case)

Event	Time [s]*
Time of Power Peak	7726
Time of IAB Release (ECCS Actuation)	7729
Time of MCFHR	7733

Note:

*Time is rounded.

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Figure 15.0-10: Return to Power - Net Reactivity (Peak Power Case)

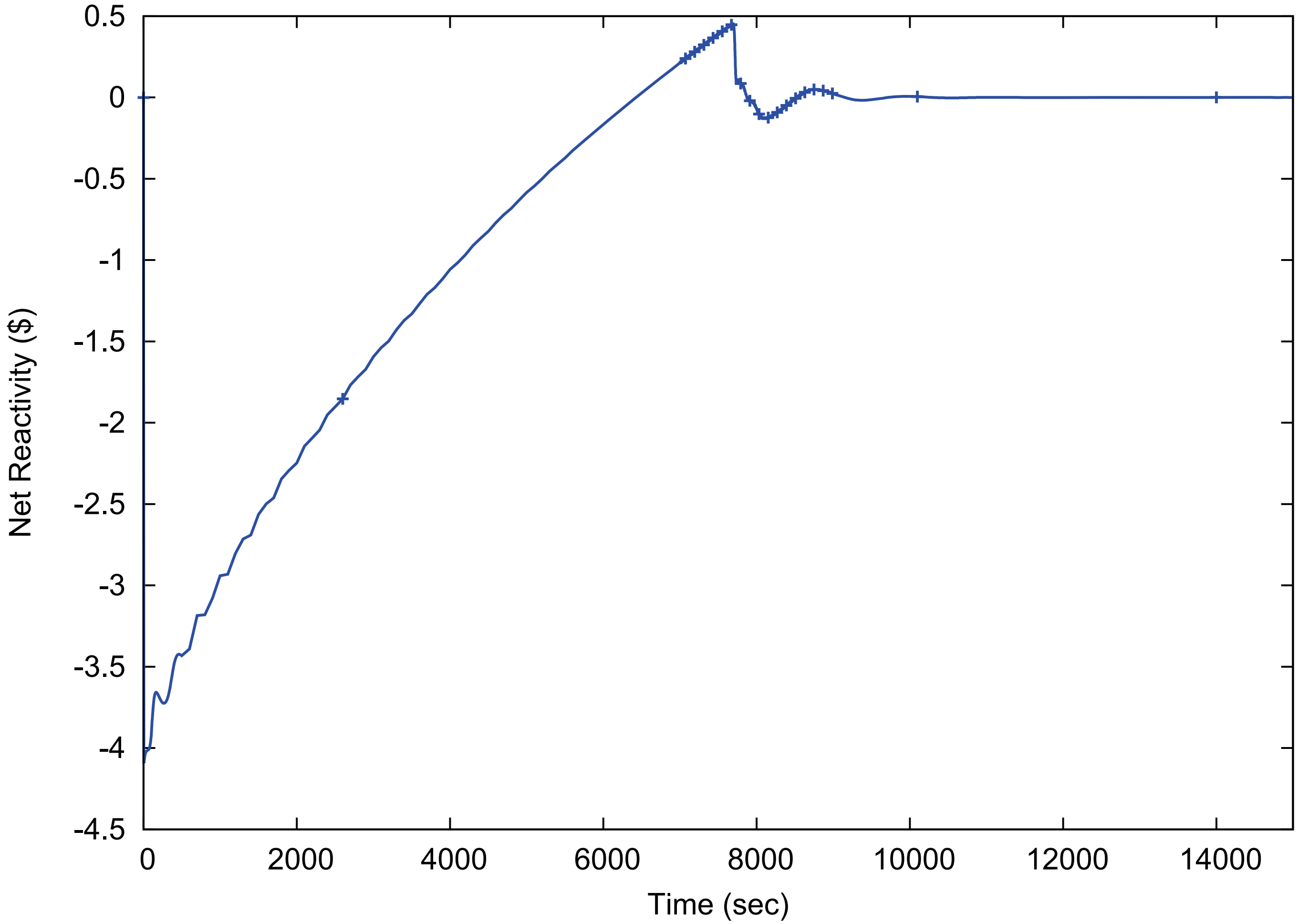

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Figure 15.0-11: Return to Power – Average Reactor Coolant System Temperature (Peak Power Case)

