

## Severe Reactivity Initiated Accidents with SIMULATE-3K and SIMULATE-3K/RELAP5 in Forsmark-3 BWR

Christian Jönsson<sup>a</sup>, Gerardo Grandi<sup>b</sup>, Jerry Judd<sup>c</sup>, Gustav Dominicus<sup>d</sup>, Henrik Bergersen<sup>e</sup>

<sup>a)</sup> Studsvik Scandpower AB, Stensborgsgatan 4, 721 32 Västerås, Sweden,

Tel: +46 (0)21-334762, Email: [christian.jonsson@studsvik.com](mailto:christian.jonsson@studsvik.com)

<sup>b)</sup> Studsvik Scandpower, Inc., 504 Shoup Avenue, Suite 201, Idaho Falls, ID 83402, USA,

Tel: +1 208 528 2398, Email: [gerardo.grandi@studsvik.com](mailto:gerardo.grandi@studsvik.com)

<sup>c)</sup> Studsvik Scandpower, Inc. 504 Shoup Avenue, Suite 201, Idaho Falls, ID 83402, USA,

Tel: +1 208 522 1720, Email: [jerry.judd@studsvik.com](mailto:jerry.judd@studsvik.com)

<sup>d)</sup> Forsmarks Kraftgrupp AB, 742 03, Östhammar, Sweden,

Tel: +46 (0)173-82414, Email: [ydb@forsmark.vattenfall.se](mailto:ydb@forsmark.vattenfall.se)

<sup>e)</sup> Forsmarks Kraftgrupp AB, 742 03, Östhammar, Sweden,

Tel: +46 (0)173-82883, Email: [hkb@forsmark.vattenfall.se](mailto:hkb@forsmark.vattenfall.se)

**Abstract** – During the autumn of 2008, two broken control rod stems were discovered at two Swedish BWRs. One of them is Forsmark 3. The discovery led to shaft replacements and careful continuous observation programs. The Swedish Nuclear Inspectorate has accepted these solutions. Even so, special studies were required in order to investigate the consequences of control rod drop, with a new assumption: *any* control rod could fall at *anytime*. Control rod drops belong to the class of severe reactivity insertion accidents. Under the new assumption, the highest rod worth, for a single control rod, can be above 3000 pcm during the control rod withdrawal sequence. These events may lead to pellet and cladding failure. They require the evaluation of the core response with multidimensional, multi-physics models. The purpose of this paper is to show the applicability of SIMULATE-3K, either standalone or coupled with the safety analysis code RELAP5, to this class of severe reactivity insertion accidents. This paper shows that SIMULATE-3K adequately captures the complicated interaction between physical processes in the core, in comparison with the more detailed RELAP5 thermal-hydraulic model, as well as the adequacy for analysis near the pellet and cladding failure limits. This paper discusses the consequences for the core in relation to the inserted reactivity worth and the amount of fuel fragmentation. This paper shows the calculated reactivity components, pin-by-pin temperature, and enthalpy distributions computed by SIMULATE-3K “explicit pin-by-pin conduction” model, as well as the pin failure statistics.

### I. INTRODUCTION

Forsmark 3 is one of three ASEA-Atom BWRs at the Forsmark site, approximately 150 km north of Stockholm, Sweden. The reactor was originally designed for 3020 MWth at a core flow rate of 12100 kg/s (700 fuel bundles). The first operation took place in 1985. A power uprate, in 1989, led to operation at 3300 MWth, up to the present cycle. The present work, in the scope of the PSAR, is performed to support a power uprate to 3775 MWth (planned to be in operation in 2014).

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and careful continuous surveillance programs. These solutions have been accepted by the Swedish Nuclear Inspectorate. Even so, special studies were required in order to investigate the consequences of control rod drop (CRD), with a new assumption: *any* control rod could fall at *anytime*.

