TRACG ANALYSIS AND RESULTS OF PHASE 1 OF THE OECD/NEA OSKARSHAMN-2 BWR STABILITY BENCHMARK

H. Cowen, S. Pfeffer, J. Vedovi, L. Ibarra, D. Vreeland, D. Miranda
GE Hitachi Nuclear Energy
3901 Castle Hayne Road, Wilmington, NC, 28402, USA
Email: haley.cowen@ge.com

ABSTRACT

On February 25, 1999, the Oskarshamn-2 (OKG2) Boiling Water Reactor (BWR) experienced an instability event that culminated in an unstable high power to flow operating region with diverging power oscillations having a decay ratio (or growth rate) of approximately 1.4. An international benchmark was organized for which the plant data collected from the OKG2 instability event was made available to participants. The best-estimate system code TRACG, the GE Hitachi Nuclear Energy proprietary version of the Transient Reactor Analysis Code (TRAC), is used to model the OKG2 reactor vessel, fuel channels and piping of the plant as well as the thermal-hydraulic conditions of the reactor prior to and during the instability event. TRACG includes a multi-dimensional, two-fluid representation of two-phase flow for the reactor thermal-hydraulics and has the capability to model three-dimensional reactor kinetics. The operating plant data is used to validate the steady state OKG2 TRACG system model. This paper presents the TRACG results for Phase 1 of the benchmark after the advancement of the core model and the 3D neutron kinetics and thermal-hydraulics solution method with respect to the preliminary analysis and results presented in a previous work[1]. This paper demonstrates how the TRACG code predicts very well the most significant aspects of the transient event. The paper incudes results for sensitivity studies performed to assess the uncertainty on the provided feedwater temperature variation.

KEYWORDS
TRACG, Thermal-hydraulics, BWR stability, Oskarshamn-2

1. INTRODUCTION AND PURPOSE

On February 25, 1999, the Oskarshamn-2 (OKG2) Boiling Water Reactor experienced an instability event. The loss of feedwater pre-heaters and a failure of the control system logic resulted in a plant operating condition with high feedwater flow and low feedwater system temperature without a reactor scram[2]. Following the onset of the event, interaction of the automatic power and flow control system caused the plant to move into a high power to flow ratio domain. The combination of the aforementioned events brought the reactor to operate in a region of instability with diverging power oscillations having a decay ratio (or growth rate) of approximately 1.4, which triggered an automatic scram at high power[2].

An International benchmark was organized by the Organization for Economic Co-operation and Development (OECD) and Nuclear Energy Agency (NEA) where the plant data collected from the OKG2 instability event was made available to participants. The goal for this OECD/NEA stability benchmark is to test Thermal-Hydraulics (T-H) and Three-Dimensional (3D) neutron kinetics codes on more challenging transient situations including diverging and unstable BWR power oscillations, with and without reactor scram. This OKG2 feedwater instability event is appropriate for code benchmarking since the reactivity insertion caused by the increasing moderator density, brought the reactor to operate in an unstable high power to flow region with power oscillations having a decay ratio greater than one.

An analysis of this feedwater stability event was performed using the best-estimate system code TRACG, the GE Hitachi Nuclear Energy proprietary version of the Transient Reactor Analysis Code

(TRAC). TRACG includes a multi-dimensional, two-fluid representation of two-phase flow for the reactor thermal-hydraulics and has the capability to model 3D reactor kinetics. As noted in the TRACG qualification License Topical Report (LTR), some of the key Nuclear Power Plant (NPP) instability events/tests the TRACG code has been qualified against are: the LaSalle-2 instability event (March 1988); the Leibstadt Cycle 1 regional instability tests; the Nine Mile Point 2 instability event (July 2003); and, the Peach Bottom Unit 2 Cycle 2 stability tests (April 1977)[3]. The February 25, 1999 stability event was modeled using the thermal-hydraulic and 3D neutron kinetics system code TRACG to further confirm the code's capability to simulate complex stability events and expand the available envelope of plant data benchmarks to include a stability event for a Nordic BWR plant. This event includes unstable power oscillations with decay ratios greater than one, such that modeling non-linear effects becomes pertinent. TRACG was used to model the OKG2 plant as well as the thermal-hydraulic conditions of the reactor prior to and during the instability event. The operating plant data given to benchmark participants was used to validate the steady state OKG2 TRACG system model.

2. EVENT DESCRIPTION

The Oskarshamn Unit 2 ABB ATOM Nuclear Power Plant experienced an instability event on February 25, 1999. The event was initiated when the power supply to a bus bar was unexpectedly interrupted for 150 milliseconds following the performance of a maintenance work task in the switchyard. The control logic for the main breaker connecting the OKG2 unit to the main grid interpreted this signal interruption as a load rejection signal and transmitted this to the turbine, causing a turbine trip.

Due to a failure in the relay circuit, the load reject signal was not received by the reactor, thus the expected automatic control such as automatic insertion of control rods and recirculation pump trips never occurred[2]. Following the turbine trip and opening of the steam line bypass valves, the feedwater preheater system was no longer functional. Without a feedwater preheating stage, feedwater temperature decreased by 75°C over a period of 150 seconds while steam flow was bypassed to the condenser. The loss of feedwater pre-heaters and a failure of the control system logic resulted in a plant operating condition with high feedwater flow and low feedwater system temperature without a reactor scram[2]. Following the onset of the event, interaction of the automatic power and flow control system (selected rod insertion) caused the plant to move into a high power to flow ratio regime. The combination of the aforementioned events brought the reactor to operate in a region of instability with diverging power oscillations, which triggered an automatic scram at high power. The key milestones and stages of the February 25, 1999 stability event timeline are summarized in Table I.

Table I. Progression of the OKG2 Stability Event.

Stage	Simulation Time (s)	Description	Power/Flow Conditions
I	0.0	Steady State	106.1%/71.2%
II	72.9	Feedwater Preheater Trip	106.1%/71.2%
III	119.9	Pump Speed Reduction	109.1%/71.2%
IV	150.0	Pump Speed Reduction	111.5%/65.3%
V	178.6	Pump Speed Reduction	111.8%/58.5%
VI	190.5	Pump Speed Reduction	104.2%/50.7%
VII	195.9	Partial Scram Control Rod Insertion	94.8%/46.8%
VIII	252.6	Reactor Scram on High Power	137.5%/34.1%

The power/flow evolution for the OKG2 stability event is shown in Figure 1.