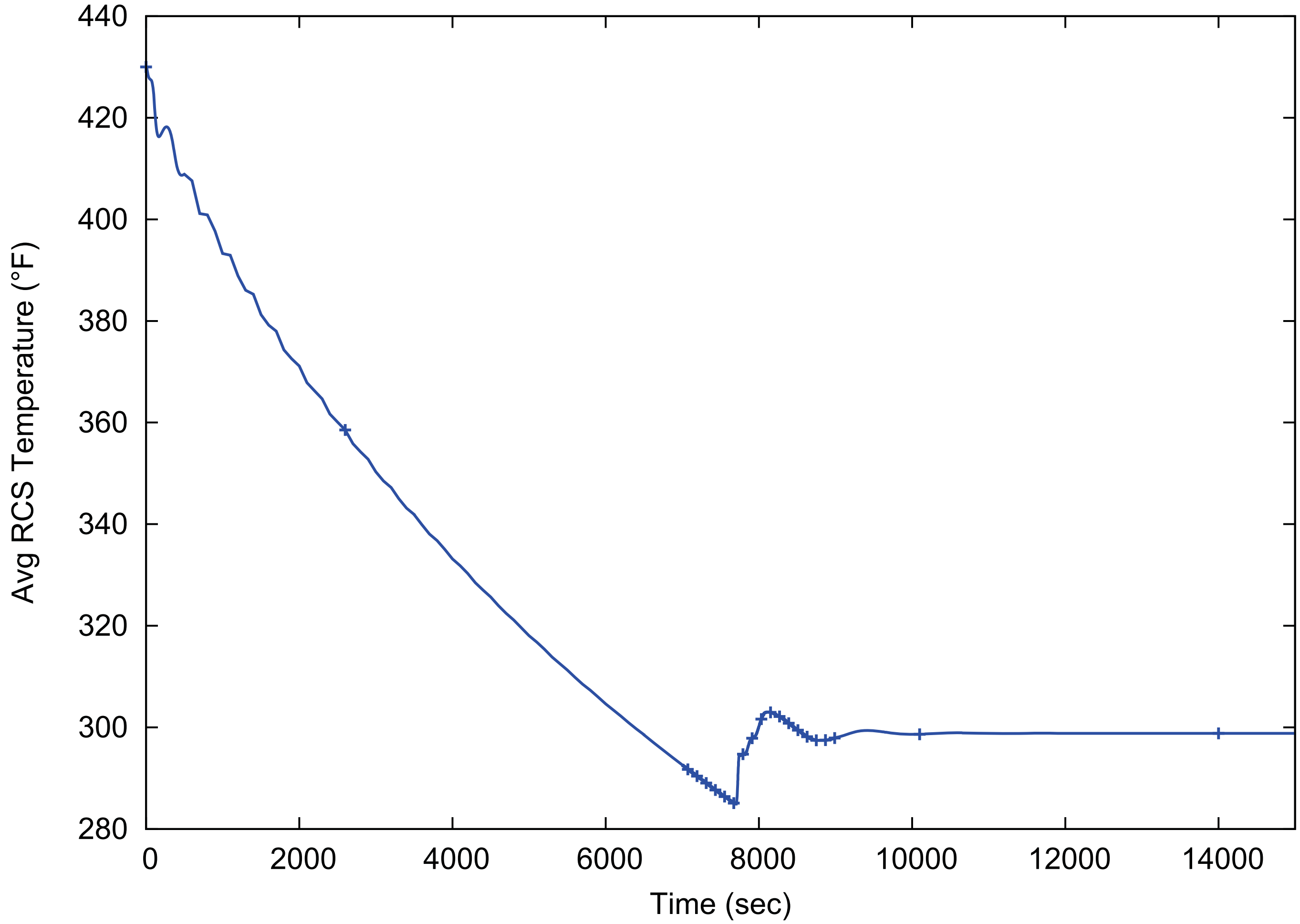

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Figure 15.0-12: Return to Power – Volume Average Fuel Temperature (Peak Power Case)

