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POLCA-T: System Analysis Code with Three-Dimensional Core Model, Appendices C and D

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POLCA-T: System Analysis Code with Three-Dimensional Core Model, Appendices C and D

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APPENDIX C
POLCA-T: APPLICATION FOR AOO TRANSIENT ANALYSIS

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ABSTRACT

POLCA-T is a best-estimate computer code for three-dimensional analysis of boiling water reactor transients (Reference 1). Appendix A of Reference 1 includes the approved POLCA-T application to control rod drop accident (CRDA) and Appendix B of Reference 1 includes the approved POLCA-T application to stability analysis.

This appendix provides the qualification basis of the POLCA-T code for application to the transient analysis and evaluation of anticipated operational occurrences (AOO) in advanced boiling water reactors (ABWR) and in boiling water reactors (BWR/2-6). Specific models required in analysis of transient events are presented and validated.

The application methodology for AOO analysis is presented in Reference 2 and Reference 3. It is demonstrated that POLCA-T can be used to perform licensing analysis of AOO events. An example of a nominal event analysis and corresponding uncertainty analysis is provided.

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ACRONYMS

Acronyms	Definition
ABWR	Advanced Boiling Water Reactor
ANS	American Nuclear Society
ANSI	American National Standards Institute
AOO	Anticipated Operational Occurrences
APRM	Average Power Range Monitor
ATWS	Anticipated Transients Without Scram
BOC	Beginning of (operating) Cycle
BWR	Boiling Water Reactor
CCFL	Counter Current Flow Limitation
CHF	Critical Heat Flux
CPR	Critical Power Ratio
CRDA	Control Rod Drop Accident
DO	Dryout (phenomenon)
ECCS	Emergency Core Cooling System
EM	Evaluation Model
EOC	End of (operating) Cycle
EPU	Extended Power Uprate
FD	Feedwater flow Decrease
FI	Feedwater flow Increase
FoM	Figures-of-Merit
FWCS	Feedwater Control System
HPDP	High Pressure heater Drain Pump
HPDT	High Pressure heater Drain Tank
ICPR	Initial Critical Power Ratio
ID	Identifier
INEL	Idaho National Laboratory
LHGR	Linear Heat Generation Rate
LOCA	Loss of Coolant Accident
LPRM	Local Power Range Monitor
LRNBP	Load Rejection with No By-Pass event
LTP	Licensed Thermal Power
MCPR	Minimum Critical Power Ratio
MELLLA+	Maximum Extended Load Line Limit Analysis Plus
MSIV	Main Steam line Isolation Valve
NRC	(US) Nuclear Regulatory Commission
[	]a,c
PB2	Peach Bottom Unit 2 Nuclear Power Plant
PD	Pressure Decrease
PI	Pressure Increase
PIRT	Phenomena Identification and Ranking Table
RCPB	Reactor Coolant Pressure Boundary
RAI	Request for Additional Information

RD	Reactor coolant flow Decrease
RFCS	Recirculation Flow Control System
RI	Reactor coolant flow Increase
RIP	Recirculation Internal Pump
RMS	Root Mean Square
RPV	Reactor Pressure Vessel
SAFDL	Specified Acceptable Fuel Design Limits
SB&PCS	Steam Bypass and Pressure Control System
SCRRI	Selected Control Rod Run-In
SLMCPR	Safety Limit Minimum Critical Power Ratio
SRP	(US NRC) Standard Review Plan, NUREG-0800
SRV	Safety Relief Valve
TBV	Turbine Bypass Valve
TCV	Turbine Control Valve
TD	Reactor coolant Temperature Decrease
TI	Reactor coolant Temperature Increase
TIP	Traversing In-core Probe
TSV	Turbine Stop Valve
TT	Turbine Trip
TT1	(PB2 EOC 2) Turbine Test 1
TT2	(PB2 EOC 2) Turbine Test 2
TT3	(PB2 EOC 2) Turbine Test 3

C.1 INTRODUCTION

C.1.1 Background

POLCA-T is a best-estimate computer code developed by Westinghouse for three-dimensional analysis of Boiling Water Reactor (BWR) transients. The POLCA-T code including physical models, correlations and numerical solution techniques are described in the main licensing topical report (Reference 1).

Appendix A of Reference 1 includes the POLCA-T application to control rod drop accident.

Appendix B of Reference 1 includes the POLCA-T application to stability analysis.

Appendix D of Reference 1 includes the POLCA-T application to Anticipated Transients Without Scram (ATWS).

This Appendix provides the assessment of the POLCA-T Evaluation Model (EM) for transient analysis with regards to the acceptance criteria associated with Anticipated Operational Occurrences (AOO).

References 2 and 3 provide the Westinghouse full-scope BWR transient analysis methodology, including the AOO analysis.

The requirements for the evaluation model utilized in this report, including the Phenomenon Identification and Ranking Table (PIRT), have been established in Reference 2. The analysis requirements and scope are summarized in Section C.2 of this report.

The POLCA-T AOO evaluation model is described in detail in Section C.3.

The evaluation of model uncertainties and the assessment base used for AOO analyses with POLCA-T is described in Section C.4.

The POLCA-T validation against plant transient tests is shown in Section C.5.

The AOO analysis application uncertainties are presented in Section C.6.

A demonstration of transient analysis including an uncertainty evaluation is included in Section C.7.

C.1.2 Summary and Conclusions

This report demonstrates the capability of POLCA-T to the analysis of transient events. Based on information contained in this report, it is concluded that POLCA-T:

1. Is an acceptable tool for analyzing ABWRs and BWR/2-6 with respect to the AOO acceptance criteria.

2. Provides analysis results that are acceptable for referencing in licensing applications for ABWR and BWR/2-6 power plants.

C.2 REQUIREMENTS AND SCOPE

C.2.1 Analysis Scope

The purpose of this topical report is to demonstrate the acceptable use of POLCA-T for calculating the reactor system response with respect to:

- Peak Reactor Coolant Pressure Boundary (RCPB) Pressure
- Specified Acceptance Fuel Design Limits (SAFDLs) such as Minimum Critical Power Ratio (MCPR) and Linear Heat Generation Rate (LHGR).

C.2.2 Nuclear Power Plant Specification

Plant designs considered in the methodology are Advanced Boiling Water Reactors (ABWR) and BWR/2 through BWR/6 plants.

C.2.3 Transient Scenario Specification

The scope of this report is to license transient analysis of Anticipated Operational Occurrences using POLCA-T.

The Westinghouse analysis methodology in References 2 and 3 describes the evaluation process of event acceptance criteria for events classified as AOOs.

C.2.4 Anticipated Operational Occurrences

AOOs are those conditions of normal operation that are expected to occur one or more times during the life of the nuclear power plant. NUREG-0800, the Standard Review Plan (SRP), uses the term AOO to refer to events as incidents of moderate frequency (events that are expected to occur several times during the plant's lifetime) and infrequent events (events that may occur during the lifetime of the plant).

The type of event is defined by its phenomenological effect on the plant. The AOO events are grouped according to type into four categories, consistent with Reference 2 and Chapter 15 of the SRP:

- Pressure Increase/Decrease (PI/PD)
- Reactor coolant flow Increase/Decrease (RI/RD)
- Feedwater flow Increase/Decrease (FI/FD)
- Reactor coolant temperature Increase/Decrease (TI/TD).

The phenomenological description of each event category is given in Section 4 of Reference 2.

C.2.5 Acceptance Criteria

The acceptance criteria for AOOs have been defined in Section 3.1 of Reference 2:

1. Radioactive effluents
2. Specified Acceptable Fuel Design Limits
3. Peak Reactor Coolant Pressure Boundary Pressure
4. Suppression Pool Temperature

C.2.6 Figures-of-Merit

The Figures-of-Merit (FoM) for AOO analysis were established in Section 3.2 of the Westinghouse analysis methodology in Reference 2, and are shown in Table C.2-1. Discussion of how these FoM meet the specified acceptance criteria is also presented in Reference 2.

Table C.2-1 Figures-of-Merit for AOO

AOO	MCPR (for clad overheating) LHGR (for clad strain) Peak RCPB pressure

C.2.7 Phenomena Identification and Ranking

A PIRT for AOO and ATWS analysis was established in Section 5 of Reference 2. The high ranked phenomena (for any of the transient types in C.2.4) listed below were identified for AOO evaluation. Individual phenomena are distinguished by their name and a unique phenomena identifier (ID) shown in parenthesis. Phenomena names and (IDs) originate from the methodology report (Reference 2).

a,c

[ ]

a,c

These phenomena are addressed in Section C.4, or in the fast transient analysis methodology (Reference 2).

C.3 EVALUATION MODEL

The POLCA-T code is described in Reference 1. Additional functionality required for AOO analysis is presented in this section. Also, validation or demonstration of the models not reviewed as part of Reference 1 is presented.

The additional code models to those contained in Reference 1 are:

- Evaluation of transient Critical Power Ratio (CPR)
- Interface to plant control system simulation tool SAFIR
- Level measurement system model
- Advanced control rod hydraulic insertion model

C.3.1 Transient Critical Power Ratio

CPR correlations are implemented by POLCA-T during the evaluation of transient dryout in licensing analyses. The correlations used are based on dryout test data and are reviewed and accepted by the NRC prior to their use. CPR data for Westinghouse BWR fuel are obtained in the [ ]^{a,c} loop which is a full-scale test facility. An example of a NRC-approved CPR-correlation can be seen in Reference 5.

This section presents the implementation and validation of POLCA-T transient CPR evaluation. The CPR evaluation method is demonstrated for [ ]^{a,c} fuel in the demonstration analysis in Section C.7. Also, the uncertainty in the transient CPR evaluation is demonstrated in Section C.7.2.

C.3.1.1 Implementation of Transient CPR evaluation

Transient CPR evaluation involves calculation of the [ ]^{a,c}. These code parameters are then used in the NRC approved CPR correlations to evaluate the transient CPR.

CPR can be evaluated [ ]^{a,c}.

The transient CPR is used to evaluate the event MCPR. The transient MCPR evaluation methodology is presented in References 2 and 3.

C.3.1.2 Transient CPR Measurements

This section presents the validation of POLCA-T transient CPR evaluation using the NRC approved CPR correlation (approved in References 4 and 5) against [ ]^{a,c} transient dryout measurements for [ ]^{a,c}.

The [ ]^{a,c} loop, test bundles, testing procedure, measurements and the test results are comprehensively described in Reference 4.

The POLCA-T model used for the simulation of the transient tests includes the [

]a,c

[ ]a,c transient dryout measurements presented in Reference 4 are included in this validation. The tests consist of [

]a,c. Validation against these tests is described in Section C.3.1.3.

[ ]a,c dryout measurements with [ ]a,c are included in the POLCA-T transient CPR validation basis. Measurements were performed for [

]a,c

described in Reference 4. Validation against these tests is described in Section C.3.1.4.

C.3.1.3 Tests with [ ]a,c Transients

The test description and measured data including the initial and boundary conditions input to test point simulation, the measured times to dryout, and the [

]a,c for each test are presented in Table 7.4 through Table 7.6 in Reference 4.

[

]a,c

The [ ]a,c transient test validation results are presented in Figure C.3-1 through Figure C.3-3 for the [

]a,c respectively. The POLCA-T comparison against these measurements shows that [ ]a,c of the data set is predicted conservatively over a wide range of transient conditions.

POLCA-T predicts the onset to (timing of) dryout with high accuracy. The average error in the predicted time of dryout is [ ]a,c seconds with a standard deviation of [ ]a,c seconds, as shown in Figure C.3-4. This provides a further demonstration of the capability of POLCA-T to predict transient CPR.

C.3.1.4 Tests with [ ]a,c

The tests with [ ]a,c were performed with the same [ ]a,c loop test setup and test bundle as described in Reference 4. The purpose of these measurements was to establish [

]a,c and measure the dryout occurrence. At a [

]a,c. Since the [ ]a,c loop is constructed for testing of dryout and is very stable at the nominal BWR operating conditions, the test pressure was [

]a,c

The tests in Table C.3-1 with [ ]a,c were chosen for POLCA-T validation. [ ]a,c

[
]a,c

Table C.3-1 [ ]a,c Dryout Measurements a,c

The measured test bundle transient flow rate, power, inlet temperature and outlet pressure are presented in Figure C.3-5 through Figure C.3-7.

The calculated CPR compared to the [ ]a,c are presented in Figure C.3-8 through Figure C.3-10. The onset of dryout is recorded at [ ]a,c. The onset of dryout is predicted when [ ]a,c. The onset of rewet is recorded at [ ]a,c. The onset of rewet is predicted when [ ]a,c.

POLCA-T predicts the onset of dryout consistent with the measurements or conservatively early. The prediction of cladding rewet is [ ]a,c. This is [ ]a,c in POLCA-T.

The results show that POLCA-T predicts the dryout conservatively during these transient scenarios. The test results with [ ]a,c.

[
]a,c

C.3.1.5 Summary of Transient CPR Validation

POLCA-T provides a conservative transient dryout prediction for [ ]a,c using the CPR-correlation approved in Reference 5. The POLCA-T comparison against [ ]a,c loop measurements reported in Reference 4 show that [ ]a,c of the data is predicted conservatively over a wide range of transient conditions.

POLCA-T can predict [ ]a,c with high accuracy. This provides a further demonstration of the capability of the POLCA-T to predict transient CPR. [ ]a,c

The validation results demonstrate the code capability to calculate fuel channel dryout and rewet [ ]a,c

a,c

Figure C.3-1 [ ]a,c Transient Dryout Measurements and POLCA-T Results

a,c

[ ]

Figure C.3-2 [ ]^{a,c} Transient Dryout Measurements and POLCA-T Results

a,c

[ ]

Figure C.3-3 [ ]^{a,c} Transient Dryout Measurements and POLCA-T Results

a,c

[ ]

Figure C.3-4 Measured and POLCA-T Calculated Onset to Dryout

a,c

Figure C.3-5 [ ]a,c Dryout Measurement for [ ]a,c Test

a,c

[ ]

Figure C.3-6 [ ]a,c Dryout Measurement for [ ]a,c Test

a,c

[ ]

Figure C.3-7 [ ]a,c Dryout Measurement for [.]a,c Test

a,c

[ ]

Figure C.3-8 POLCA-T Predicted CPR for [ ]^{a,c} Dryout Test

a,c

[ ]

Figure C.3-9 POLCA-T Predicted CPR for [ ]^{a,c} Dryout Test

a,c

[ ]

Figure C.3-10 POLCA-T Predicted CPR for [ ]^{a,c} Dryout Test

C.3.2 Interface to Plant Control System Simulation Tool SAFIR

The application of SAFIR with POLCA-T for applications in AOO analysis is described and demonstrated in this Section.

SAFIR is used to provide transient boundary conditions to POLCA-T, not to perform actual transient calculations. POLCA-T communicates with SAFIR through an interface where the signals are specified and translated into the corresponding POLCA-T input.

In transient calculations, POLCA-T [ ]^{a,c} and calls on SAFIR to perform time integration for each modeled component. Each component updates its state in time until all components are updated and have [ ]^{a,c}. SAFIR returns the evaluated values for signals to POLCA-T.

The use of SAFIR with POLCA-T is demonstrated against the Chubu Electric Power Co. Inc. Hamaoka 5 ABWR start-up tests in Reference 7. Eight Hamaoka 5 start-up tests of interest for AOO analysis were simulated with POLCA-T and are shown in detail in Section C.5.3. The plant control systems were simulated with SAFIR.