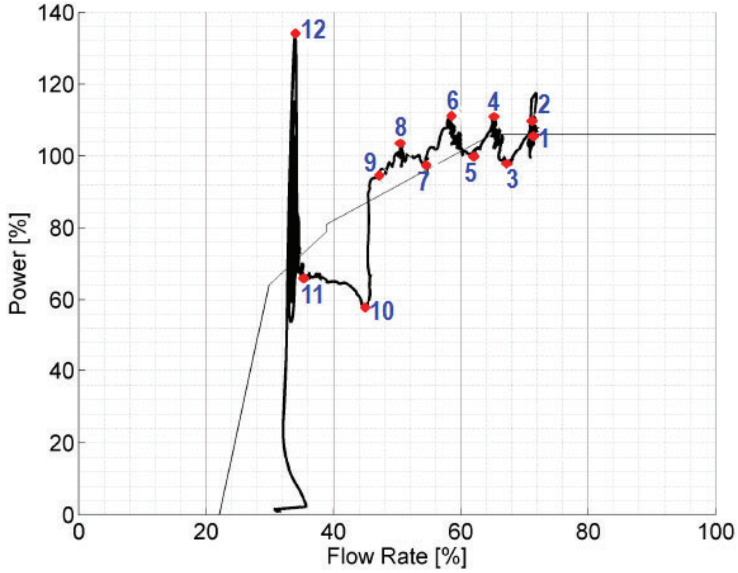

Figure 1. OKG2 Feedwater Transient Power-Flow Map[2].

The finer, gray line shown on the power-to-flow map above represents the licensed domain boundary for reactor operation. State point 1 in Fig. 1 depicts the time at which the load rejection signal was transmitted to the turbine. Because of turbine trip and associated opening of the steam line bypass valves, the feedwater preheater system tripped (Stage II in Table I) and was no longer operable and thus, the feedwater temperature decreased. As feedwater temperature decreased, inlet subcooling increased causing reactor power to increase. The initial increase in core power due to the positive reactivity feedback from the decrease in feedwater temperature is depicted in the movement from state point 1 to state point 2. A pump controller, which controlled the rotation of the recirculation pumps, reduced the recirculation speed when reactor power increased to more than 108% rated power (Stage III). Recirculation pump speed was reduced at 119.9 seconds to decrease core flow and reactor power, as shown in the movement from state point 2 to state point 3 in Fig. 1.

However, the inflow of cold feedwater continued, causing reactor power to again increase above 108% (as shown from state point 3 to state point 4 in Fig. 1). This increase in reactor power activated the pump controller a second time at state point 4 at approximately 150 seconds following the start of the event (Stage IV in Table I). This recirculation pump controller activation sequence was repeated a third time at approximately 178.6 seconds (Stage V) at state point 6 in Fig. 1. Power and flow were shown to decrease to state point 7, but then reactor power is shown to increase again (to state point 8) as cold feedwater enters the vessel adding positive reactivity feedback to the core.

In order to control plant operating conditions and bring reactor operation away from the unstable high power to flow region on the power to flow map, recirculation pump speed was again reduced to decrease power as shown at point 8 (Stage VI in Table I) and subsequentially, a Selected Rod Insertion (SRI) was performed. Plant operators reduced reactor power by fully inserting 7 predefined control rods as shown at point 9 in Fig.1 (Stage VII). With the SRI, the power was reduced to 65% and flow was reduced to 3200 kg/sec (from state point 10 to state point 11).

Following the SRI and reduction in core flow, cold feedwater continued to enter the reactor vessel. This caused reactor power to again increase from point 10 to point 11 in Fig. 1 and the plant to enter the unstable region of the power to flow map. As shown in the movement from point 11 to point 12 in Fig. 1, reactor power started to oscillate with successive increasing amplitudes over a period of 20 seconds and continued oscillating with a decay ratio greater than one, until the reactor scrammed (Stage VIII in Table I) due to high power when the power exceeded 132% at 2500 kg/s recirculation flow (point 12 in Fig. 1).

3. TRACG MODEL DESCRIPTION

The Oskarshamn-2 Nuclear Power Plant is a 1700 MWth BWR unit designed by ABB ATOM and is located in Oskarshamn, Sweden. Oskarshamn unit 2 was originally licensed to operate at a rated power of 1700 MWth but was uprated in the early 1980's to 106% of its original power to a rated power of approximately 1802 MWth[2]. The power in the OKG2 NPP is generated in 444 Fuel Assemblies (FA) and it is removed by a nominal total recirculation mass flow rate of about 7700 kg/s.

During the 1999 event, the OKG2 core was loaded with 444 Fuel Assemblies from four different fuel vendors and included: 232 SVEA64 8x8 FAs; 186 KWU 9x9-9 FAs; 4 ATRIUM10 10x10 FAs; and, 22 GE12 10x10 FAs. Coolant recirculation is obtained by four external recirculation loops, each one equipped with a recirculation pump[2]. There are no internal jet pumps employed in this Nordic BWR design, such that all of the core flow passes through the four recirculation loops. Additionally, there are a total of four feedwater pipes and four steam lines connected to the vessel.

TRACG includes a multi-dimensional, two-fluid representation of two-phase flow for the reactor thermal-hydraulics and has the capability to model three-dimensional reactor kinetics. The best-estimate system code TRACG was used to model the OKG2 reactor vessel, fuel channels and piping of the plant as well as the thermal-hydraulic conditions of the reactor prior to and during the instability event. The thermal-hydraulic system code implements a 3D neutron kinetics model consistent with the GE 3D core simulator, PANAC11. A PANAC11 core design wrapup at rated OKG2 power and flow conditions consistent with those at the initiation of the stability event was used to initialize the TRACG 3D kinetic model and power conditions. The given benchmark plant data was used to validate the steady state OKG2 TRACG system model for operation prior to the instability event.