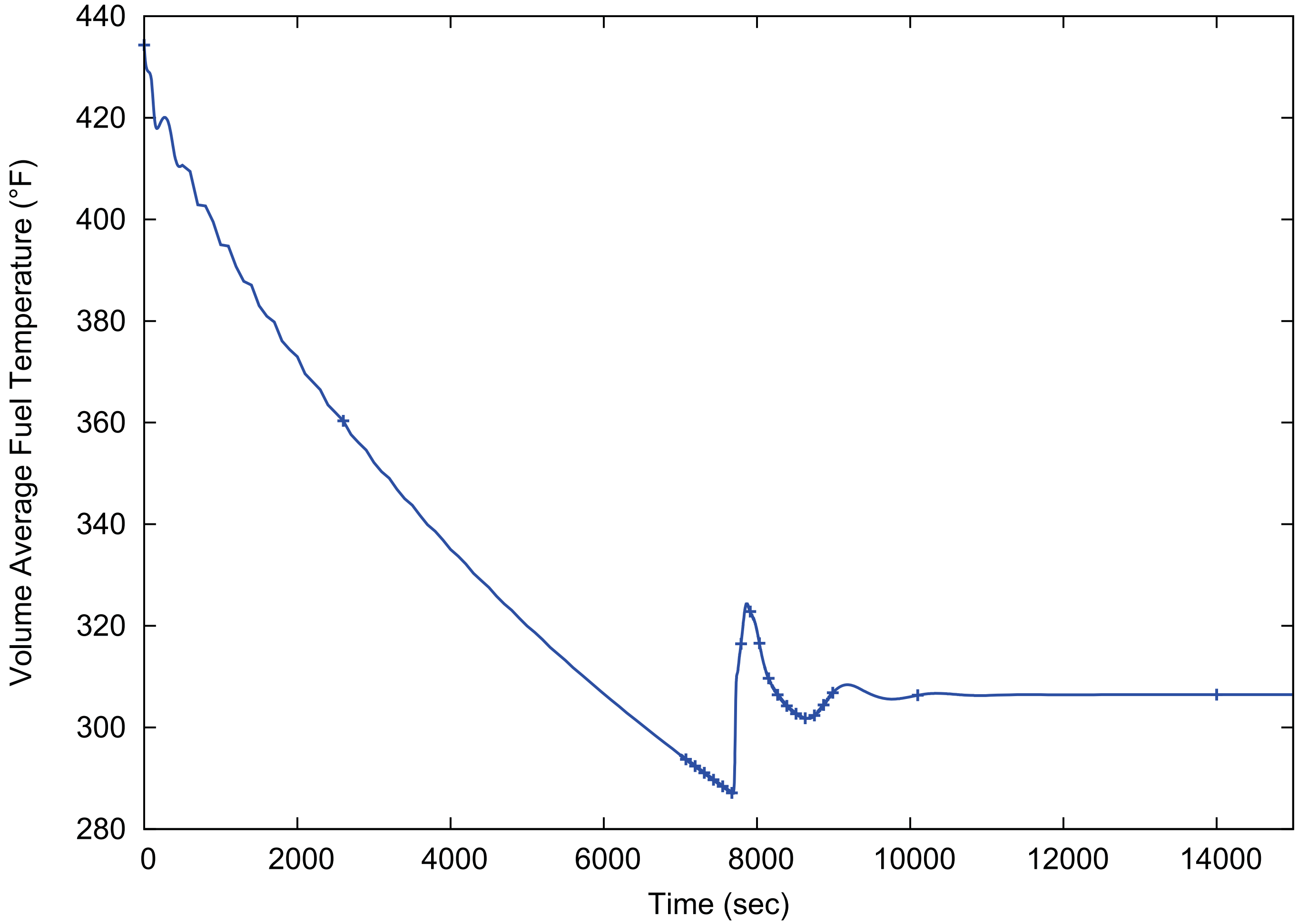

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Figure 15.0-13: Return to Power – Reactor Pressure Vessel Lower Plenum Pressure (Peak Power Case)