Scenarios like the one described above, belong to the class of severe Reactivity Insertion Accidents (RIA). The typical cycle-specific analyses involve possible control rod drops for control rods where the control rod drive has been withdrawn. The control rod is thereby assumed to drop to the position of the control rod drive. If the control rod stem brakes, the control rod can drop at any condition. Under the new assumption, that *any* control rod could fall at

*anytime*, the highest rod worth for a single control rod can be above 3000 pcm during the control rod withdrawal sequence. In the present study, a startup situation at End of Cycle (EOC) was found to be limiting. Furthermore, it was found that most of the reactivity release comes in the first meter of the control rod drop.

Events that may lead to pellet and cladding failure require the evaluation of the core response with multidimensional, multi-physics models. The purpose of this paper is to show the applicability of SIMULATE-3K.<sup>1</sup> (S3K), either standalone or coupled with the safety analysis code RELAP5,<sup>2</sup> to RIA. S3K is a best-estimate nodal reactor analysis tool that employs advanced core neutronic model coupled with detailed thermal-hydraulic channel and fuel pin models. Utility licensing approval from the U.S. NRC has been obtained for RIA.

Recently, there has been considerable focus on the licensing basis for RIA limits, which impacts both PWR and BWR RIA analyses. In Europe and Japan, exposure-dependent RIA fuel enthalpy criteria have been adopted. Similarly, a recent NRC recommendation for an interim RIA acceptance criterion has also proposed exposure-dependent limits on fuel enthalpy rise. One consequence of these new criteria is that a more detailed analysis of RIA events may be required because fuel pins with highest enthalpy may no longer be the most limiting. The new RIA acceptance criteria led to the development of an S3K “explicit pin-by-pin conduction” model that allows the detailed tracking of individual fuel pins in the core.

Section II discusses the effect of high reactivity transient by describing fuel fragmentation experimental results in a test reactor. Section III provides an overview of the observations of the control rod cracks that led to the present study. Section IV provides an overview of S3K models, RELAP5 models, and the linkage between the two codes. Special emphasis will be paid to the fuel pin model due to its importance to RIA. Finally, section V summarizes the S3K standalone and S3K/RELAP5 RIA results.

## II. RIA – FUEL FRAGMENTATION

At a condition of rapid power bursts associated with high worth RIA, the amount of power release is such that fuel fragmentation can take place. Results presented in Section V will demonstrate that a number of fuel pins are at risk of fuel melt or fuel fragmentation during RIA under the new assumption that *any* control rod could fall at *anytime*.

The consequences of severe RIA on LWR fuel have been measured in reactor tests.<sup>3</sup> By using a highly enriched test section, in a test reactor, power pulses of the same magnitude as the ones predicted in severe RIA scenarios have been generated. The inserted reactivity reached 4 to 5\$ with minimum reactivity periods of 1.1 ms. The

maximum fuel enthalpy was above 2000 kJ/kg, depending on the exact level of test sample enrichment. Fig. 1 shows a schematic of the test capsule.

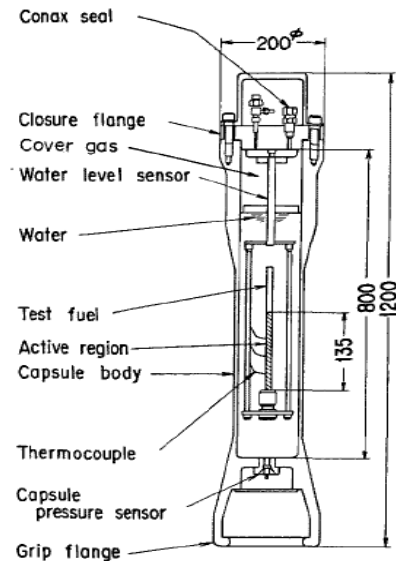


Fig. 1. Test capsule - 2180 kJ/kg case.<sup>3</sup>

The subcooling was varied between 60° and 85°C. These experimental conditions resemble cold cases with high subcooling analyzed later in this paper. At each measurement the cladding temperature, water pressure, and water level (in the test capsule) were measured.