3.1. Thermal-Hydraulic Model

Modeling with TRACG included a T-H system model for the vessel, core channels, and associated balance of plant system nodalization. The vessel was modeled with sixteen axial levels and one azimuthal sector. In addition, the vessel component was designed to include four radial rings - three inner rings for the core region and one outer ring for the downcomer region. The spatial nodalization for the plant system model included modeling simplifications such as the lumping together of individual recirculation loops, feedwater lines, and steam line elements into a single model component. An assessment of these kinds of simplifications, along with the sensitivities to spatial nodalization, is included in the TRACG qualification LTR (Ref. 3)[4]. The four external recirculation loops were collapsed into two separate T-H loops, scaled to model the equivalent pump characteristics and piping cross-sectional flow area. One recirculation loop was scaled to model three external recirculation pump loops while the second loop modeled a single recirculation piping line. With this recirculation loop system model, the recirculation flow area was scaled to model an approximate three to one flow area ratio for the two recirculation pump loops, respectively. The two separate recirculation loops were connected to the vessel at vessel junctions at axial level 3.

Additionally, the four feedwater lines were collapsed into one equivalent feedwater system line by scaling the pipe characteristics and cross-sectional flow area. The feedwater line is connected to the vessel at axial level 11. The four steam lines were scaled to model two equivalent steam lines, connected to the vessel towards the top of axial level 15. The turbine inlet conditions were simulated by a break component. The turbine break pressure boundary was set by the pressure controller, which maintained steam dome pressure at its reference condition.

The OKG2 core model consists of 444 bundles and 109 control rods (CR). TRACG Channel templates, using CHAN components, were generated to use as TRACG inputs using given fuel dimensions, fuel rod loading patterns, water rod locations and dimensions, spacer grid arrangements and locations, and loss coefficient specifications for each of the four fuel bundle types present in the core. A detailed core channel grouping scheme was selected for the TRACG model such that the flow distribution at the core entrance could be modeled for the different fuel channel types. Both the ATRIUM 10 and GE12 fuel types have partial length rods. The channel templates for these 10x10 assemblies have axial zones with different flow areas to account for the lattice transition from including both partial and full length rods at the channel inlet to including only full length rods (above the partial length rod height in the channel) along the bundle length. Additionally, the four fuel types have water rods, which are modeled explicitly using TRACG's water rod option for the channel component.

The 444 fuel bundles for the OKG2 core were grouped using a sufficient number of T-H channels to model the stability event according to the physical location (vessel radial ring), orifice type, radial peaking factor (RPF) and CR positions. Each of the T-H channel components used in the TRACG core model groups a certain number of fuel assemblies. The selected channel grouping scheme is considered sufficient for the initial T-H modeling of this core-wide oscillations event. However, the current state-of-the-art model[4] employed by GEH for stability applications is to use a full core individual bundle model that eliminates the T-H lumping due to channel grouping, hence reducing the approximation caused by the selected T-H grouping model[3]. Using the full core individual bundle model T-H channel mapping scheme provides the most accurate core modeling method for simulating instability stability events and will be used in future benchmark modeling efforts for the February 25, 1999 instability event. The resulting OKG2 T-H channel grouping used as input to TRACG for this analysis is shown in Fig. 2.

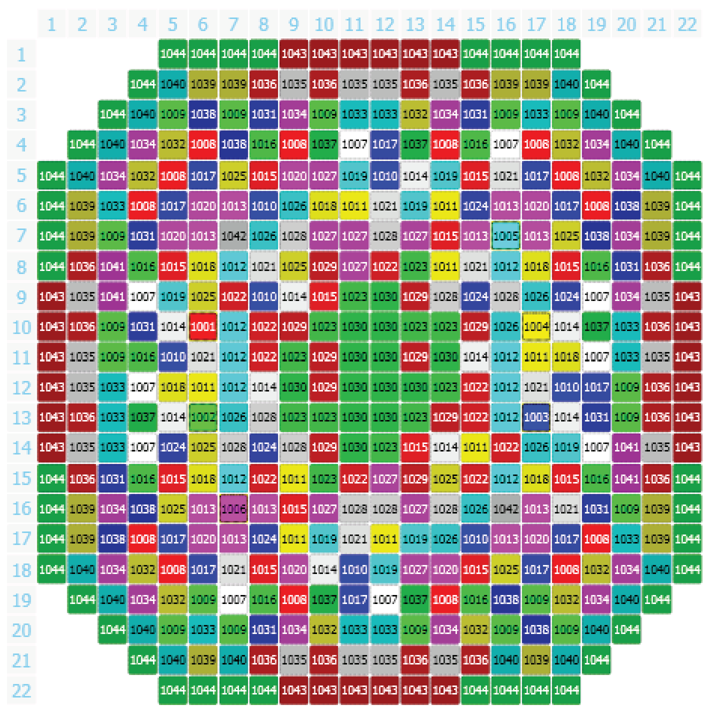

Figure 2. TRACG T-H Channel Grouping Map.

Bundles that belong to the same channel group, which are represented with the same channel component, for the TRACG T-H channel grouping scheme are pictured in the same color in Fig. 2. To model the core, the TRACG T-H channel components are connected to the reactor vessel across the upper plenum and lower plenum regions of the vessel component using vessel junctions. Junction 1 of the channel component represents the channel inlet connection to the lower plenum and Junction 2 represents the channel exit to the upper plenum[5]. The secondary section of the channel component is used to model the leakage path from the channel to the core bypass region through Junction 3. The TRACG channel nodalization included an axially refined node scheme with additional details such as the modeling of the water rods, part-length fuel rods and multiple fuel rod groups. The channel nodalization accounts for the lower plenum unheated length, the active fuel length, and the upper unheated length of the channel and uses the appropriate nodalization scheme required of each region to capture the important phenomena needed for stability applications.

The TRACG T-H system model of the OKG2 plant, developed with the T-H channel grouping core model, was then paired with the 3D neutron kinetics core model inputs from PANAC11 to complete the 3D neutron kinetics model. Using TRACG, a steady state case was run for 500 seconds to initialize the T-H and 3D neutron kinetics TRACG model, which implements the PANAC11 code 3D core neutronics model. Plant parameters, including setpoints and controls, were defined to model the transient response and control system characteristics for the OKG2 stability event. Thermal-hydraulic boundary conditions for feedwater flow, feedwater temperature, and steam line pressure versus time were provided and implemented in the TRACG model using control systems and pressure boundary BREK components to model the stability event.