The tests consisted of control system set point perturbations and actual flow and pressurization transient events. POLCA-T and SAFIR models provide an accurate prediction of plant behavior during the transient tests reported in Reference 7.

Based on the results from these demonstrations, it is concluded that the SAFIR tool can be used to model control systems response to be used as input to POLCA-T.

C.3.3 Level Measurement System Model

The POLCA-T code has the capability to model [
]^{a,c}. The [ ]^{a,c} is measured in the reactor.

POLCA-T can model the [
]^{a,c}. The [ ]^{a,c} is used for studies and validations where accurate simulations are not required.
The [ ]^{a,c} is used when the dynamic response of the level measurement system is important to the transient scenario.

The POLCA-T level measurement model makes it possible to [
]^{a,c}, see Figure C.3-11.

a,c

Figure C.3-11 POLCA-T Water Level Measurement System

Similar to physical measurement process, the measured water level is determined from [

]^{a,c}

[a,c]

[

]a,c

An equivalent model to [
]a,c in POLCA-T can be made using SAFIR. [

]a,c

A demonstration of the use of the SAFIR level measurement model, [
]a,c, is a part of the transient validation against Hamaoka 5 start-up tests in Section C.5.3. Of these eight demonstration cases, the FWCS set point perturbation test in Section C.5.3.4, generator load rejection test in Section C.5.3.7 and all Main Steam line Isolation Valve (MSIV) closure test in Section C.5.3.8 have the most significant change in RPV level. Since the level measurement system is a part of the FWCS control, the FWCS set point perturbation directly verifies the level measurement system adequacy.

The POLCA-T calculated RPV water level is in good agreement with the measured water level in all validation cases presented in Section C.5.3.

C.3.4 Advanced Control Rod Insertion Model

Two different hydraulic control rod insertion (scram) simulation models exist in POLCA-T, a simple model and an advanced model.

The simple model uses [ ]a,c according to the [
]a,c evaluation methodology in References 1.

The Westinghouse advanced control rod hydraulic insertion model described in Reference 8 is also implemented in POLCA-T for calculation of the control rod insertion times by [
]a,c.

The model includes [
]a,c. Given input of these components' physical properties, the model simulates a scram by calculating how [

]a,c.

The POLCA-T input to the advanced scram model is [ ]a,c. The scram insertion time is accurately determined from [
.]a,c

C.4 MODEL UNCERTAINTIES

In this Section individual uncertainties and biases of high ranked phenomenon and component are determined from a combination of comparisons of code predictions against separate and integral effects tests, component performance tests, plant data tests and code benchmarks.

The model uncertainty evaluation includes both the random (standard deviation) and systematic uncertainties (bias). [

]a,c

C.4.1 High Ranked Phenomena

The PIRT in Reference 2 lists [ ]a,c high ranked phenomena and plant components for AOO evaluation in ABWR and BWR/2-6 plants. This section determines the corresponding model parameters for these and their uncertainties.

The high ranked phenomena listed in Section C.2.7 were identified for AOO evaluation. Modeling of these phenomena or components in POLCA-T, determination of model uncertainties and the treatment of these uncertainties is presented in this Section.

C.4.2 [ ]a,c

[

]a,c

Table C.4-1 [ ]a,c a,c

]

C.4.3 [ ]a,c

[

]a,c

C.4.4 [ ]a,c

[

]a,c

a,c

[ ]

Figure C.4-1 [ ][a,c]

a,c

[ ]

Figure C.4-2 [ ][a,c]

Table C.4-2 [ ]a,c

a,c

[

]a,c

C.4.5 [ ]a,c

[

]a,c

[

]a,c

C.4.6 [ ]a,c

[

]a,c

C.4.7 [ ]a,c

[

]a,c

C.4.8 [ ]a,c

[

]a,c

[

]a,c

Table C.4-3 [ ]a,c a,c

[

]a,c

C.4.9 [ ]a,c

[

]a,c

C.4.10 [ ]a,c

[

]a,c

C.4.11 [ ]a,c

[

]a,c

C.4.12 [ ]a,c

[

]a,c

C.4.13 [ ]a,c

[

]a,c

C.4.14 [ ]a,c

[

]a,c

C.4.15 [ ]a,c

[

]a,c

C.4.16 [ ]a,c

[

]a,c

C.4.17 [ ]a,c

[

]a,c

C.4.18 [ ]a,c

[

]a,c

C.4.19 [ ]a,c

[

]a,c

C.4.20 [ ]a,c

[

]a,c

C.4.21 [ ]a,c

[

]a,c

[

]a,c

C.4.22 [ ]a,c

[

]a,c

C.4.23 [ ]a,c

[

]a,c

a,c

Figure C.4-3 [ ]a,c

[

]a,c

[

]a,c

a,c

[ ]

Figure C.4-4 [ ]a,c

[ ]a,c

a,c

[ ]

Figure C.4-5 [ ]a,c

C.4.24 [ ]a,c

[

]a,c

[

]a,c

Table C.4-4 [ ]a,c

a,c

[

]a,c

[

]a,c

[

]a,c

a,c

Figure C.4-6 [ ]a,c

a,c

[ ]

Figure C.4-7 [ ]a,c

C.4.25 [ ]a,c

[

]a,c.

C.4.26 [ ]a,c

[

]a,c

C.4.27 [ ]a,c

[

]a,c

C.4.28 [ ]a,c

[

]a,c

C.4.29 [ ]a,c

[

]a,c

C.5 PLANT DATA QUALIFICATION

This section provides qualification of POLCA-T using BWR transient plant data. Comparison against plant measurements provides an integral evaluation of POLCA-T's accuracy, which confirms the code's capability to model the interplay of important basic models and correlations as well as the codes interaction to system models (SAFIR). The test cases chosen demonstrate the capability of simulating the basic thermal-hydraulic, thermal-mechanical and kinetics related phenomena that govern the response of plant transients.

BWR plant data for the following test cases are evaluated:

- [ ]^{a,c} Beginning of Cycle (BOC) 1 all recirculation pump trip test
- Peach Bottom 2 End of Cycle (EOC) 2 turbine trip tests
- Hamaoka 5 ABWR BOC 1 start-up tests.

The previous qualifications presented in Appendix A and Appendix B of Reference 1 have already been approved by the NRC for simulation of BWR transient behaviors involving CRDA and stability transients. The validation presented here enhances the extension of POLCA-T qualification basis to apply for fast transient analysis. All together, these tests capture the important phenomena needed in transient analysis.

The code options for the AOO transients are slightly different from the code options for CRDA analysis. In comparison to the validation presented in Appendix A, the following options are used:

- [

]^{a,c}

C.5.1 [ ]a,c All Recirculation Internal Pump Trip Test

During commissioning of the ASEA Atom designed BWR [ ]a,c in [ ]a,c one of the tests performed was a trip of all six internal recirculation pumps. This transient has historically been one of the most limiting – and analyzed – ones for the plant operation with regards to Minimum Critical Power Ratio (MCPR).

The test results have been used extensively for the validation of the transient codes. A detailed description of the plant model and the test description can be found in Reference 15. This section provides a brief summary of the plant characteristics, transient test and POLCA-T simulation results.

This test is used to demonstrate POLCA-T's ability to predict [ ]a,c a recirculation pump trip transient. Also the capabilities to predict [ ]a,c are verified against measured [ ]a,c during the test.

The conclusions from this test case are:

a,c

[ ]

C.5.1.1 Plant Description

[ ]a,c is an internal pump reactor of ASEA Atom design. The reactor features, with the nominal data at the start of the operation, are shown in Table C.5-1.

Table C.5-1 General Information of [ ]a,c Initial Design

a,b,c

[ ]

Table C.5-1 General Information of [ ]a,c Initial Design a,b,c

[ ]

C.5.1.2 POLCA-T Model

The POLCA-T model consists of the primary coolant loop model based on a geometry described in Reference 15, a recirculation pump model, and a core model based on POLCA7 simulation.

The core response to recirculation flow reduction was studied for this transient test. For simplicity, [ ]a,c

The reactor pressure variation during the test was [ ]a,c

The recirculation pumps were described by homologue pump curves and were tripped by [ ]a,c

The RPV (narrow range) condensed water level measurement system was modeled by the POLCA-T [ ]a,c in the simulations. No other control or safety systems were simulated since these were not initiated during the test.

The core model simulates [ ]a,c

C.5.1.3 Test Conditions

All six internal recirculation pumps were tripped simultaneously at the operating conditions shown in Table C.5-2.

Table C.5-2 [ ]a,c All Recirculation Pump Trip Test Initial Conditions

a,b,c

[ ]

C.5.1.4 Fission Power

The measured Average Power Range Monitor (APRM) is based on [ ]^{a,c} Local Power Range Monitors (LPRM) with [ ]^{a,c} axial devices in each location. The measured APRM is presented using a [ ]^{a,c}. The recirculation pumps were tripped at time = [ ]^{a,c} in the simulation.

The calculated APRM is compared to the measured APRM. The [ ]^{a,c} signal was chosen for the comparison [ ]^{a,c}.

During the initial phase the calculated fission power matches the measured signal well. The dominating [ ]^{a,c} during the [ ]^{a,c}. Based on the excellent agreement between the calculated and measured fission power, the [ ]^{a,c} is accurately captured by POLCA-T.

[ ]^{a,c}

a,b,c

Figure C.5-1 Measured and Calculated APRM for [ ]^{a,c} RIP trip test

C.5.1.5 Channel Flow and Core Flow Distribution

Eight individual fuel channel flows are measured by use of differential pressure cells over the bundle inlet orifice. The core orifice zones and positions of the individual flow measurement are shown in Figure C.5-2. The bundles are evenly distributed in the central orifice zone.

The flow measurement positions are 211K301 through 211K304 and 211K311 through 211K314. The measured and calculated channel flow rates are shown in Figure C.5 3 and Figure C.5 4. The measured channel flows are presented using [ ]^{a,c}.

It is concluded that the channel flows are calculated accurately during transient conditions. The accuracy of channel flow calculation is independent of the location in the core. It demonstrates the capability of POLCA-T to accurately predict the core flow distribution.

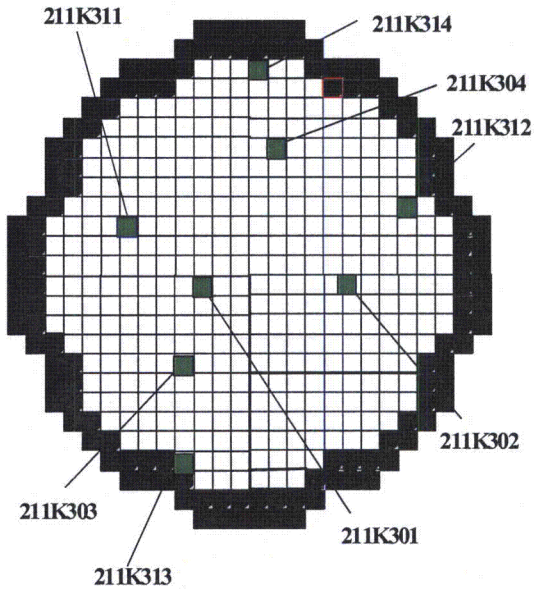

Figure C.5-2 Channel Flow Measurements for [ ]^{a,c} all RIP trip test

a,b,c

[ ]

Figure C.5-3 Measured and Calculated Channel Flows for [ ]a,c all RIP trip test

a,b,c

Figure C.5-4 Measured and Calculated Channel Flows for [ ]^{a,c} all RIP trip test

C.5.1.6 Recirculation Flow

The core recirculation flow (in kg/s) is evaluated from the fuel channel flow measurements by the following formula:

a,c

[ ]

The measured recirculation flow is compared to the calculated recirculation flow in Figure C.5 5. The measured recirculation flow is presented using [ ]a,c.

As can be seen the calculated and measured recirculation flows agree well [ ]a,c After that POLCA-T [ ]a,c the total recirculation flow by [ ]a,c

The measured and calculated recirculation pump speeds are also shown in Figure C.5 5. The measured pump speed is presented using [ ]a,c. The measured pump speed is an average of the six measured individual pump speeds. The calculated pump speed is determined from [ ]a,c

It is concluded that POLCA-T simulates the recirculation flow reduction accurately from forced circulation down to natural circulation. This conclusion is supported by the good agreement in the core power decay along with the flow reduction.

a,b,c

[ ]

Figure C.5-5 Measured and Calculated Recirculation Flow for [ ]a,c all RIP trip test

C.5.1.7 Moderator (Void) Reactivity Uncertainty

The [ ]a,c all RIP trip test provides a benchmark for the void reactivity effect in a BWR. Following the pump trip, the reactor coolant flow decreases and steam void in the core increases. Consequently, reactivity and power decreases due to a negative void reactivity feedback. The time from the trip of the pumps down to minimum power [ ]a,c is studied to determine the moderator feedback accuracy.

The initial power response is governed by the moderator feedback. Other reactivity effects exist but play a minor role. No scram or other safety functions were initiated during the pump trip test. [

]a,c

The fuel temperature reactivity feedback (Doppler) was estimated as well. The calculated average fuel temperature decrease from the initial conditions to minimum power was [ ]^{a,c}. This change in fuel temperature corresponds to a Doppler feedback of about [ ]^{a,c} which is compared to the total reactivity change of [ ]^{a,c} The evaluated Doppler reactivity corresponds to a relative reactivity contribution of [ ]^{a,c} This contribution has a [ ]^{a,c} on the overall power response. [

]^{a,c}

The measured power response is compared to the POLCA-T simulation in Figure C.5-6. [

]^{a,c}

a,b,c

[ ]

Figure C.5-6 The Moderator Reactivity Uncertainty for [ ]^{a,c} all RIP Trip Test

C.5.2 Peach Bottom 2 Turbine Trip Tests

Peach Bottom 2 (PB2) End of Cycle (EOC) 2 Turbine Trip (TT) tests were included in the POLCA-T qualification base for CRDA analysis in Appendix A of Reference 1. The analytical results of the initial conditions, sequence of events, transient pressure and power response to these tests were already approved by the NRC.

The PB2 TT3 test is not included in this report since it doesn't provide additional validation compared to TT1 and TT2. The reactivity feedback and other kinetic phenomena are cut down by the scram early during the TT3 test and make the case less interesting from the code kinetic validation point of view.

The PB2 EOC2 TT steady-state evaluation results in Appendix A were reviewed in regards to the validation of POLCA-T to predict the initial transient conditions. POLCA-T was found to provide acceptable power distribution for initialization of BWR cores for transient simulation.

The predicted core power transient response was compared to the measured transient response in Appendix A. The comparison verified that the transient response peak power is modeled accurately.

The comparison of the LPRM measurements verified that the core transient power following the initiation of core void collapse was modeled adequately. This verifies POLCA-T's capability to model transient coolant conditions and reactivity feedback.

The qualification analyses in Appendix A demonstrated an acceptable coupling between the thermal-hydraulic and neutronic models to determine core reactivity, transient flux distribution, and local heat flux for fast transients (on the time scale of seconds).

This section includes additional results for the PB2 EOC2 TT1 and TT2 to demonstrate the accuracy of predicted energy in the power peak, the system pressure and the RPV water level. The TT2 peak power simulation, together with the [ ]^{a,c} all RIP trip test, is used to evaluate the uncertainty in moderator feedback for AOO transient uncertainty analysis.

Unlike the CRDA calculation options used in Appendix A, calculation options for AOO and ATWS transients are used in the simulations presented in this section.