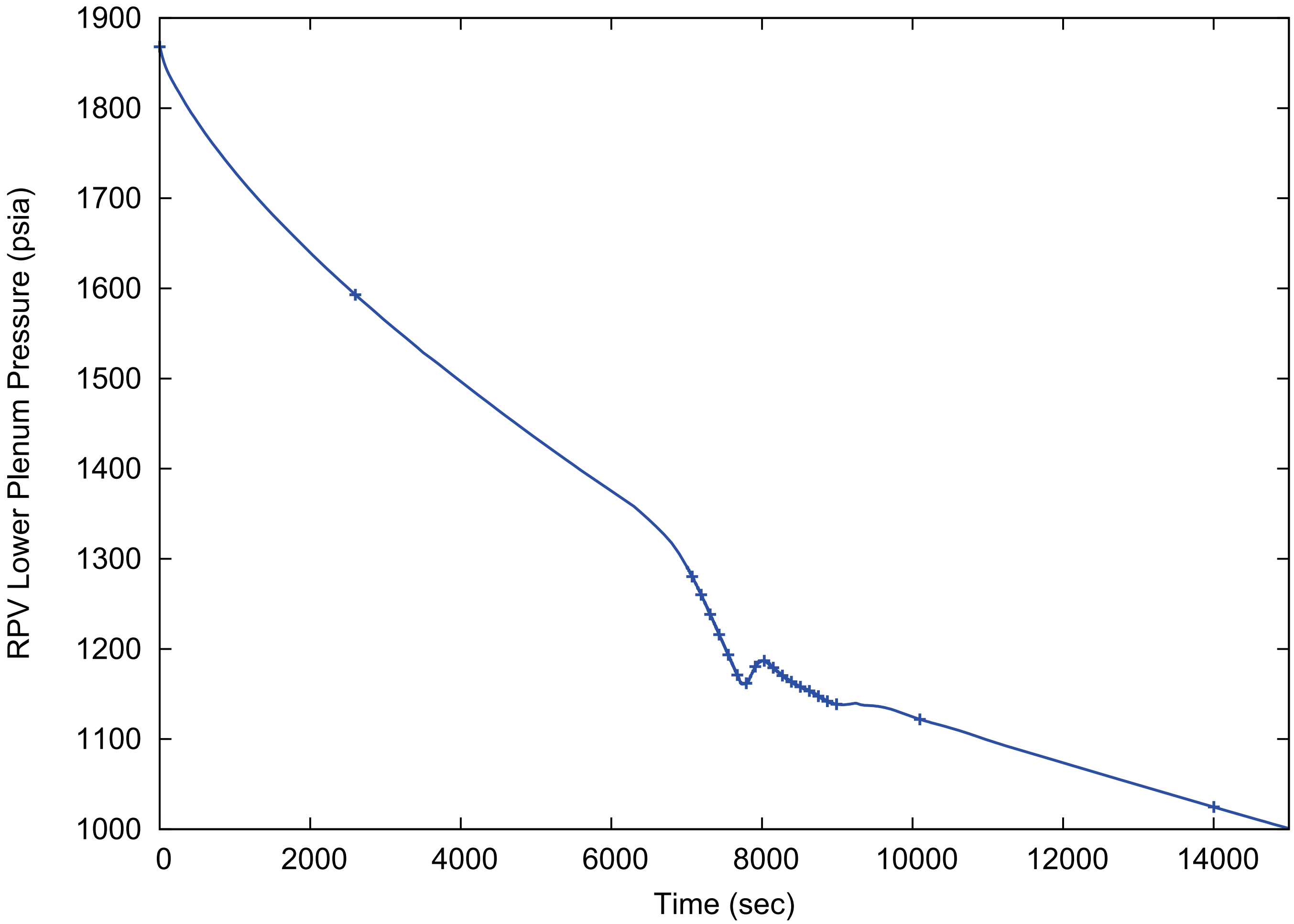

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Figure 15.0-14: **Return to Power ECCS Transition Case – Critical Heat Flux Ratio (MCHFR Case)**