3.2. Neutron Kinetics Model

The 3D neutron kinetics model of the core simulator PANAC11 code was implemented with the TRACG system code and provided the initial 3D power distribution and cross-sections for the T-H channel grouping core model defined in TRACG. The PANAC11 core wrapup was generated at rated power and flow conditions consistent with those at the initiation of the stability event. The results presented in this paper include a 3D kinetics core model which approximates neutronics conditions for the core based on generic GE 8x8, 9x9, and 10x10 fuel bundle types. The PANAC11 transient cross-sections are derived from the plant-specific core design depletion and take into account moderator density, fuel temperature and control rod state. However, with limited usable information on bundle neutronics inputs for the different OKG2 bundle types, a generic BWR core design with the same core size was used as a baseline for developing the OKG2 3D kinetics model. The neutron kinetics model individually modeled each bundle in the core using 25 equally sized axial nodes. The neutronics model included one node per each fuel bundle (radially) times the number of axial nodes. The core active length is simulated using 25 uniform axial meshes for each fuel type.

OKG2 control rods were defined using two control rod groups – one for the Selected Rod Insertion and one for the Reactor Scram) and were placed in the given initial positions (or notches of insertion into the core) and physical locations specified in the benchmark specifications.

OKG2 plant cross-section data was supplied by the benchmark organizers in the CASMO format, which is not compatible with the PANAC11 format. Tabular format of these cross-sections (NEMTAB rodded and unrodded) are yet to be provided to the benchmark participants and it is likely that these will not be released for the duration of the benchmark. Therefore, in order to initialize the PANAC11 3D neutron kinetics model and check the coupling with the TRACG T-H model, some OKG2 neutronics characteristics were approximated using generic fuel bundle and blade design inputs in the current neutronics model.

4. TRACG ANALYSIS AND RESULTS

The process to perform the TRACG simulation included the generation of a plant-specific TRACG basedeck and of all T-H channels representing the fuel bundles for the analyzed core design, which was done with the ALGCH code. These inputs are based on the TRACG modeling methodology described in Section III. A steady state case was run to initialize the coupled T-H and 3D neutron kinetics models between TRACG and PANAC11 codes. Following the achievement of a proper OKG2 steady state, as shown in Figure 3, the actual TRACG transient was run and the February 25, 1999 instability event was initiated via the TRACG modeled control system.

4.1. Steady State TRACG Model Validation

The TRACG simulation was initialized at state point 1 in Fig. 1 with the OKG2 core operating at rated power and flow. The control blade pattern was specified to match the given control rod positions prior to the instability event. Boundary and initial steady state conditions, most of them measured, some calculated and used by benchmark organizers, were provided to benchmark participants to model the OKG2 NPP operation prior to the stability event. A steady state case was run for 500 seconds to initialize the T-H and 3D neutron kinetics TRACG model, implementing the PANAC11 code 3D core neutronics model. A TRACG steady state calculation of 500 seconds was performed to make sure that key parameters such as power, core flow, feedwater temperature and flow, steam line flow, steam line pressure, inlet subcooling, etc., were matching the available plant data.

The plant operating data was used to validate the steady state OKG2 TRACG system model for operation prior to the instability event. The TRACG results of the steady state analysis are given in Table II. The results and comparison are provided for TRACG with the 3D Kinetics model. When available, measured plant data was used for comparison with TRACG results. Boundary and initial conditions used for the benchmark organizer's NPP code were used for comparison purposes when measured plant data was not available. Comparisons with the NPP measured data and with the data obtained by the NPP code calculations are reported in Table II.

Table II. Steady State Analysis Results.

Steady State Parameter	NPP Measured Data	TRACE/ PARCS Results[2]	TRACG Results	TRACG vs. NPP Measured Data Relative Error	TRACG vs. TRACE/ PARCS[2] Relative Error
Reactor Power (MW)	1798.6	1802	1800	0.1%	-0.1%
Steam Dome Pressure (MPa)	6.93	7	6.94	0.1%	-0.9%
Core Inlet Pressure (MPa)	N/A	7.12	7.14	N/A	0.4%
Core Outlet Pressure (MPa)	N/A	7.01	7.02	N/A	0.1%
Orifice & Lower plate dP (kPa)	N/A	55.1	56.4	N/A	2.3%
Core Average Void	N/A	0.42	0.38	N/A	-9.8%
Core Average Fuel Temp (K)	N/A	823.9	851.1	N/A	3.3%
Feedwater Temperature (K)	457.65	456.62	457.7	0.0%	0.2%
Core Inlet Temperature (K)	547.3	543.57	543.3	-0.7%	-0.1%
Inlet Subcooling (K)	N/A	16.6	17.1	N/A	3.2%
Steam Temperature (K)	N/A	558.5	558.5	N/A	0.0%
Total Core Flow Rate (kg/s)	5474	5515.9	5358.5	-2.1%	-2.9%
Active Core Flow Rate (kg/s)	N/A	4800.4	4746.8	N/A	-1.1%
Steam Flow Rate (kg/s)	900	903.1	903.5	0.4%	0.0%

Calculated parameters are in reasonable agreement with the given boundary and initial condition code inputs and plant data for the approximated 3D neutron kinetics core model. Some of the deviations were expected considering that some proprietary fuel parameters were not given from measured NPP data and were instead approximations used by the benchmark organizers in their code model.

Deviations in the channel model thermal-hydraulics results between the 3D kinetics model and the TRACG system model are currently being addressed in the ongoing work on this benchmark analysis.

Results for TRACG key parameters including power, core flow, inlet subcooling, feedwater flow, steam line flow, and steam line pressure for the TRACG steady state calculation of 500 seconds are shown and compared with NPP Data in Fig. 3.