In addition to the conclusions presented in Appendix A, the following conclusions can be made for these test cases:

- Both the POLCA-T calculated peak power and the calculated fission energy are in a good agreement with the measured data.
- The POLCA-T calculated turbine inlet, steam dome and core exit pressures are in a good agreement with the measured pressures. [ ]^{a,c}
- Even though the [ ]^{a,c} level measurement model was used, the POLCA-T calculated condensed RPV water level is in good agreement with the measured level. [ ]^{a,c}

C.5.2.1 Fission Power and Energy

The measured and calculated fission power for PB2 TT1 and TT2 are shown in Figure C.5-7 and Figure C.5-8. POLCA-T calculated power peaks are in good agreement with the measured power. The agreement between the predicted and the measured timing of power peaks is good. The calculated TT2 power peak time is [ ]a,c the measured time. The TT2 Turbine Stop Valve (TSV) closure profile and timing was used also in the TT1 simulation since the TSV position recording failed during the test. [ ]a,c

a,b,c

Figure C.5-7 Measured and Calculated Fission Power for PB2 TT1

a,b,c

Figure C.5-8 Measured and Calculated Fission Power for PB2 TT2

For the evaluation of transients, the generated total fission energy is of interest. The [ ]a,c in Figure C.5-9 and Figure C.5-10 for the TT1 and TT2 cases respectively. The calculated generation of fission energy is in good agreement with the measured fission energy for TT2 and [ ]a,c for TT1.

a,b,c

Figure C.5-9 Measured and Calculated Fission Energy for PB2 TT1

a,b,c

Figure C.5-10 Measured and Calculated Fission Energy for PB2 TT2

C.5.2.2 System Pressure

The measured and calculated turbine inlet pressure for PB2 TT1 and TT2 are shown in Figure C.5-11 and Figure C.5-12. POLCA-T simulated pressure levels are in good agreement with the measured pressures.

a,b,c

Figure C.5-11 Measured and Calculated Turbine Inlet Pressure for PB2 TT1

a,b,c

Figure C.5-12 Measured and Calculated Turbine Inlet Pressure for PB2 TT2

The measured and calculated steam dome pressure for PB2 TT1 and TT2 are shown in Figure C.5-13 and Figure C.5-14. POLCA-T calculated pressure levels are in good agreement with the measured pressures.
[

]a,c

a,b,c

Figure C.5-13 Measured and Calculated Steam Dome Pressure for PB2 TT1

a,b,c

Figure C.5-14 Measured and Calculated Steam Dome Pressure for PB2 TT2

The measured and calculated core exit pressure for PB2 TT1 and TT2 are shown in Figure C.5-15 and Figure C.5-16. [

]a,c

a,b,c

Figure C.5-15 Measured and Calculated Core Exit Pressure for PB2 TT1

a,b,c

Figure C.5-16 Measured and Calculated Core Exit Pressure for PB2 TT2

C.5.2.3 RPV Water Level

The measured and calculated RPV narrow range water level (above the axial vessel zero) for PB2 TT1 and TT2 are shown in Figure C.5-17 and Figure C.5-18. [ ]a,c the POLCA-T calculated condensed RPV water level is in good agreement with the measured level.

a,b,c

Figure C.5-17 Measured and Calculated RPV Water Level for PB2 TT1

a,b,c

Figure C.5-18 Measured and Calculated RPV Water Level for PB2 TT2

C.5.2.4 Moderator (Void) Reactivity Uncertainty

The PB2 TT2 test provides an excellent benchmark for the void reactivity effect in a BWR. The transients were initiated by the closure of the TSV. The resulting pressure wave propagates to the core. It is attenuated by opening of the bypass valve. When the pressure wave reaches the core the steam void reduction results in reactivity and core power increase.

The initial power response is governed by the moderator feedback. Other reactivity effects exist but play a minor role. Control rods were inserted during the TT2 but the reactivity peak was already reached when the control rods started to move. The core reactivity and power response was not significantly influenced by the control rods early (from time 0 to 0.74 seconds) in the transient.

[ ]a,c

The fuel temperature reactivity feedback (Doppler) was estimated as well. The calculated average fuel temperature increase from initial conditions up to the peak power was [ ]a,c. This change in fuel temperature corresponds to a Doppler feedback of about [ ]a,c which is compared to the total reactivity change of [ ]a,c. The evaluated Doppler reactivity corresponds to a relative reactivity contribution of [ ]a,c. This contribution has [ ]a,c on the overall power response. [ ]a,c

The measured power response is compared to the POLCA-T simulation in Figure C.5-19. [ ]a,c

a,b,c

Figure C.5-19 The Moderator Reactivity Uncertainty for PB2 TT2

C.5.3 Hamaoka 5 Start-Up Tests

POLCA-T is validated against the Chubu Electric Power Co., Inc. Hamaoka 5 ABWR start-up tests in Reference 7. The start-up test program was performed prior to commercial operation in order to demonstrate a safe and stable plant response to AOO's.

The Hamaoka 5 start-up tests that involve phenomena interesting from a transient analysis perspective are simulated with POLCA-T. Eight start-up transient tests are simulated and results are compared against the recorded test data in order to demonstrate POLCA-T's applicability for AOO analyses:

1. Pressure Control System Step Change Test (Section C.5.3.3),
2. Feedwater Control System Step Change Test (Section C.5.3.4),
3. Recirculation Flow Control System Ramp Change Test (Section C.5.3.5),
4. High Pressure Heater Drain Pump Trip Test (Section C.5.3.6),
5. Reactor Internal Pump Trip Test (Section C.5.3.7),
6. Generator Load Rejection Test (Section C.5.3.8),
7. All Main Steam Isolation Valves Closure Test (Section C.5.3.9),
8. Selected Control Rod Run-In Test (Section C.5.3.10).

POLCA-T and SAFIR have a good capability to model an ABWR plant and capture the transient characteristics of it. The conclusions from these test cases are:

- POLCA-T accurately captures the transient change of core power, recirculation flow, feedwater and steam flow, system pressure and RPV water level.
- POLCA-T and SAFIR level measurement system modeling, described in Section C.3.3, accurately predict the condensed RPV water level.
- POLCA-T and SAFIR recirculation pump modeling accurately predicts the RIP runback (tests 4, 6 and 7) and the RIP trip (tests 5 and 7).
- POLCA-T and SAFIR accurately models the TCVs and TBVs (tests 1 and 6) and SRVs (test 7).
- POLCA-T accurately models both the (scram) hydraulic control rod insertion (tests 6 and 7) and the electrical control rod insertion (test 8).

C.5.3.1 Plant Description

The ABWR Hamaoka Unit-5 of Chubu Electric Power Co. Inc., began commercial operation in January 2005. At the Hamaoka site, which is located in Shizuoka Prefecture on Japan's east coast facing the Pacific Ocean, there are three operating reactor units with a total generator output of 3.6 GWe. Table C.5-3 summarizes the general features of Hamaoka 5 (Reference 7):

Table C.5-3 General Information of the Hamaoka 5 ABWR Plant *

Item	Nominal Data
Reactor Thermal Output	3,926 MWt
Reactor Pressure	7.17 MPa
Steam Flow	2,120 kg/sec
Core Flow	14,500 kg/sec
Number of Fuel Assemblies	872
Initial Cycle Fuel Design	9x9
Number of Control Rods	205
Coolant Recirculation	10 RPV Internal Pumps
Control Rod Drive	Electric Motor (Normal) Hydraulic Drive (Scram)

The initial core was loaded with 9x9 fuels, which consists of two different U-235 enriched bundles, 2.2 and 3.7 w/o average. The fuel design features high burnable absorber concentrations up to 10.0 w/o Gadolinium. Compared, for example, to a conventional BWR/5 the fuel lattice pitch is 2.54 mm wider in an ABWR to increase the amount of non-boiling water in the inter-assembly bypass. This design leads to less negative moderator (void) reactivity during transients.

C.5.3.2 POLCA-T Model

The POLCA-T model of Hamaoka 5 was generated based on the first cycle core loading and ABWR geometry of the primary coolant loop (RPV), main steam lines and control systems.

The PHOENIX4 cross-sections and the POLCA7 core model were validated against cold critical tests. The cold critical tests were performed for local critical conditions. A few adjacent control rods were withdrawn at a peripheral or central position of the core to achieve several local critical conditions. The standard deviation of the calculated eigenvalue among the cold critical tests was evaluated to 0.3 %. Also, the calculated core average axial power shape was compared to the on-line core monitor, which is based on measured data. Figure C.5-20 shows the comparison between the predicted results and on-line monitoring at the beginning of the cycle. POLCA7 results are in good agreement with the on-line monitoring.

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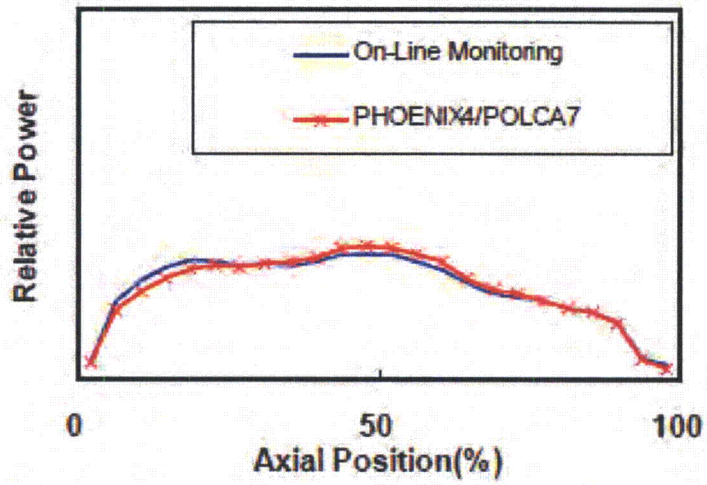

Figure C.5-20 Calculated and Measured Axial Power Shape for BOC 1 Hamaoka 5 *

A schematic of the POLCA-T Hamaoka 5 primary coolant loop model, steam line connections and feedwater boundary location are shown in Figure C.5-21.

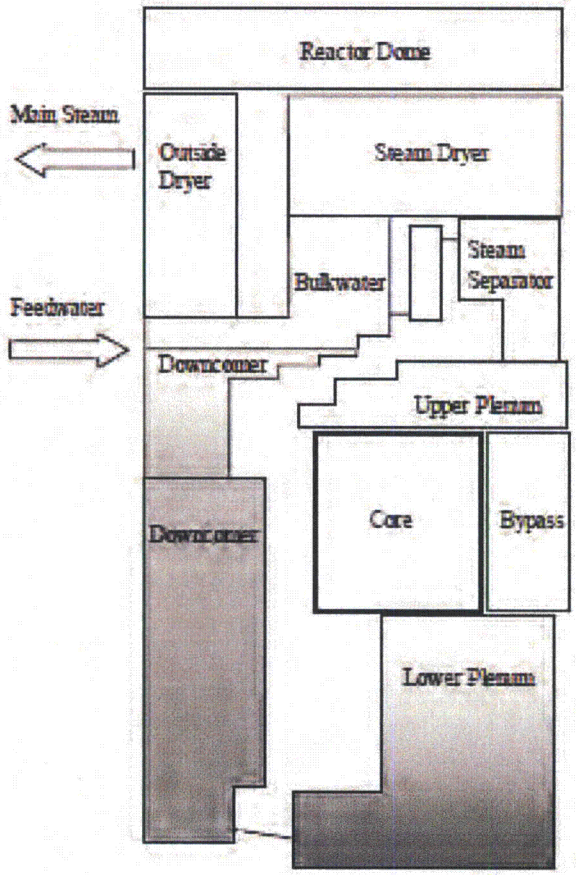

Figure C.5-21 Schematic of the POLCA-T Hamaoka 5 ABWR RPV Model *

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The plant control systems were simulated with SAFIR. The model consists of control systems such as SB&PCS, FWCS and RFCS.

C.5.3.3 Pressure Control System Step Change

The primary function of the SB&PCS is to control reactor vessel pressure by controlling the turbine control and/or steam bypass valves. For normal operation, the Turbine Control Valves (TCVs) are used to control the steam dome pressure. However, if the total steam flow demand from the pressure controller exceeds the effective TCV steam flow capacity, the excess steam flow is then bypassed to the main condenser through the turbine bypass valves.

This test was performed by a 0.069 MPa stepwise decrease/increase of the reactor pressure set point which resulted in a reactor dome pressure change.

Figure C.5-22 and Figure C.5-23 show comparison of the simulation results and the test data. The pressure set point reduction (starting at time = 0 seconds) results in an increased steam flow and decreased reactor dome pressure. A reduction of reactor pressure increases the steam generation in the core which causes a decrease in reactor power and an increase in the water level. Finally, the reactor pressure stabilizes 0.069 MPa below the initial pressure. The increase of set point (at time = 50 seconds) has an inverse response to the set point reduction. The POLCA-T results agree well with the test data.

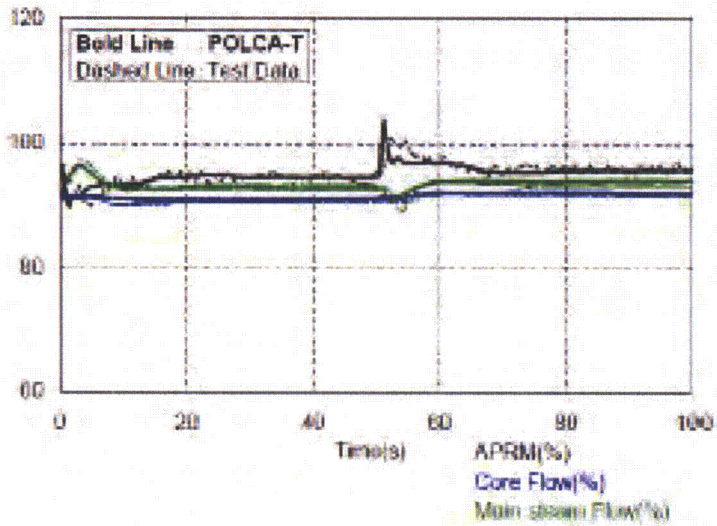

Figure C.5-22 Results of Pressure Control System Step Change Test *

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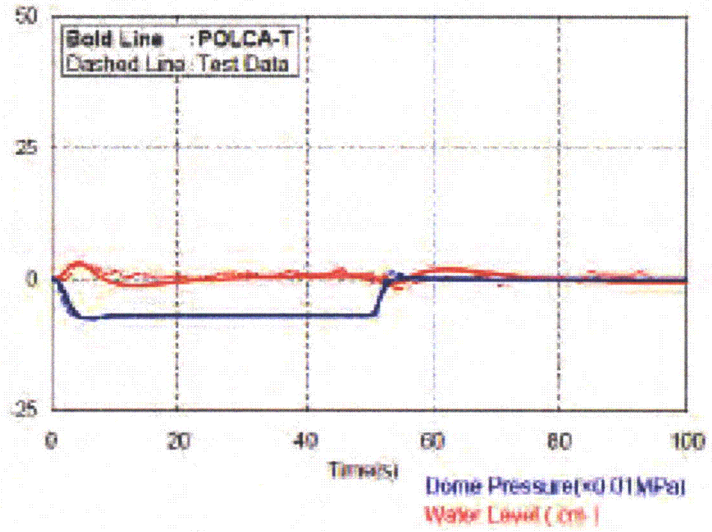

Figure C.5-23 Results of Pressure Control System Step Change Test *

C.5.3.4 Feedwater Control System Step Change

The main objective of the FWCS is to control the feedwater flow into the RPV to maintain an appropriate vessel water level. At rated conditions, FWCS is in a three-element control mode and uses the measured water level, feedwater flow and main steam flow to control the feedwater flow.