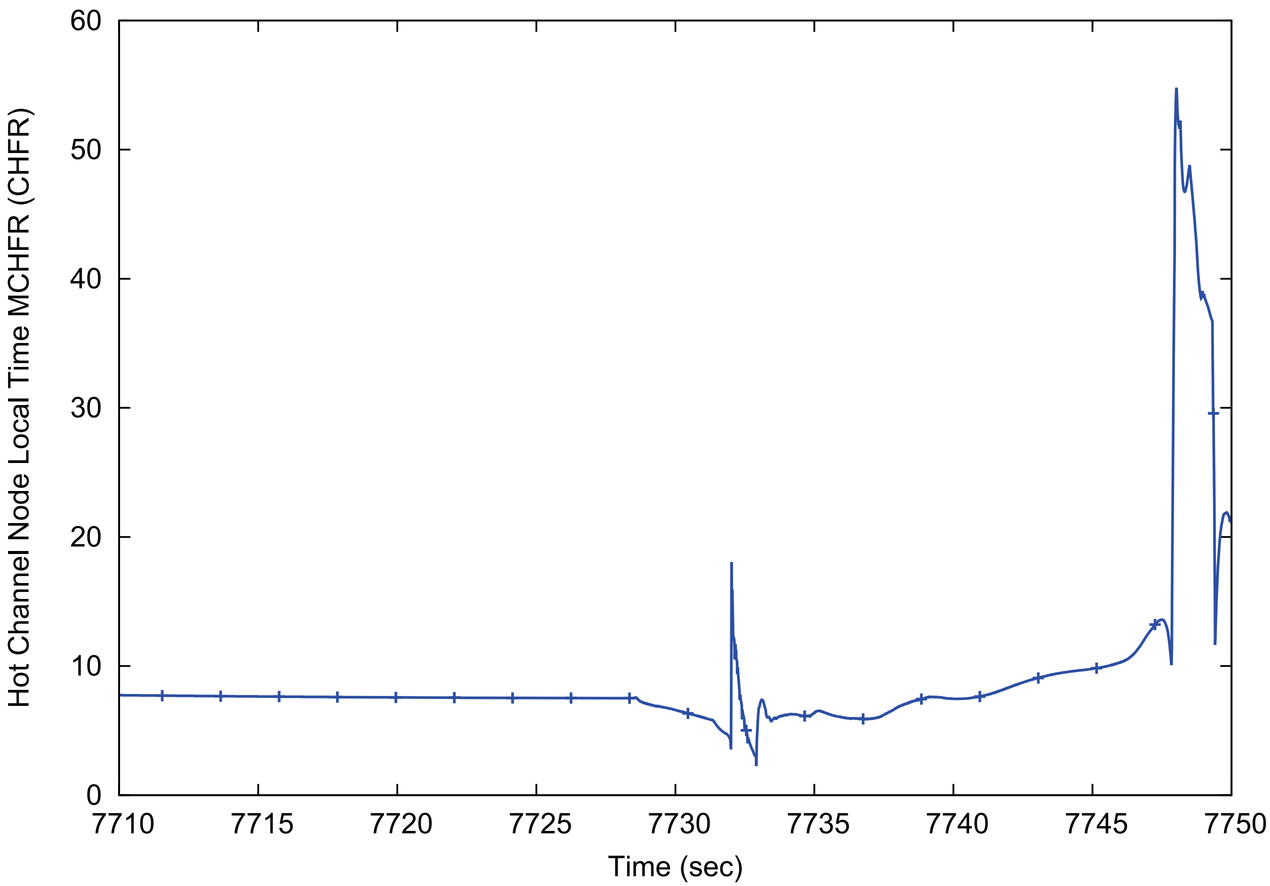

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Figure 15.0-15: Return to Power ECCS Transition Case – Reactor Power (MCHFR Case)