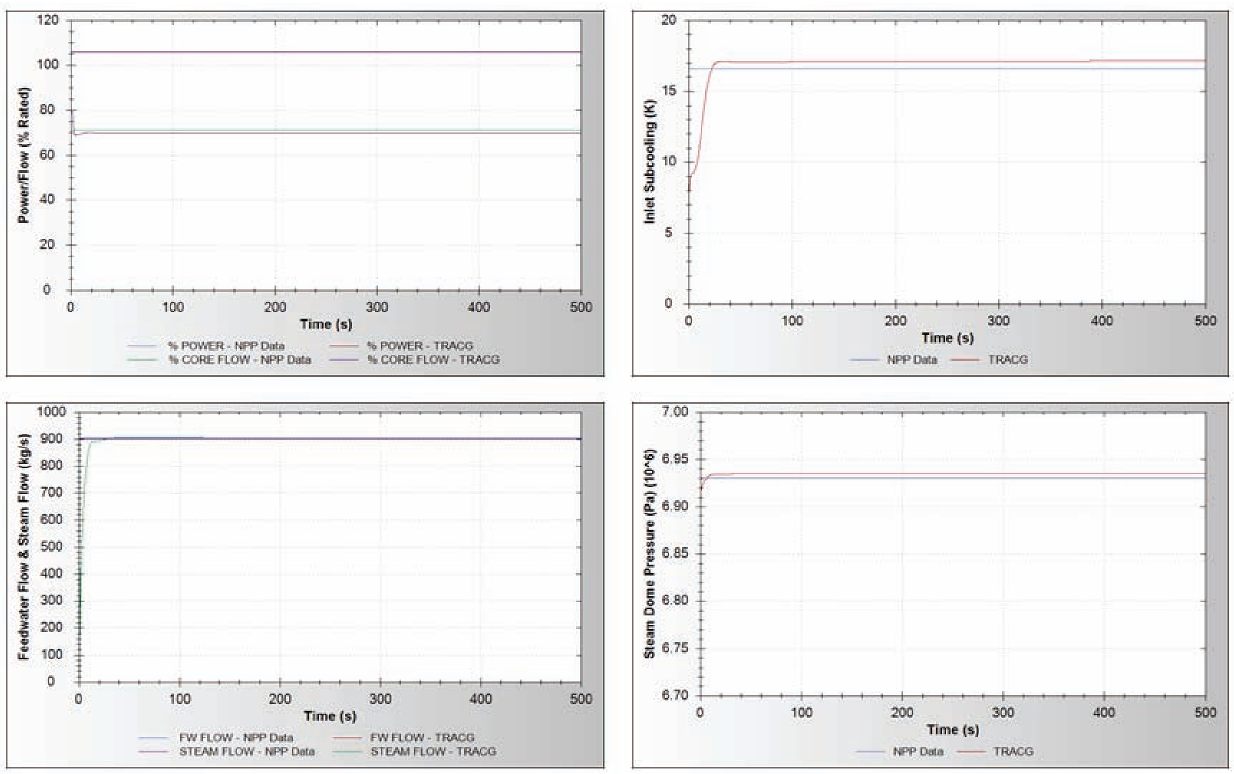

Figure 3. TRACG Steady State Results with the 3D Kinetics Model.

Steady state results show that the TRACG T-H model with 3D kinetics can be properly initialized and run to simulate nominal plant operation for Oskarshamn-2 prior to the stability event. Results for the above TRACG key parameters are shown to stabilize at steady state conditions approximately matching those specified in the NPP code and given as measured plant data listed in Table II, validating the TRACG Oskarshamn-2 plant model.

4.2. TRACG Stability Event Results

Plant parameters, including setpoints and controls, were defined to model the transient response and control system characteristics for the OKG2 stability event. Following the achievement of a proper OKG2 steady state, the actual TRACG transient was run and the February 25, 1999 stability event was initiated via the TRACG modeled control systems. In modeling an instability, it is important to predict the correct inlet subcooling.

Thermal-hydraulics boundary conditions for feedwater flow, feedwater temperature, and steam line pressure over the 300-second transient duration were provided to model the stability event. The feedwater temperature and flow curves were imposed and used as the inlet boundary condition. The steam line pressure was used as an outlet boundary condition. Additionally, the recirculation pump speed data versus time was given and used as a third inlet boundary condition for the transient OKG2 model. The control rod pattern for both operation before the event and after the SRI was given to model the control movement of seven predefined control rods during the event.

The transient TRACG simulation was run according to the time history detailed in Table I. Boundary conditions for recirculation pump speed, feedwater temperature, and feedwater flow were used to simulate the recirculation pump controller pump speed reductions, the decrease in feedwater temperature due to the feedwater preheater trip and the corresponding feedwater flow conditions, respectively, involved in the evolution of the February 25, 1999 stability event. Initially the feedwater temperature and flow rate were decreased to simulate the feedwater preheater trip that occurred at the initiation of the OKG2 transient event from state point 1 to state point 2 in Fig. 1. The partial scram control rod insertion movement is modeled in the transient case by inserting the group of control rods from the known initial control rod pattern at state point 1 to the given partial scram rod pattern at state point 9 in Fig. 1 at approximately 196 seconds. For the final stage of the stability event, all remaining control rods were fully inserted to model the reactor scram at 252.6 seconds shown at Stage VIII in Table I and correspondingly, state point 12 in Fig.1.

A TRACG transient case of 300 seconds with coupled T-H/3D kinetics models was performed to simulate the February 25, 1999 feedwater event. Key parameters such as power, core flow, feedwater temperature and flow, steam line flow, steam line pressure, and inlet subcooling from the transient run were compared with measured plant data from the event. Results for the key parameters of reactor power and core flow for the TRACG transient analysis are reported and compared with measured plant data in Figures 4 and 5, respectively.