At the condition of fragmentation, fuel melt takes place, which creates immediate pellet expansion and the pressure inside of the fuel rod increases. The cladding heats up and eventually ruptures. Melted UO<sub>2</sub> is pressed out (ejected) into the water. The droplets cool directly and the size of the fragments is coupled to the energy transfer to the water. If the period is very small (fast pulse) a higher pressure builds up before the fragmentation and the size of the particles becomes smaller. The energy transfer is higher. The measured parameters vary according to Fig. 2.

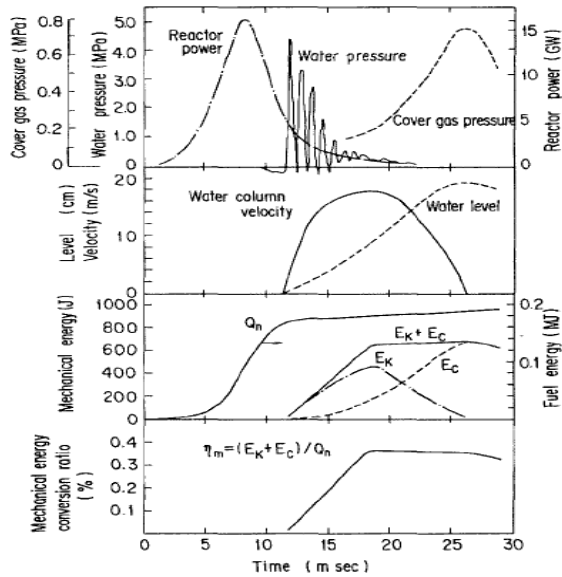


Fig. 2. Experimental result at fuel fragmentation – 2180 kJ/kg case.<sup>3</sup>

A high frequency pressure pulse is observed with a 1 ms oscillation period. The high pressure oscillation amplitudes have decreased to zero in a time period of 10 ms. The energy transfer from the fragmented material is between 0.05% to 1% depending on the maximum enthalpy. It is interesting to observe that the cladding failure occurred at approximately 1465 kJ/kg, which is approximately the same level at which fuel melts starts.

### III. CONTROL RODS MECHANICAL INTEGRITY

During the fall of 2008 two control rod stem breaks occurred at Oskarshamn 3 and Forsmark 3. Initial cracks, that can cause rupture after long term exposure to thermal fatigue, originate from cold (60°C) crud-cleaning water flow that enters the control rod guide tubes and travels upwards inside the tubes and mixes with the warm lower plenum water flow that enters the control rod guide tubes through small holes. The cleaning water temperature is too cold in comparison with the plenum water and crack indications have evolved slowly into cracks since the start of the operation in 1985-1986, due to thermal fatigue, probably in combination with mechanical vibrations.<sup>4</sup>

From January to August 2009 operation was carried out with control rods in a 86% withdrawn position in order to change the axial zones of the control rod shaft that experience the cold water flow stream. The new position of the control rods resulted in crack indications at new locations. These new crack indications are very small and just possible to detect. The position of the cracks is shown in Figs. 3 and 4. An indication of a crack is shown in Fig. 5.

The continued operation includes inspection, material surface treatment, and extended safety analysis, before a long-term solution will be chosen.

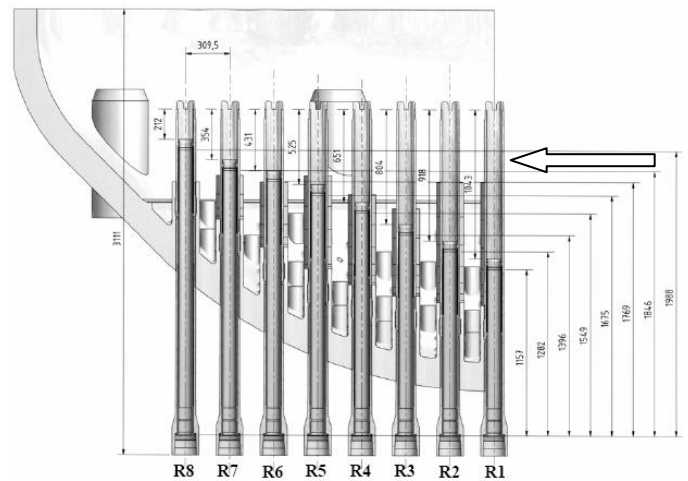


Fig. 3. Bottom part of lower plenum with control rod housing and shafts. Arrow marks level of lower plenum water inlet.