This test was performed by a 0.15 m stepwise decrease/increase of the reactor water level set point which resulted in a reactor water level change.

Figure C.5-24 and Figure C.5-25 show comparisons of the simulation results and the test data. After the vessel water level set point was decreased (starting at time = 0 seconds), the water level starts to decrease due to the feedwater flow reduction. This also causes a decrease in the core inlet sub-cooling which causes a decrease in the neutron flux. The RPV water level decreases until it stabilizes 0.15 m below the initial level. After 20 seconds, the feedwater flow stabilizes close to the initial value. The set point increase perturbation (at time = 100 seconds) shows an inverse response to the decrease perturbation. The POLCA-T results agree well with the test data.

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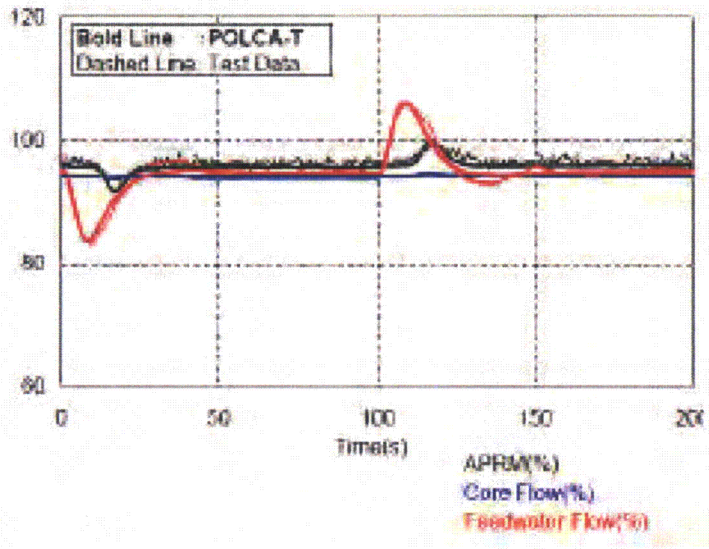

Figure C.5-24 Results of Feedwater Control System Step Change Test *

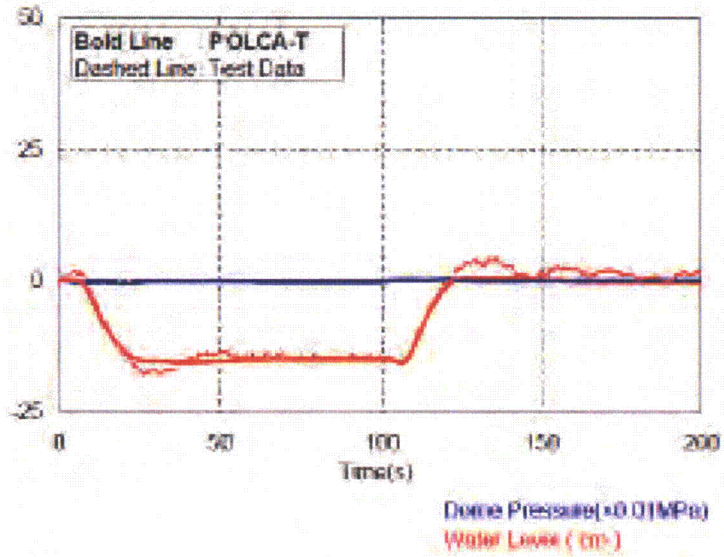

Figure C.5-25 Results of Feedwater Control System Step Change Test *

C.5.3.5 Recirculation Flow Control System Ramp Change

The function of the RFCS is to control reactor power by controlling the reactor core flow rate. This test was performed using a 10 % ramp decrease/increase of the reactor power (load) set point which resulted in a reactor power change. The objective of this test was to determine the plant response (core flow, neutron flux) and to adjust the performance of the recirculation control system in order to achieve a stable response.

Figure C.5-26 and Figure C.5-27show comparisons of the simulation results and test data. After the load set point was decreased 10 % (starting at time = 0 seconds), the core flow decreases in order to reduce the reactor power, which also causes a decrease in reactor pressure. Due to the pressure decrease, more steam

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is generated in the core, and the vessel water level initially increases. After about 60 seconds, the reactor equilibrates to a new stable power level. Increase of the load set point (at time = 100 seconds) has an inverse response to the set point reduction. The POLCA-T results agree well with the test data.

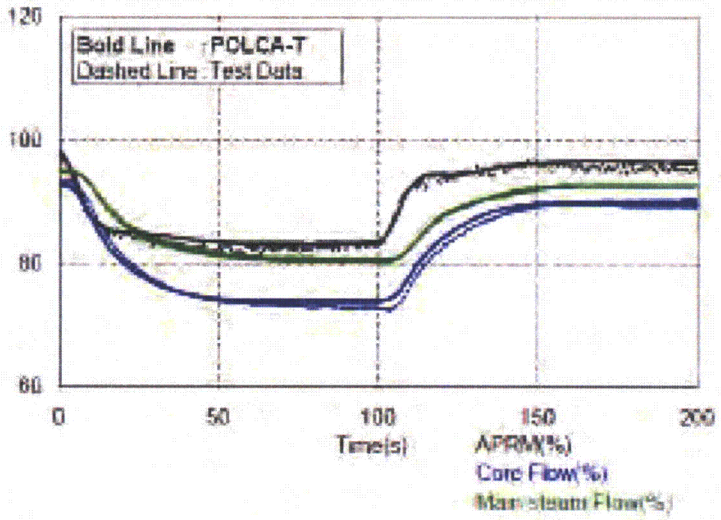

Figure C.5-26 Results of Recirculation Flow Control System Ramp Change Test *

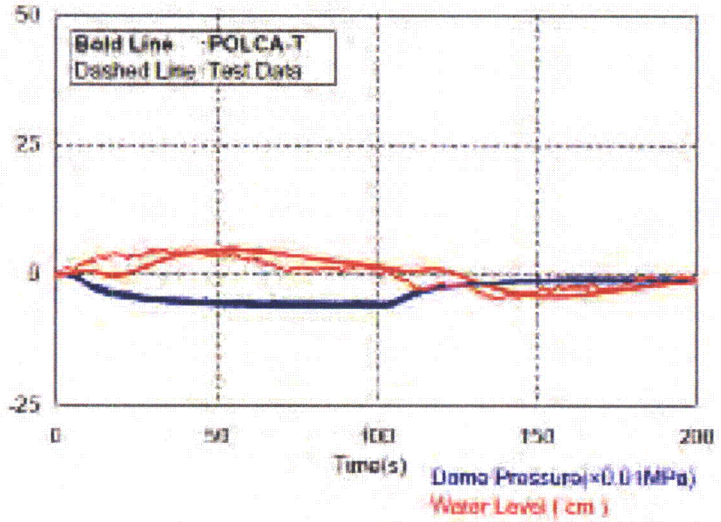

Figure C.5-27 Results of Recirculation Flow Control System Ramp Change Test *

C.5.3.6 High Pressure Heater Drain Pump Trip Test

The ABWR is equipped with three High Pressure heater Drain Pumps (HPDP). Two pumps are operating during normal operation and one pump is on standby. The HPDP feeds water from the High Pressure Drain Tank (HPDT) to the suction side of the reactor feedwater pump.

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This test was initiated by a manual trip of two HPDPs. In order to avoid the decrease in pressure at the feedwater pump suction side, one feedwater pump was tripped automatically and start-up of the standby feedwater pump was bypassed.

Figure C.5-28 and Figure C.5-29 show comparisons of the simulation results and the test data. In the simulation, the transient was initiated by trip of one feedwater pump. In order to reduce the reactor power and to prevent the water level decrease, the core flow starts to decrease due to the runback of all RIPs about two seconds after the HPDP is tripped. At the start of the transient, the reactor vessel water level decreases due to the feedwater pump trip. The water level starts to rise again since the main steam flow becomes lower than the feedwater flow. Finally, when the feedwater flow, controlled by the FWCS, matches the steam flow, the plant equilibrates at new conditions. The POLCA-T results agree well with the test data.

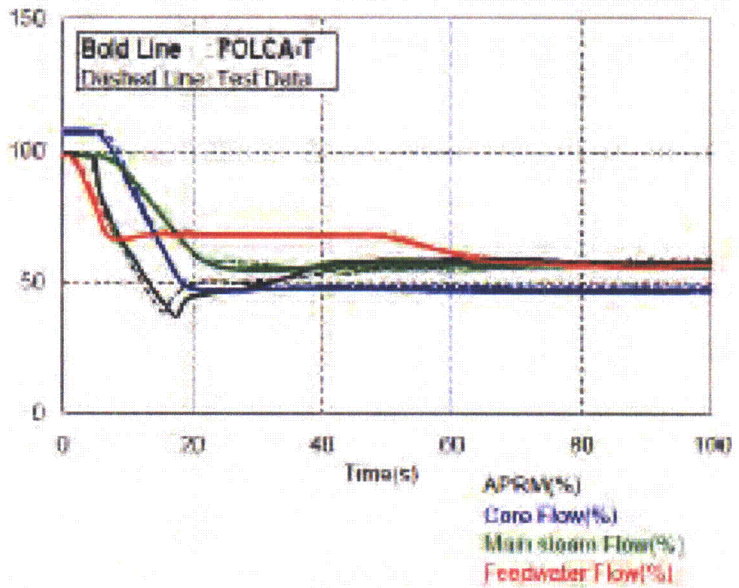

Figure C.5-28 Results of High Pressure Heater Drain Pump Trip Test *

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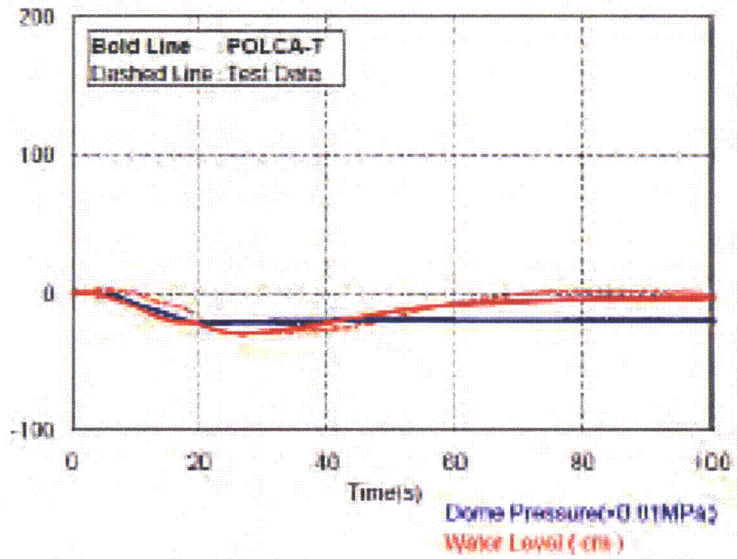

Figure C.5-29 Results of High Pressure Heater Drain Pump Trip Test *

C.5.3.7 Reactor Internal Pump Trip Test

The ABWR is equipped with 10 RIPs. This test was performed by tripping three RIPs in order to confirm the reactor response to a RIP trip. Trip of three RIPs is the worst case caused by a single failure since only two or three RIPs are connected to one power supply bus in an ABWR. The test corresponds to a transient initiated by a trip of three RIPs.

Figure C.5-30 and Figure C.5-31 show comparisons of the simulation results and the test data. After the trip of three pumps, the core flow decreases rapidly for a few seconds and then equilibrates to a new level. Neutron flux also shows a rapid reduction because of the void generation caused by the core flow decrease.

The steam (void) generation causes the RPV level to rise due to swelling. The water level is controlled by the FWCS. The system mitigates the water level rise and the plant equilibrates to new conditions. The POLCA-T results agree well with the test data.

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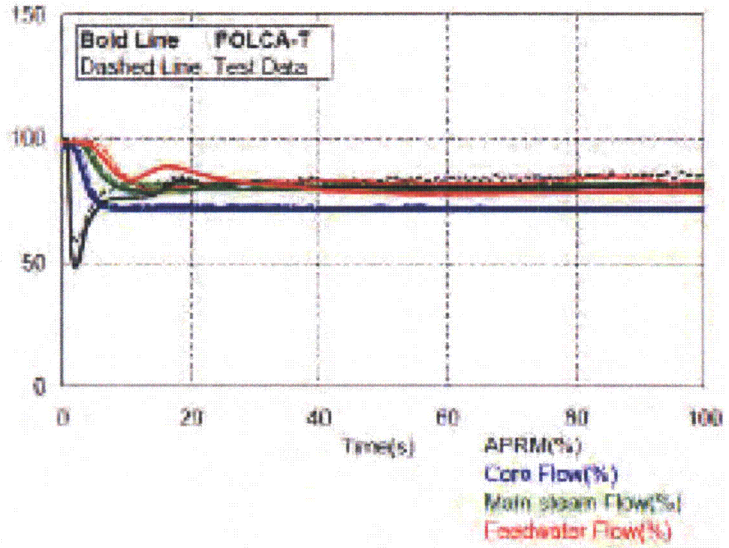

Figure C.5-30 Results of Reactor Internal Pump Trip Test *

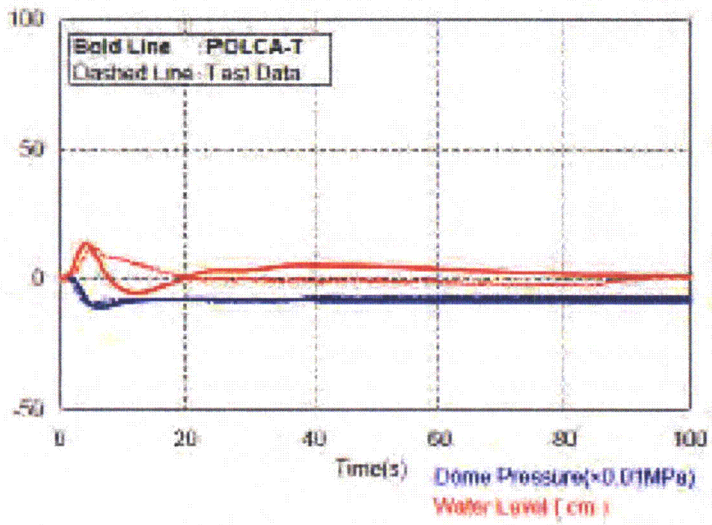

Figure C.5-31 Results of Reactor Internal Pump Trip Test *

C.5.3.8 Generator Load Rejection Test

A load rejection is initiated whenever electrical grid disturbances, resulting in a significant loss of generator electrical load, occur. In this event, fast closure of the TCVs is activated by the power load unbalance relay. Fast closure of TCVs initiates reactor scram and trip of four RIPs.

In this test, the transient was initiated by simulating the generator load rejection with bypass valve opening.

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Figure C.5-32 and Figure C.5-33 show comparisons of the simulation results and the test data. A fast closure of the TCVs causes a rapid reduction of turbine steam flow which results in an increased reactor pressure. At the same time, the TBVs are signaled to open by the SB&PCS in order to bypass steam from the reactor. The opening of the TBVs significantly mitigates the increase in reactor pressure. The power generation ends when the control rods are fully inserted into the core by the hydraulic scram system. The safety relief valve does not initiate to open. The core flow initially decreases by trip of four RIPs and the runback of the remaining six RIPs. The load rejection also causes a relatively large drop in water level due to the scram. Reactor pressure is set down at lower level than the initial value by the SB&PCS. The POLCA-T results agree well with the test data.