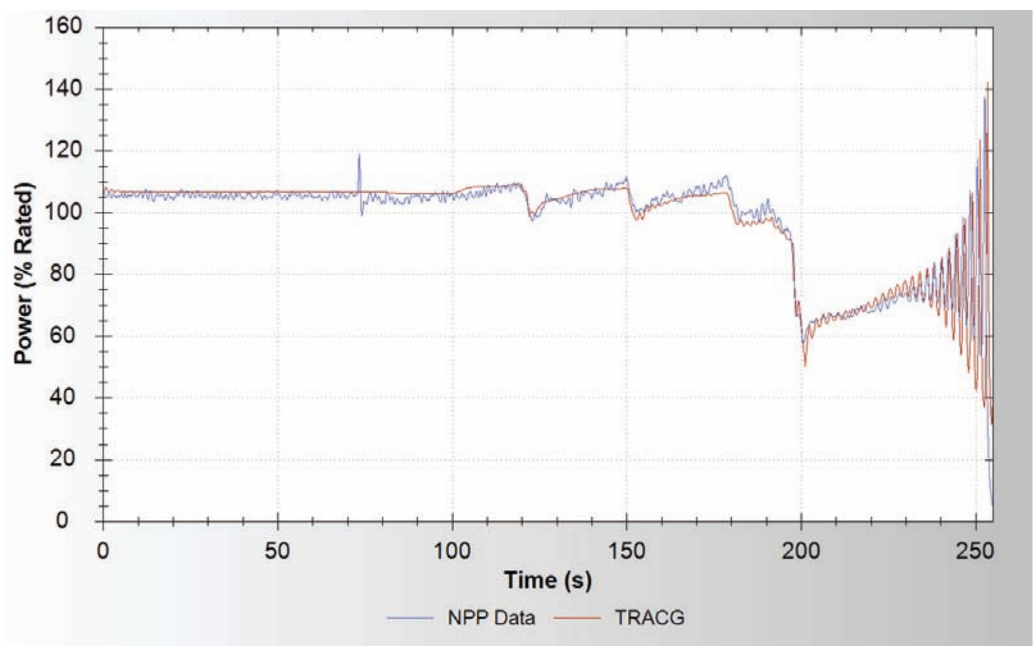

Figure 4. TRACG Transient Analysis Power Comparison with NPP Measured Data.

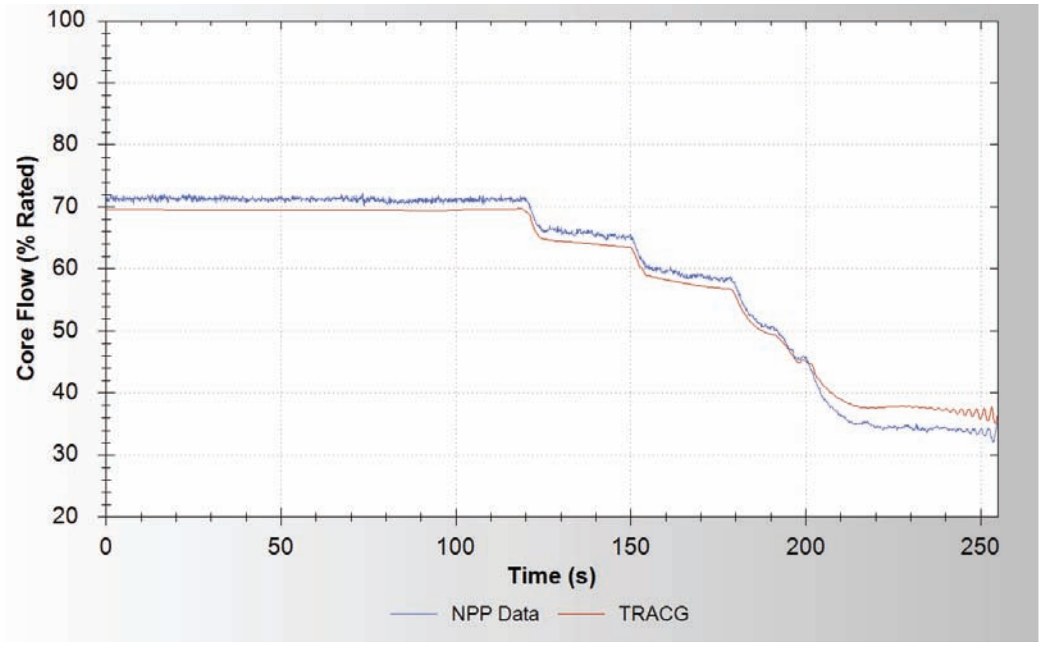

Figure 5. TRACG Transient Analysis Core Flow Comparison with NPP Measured Data.

Calculated transient parameters are in good agreement with the measured NPP plant data to capture the evolution of the transient. Transient case results are presented in Figures 4 and 5 above for operation up to the reactor scram, or the culmination of the instability event, at 252.6 seconds. The current results show that the TRACG T-H model with its coupled 3D kinetics can be properly initialized and run to simulate this complex instability event. Notwithstanding some approximation to the core neutronics and blade design used in the TRACG 3D kinetics model, the results in Figures 4 and 5 demonstrate an excellent agreement of the calculated TRACG simulation and the plant measured data for the instability event. One identified source of uncertainty is the time lag between various plant instrumentation and signal measurements, such as the time delay in the Resistance Temperature Detector (RTD) instrumentation measurement of feedwater temperature and the actual change in signal during the event. In particular, a feedwater adjustment was provided from measured data as part of the benchmark and the next section addresses this sensitivity. The results provided in Figures 4 and 5 include this adjustment.

5. TRACG STABILITY EVENT FEEDWATER TEMPERATURE SENSITIVITY

A sensitivity study was performed to analyze the impact of the feedwater temperature correction on the power oscillations response for this instability event. Transient results shown in Figures 4 and 5 above were determined using the given modified feedwater temperature boundary condition for the instability event. The modified feedwater temperature boundary condition given in the benchmark specifications was adjusted by decreasing the measured plant data for feedwater temperature during the event to account for the time lag for the RTD temperature measurement[6].

The feedwater temperature was measured by the RTDs, which are mounted on thermowells which contain the temperature sensors that go inside the feedwater flow stream. The RTDs do not penetrate the feedwater pipe and thus, are not a part of the pressure boundary[7]. RTDs work well for steady state operation because reactor operators are mainly interested in the steady state temperature thus the sensor delay is not an issue since normal operating conditions stay relatively constant. While the RTDs are accurate, there is a time delay in the transient response and feedwater temperature measurement due to the time required to conduct heat through the steel separating the feedwater flow from the detector mounted on the well[7]. The benchmark organizers modified the time delayed feedwater temperature measurement based on the inverse solution of the one-dimensional time-dependent heat conduction equation for the one-dimensional semi-infinite slab of steel. The temperature of the feedwater flow was solved for as the boundary condition at the steel wall and the time-dependent temperature at a certain depth into the steel slab was known using the RTD measurement[7]. The steel pipe thickness was assumed to be 0.5 inches in the benchmark organizers' correction method for feedwater temperature[7]. The typical material thickness for a thermowell approximately 0.25 inches. Since the actual material thickness for the steel is unknown, a sensitivity study was performed to further analyze the impact of the time-dependent feedwater temperature correction method on the transient response.