## IV. MODELS

A summary of the S3K and RELAP5 models is provided in what follows, with special attention to the fuel pin model and the heat transfer package. In addition, a brief description of the coupled S3K/RELAP5 system is given.

### IV.A. S3K Core Model

The neutronic model used in S3K solves the transient three-dimensional, two-group neutron diffusion equations, including a six-group model for delayed neutron precursors. S3K tracks the nodal concentrations of fission products and accounts for the extraneous neutron sources due to spontaneous fissions, alpha-n interactions from actinide decay, and gamma-n interactions from long-term fission product decay.

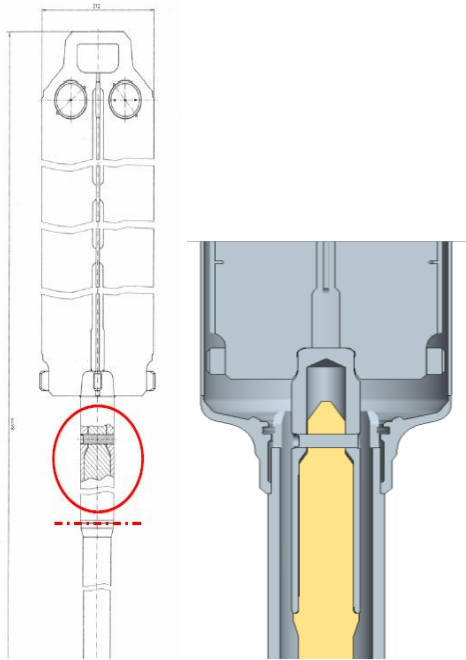


Fig. 4. Control rod guide tubes and blade with highlighted crack zone.

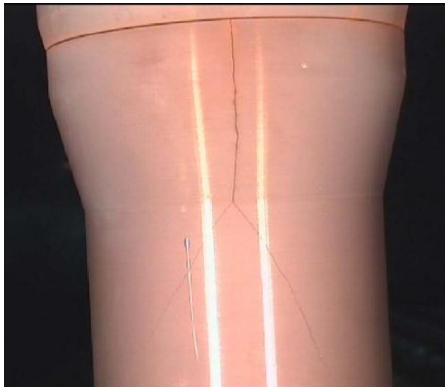


Fig. 5. Control rod showing an indication of a crack.

The basic spatial integration model of S3K is formed via transverse integration of the 3-D equations separately over each spatial direction. This procedure creates an equivalent set of three one-dimensional equations coupled via a transverse leakage term. The flux distribution is expanded in terms of fourth-order polynomials (or analytical functions) in each direction and thus the spatial gradient of the flux can be analytically represented by a third-order function. The frequency transform method is used for the time integration. This method separates the flux into two components, one with a pure exponential time dependence, and the other with primarily spatial (and weak temporal) dependence. The frequency transform method is especially suitable for the analysis RIA.

The knowledge of the intranodal flux and power distributions within each node can be used to compute the

pin-by-pin power for every axial level of every fuel pin in the core. The basic assumption of the pin power reconstruction method is that the core and assembly flux distributions can be approximated as spatially separable. The pin-by-pin “homogeneous” pin power distribution is evaluated by integrating the product of the fission cross sections and the homogeneous flux distributions over each pin location. The pin-by-pin “heterogeneous” pin power distribution is evaluated by multiplying the “homogeneous” pin power distribution by the CASMO heterogeneous form functions. The pin power reconstruction capability during the transient provides the source term for the explicit calculation of pin-by-pin fuel temperatures and enthalpies.