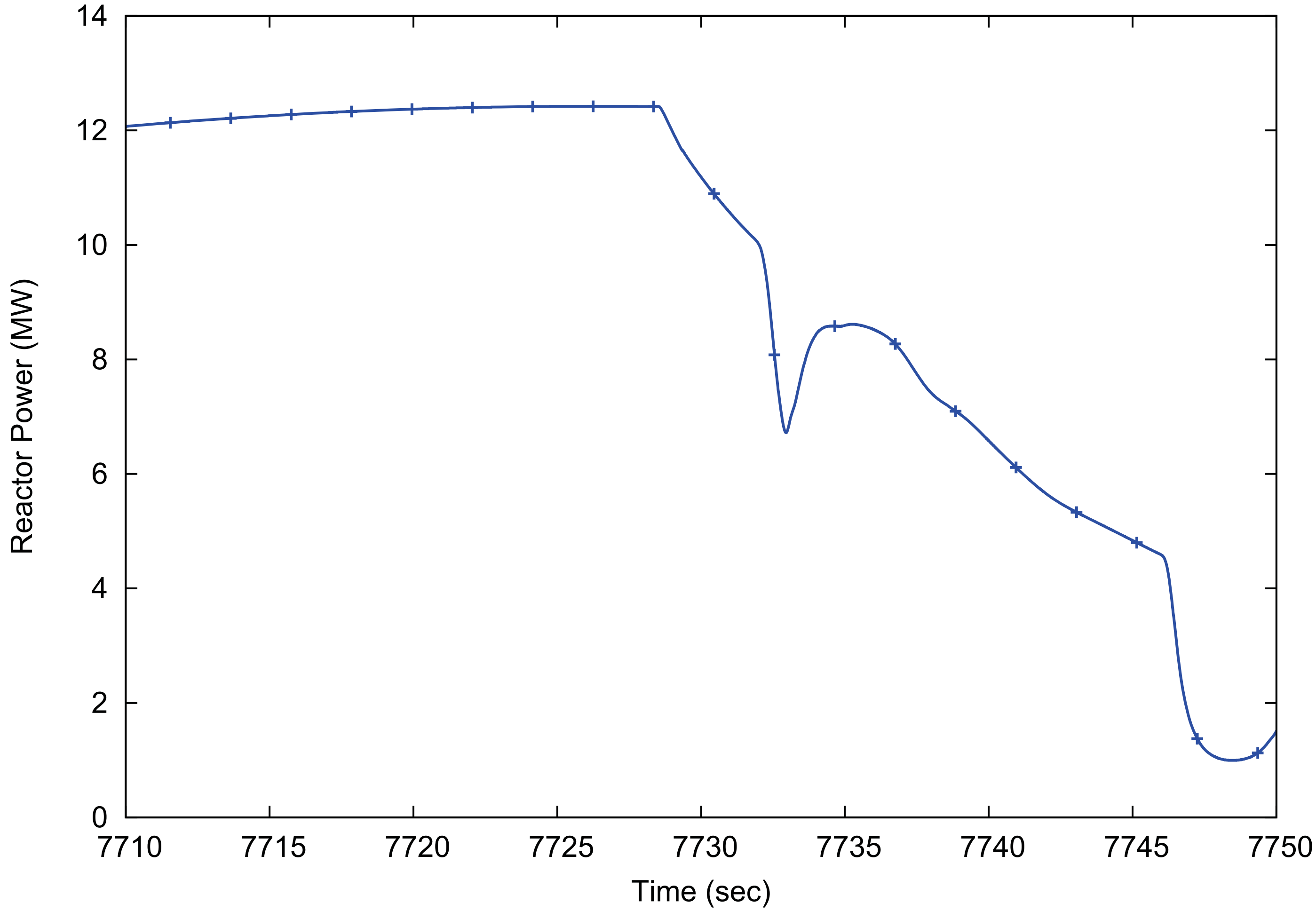

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Figure 15.0-16: Return to Power ECCS Transition Case – RCS Flowrate (MCHFR Case)