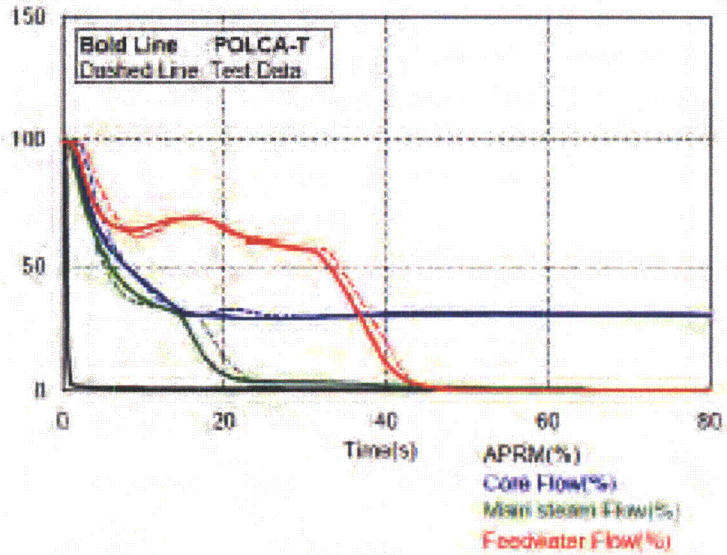

Figure C.5-32 Results of Generator Load Rejection Test *

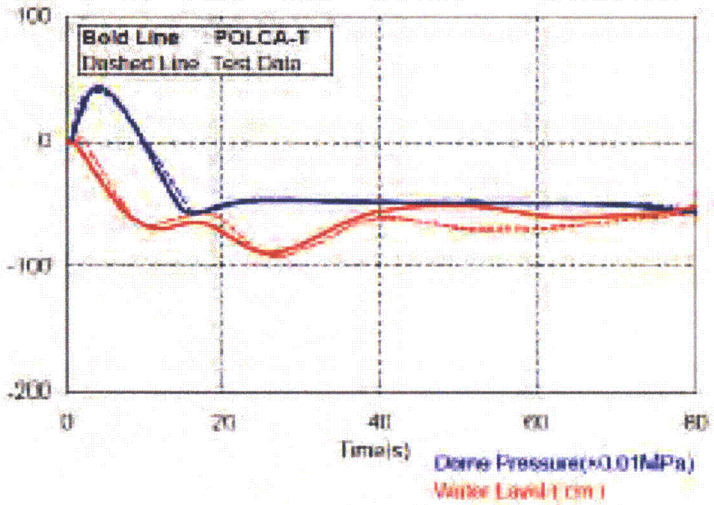

Figure C.5-33 Results of Generator Load Rejection Test *

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C.5.3.9 All Main Steam Isolation Valves Closure Test

MSIV closure is initiated by various steam line and nuclear system malfunctions, or operator actions. MSIV closure initiates a hydraulic reactor scram via valve position signal to the reactor protection system.

In this test, the transient was initiated by a manual closure of all MSIVs.

Figure C.5-34 and Figure C.5-35 show comparisons of the simulation result and the test data. The MSIVs closure causes a pressure increase in the reactor vessel. The pressure increase is mitigated by the opening of the SRVs when the pressure exceeds the opening set points. Core flow initially decreases by the RIPs runback following the hydraulic reactor scram. The hydraulic reactor scram also causes a large decrease in RPV water level, which causes the water level to reach the low water level set point. This set point initiates the trip of four RIPs. The POLCA-T results agree well with the test data.

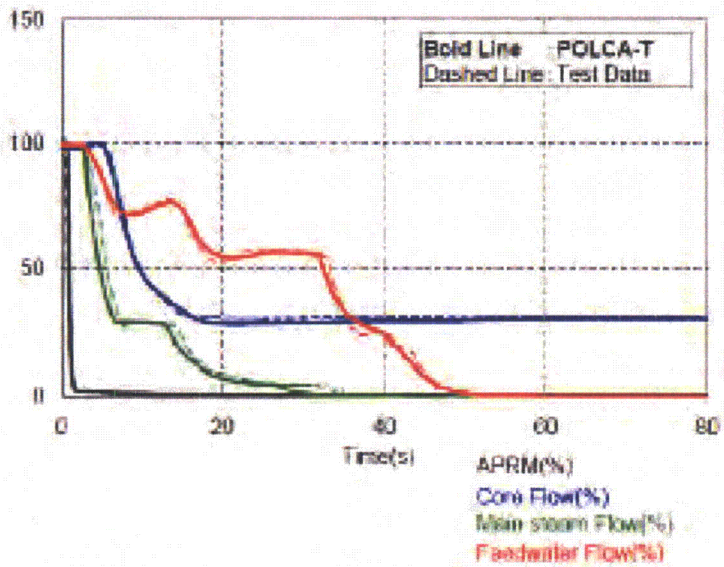

Figure C.5-34 Results of All MSIV Closure Test *

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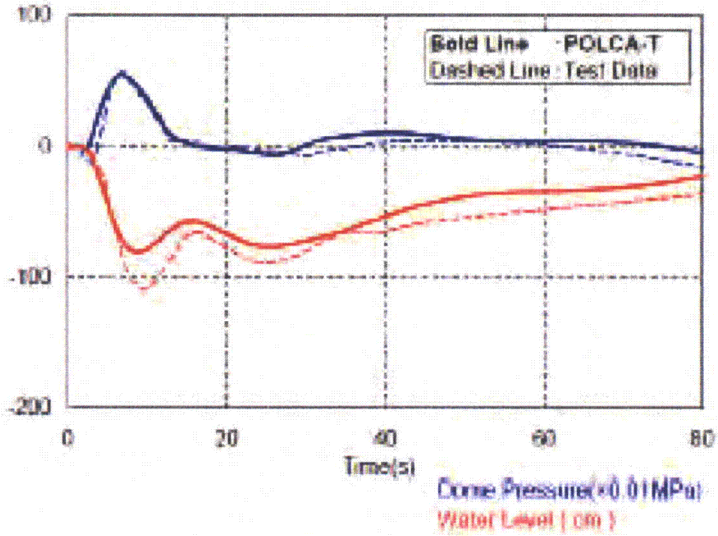

Figure C.5-35 Results of All MSIV Closure Test *

C.5.3.10 Selected Control Rod Run-In Test

The Selected Control Rod Run-In (SCRRI) function is provided as a core instability countermeasure.

A trip of two or more RIP is followed by an electrical insertion of pre-assigned control rods if the pump trips leads to operation in the high power and low flow region.

In this test, the transient was initiated by a manual SCRRI activation at 50% power level.

Figure C.5-36 and Figure C.5-37 show comparisons of the simulation results and the test data. The selected control rods are inserted by the drive motor and full insertion takes approximately two minutes. The reactor power decreases gradually with the control rod insertion. Due to the power reduction, steam (void) generation in the core becomes lower and causes a decrease in water level. The feedwater flow is controlled by the FWCS to maintain the water level. The feedwater flow is higher than the main steam flow which results in a water level increase. The POLCA-T results agree well with the test data.

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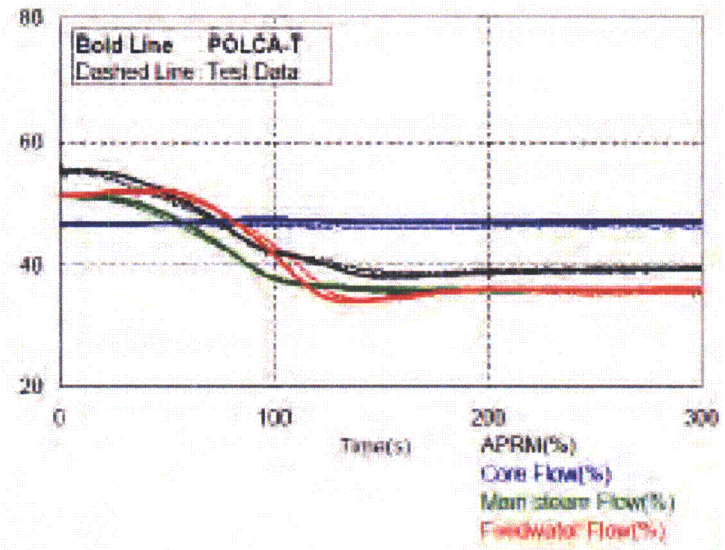

Figure C.5-36 Results of SCRRI Test *

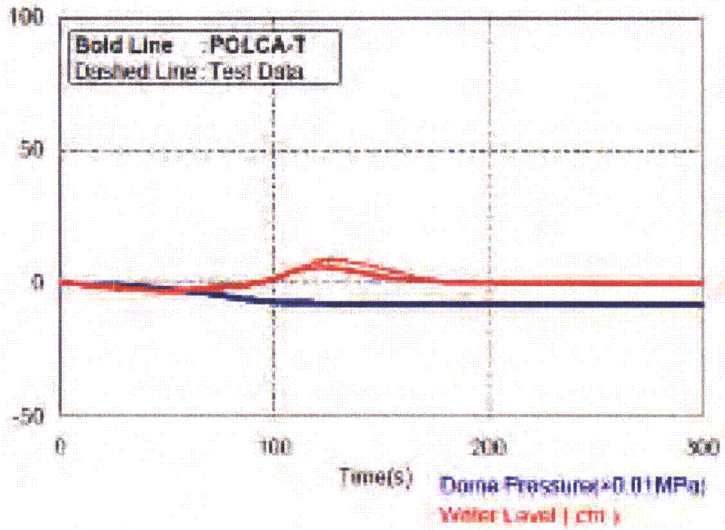

Figure C.5-37 Results of SCRRI Test *

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C.6 APPLICATION UNCERTAINTIES

Reference 2 provides the fast transient uncertainty analysis methodology. This section is limited to a discussion of how input parameters are treated to quantify the uncertainty in the AOO FoM.

The analysis initial conditions and their uncertainty in AOO analysis are discussed in Section C.6.1.

The plant parameters and their uncertainty in AOO analysis are discussed in Section C.6.2.

The high ranked code input parameters used to quantify the sensitivity of the FoM to individual model uncertainties are summarized in Section C.6.3.

C.6.1 Initial Conditions

Initial conditions are the key plant inputs that define the (steady-state) operating conditions prior to a specific transient event. Initial conditions may vary due to the allowable operating range and/or the uncertainty in the process measurements at a given operating condition.

The methodology for choosing the initial conditions for AOO events is described in Reference 2.

For AOO, event analysis will be initiated from the limiting initial conditions. A single operating state or operating condition can conservatively be used to cover all other possible states. Therefore, no uncertainty needs to be evaluated in the initial conditions for the AOO analysis.

C.6.2 Plant Parameters

A plant parameter is a plant-specific system or component quantity, such as a protection system set point, valve capacity and/or stroke time, coolant system capacity and temperature, pump head or inertia, etc.

The high ranked plant systems and components with respect to the AOO FoM are [

]a,c

C.6.3 Model Parameters and Uncertainties

The high ranked phenomena for AOO analysis are established in Reference 2. These high ranked phenomena are coupled to POLCA-T code models and model parameters in Section C.4.

The model uncertainty parameters and their distributions for the high ranked phenomena to be regarded in the transient uncertainty analysis are also evaluated in Section C.4. Also, the models or model parameters that shall be treated conservatively in AOO analysis are identified.

The high ranked phenomena [

]a,c

[

]a,c

The treatment of the high ranked models and parameters for AOO events are summarized in Table C.6-1.
[

]a,c

Table C.6-1 Model Parameters and Uncertainties for All High Ranked AOO Phenomena a,c

a,c

a,c

[ ]

C.7 DEMONSTRATION ANALYSIS

This section describes an AOO demonstration analysis using POLCA-T and the analysis methodology presented in Reference 2. The purpose of the analyses is to demonstrate the application of the methodology of evaluating the operating limits using POLCA-T.

Two types of analysis are performed; a nominal and a statistical. The statistical Monte-Carlo analysis applies the model uncertainties evaluated in Section C.4 and summarized in Section C.6.3.

The models that are to be treated conservatively [ ]^{a,c}. This is done to demonstrate the level of uncertainty in POLCA-T evaluated MCPR.

The calculated results are only an example and shall not be applied in a real plant safety analysis.

C.7.1 Nominal Analysis

A load rejection without bypass event is chosen for the demonstration. It is a transient that may challenge the AOO acceptance criteria.

The core SLMCPR is assumed to = [ ]^{a,c} in the demonstration of MCPR evaluation. The CPR-correlation for [ ]^{a,c} fuel, validated with POLCA-T in Section C.3.1, is used in the analysis.

C.7.1.1 Initial Conditions

The demonstration is done for a typical ABWR plant with a [ ]^{a,c} The fuel has an [ ]^{a,c}.

The core loading pattern is of [ ]^{a,c} with a maximum bundle power peaking of [ ]^{a,c} and maximum nodal power peaking of [ ]^{a,c} compared to the core average power.

The ABWR core flow window at rated power is [ ]^{a,c} of the nominal flow. The operating point used in the analysis is [ ]^{a,c} power and [ ]^{a,c} flow at [ ]^{a,c} conditions. The demonstration analysis is initiated at [ ]^{a,c} power level.

Table C.7-1 LRNBP Demonstration Case Initial Conditions a,c

[ ]

The simulated core average axial power profile at the initial conditions is shown in Figure C.7-1.

a,c

Figure C.7-1 Simulated initial axial power profile

C.7.1.2 Conservative Models and Parameters

A load rejection with no available steam bypass is a pressure increase event. The high ranked plant models and parameters that shall be treated conservatively for a pressure increase AOO event are found in the PIRT in Reference 2. The following phenomena are treated conservatively in the analysis:

a,c

a,c

C.7.1.3 Nominal Results

The transient is simulated by a closure of all TCVs together with a failure to open all the TBVs. The sequence of events for the simulated load rejection transient is shown in Table C.7-2.

After sensing a significant loss of electrical load on the generator, the TCVs are commanded to close fast to prevent excessive speed of the turbine-generator rotor. The fast closure of the TCVs initiates reactor scram and a trip of four RIPs, thereby reducing the core flow. Closure of the TCVs causes a sudden reduction in the steam flow, which results in an increase in system pressure and the hydraulic scram is initiated. At the same time, the TBVs are signaled to open by the SB&PCS in order to bypass steam from the reactor. When all turbine bypass valves fail to open the reactor pressure continues to increase rapidly and cause an increase in reactor power. The SRVs open to lower the reactor pressure and the event is over when the control rods are fully inserted into the core.

Table C.7-2 LRNBP Demonstration Case Predicted Sequence of Events

a,c

The calculated APRM, core average heat flux, scram insertion, steam dome pressure, recirculation flow and RPV condensed water level of the nominal case are shown in Figure C.7-2 through Figure C.7-5 . The calculated minimum CPR in the core during the event is shown in Figure C.7-6. The analysis results are summarized in Table C.7-3.

Table C.7-3 LRNBP Demonstration Case Results

a,c

a,c

[ ]

Figure C.7-2 Simulated APRM, Scram and Average Heat Flux in an ABWR LRNBP event

a,c

[ ]

Figure C.7-3 Simulated Reactor Pressure in an ABWR LRNBP event

a,c

Figure C.7-4 Simulated Recirculation Flow in an ABWR LRNBP event

a,c

Figure C.7-5 Simulated Condensed Water Level in an ABWR LRNBP event

a,c

[ ]

Figure C.7-6 Simulated Minimum CPR in an ABWR LRNBP event

C.7.2 Uncertainty Analysis

The statistical uncertainty evaluation methodology described in Reference 2 is applied to account for the uncertainty in input and modeling parameters for calculating the core and event specific MCPR for the generator load rejection with no bypass transient.