#### IV.B. S3K Hydraulic Channel Model

The core is represented with one thermal-hydraulic channel per fuel bundle with no cross flow. The hydraulic model uses a five-equation model: vapor and liquid mass conservation, vapor and liquid energy conservation, and averaged momentum conservation. The hydraulic primitive variable set solved for, is comprised of phasic mass fluxes, phasic enthalpies, pressure, and void fraction. In addition to the conservation equations, closure relationships exist for each phasic density, defined as a function of the pressure and phasic enthalpy. It is important to mention that steam/water properties are evaluated at the core exit pressure. The general drift formulation for the void fraction completes the set of equations to be solved. The concentration parameter and the void-weighted drift velocity are calculated using the EPRI correlations. The subcooled boiling model is taken from Lahey’s mechanistic model.

#### IV.C. S3K Fuel Pin Model

The heat conduction in the fuel pin is governed by the one-dimensional, radial heat conduction equation. Material properties depend on temperature and burnup. The S3K gap conduction model is based on the models of the INTERPIN code and is functionalized versus exposure and fuel temperature. The heat source intrapellet distribution is evaluated based on the pellet average exposure by interpolation in pre-computed tables generated with CASMO. The boundary conditions for solution of the conduction equation are the imposition of symmetry about the fuel pin centerline (the origin) and the specification of the clad outer wall heat flux. The wall-to-fluid heat transfer coefficient is determined from one of several different possible modes of heat transfer according to a classical boiling formulation: (a) single phase liquid forced convection, (b) nucleate boiling, (c) transition boiling, (d) film boiling, and (e) single phase vapor forced convection.

For single phase forced convection, the Dittus-Boelter correlation is employed. In the nucleate boiling regime, the

Chen correlation is utilized. The Chen correlation is valid up to the point where Critical Heat Flux (CHF) is reached. The 2005 CHF look-up table is used to predict CHF. Beyond CHF, transition boiling occurs which is characterized by unstable regions of both nucleate boiling and film boiling. The transition region is unstable, with the wall temperature experiencing a rapid increase en route to film boiling. Transition boiling is assumed to persist until the wall temperature exceeds the Minimum Stable Film Boiling temperature (MSFB) at which the onset of film boiling occurs. However, a direct jump from nucleate boiling to film boiling was used for the specific analysis presented in this paper. The film boiling heat transfer coefficient is computed using the modified Bromley correlation. Film boiling persists until the equilibrium quality is greater than 0.95, at which point the transition to single phase vapor heat transfer begins, and the heat transfer coefficient is linearly interpolated in quality between the film boiling heat transfer coefficient and the single phase vapor heat transfer coefficient.

For purposes of modeling thermal hydraulic feedback effects, S3K historically has modeled all fuel pins within a node by the *average* pin. The average fuel pin was supplemented in the original S3K by tracking of the *peak* pin; permitting an accurate estimation of peak fuel pin temperatures and enthalpies. The capability to explicitly represent every fuel pin in the reactor core has been recently added to S3K. This capability is referred to as the “explicit pin-by-pin conduction model.” The following assumptions are made:

- Fuel pin heat sources are taken directly from time-dependent pin-by-pin energy deposition rates.
- Bulk fluid thermal-hydraulic conditions are the same for all pins within a node.
- Fuel pin and cladding material properties are computed using pin-by-pin burnups and temperatures.
- Heat transfer regime and heat transfer coefficients are computed for individual fuel pins.

With the explicit pin-by-pin conduction model, S3K can explicitly follow the complete radial distribution of fuel pin temperatures and enthalpies (in every fuel pin) throughout a transient. Moreover, at the end of a transient simulation, S3K compares individual fuel pin temperatures and enthalpies to the fuel safety criteria provided by the user. Detailed information is then summarized, including core-wise, assembly-wise, maximum and limiting values, and the total number of fuel pins failing the safety criterion.