The measured NPP Data for feedwater temperature ("NPP Measured") is shown versus the NPP benchmark organizers' modified version of the feedwater temperature data ("NPP Modified"), which was given to benchmark participants to use as a boundary condition for the event, is shown below in Fig. 6.

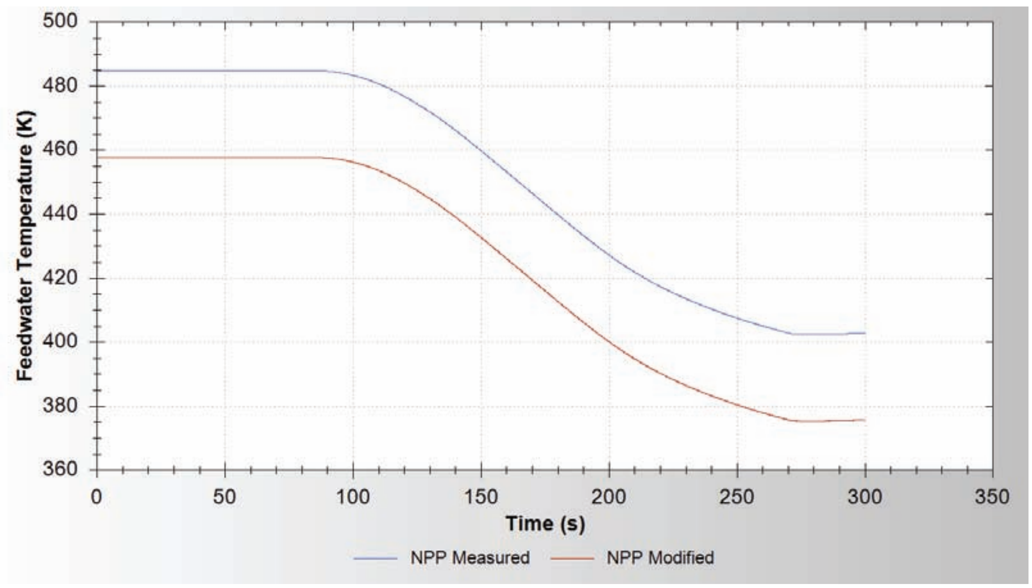

Figure 6. NPP Feedwater Temperature Boundary Condition Modification.

Decreasing the measured feedwater temperature data to develop the modified feedwater temperature boundary condition causes the core power to increase for the case using the modified feedwater temperature boundary condition. Since there is uncertainty in the feedwater temperature measurement itself and in the correction method used to account for the RTD time lag, driven by the thermal inertia effect, for the given modified feedwater temperature boundary condition ("NPP Modified") from the benchmark specifications for the transient, a sensitivity study was performed to assess the impact of this boundary condition on the TRACG model's ability to capture the entire behavior of the stability transient. This issue was identified and documented in previous work by T. Kozlowski[7] et al. and similar sensitivity performed with TRACE/PARCS.

The given modified feedwater temperature boundary condition used for the original TRACG transient case (displayed in Figures 4 and 5) was further adjusted by decreasing the feedwater temperature, adding to the correction factor methodology used to correct the measured plant data by benchmark organizers. A comparison of the different feedwater temperature boundary condition variations is shown in Fig. 7 below.

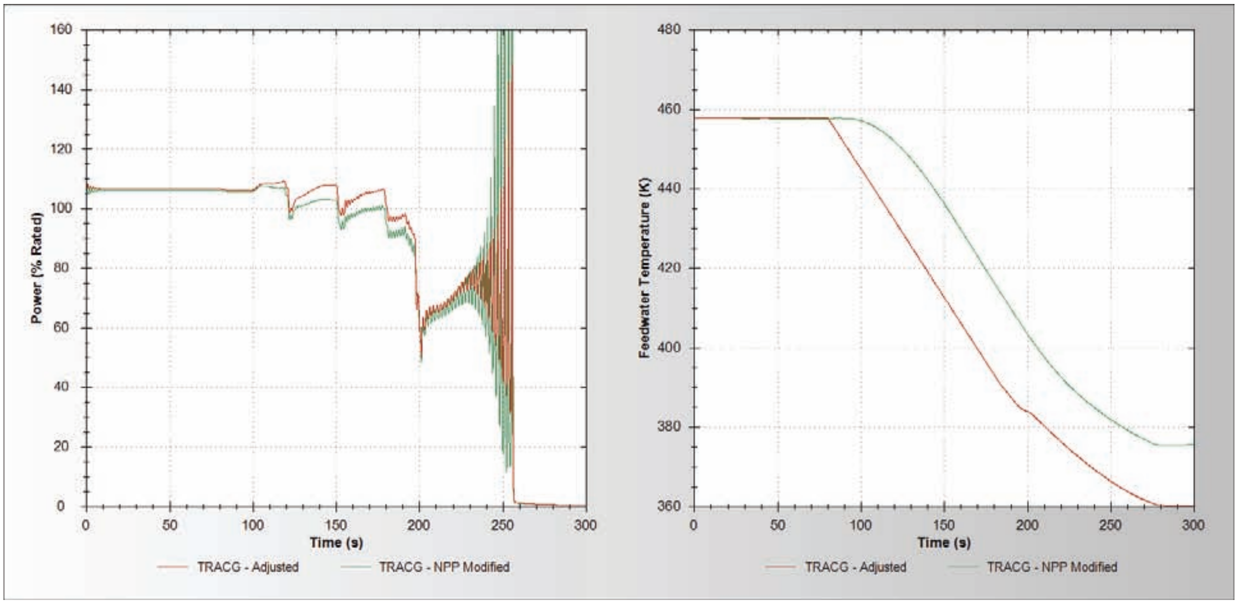

Figure 7. TRACG Transient Case Feedwater Temperature Boundary Condition Sensitivity.