Upper probability bound for MCPR at 95% is estimated with 95% confidence. As the analysis requires the evaluation for one parameter at the limits of 95/95, the minimum amount of code runs is 59 assuming that the MCPR is not normally distributed and that the uncertainty is calculated by the order statistics method.

The individual uncertainty contributors, as defined in Table C.6-1, are sampled according to their probabilistic distribution functions. A run matrix of 59 components (individual transient cases) is created.

The rest of the code input and modeling parameters are treated like in the nominal case analysis.

C.7.2.1 Model Uncertainties

The evaluated model uncertainties for pressure increase event in Table C.6-1 are used to quantify the uncertainty in the evaluated MCPR for the simulated transient. Table C.7-4 shows the parameters used in this demonstration of uncertainty analysis for the LRNBP transient. The fuel specific values are examples of typical uncertainties for [ ]^{a,c}.

Table C.7-4 Uncertainty Parameters and Distributions for an ABWR LRNBP

a,c

a,c

C.7.2.2 Statistical MCPR

The ICPR and CPR_{min} values are extracted and MCPR is calculated for each of the 59 uncertainty code runs.

An Anderson-Darling normality test is performed and Anderson-Darling statistics A*2 is calculated to be lower than 0.751. Hypothesis of normality is therefore not rejected for a 5% level test and analysis of variance method is used to calculate the MCPR for the generator load rejection with no bypass transient event according to:

$$MCPR_{95,95} = \overline{MCPR} + z_{95,95}\sigma_{MCPR} \qquad \text{(C.7-1)}$$

Where:

$MCPR_{95,95}$	95th percentile estimate of MCPR with 95% confidence,
$\overline{MCPR}$	Average value for MCPR calculated from the code runs,
$z_{95,95}$	Factor for one sided normal tolerance limit,
σ_{MCPR}	Sample standard deviation calculated from the code runs.

The results from the uncertainty code runs with randomly sampled input parameters (according to uncertainties and their distributions from Table C.7-4) are shown in Figure C.7-7 where the x-axis shows the number of run (1-59) and y-axis shows the calculated MCPR for each case.

a,c

[ ]

Figure C.7-7 Uncertainty Analysis of an ABWR LRNBP event

The resulting MCPR including the model uncertainties for the demonstrated event is [ ][a,c].

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APPENDIX D
POLCA-T: APPLICATION FOR ANTICIPATED TRANSIENTS WITHOUT SCRAM

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ABSTRACT

This appendix provides the qualification basis of the POLCA-T code for application to the analysis of anticipated transients without scram (ATWS) in advanced boiling water reactors (ABWR) and in boiling water reactors (BWR/2-6). It is demonstrated that POLCA-T can be used to perform licensing analysis of ATWS events for these reactor types. The information in this appendix is an extension to the qualification studies presented for anticipated operational occurrences (AOOs) in Appendix C of the base topical report (Reference 1). The application methodology for ATWS analysis is presented in Reference 2. Appendix A (in Reference 1) provides the qualification studies for the application to control rod drop accidents (CRDA) and Appendix B (in Reference 1) contains the documentation of the qualification for BWR stability analysis.

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ACRONYMS

Acronyms	Definition
ABWR	Advanced Boiling Water Reactor
ADS	Automatic Depressurization System
AOO	Anticipated Operational Occurrences
APRM	Average Power Range Monitor
ARI	Alternate Rod Insertion
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transients Without Scram
BAF	Bottom of Active Fuel
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
CRDA	Control Rod Drop Accident
ECCS	Emergency Core Cooling System
EOC	End-of-Cycle
FMCRD	Fine Motion Control Rod Drive
HPCF	High Pressure Core Flooder
LWR	Light Water Reactor
MSIVC	Main Steam Line Isolation Valve Closure
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
PCT	Peak Cladding Temperature
PDF	Probability Distribution Function
PIRT	Phenomena Identification and Ranking Table
RCIC	Reactor Core Isolation Cooling
RCPB	Reactor Coolant Pressure Boundary
RPV	Reactor Pressure Vessel
SLCS	Standby Liquid Control System
SRNM	Startup Range Neutron Monitor
SRP	Standard Review Plan, NUREG-0800
SRV	Safety Relief Valve
STUK	Finnish Radiation and Nuclear Safety Authority
TAF	Top of Active Fuel

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D.1 INTRODUCTION

D.1.1 Background

POLCA-T is a computer code developed by Westinghouse for three-dimensional analysis of Boiling Water Reactor (BWR) transients. It is a best-estimate computer code that can be applied for analysis of transients including core stability, control rod drop accidents (CRDA), anticipated operational occurrences (AOOs), and anticipated transients without scram (ATWS). It has been extensively used for licensing applications of BWR's in Europe. POLCA-T incorporates a two-fluid approach for thermal-hydraulics and a three-dimensional nodal description of neutron kinetics. The energy and mass balance equations are solved for both the liquid and vapor/gas phases that may be in thermal non-equilibrium. The three-dimensional kinetics model is consistent with the NRC approved core simulator POLCA7 (Reference 3). The thermal-hydraulics simulation is based on a five-equation drift-flux model and one-dimensional fluid-dynamics approach. Multiple parallel channels are employed for a whole-core representation. The physical models, correlations and the numerical solution techniques are described in the main topical report (Reference 1). POLCA-T has been qualified against separate effects tests, integral effects tests, component performance data, and operating BWR plant data. Detailed documentation of the qualifications studies are reported in Appendices A-D of the base topical (Reference 1) corresponding to the different analysis applications.

Appendix A includes the methodology and qualification description for the application of POLCA-T to CRDA. Appendix B includes the methodology and qualification description for the application of POLCA-T for stability evaluation. Appendix C covers the qualification work for the application to the analysis of AOOs. Appendix D describes the code qualification of POLCA-T for analyzing ATWS events. POLCA-T has already been approved by the NRC for CRDA and BWR stability analysis. Appendix C is submitted for NRC review concurrently with the submission of Appendix D. The qualification work in Appendix D is an extension of the qualification studies presented in Appendix C. The associated methodology (Reference 2) for the analysis of fast transients and ATWS events is currently under review by the NRC. The scope of the qualification is limited to cover analyses of ABWRs and BWR/2-6.

D.1.2 Summary

This report demonstrates the capability of POLCA-T to the analysis of ATWS events. Based on information contained in this report, it is concluded that POLCA-T:

1. Is an acceptable tool for analyzing ABWRs and BWR/2-6 with respect to the ATWS acceptance criteria.

2. Provides analysis results that are acceptable for referencing in licensing applications for ABWR and BWR/2-6 power plants.

D.2 REQUIREMENTS AND SCOPE

D.2.1 Analysis Scope

The purpose of this Appendix is to demonstrate the acceptable use of POLCA-T for its application to the analysis of ATWS events in ABWRs and BWR/2-6s. This appendix provides the code qualification basis for the evaluation against the ATWS licensing acceptance criteria, including calculation of:

- Fuel integrity
- Peak Reactor Coolant Pressure Boundary (RCPB) system pressure
- Containment integrity (mass and energy release to the containment)
- Long-Term Shutdown Cooling

D.2.2 Nuclear Power Plant Specification

Plant designs considered are the ABWR and BWR/2-6 plants.

D.2.3 Transient Scenario Specification

The transient scenarios considered are those events associated with ATWS.

D.2.4 Anticipated Transients Without Scram

An ATWS is an AOO followed by the failure of the reactor trip portion of the protection system, as specified in General Design Criterion 20 of Appendix A of 10CFR50. The probability of an AOO, in coincidence with a complete failure of the reactor protection system, is much lower than the probability of any of the other events that are evaluated under NUREG-0800, the Standard Review Plan (SRP) in Chapter 15. As such, ATWS events are classified as beyond design basis accidents, and consequently, ATWS events are treated separately in Chapter 15.8 of the SRP.

ATWS events are mitigated by the following manual and automatic reactor shutdown scenarios:

a) Reactor shutdown by Alternate Rod Insertion (ARI), as required by 10CFR50.62. This scenario is intended to show the effectiveness of the ARI design.

b) Reactor shutdown by Fine-Motion Control Rod Drive (FMCRD) run-in*. This scenario assumes a total failure of ARI and is analyzed to show the backup capability of FMCRD run-in.

c) Reactor shutdown by manual or automatic activation of Standby Liquid Control System (SLCS). In this scenario, both ARI and FMCRD run-in are assumed to fail. Boron injection is relied upon to mitigate the transient. This case is analyzed to show the in-depth ATWS mitigation capability.

D.2.5 ATWS Acceptance Criteria

The following acceptance criteria were presented in Reference 2 to demonstrate compliance with 10CFR50.62:

1. Fuel Integrity - The long-term core cooling capability is assured by meeting the cladding temperature and oxidation criteria of 10CFR50.46 (i.e., peak cladding temperature not exceeding 1204°C (2200°F), and the local oxidation of the cladding not exceeding 17 % of the total cladding thickness).

2. Primary System - The reactor pressure vessel integrity is assured by limiting the maximum primary stress within the RCPB to the emergency limits as defined in the ASME Code, Section III.

3. Containment Integrity - The long-term containment capability is assured by limiting the maximum containment pressure to the design pressure of the containment structure and the suppression pool temperature to the wet well design temperature.

4. Long-Term Shutdown Cooling - Subsequent to an ATWS event, the reactor shall be brought to a safe shutdown condition, and be cooled down and maintained in a cold shutdown condition.

D.2.6 Figures-of-Merit

The figures-of-merit for ATWS analysis were established in Chapter 3.2 of the Westinghouse analysis methodology (Reference 2). They are presented in Table D.2-1. The figures-of-merit are derived from the regulatory requirements corresponding to the acceptance criteria in D.2.5.

* If this feature is included in the design

Table D.2-1 Figures-of-Merit for ATWS	
ATWS	Cladding temperature Reactor vessel pressure (RVP) Mass and energy release to containment

D.2.7 Phenomena Identification and Ranking

A Phenomena Identification and Ranking Table (PIRT) for AOO and ATWS analysis was established in Chapter 5 of the analysis methodology report (Reference 2). A total of [ ]a,c high ranked phenomena were identified for ATWS evaluation. Specific ATWS phenomena not considered for AOOs are identified in Table D.2-2. [ ]a,c Individual phenomena are distinguished by their name and a unique identifier (ID) shown in parenthesis. Phenomena names and (IDs) originate from the methodology report (Reference 2). The following high ranked phenomena have been identified for ATWS analysis:

a,c

[ ]

a,c

Table D.2-2 Specific ATWS Phenomena/Components

a,c

D.3 EVALUATION MODEL

This Chapter includes a description of models specifically employed for ATWS analysis which have not been previously described in WCAP-16747-P-A (Reference 1).

D.3.1 Boron Transport Model

In the absence of control rod insertion, reactivity control is established using the SLCS. The SLCS injects a sodium pentaborate solution into the reactor vessel. For the neutron absorbing poison to become effective in controlling the reactivity it must reach the reactor core. POLCA-T includes a model to calculate the boron distribution in the reactor pressure vessel (RPV). The boron transport model is described as part of the Hydrodynamic model in Section 7.1.1 of the main document (Reference 1). Justification for its use in ATWS analysis is discussed in Section D.4.6. Boron stratification is handled by the [ ]^{a,c} model described in Section D.3.2.

D.3.2 Boron Settling Model

At very low flow rates (e.g. natural circulation), boron stratification may become important. For those conditions an empirical boron mixing efficiency function is introduced for the boron mass flow rate as follows:

[

]^{a,c}

The mixing efficiency function applied in POLCA-T is shown qualitatively in Figure D.3-1. The boron mixing efficiency is the fraction of boron present in the lower plenum that is mixed with the coolant flow and thus, depending on the flow direction, transported to the neighboring cell. The mixing efficiency is controlled by two input parameters, UBOR_LIMIT and UBOR_LIMIT_LOW. UBOR_LIMIT provides the upper flow velocity limit. For flow velocities above UBOR_LIMIT [ ]^{a,c} Below this velocity [ ]^{a,c} UBOR_LIMIT_LOW is the lower flow velocity limit. Below this limit [ ]^{a,c}

[

]a,c

a,c

Figure D.3-1 Mixing Efficiency Function in POLCA-T (shown qualitatively)

D.3.3 Reactivity Model of Bypass Channel Coolant Density

The coolant flow inside a BWR fuel assembly is divided into an active coolant region (coolant in contact with the fuel rods) and assembly bypasses [

]a,c. In addition, water between the fuel assemblies and water outside the core periphery but inside the core shroud is considered an inter-assembly bypass.

Nuclear cross-section data are typically tabulated as function of the active coolant density (among others) while keeping bypass coolant density constant (saturated conditions). The reduction of the reactor pressure vessel water level is normally part of the emergency procedures of a BWR plant during an ATWS and under these conditions some boiling may occur in the bypass channels. [

]a,c

D.3.3.1 Effective coolant density model

[ ]a,c

[a,c]

[

]a,c

D.3.3.2 Heterogeneous coolant density model

In order to account for heterogeneous distributions of coolant density as well as soluble boron concentration within the nodal domain of each fuel assembly, [
]a,c in the computed nodal cross-sections according to:

a,c

[ ]

[

]a,c

D.3.4 Post Dryout and Rewet Model

The post dryout and rewet models are used to determine the Peak Cladding Temperature (PCT). Evaluation of PCT in ATWS is done using the same cladding temperature model described in Supplement 4 to the BISON Topical (Reference 4). The occurrence of boiling transition during the transient is predicted using approved dryout correlations as investigated in Appendix C. For ATWS analysis, [ ]a,c Validation of the model is provided in section D.4.8.

D.4 MODEL UNCERTAINTIES

In this Chapter individual uncertainties and biases of each high ranked phenomena/component are determined from a combination of comparisons of code predictions against (1) Separate Effects Tests, (2) Integral Effects Tests, (3) Component Performance Tests, (4) Plant Transient Simulation Tests, (5) Code Benchmark (Code) and (6) Analytical Tests. The uncertainty estimation includes all the relevant random and systematic uncertainties (bias).

D.4.1 High ranked phenomena

The PIRT in Reference 2 classifies [ ]a,c phenomena as being important for ATWS evaluation. Of these, model parameters and uncertainties for [ ]a,c phenomena have already been determined in Appendix C for AOO's. [

]a,c These are listed with their PIRT table ID's below:

a,c

[ ]

D.4.2 [ ]a,c

[

]a,c

[

]a,c

D.4.3 [ ]a,c

[

]a,c

a,c

Figure D.4-1 [

]a,c

D.4.4 [ ]a,c

[

]a,c

Table D.4-1 [ ]a,c a,c

D.4.5 [ ]a,c

[

]a,c

D.4.6 [ ]a,c

[

]a,c

Table D.4-2 [ ]a,c

a,c

[ ]

D.4.7 [ ]a,c

[

]a,c

Table D.4-3 [ ]a,c a,c

D.4.8 [ ]a,c

[

]a,c

D.4.8.1 [ ]a,c transient tests

The [ ]a,c, test bundles, testing procedure, measurements and the test results are comprehensively described in Reference 7. Appendix C includes qualification of POLCA-T's transient [ ]a,c calculation against [ ]a,c transient dryout measurements for [ ]a,c. It was demonstrated that POLCA-T predicts [ ]a,c with a high degree of accuracy and [ ]a,c. On the average POLCA-T predicts dryout [ ]a,c. The objective of this section is to extend the discussion presented in Appendix C to validate POLCA-T's [ ]a,c. The transient [ ]a,c presented in Reference 7 are the basis for the comparison.