#### IV.D. RELAP5 Model

The RELAP5<sup>5</sup> computer code is formulated upon a transient, one-dimensional, two-phase flow model allowing both thermal non-equilibrium and non-homogeneous two-phase flow. The field equations are tightly coupled by efficient, semi-implicit or “Courant violating” nearly-

implicit numerical solution schemes. As shown in Table I, a comprehensive two-phase flow regime map (e.g., bubbly, slug, mist, etc.) has been formulated to span all possible operating regimes. Once the local nodal flow regime has been determined, constitutive equations are used to compute interfacial surface area, heat transfer coefficients, and interphase drag or slip. Likewise, wall-to-fluid heat transfer and friction coefficients couple the fluid behavior to structures. The code includes many “specialized components” like pumps, accumulators, valves, turbines, and steam separators. RELAP5 contains other special process models like critical flow, non-condensable gas transport, thermal stratification, and water level tracking.

TABLE I  
 RELAP5 Flow Regime Map for Thermal-Hydraulic  
 Constitutive Models

HYDRAULIC FLOW REGIME “MAP”						
VERTICAL			HORIZONTAL			CORE
Pre-CHF	Post-CHF	Transition	Stratified	Low Flux	High Flux	
Bubbly	Inverted Annular	Linear Weighted Transition	Wavy Surface	Bubbly	Bubbly	EPRI Void Fraction Correlation (Chexal-Lellouche)
Slug	Inverted Slug			Slug	Slug	
Annular	Mist			Annular Mist	Annular Mist	

RELAP5 solves the complete set of conservation laws for two-phase fluid dynamics. Specifically, the conservation of mass, energy, and momentum are ensured for the liquid and vapor components individually within each node (control volume). This produces a true “six-equation,” full two-velocity non-equilibrium hydrodynamic model. Mathematical closure results from the flow regime dependent interfacial phase coupling equations. The six-equation set of field equations results in a numerically robust model that ensures the conservation of vapor mass, liquid mass, vapor energy, liquid energy, vapor momentum (vector), and liquid momentum. A self-consistent set of interfacial closure equations determines the exchange of mass, momentum, and energy between the vapor and liquid fields at each node. Additionally, transport equations are solved for the conservation of non-condensable gas and of material species (e.g., boron).

The above equation set represents the most comprehensive possible model for transient two-phase flows. Depending upon the flow regime in the node, the phases may be completely coupled or completely decoupled, both in relative motion and energy interchange. It is precisely because of this “first principles approach” that RELAP5 has truly predictive capability. This results in a high-fidelity simulation of all transient behavior under normal, abnormal, and emergency situations.

RELAP5 has a general purpose heat conduction model to calculate the energy state (temperatures) of metal mass (structures) throughout the system. Based upon Fourier’s

Law, the model is transient and generally one-dimensional with three possible coordinate frames (rectangular, cylindrical, and spherical) for each structure. RELAP5 heat structures may have variable thermo-physical properties and internal heat sources. Boundary conditions can be convective, boiling, radiative, adiabatic, specified temperature and/or specified heat flux. Structural (wall) boundaries can be coupled directly to the fluid flow field; this allows for representation of internal walls and plates, as well as the external environments of primary and secondary containment atmospheres.

The heat transfer coefficient at a structure's surface is computed with a very comprehensive "boiling map" that considers all possible heat transfer regimes, such as subcooled single phase, nucleate, CHF/DNB, transition, and film boiling. In addition, the influence of non-condensable gases on these regions is mechanistically considered. Table II illustrates the possible regimes of boiling heat transfer. The heat transfer regime map has been subjected to extensive experimental validation under a broad spectrum of typical reactor conditions and thermodynamic states (e.g., pressures). RELAP5's fuel rod model has been well validated to give accurate prediction of the parameters of interest in safety-related simulations, the peak clad temperature (PCT). Experimental programs serving to validate these models include Semiscale, THTF, LOFT, FIST, and CCTF.