The TRACG results for the case that uses the given modified feedwater temperature boundary condition (adjusted by benchmark organizers) are listed as "TRACG - NPP Modified" and the TRACG results for the case that uses an adjusted (decreased) temperature boundary condition, which was modified as described above to address the transient effect on the RTD, is labeled as "TRACG - Adjusted". The "TRACG – Adjusted" feedwater temperature was adjusted by decreasing the temperature starting from around 80 seconds, just after the time of the feedwater preheater system trip (Stage II in Table I), to account for the time delay in the transient response for the detector measurement. Decreasing the feedwater temperature boundary condition input for the "TRACG - Adjusted" case is shown to increase the core power response for the event in comparison to the previous TRACG case (labeled "TRACG - NPP Modified"). It is important to note that the TRACG transient response curves shown in Figures 4 and 5 also used the adjusted modified feedwater temperature boundary condition.

As shown in Fig. 7, decreasing the given modified feedwater temperature boundary condition causes the TRACG predicted core power to increase following the SRI movement which more closely follows that for the measured OKG2 plant data for the stability event. The results of the feedwater temperature study suggest that the time delay in feedwater temperature measurement by the RTD instrumentation is a limitation and that the measured feedwater temperature data should be adjusted (decreased) to more accurately model the transient response during the instability event. The results of this sensitivity study performed with TRACG are similar to the findings obtained with TRACE/PARCS[7].

6. CONCLUSIONS

A T-H/3D neutron kinetics coupled model of TRACG was developed to simulate the 1999 instability transient at the Oskarshamn 2 plant. The TRACG results for both the steady state and transient compare well with measured OKG2 plant data. The TRACG transient analysis predicts the same core power and flow oscillation behavior as that observed for the feedwater instability event, with the magnitude of oscillation growth slightly greater for the TRACG response than that for measured data. The results documented in this paper demonstrate the capability for TRACG to properly simulate the Oskarshamn instability event.

Since there is inherent uncertainty in the RTD feedwater temperature measurement and in the correction method used by benchmark organizers to account for the RTD thermal inertia driven time lag effect for the given modified feedwater temperature boundary condition from the benchmark specifications for the transient, a sensitivity study was performed to assess the impact of this boundary condition on the TRACG model's ability to capture the behavior of the stability transient. This method of feedwater temperature boundary condition adjustment was found to be relevant for capturing more closely the trend of the oscillations during the instability event. This result confirms the importance of time-dependent inlet subcooling in determining the onset of oscillations.

Future work will involve the refinement of the 3D neutron kinetics model in TRACG by possibly implementing the NEMTAB cross-sections once they will be provided to benchmark participants and also to include a full core individual bundle T-H model.

ACKNOWLEDGMENTS

The author appreciates the cooperation of OECD/NEA Benchmark organizers and the Oskarshamn-2 team for organizing this International benchmark, providing plant data from the February 25, 1999 feedwater stability event, and for allowing this work to be published. The author appreciates the cooperation of the BWR Stability Analysis team at GEH for providing guidance and lending their expertise to help advance this benchmark project forward and for allowing the TRACG code to be used in this stability benchmark application. The author appreciates the GE Edison Engineering Development Program for sponsoring and supporting efforts towards completing this stability analysis project. The author appreciates the cooperation of Dr. Juswald Vedovi for serving as dedicated Technical Mentor and providing guidance throughout the completion of this benchmark project.

NOMENCLATURE

3D	Three-Dimensional
BWR	Boiling Water Reactor
CR	Control Rod
FA	Fuel Assembly
GE	General Electric
LTR	License Topical Report
NEA	Nuclear Energy Agency
NPP	Nuclear Power Plant
OECD	Organization for Economic Co-operation and Development
OKG2	Oskarshamn-2
RPF	Radial Peaking Factor
RTD	Resistance Temperature Detector
SRI	Selected Rod Insertion
T-H	Thermal-Hydraulic
TRAC	Transient Reactor Analysis Code
TRACG	GE Hitachi Transient Reactor Analysis Code

REFERENCES

1. H. M. Cowen, S. L. Pfeffer, J. Vedovi, L. Ibarra, D. G. Vreeland, D. C. Miranda, "TRACG Analysis of the OECD/NEA Oskarshamn-2 BWR Stability Benchmark Based on February 25, 1999 Feedwater Transient Event," *Proceedings of the 2014 International Congress on the Advances in Nuclear Power Plants Meeting*, Charlotte, USA, Paper 14363, April 6-9, 2014 (2014).

2. T. Kozlowski, S. Roshan, T. Lefvert, C. Netterbrant, T. Downar, Y. Xu, A. Wysocki, K. Ivanov, J. Magedanz, M. Hardgrove and J. March-Leuba, *BWR Stability Event Benchmark based on Oskarshamn-2 1999 Feedwater Transient*, NEA/NSC/DOC (2013).
3. GE Hitachi Nuclear Energy, *TRACG Qualification*, NEDO-32177, Revision 3 (2007).
4. J. Vedovi, J. Yang, L. Klebanov, D. G. Vreeland, J. F. Zino, "Best-Estimate Plus Uncertainty Thermal-Hydraulic Stability Analysis of BWRs Using TRACG Code," *Proceedings of the 2012 International Congress on the Advances in Nuclear Power Plants Meeting*, Chicago, Illinois, USA, Paper 12261, June 24 – 28, 2012 (2012).
5. GE Hitachi Nuclear Energy, *TRACG Model Description*, NEDO-32176, Revision 4 (2008).
6. T. Kozlowski, A. Wysocki, I. Gajev, Y. Xu, T. Downar, K. Ivanov, J. Magedanz, M. Hardgrove, J. March-Leuba, N. Hudson, W. Mac, *Analysis of the OECD/NRC Oskarshamn-2 BWR stability benchmark*, Annals of Nuclear Energy 67, pp. 4–12 (2014).
7. T. Kozlowski, T. Downar, Y. Xu, A. Wysocki, K. Ivanov, J. Magedanz, M. Hardgrove, J. March-Leuba, N. Hudson and D. Woodyatt, *TRACE/PARCS analysis of the OECD/NEA Oskarshamn-2 BWR stability benchmark*, Knoxville, Tennessee, USA (2012).