In the calculations, the [ ]a,c model is activated whenever [ ]a,c The predicted versus measured [ ]a,c increases are shown in Figure D.4-2 for [ ]a,c The bias is [ ]a,c and the corresponding standard deviation is [ ]a,c. The comparison shows that the calculated [ ]a,c is in good agreement with experimental data. Based on these results, it can be concluded that POLCA-T simulates [ ]a,c

a,c

[ ]

Figure D.4-2 [ ]a,c

D.4.8.2 [ ]a,c experiments

POLCA-T's capability to predict [ ]a,c test data has been used [a,c. Previous studies compared measured steady-state [ ]a,c as well as transient [ ] for a pressure increase test. In this section, a comprehensive comparison against [ ]a,c test data is provided comprising both pressure increase and pump trip events. The purpose is to add confidence to POLCA-T's capability to predict [ ]a,c over the full range of ATWS conditions. [ ]a,c

[ ]a,c

[ ]a,c The aim of the work was to increase the knowledge of and improve the basis for predictions of the consequences of certain transient conditions in a BWR. The validation studies reported here include pump trip transients and pressurization transient tests. Such transients occur during a turbine trip (or a load rejection) without bypass or during an all recirculation pump trip tests. Table D.4-4 summarizes the [ ]a,c experiments simulated with POLCA-T along with the initial conditions before the transient. The initial conditions correspond to realistic reactor conditions, scaled to the [ ]a,c fuel bundle.

Geometrical data for the fuel bundle section is provided in Table D.4-5. The fuel bundle consists of [ ]a,c fuel rod simulators. A cross-section of the fuel bundle test section is provided in Figure D.4-3. [ ]a,c

Table D.4-4 [ ]a,c Experiments Simulated with POLCA-T a,c

a,c

Table D.4-5 Fuel Bundle Geometry

a,c

a,c

Figure D.4-3 Cross-section of the Test Section at Heated Part

POLCA-T model

[

]a,c

a,c

Figure D.4-4 POLCA-T Model of the [ ]a,c Test Section

a,c

[ ]

Figure D.4-5 Relative Axial Power Distribution for the Fuel Bundle

POLCA-T simulation

The start time of the transient is defined as t=0. The initial conditions are defined according to the description in Table D.4-4. [

]a,c

a,c

[ ]

Figure D.4-6 [ ][a,c]

a,c

[ ]

Figure D.4-7 [ ][a,c]

a,c

[ ]

Figure D.4-8 [ ][a,c]

Results

In Figure D.4-9, the experimental value of [ ]^{a,c} at each node height is plotted against the calculated [ ]^{a,c}

The main results of interest are [ ]^{a,c}. These results are compiled in Table D.4-6 for [ ]^{a,c}, where the calculated value is also given. [ ]^{a,c}

Conclusion

The [ ]^{a,c} series of tests simulate pressure increase and pump trip transients in BWRs. These tests cover the [ ]^{a,c} and therefore cover the anticipated range of application for ATWS conditions. The results of the validation show that

POLCA-T predicts [ ]a,c conservatively. This leads to conservatively [ ]a,c compared to the measurements. Based on these comparisons, it is concluded that POLCA-T calculates a conservatively [ ]a,c during ATWS and uncertainties in [ ]a,c

a,c

Figure D.4-9 [ ]a,c

Table D.4-6 Summary of Results From [ ]a,c Simulation

a,c

a,c

[ ]

D.4.9 [ ]a,c

[

]a,c

D.4.10 [ ]a,c

[

]a,c

D.4.11 [ ]a,c

[

]a,c

D.4.12 [ ]a,c

[

]a,c

D.4.13 [ ]a,c

[

]a,c

D.4.14 [ ]a,c

[

]a,c

D.4.15 [ ]a,c

[

]a,c

D.4.16 [ ]a,c

[

]a,c

D.4.17 [ ]a,c

[

]a,c

D.5 PLANT DATA QUALIFICATION

This section provides qualification of POLCA-T based on BWR transient plant data. Comparison against plant measurements provides an integrated evaluation of POLCA-T's accuracy, which confirms the code's capability to model the interplay of basic models and correlations. The test cases are chosen to demonstrate the capability of simulating the important thermal-hydraulic and kinetics-related phenomena that govern the response of ATWS events. This qualification, when combined with the model uncertainties based on separate effects, component, and integral effects testing in Section D.4, provides a comprehensive qualification basis for the application of POLCA-T to BWR plant transient evaluations, including both AOO and ATWS events.

The performance of POLCA-T has been qualified against transient and control rod drop tests from test facilities and BWR plants as well as stability tests. Descriptions of these qualification studies are provided in Appendices A, B, and C. BWR plant data for the following test cases have been evaluated:

- Peach Bottom 2 Nuclear Power Plant end-of-cycle turbine trip tests (Appendix C).
- [ ]^{a,c} Nuclear Power Plant all recirculation pump trip test (Appendix C).
- ABWR Hamaoka-5 Nuclear Power Plant start-up tests (Appendix C).
- SPERT power excursion tests (Appendix A).
- Core stability tests conducted at [ ]^{a,c} Nuclear Power Plants (Appendix B).

These tests also cover many of the phenomena that are relevant for ATWS response, although the exact temporal response and absolute values of variations may differ. This chapter includes qualification cases that are specifically aimed to extend the scope of the qualification to support the application of POLCA-T for prediction of plant response during an ATWS event. Taken together, these tests along with the above mentioned qualification studies cover the range of phenomena which are important for ATWS predictions. The following ATWS related plant data comparisons are discussed in this section:

- [ ]^{a,c} Nuclear Power Plant FMCRD insertion event
- [ ]^{a,c} full-scale natural flow tests
- ATWS analysis of Olkiluoto 1 and 2 Nuclear Power Plants
- Qualification for ATWS stability analysis

D.5.1 [ ]^{a,c} Fine Motion Control Rod Drive Insertion Event

POLCA-T validation record includes comparison against a FMCRD insertion event that occurred in the [ ]^{a,c} BWR plant. This event represents a qualified validation test with regards to ATWS

conditions as it involves a partly malfunctioning scram system. The main focus of the comparison is to provide relevant validation on the effects of a slow insertion of control rods.

D.5.1.1 Test conditions

The event occurred on [ ]a,c. Operating conditions prior to the transient are shown in Table D.5-1. The plant was approaching end-of-cycle at coast-down operation. The operation was stable the week before the event. The event was triggered by a false scram signal. This initiated a pump run-back, a turbine trip and an insertion of the control rods through the FMCRD system, but not the hydraulic scram system. At [ ]a,b,c a manual partial scram was initiated by the operator, in which [ ]a,b,c control rods were inserted hydraulically. After [ ]a,b,c all the control rods were inserted and the reactor was shut down. The event has been simulated with POLCA-T. [

]a,c

Table D.5-1 Initial Operating Conditions for [ ]a,c FMCRD Event a,b,c

D.5.1.2 Fission power

The most important and demanding test of the kinetics model is the neutron flux response (APRM). The comparison is shown in Figure D.5-1. There are only small differences between the APRM signals. [ ]a,c has been chosen as 'reference' since [ ]a,c The severity of the flux transient determines the fission power and its prediction requires not only good core pressure calculations, but accurate reactivity feedback models. The agreement between the calculated fission power and the measured APRM is generally very good. [

]a,c

a,b,c

[ ]

Figure D.5-1 Comparison of Reactor Fission Power for [ ]a,c FMCRD Event

D.5.1.3 Recirculation flow

The coolant flow is primarily a function of pump speed even if parameters like the core pressure drop also have an impact. To compare the core flow therefore primarily serves to check the modeling of the recirculation pumps. The comparison is shown in Figure D.5-2. [ ]a,c As shown, the agreement between measured and calculated inlet flow is very good.

a,b,c

[ ]

Figure D.5-2 Comparison of Coolant Inlet Flow for [ ]a,c FMCRD Event

D.5.1.4 Reactor pressure

In Figure D.5-3 a comparison between the measured and calculated stem dome pressure is shown. The agreement between the measured and calculated steam dome pressure is generally very good. [ ]a,c

a,b,c

[ ]

Figure D.5-3 Comparison of Reactor Pressure for [ ]^{a,c} FMCRD Event

D.5.1.5 Generated energy

Figure D.5-4 contains a comparison of the "measured" and calculated energy release during the transient. The energy is not measured directly but obtained by time integrating the measured fission power response (relative neutron flux multiplied with initial fission power) retrieved from data files logged during the event. This is then compared with the time integrated fission power from the calculation. Calculation of the amount of fission energy released is an important objective of ATWS analysis as it has great impact on the suppression pool heat-up. In addition, [ ]^{a,c} It is seen that POLCA-T predicts the energy release very well [ ]^{a,c}

a,b,c

[ ]

Figure D.5-4 Comparison of Generated Fission Energy for [ ]a,c FMCRD Event

D.5.1.6 Conclusion

The comparison shows that POLCA-T is able to reproduce APRM, coolant flow, steam dome pressure, and integrated energy with good precision. For the reactor power level, peak pressure and energy release POLCA-T [

]a,c The results provide strong support for the use of POLCA-T to predict [

]a,c which are key variables for the simulation of an ATWS.

D.5.2 [ ]a,c full-scale natural flow tests

The purpose of this validation is to evaluate the adequacy of POLCA-T for the prediction of natural circulation flow by comparison to experimental data. Test simulations have been carried out on natural circulation flow in [

]a,c The purpose is to verify the coupled thermal-hydraulic models, so that POLCA-T can be confidently used for ATWS simulations under pump trip conditions.

D.5.2.1 Test facility

A test rig of the [ ]a,c. A layout of the test rig is shown in Figure D.5-5. [ ]a,c Results of the tests provide validation data for simulation codes.

A total of [ ]a,c tests were carried out at the facility. The tests were conducted in [ ]a,c. Tests [ ]a,c were aimed at testing of the [ ]a,c. Tests [ ]a,c were made with [ ]a,c. Test [ ]a,c has been used for the POLCA-T qualification study described herein.

a,c

[ ]

Figure D.5-5 Layout of [ ]a,c Test Rig

D.5.2.2 POLCA-T model

The test rig is modeled using regular POLCA-T volume cells in conjunction with fluid paths (junctions). [ ]a,c
POLCA-T nodalization is shown in Figure D.5-6. [

]a,c

a,c

Figure D.5-6 POLCA-T Nodalization of [ ]a,c Test Rig

The model is fed with boundary conditions and initial conditions from [ ]a,c as input conditions. The boundary conditions were:

a,c

D.5.2.3 Test conditions

[

]a,c

[

]a,c

Table D.5-2 Heating Powers and Heat-fluxes in [ ]a,c

a,c

[

]

D.5.2.4 Test results and comparison with POLCA-T

[

]a,c The flow increases when the heated water flows up through the hot leg, because the available pressure head increase due to the added heat and hot water column length increase as the hot water flows up through the riser. When the suppression pool starts to heat up the flow rate decreases due to smaller density difference and reaches its lowest flow rate when the pool temperature is stabilized due to equilibrium in heat balance of the suppression pool.

[

]a,c POLCA-T simulates the transient natural circulation flow very well during startup period and during the coast down period when [ ]a,c natural circulation is established. The calculation of [ ]a,c natural circulation flow shows the same trend compared to experimental result. POLCA-T correctly predicts [

]a,c .

[

]a,c

a,c

Figure D.5-7 Comparison of the Mass Flow Rate Through the Test Rig

[

]a,c

a,c

Figure D.5-8 Comparison of the Inlet Temperature of the Test Rig

[

]a,c

a,c

Figure D.5-9 Comparison of the Outlet Temperature of the Test Rig

D.5.2.5 Conclusions

As part of POLCA-T qualification, natural circulation in the [ ]^{a,c} test rig was simulated using POLCA-T. The test data and POLCA-T calculations were found to be in good agreement. [ ]^{a,c} and the [ ]^{a,c} natural circulation flow rate through the loop is predicted accurately in the test. Inlet and outlet temperatures are also very well simulated by POLCA-T. [

]^{a,c} As such, the test provide further support for the conclusion that the hydraulic model and coupled heat transfer models in POLCA-T are adequate for the prediction of ATWS response when forced circulation is lost.

D.5.3 Qualification for ATWS analysis involving automatic depressurization

Westinghouse has applied POLCA-T to licensing evaluation of ATWS events for various Nordic BWR plants, including Olkiluoto 1 and 2, [ ]^{a,c}. A common feature in these plants is the actuation of the Automatic Depressurization System (ADS) during some ATWS scenarios that leads to very low RPV water levels. Although these evaluations include no comparison with measurements for the events considered, they demonstrate that POLCA-T can be used to analyze complex ATWS safety analyses.

This section includes a demonstration example of the ATWS analyses performed for Olkiluoto Nuclear Power Plant Units 1 and 2. The example is aimed to demonstrate the code capability to simulate the plant response during accidents scenarios that involve ADS activation as a part of the mitigating safety features. The analyses were carried out to demonstrate compliance with the requirements of the Finnish Radiation and Nuclear Safety Authority (STUK).

The analyzed initiating event is a loss-of-feedwater transient. It is assumed that both the hydraulic insertion and the fine motion control rod insertion malfunction. Reactor shutdown is achieved by SLCS injection of liquid boron solution. ADS is activated on low water level by the plant protection systems. The ADS employs pressure relief valves for steam discharge to the pressure suppression pool and the controlled reduction of primary system pressure. In compliance with Finnish regulations, the ATWS event is analyzed against fuel and cladding integrity criteria as well as primary system pressure acceptance criteria. The analyses have been presented in Reference 8. The simulation covers all critical phases of the event, including boron injection phase, trip of all recirculation pumps, depressurization down to near atmospheric pressure levels, core flooding by the low-pressure coolant system, void collapse, cool-down, boron dilution, re-criticality, and boron re-entrainment leading up to stable shutdown conditions. Figure D.5-10, Figure D.5-11, and Figure D.5-12 illustrate the APRM level, the dome pressure, and the reactivity/boron concentration, respectively, associated with the loss-of-feedwater ATWS event in Olkiluoto 2.

In conjunction with the ATWS qualification studies for ABWR and BWR/2-6, the analyses conducted at the Nordic BWRs support the application of POLCA-T for evaluation of ATWS events in BWR plants

involving ADS actuation. Westinghouse concludes that realistic calculations with POLCA-T can be used to support licensing evaluations of such plants.