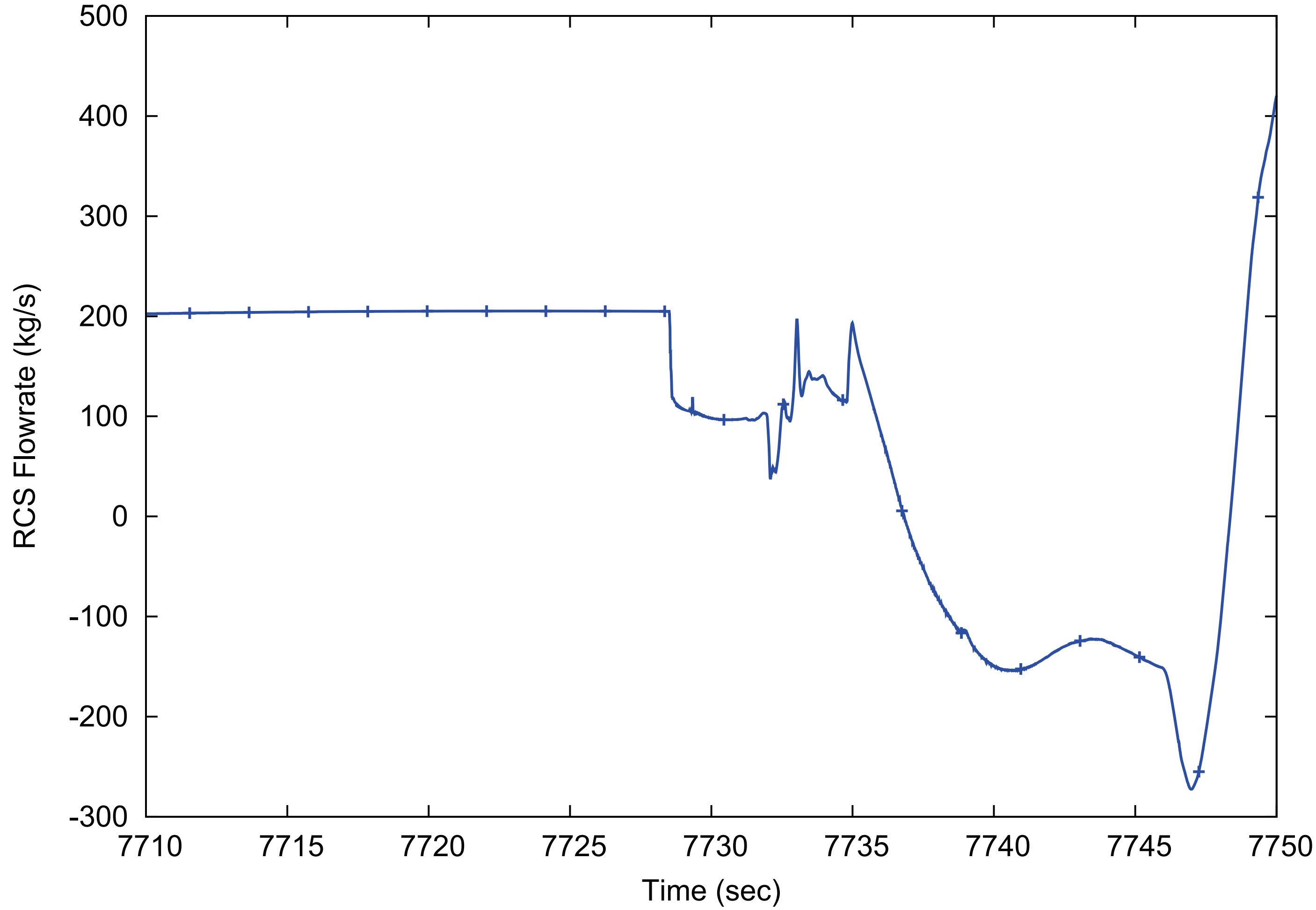

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Figure 15.0-17: Return to Power ECCS Transition Case – Hot Channel Heat Flux (MCHFR Case)