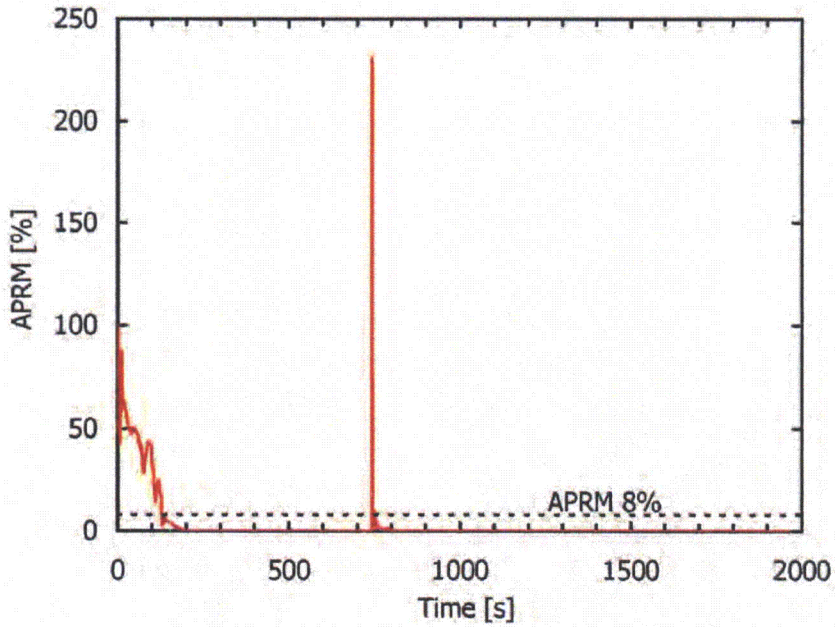

Figure D.5-10 APRM Levels for Olkiluoto 2 Loss-of-Feedwater ATWS Event with SLCS Initiation *

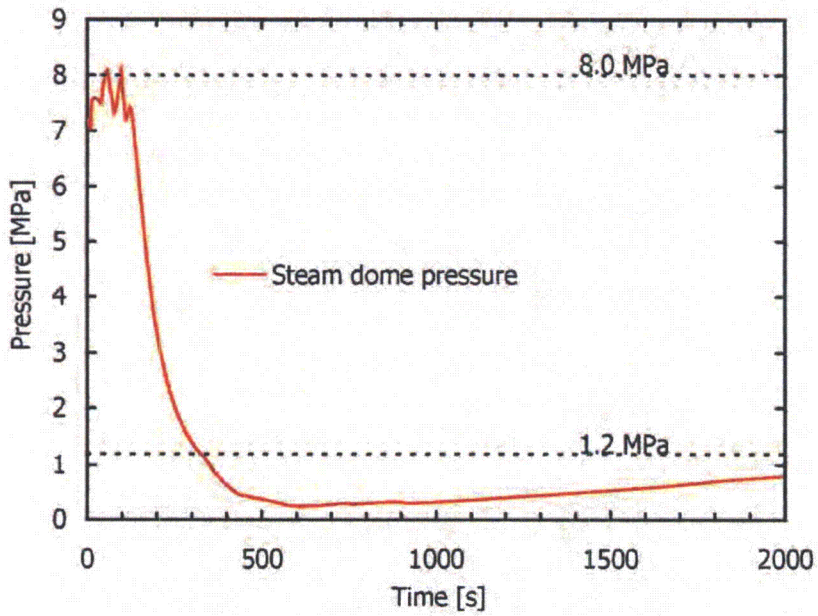

Figure D.5-11 Dome Pressure for Olkiluoto 2 Loss-of-Feedwater ATWS Event with SLCS Initiation *

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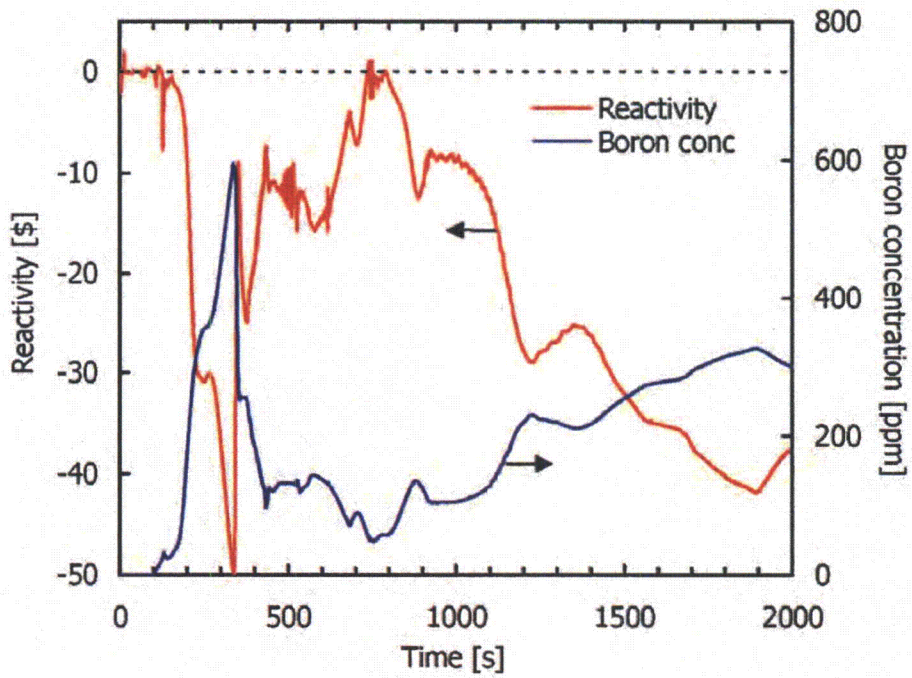

Figure D.5-12 Reactivity and Boron Concentration for Olkiluoto 2 Loss-of-Feedwater ATWS Event with SLCS Initiation *

D.5.4 Qualification for ATWS stability analysis

On the basis of the following qualification studies, Westinghouse finds that POLCA-T is capable of performing ATWS instability analysis:

- Previous qualification of POLCA-T for BWR stability analysis, presented in Appendix B. POLCA-T has been extensively validated for stability purposes by comparing POLCA-T predictions against results from stability tests and instability events. These tests cover the relevant thermal-hydraulic conditions (operating conditions) where ATWS instability is important.

- POLCA-T comparison to [ ]^{a,c} tests for [ ]^{a,c} These tests are presented in Appendix C. POLCA-T simulations show good agreement between the measured and calculated [ ]^{a,c} occurrences. This work provides excellent experimental validation for ATWS stability analyses in which [ ]^{a,c} conditions are expected.

- POLCA-T comparison to SPERT-III-E power excursion test. These tests were part of the validation base of POLCA-T for CRDA analyses discussed in Section A.3.2.2 of Appendix A. The analyses demonstrate the capability of POLCA-T to accurately simulate fast reactivity pulses similar to those observed in ATWS power oscillations. The resulting values of the inserted reactivity, power shape, integrated energy, and time-to-peak power are well represented by POLCA-T.

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- POLCA-T comparison against [ ]^{a,c} FMCRD event. This validation is discussed in Section D.5.1. In this case, the recirculation pumps are run-back to [ ]^{a,c} of initial pump speed and the fine motion control rod run-in is activated. Power oscillations can be observed [ ]^{a,c}. Good agreement between POLCA-T and measurement for the average power level is observed. The amplitude and frequency of the power oscillations are well represented. In addition, POLCA-T has been validated, with good accuracy, to other reactivity transients where the reactivity insertion is caused by changes in the thermal hydraulics, e.g. Peach Bottom 2 turbine trip transients presented in Appendix C.

D.6 APPLICATION UNCERTAINTIES

Reference 2 provides the basis for the analysis methodology for ATWS analysis. This section is limited to a discussion of how application input parameters are treated with respect to quantifying the uncertainty in the calculation of the figures-of-merit. As such, it provides the basis for estimating the uncertainties associated with inaccuracies in the basic models. Section D.6.1 and Section D.6.2 provide the basis for ATWS initial conditions and plant parameters. D.6.3 contains a summary of the model input parameters used to quantify the sensitivity of the figures-of-merit to individual model uncertainties.

D.6.1 Initial Conditions

Initial conditions are key plant inputs that define the steady-state operating conditions prior to the transient. Initial conditions may vary due to the allowable operating range and/or to uncertainty in the measurements at a given operating condition. For ATWS the uncertainty analysis will be initiated from the limiting point in the allowable operating domain. Due to the extremely low probability of occurrence for ATWS, initial conditions will not be adjusted to account for measurement or instrumentation uncertainties. The methodology for choosing the initial conditions for ATWS events is described in Reference 2.

D.6.2 Plant Parameters

This section defines the plant input parameters used for the model uncertainty analysis. A plant parameter is a plant-specific quantity such as a protection system setpoint, SRV capacity, coolant systems capacity and temperature, MSIV stroke time, pump setpoints, etc. Due to the low probability of occurrence for ATWS, nominal plant input parameters are used. Conservative plant input values (e.g. plant technical specification) may be used if a nominal plant parameter is difficult to define. The absolute values of these inputs will vary from plant to plant.

D.6.3 Model parameters and uncertainties

Model inputs vary due to uncertainty in the phenomenological models. The assessment of the accuracy of incorporated models, along with the bias and uncertainties involved was established in Section D.4. Uncertainties in the high ranked phenomena are accounted for through variation in special "code input parameters". These are special inputs that allow coefficients in POLCA-T models or correlations to be varied for the purpose of the uncertainty analysis. The parameter ranges are selected to cover the uncertainties in the models. Once established, [ ]^{a,c}. The uncertainty values presented in Table D.6-1 [ ]^{a,c}. They provide the bases for determining the overall uncertainty on the appropriate figures-of-merit in the application of POLCA-T to ATWS analyses. [ ]^{a,c}

Table D.6-1 summarizes model parameters and their associated uncertainties for all high ranked phenomena for ATWS analysis. [

]a,c

Table D.6-1 Model Parameters and Uncertainties for All High Ranked ATWS Phenomena a,c

a,c

a,c

a,c

[ ]

D.7 DEMONSTRATION ANALYSIS

With the background information provided in Section D.6, the accuracy of the POLCA-T code can be demonstrated. The analyses provided in this section are used to demonstrate the procedure for determining the overall model uncertainty for ATWS application. The procedure is consistent with process described in the methodology report (Reference 2). The calculations performed in this section are for demonstration purposes only. Uncertainty analysis of this kind will be [

]a,c.

Selection of the limiting ATWS event is conducted in section D.7.1. A nominal case analysis (with nominal POLCA-T model input parameters) is performed in Section D.7.2 for the limiting ATWS event. The sensitivity of the figures-of-merit to the individual model uncertainties is determined in section D.7.3. In this case, each model parameter is varied randomly as described in Reference 2. The results of the uncertainty analysis are deviations in the figures-of-merit from the nominal case. The deviations represent the range of uncertainty in the calculated figures-of-merit from considering uncertainties in the phenomenological models.

D.7.1 Selection of event

The POLCA-T modeling uncertainty analysis is performed on one limiting ATWS event. Generic studies have shown that the Main Steam Isolation Valve Closure (MSIVC) event with failure of rod insertion results in high cladding temperature, RCPB pressure, and suppression pool temperature. The maximum values from this event are, in most cases, bounding of all ATWS events considered. The MSIVC event is the only event considered in the demonstration analysis.

D.7.2 Nominal case analysis

This section includes the demonstration of a nominal calculation of the limiting ATWS event. The demonstration analysis is performed on a representative ABWR plant. The initiating event corresponds to MSIVC with failure of rod insertion (SLCS case). Nominal initial conditions and plant parameters are selected and using nominal POLCA-T model input parameters.

The representative ABWR model is an internal recirculation pumps BWR with a rated power of [ ]a,c. The reactor core has [ ]a,c. It is assumed that the core is an [ ]a,c. The nominal specific power density is [ ]a,c. The ATWS demonstration analysis is performed at [ ]a,c Data used for plant initialization prior to the event are summarized in Table D.7-1.

Table D.7-1 ABWR Initial Operating Conditions for ATWS Demonstration Analysis a,c

The transient is initiated by the inadvertent closure of all main steam isolation valves. All scram signal paths are assumed to fail. The RPV isolation causes a rapid increase in reactor vessel pressure, which results in core void reduction and associated power increase. The ATWS high pressure setpoint is reached which initiates trip of four of ten recirculation pumps and the remaining six pumps start to runback. The ATWS high pressure signal actuates the ARI and the FMCRD, both are assumed to fail. The pressure increase is limited by the reduction in core flow and the opening of safety/relief valves. The reduction in feedwater temperature by the loss of steam flow to the feedwater pre-heaters pushes the reactor into the unstable operating region. At [ ]a,c seconds the power starts to oscillate with a frequency of [ ]a,c Hz. After two minutes, automatic feedwater runback is initiated. This action stops the cold water injection, and the core inlet subcooling is subsequently reduced and the power oscillations start to recede. Makeup water from ECCS is activated when water level drops below Level 2 and Level 1.5, which prevent core uncover. At water Level 2 the remaining six recirculation pumps trip and natural circulation mode is established. Because of low reactor water level the natural circulation driving density head for the flow path including downcomer, recirculation pumps, and core inlet drops to near zero. An internal circulation flow path is created by the fluid density differences inside versus outside the active channels. The core flow distributes itself in such a way that the coolant flows down into the bypass and up through the active fuel channels.

High RPV pressure together with startup range neutron monitors (SRNM) ATWS permissive for [ ]a,c minutes activates the SLCS. The boron reaches the core inlet approximately [ ]a,c seconds later. It takes [ ]a,c seconds for the pumps to start and for the boron solution to fill the pipe space and [ ]a,c seconds to reach the core inlet from the injection spargers. The SLCS connects to the high pressure core flooders (HPCF). The HPCF spargers inject to the upper plenum and the boron spreads into the bypass in the direction of the flow and then to the active fuel channels. As the boron reaches the core, the power level drops due to the negative reactivity insertion and reaches decay heat levels at about [ ]a,c seconds. When the water level increases above Level 8 the HPCF and RCIC are shutoff. Due to low flow rates in the downcomer there is little carry-over of boron into this region. The first major opportunity for borated water to reach the downcomer is when the water level reaches the upper end of the separator where the separated water exits (to join the downcomer annulus flow). At around [ ]a,c seconds, there is a temporary increase in the coolant flow, when the relatively cold (borated) water is transferred from the separator to the pool that surrounds the standpipes (i.e. bulkwater). Steam condensation causes pressure reduction and void increase in the downcomer and lower plenum and this pushes a slug of unborated water from the downcomer into the core region. This leads to a rise in reactivity and return-to-power. The power increase is mitigated by negative feedbacks from void generation. The fuel bundles are well cooled

during this phase. At around [ ]a,c seconds the reactor is shutdown and the boron concentration steadily builds up.

The simulation is run in the transient mode for [ ]a,c seconds. A selection of figures illustrating the transient is given in Figure D.7-1 to Figure D.7-6. Key results are summarized in Table D.7-2.

a,c

Figure D.7-1 Reactor Power Level – ABWR ATWS MSIVC SLCS Case

a,c

Figure D.7-2 Reactor Pressures – ABWR ATWS MSIVC SLCS Case

a,c

Figure D.7-3 Core Flow ABWR ATWS MSIVC SLCS Case

a,c

Figure D.7-4 Boron Concentration – ABWR ATWS MSIVC SLCS Case

a,c

Figure D.7-5 Energy Release to Containment – ABWR ATWS MSIVC SLCS Case

a,c

Figure D.7-6 Peak Cladding Temperature – ABWR ATWS MSIVC SLCS Case

Table D.7-2 Key Results From ATWS Analysis of Nominal ABWR MSIVC SLCS Case

a,c

D.7.3 Analysis of model uncertainties

Statistical uncertainty analyses (Monte Carlo method) are performed for the MSIVC SLCS event. The relevant model parameters listed in Table D.6-1 are randomly varied within their range of uncertainties, all of them at a time for each simulation. The analyses include 59 trials. An overall uncertainty for the figure-of-merits is obtained. The procedure is in accordance with Reference 2 and will not be repeated here. An example demonstration is included in Appendix C for AOO model uncertainties. Similar procedure is applied for ATWS.

D.8 CONCLUSIONS

The POLCA-T code has been evaluated for the purpose of determining its adequacy for analyzing ATWS events in ABWRs and BWR/2-6. It is concluded that:

- POLCA-T is capable of simulating BWR ATWS events. It adequately predicts the ATWS phenomena of importance, and the physical models and correlations applied for simulating the important phenomena are qualified.
- The nominal POLCA-T calculation combined with deviations from the model uncertainty analyses provide conservative estimates of the PCT, peak RCPB pressure, and mass- and energy release to the containment.

D.9 REFERENCES

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