TABLE II  
 RELAP5 Heat Transfer Model Regimes

HEAT TRANSFER REGIME "MAP"	
REGIME	CRITERIA—PURPOSE
Boiling	$T_w > T_{sat}$ Subcooled boiling Nucleate CHF Dryout Film Transition
Convection	$T_{sat}(P_{steam}) < T_w < T_{sat}$ Single phase Two-phase Natural convection
Condensing	$T_w < T_{sat}(P_{steam})$ Steam condensation
Noncondensibles	$X_{NC} > 0$ Reduced heat transfer coefficient due to presence of noncondensable gas
Thermal Radiation	$\epsilon\sigma T^4$ Wall-to-wall interchange

#### IV.E. S3K/RELAP5 Linkage Code

The S3K and RELAP5 linkage is a direct, explicit coupling of the two codes on a synchronous time-step basis. The coupling provides an execution method for the S3K three-dimensional neutronic model using the Nuclear Steam Supply System (NSSS) boundary conditions calculated by the RELAP5 code. Also, it allows the S3K

calculated total core power and core power distributions to drive the RELAP5 system model core.

Detailed calculations from the component codes result in a methodology for analyzing limiting transients such as steam line breaks, rod drops/ejections, and ATWS scenarios. These transient events require detailed three-dimensional core data and information about the behavior of NSSS components. A coupled analysis of these transients is important because the core behavior is closely tied to the NSSS system. For example, to capture the timing and characteristics of the important thermal-hydraulic phenomena and/or operations events, such as valve closures, safety injection, or control system interactions, requires a detailed plant model.

The coupling between S3K and RELAP5 is as follows. The thermal-hydraulic conditions in the core and plenum regions are passed to the S3K model, which performs a calculation of detailed core power, which is then passed back to the RELAP5 model to use for the next time step. There are different coupling options that are available for the linkage between RELAP5 and S3K. The so-called "nodal" coupling option, which is used for the analysis in this paper, does not use the S3K thermal-hydraulics calculation. However, an estimate of the three-dimensional moderator density and fuel temperature distributions is made utilizing the current nodal powers. The fuel temperature is estimated from the coarse value calculated by RELAP5 using a weight factor that is the ratio of the nodal power to the average power in the channel in that plane. The density for a given fuel assembly is calculated using a simple enthalpy rise calculation and the same weight factor described for the fuel temperature calculation. The density calculation also includes a normalization step that preserves the mass of liquid for each RELAP5 channel.

The Forsmark 3 RELAP5 model accurately represents internal vessel components and is based upon design data to the maximum extent possible. Nuclear fuel rods are modeled by subdividing their axial length into regions corresponding to the core hydrodynamic nodes. Eight radial conduction nodes are used to model the fuel pellet: one for the gas gap and two for the cladding, which yields sufficient detail to accurately model fuel thermo-physical and gap conductance phenomena. The RELAP5 channel assignments for the S3K model applied in the RIA calculations use 56 thermal-hydraulic channels centered on the region where the control rod drops. The S3K standalone calculation models all 700 bundles with regard to neutronic and thermal-hydraulic equations. Both S3K and RELAP5 use 25 axial nodes in each channel.

## V. EXTENDED SAFETY ANALYSIS

The standard safety analyses of BWRs include RIA. One group of RIA is the CRD events. The most important CRD is the event where a group of control rods are believed to be completely withdrawn. One rod, however, gets stuck between the fuel boxes and drops at a later occasion. The control rod falls with a speed determined by the gravitational constant corrected for buoyancy and water friction. Typical values are between 7 and 8.5 m/s<sup>2</sup>.

The RIA event becomes more severe for a given control rod worth if the operating condition is "cold," meaning that the core subcooling is high (~80°C) and the power is low (~0.001 %). However, the possible reactivity insertion in the "last rod in a group" CRD scenario is typically very small and prompt criticality is either avoided or is just barely reached.

At the typical reactor "warm up" condition, the power is around 1 to 2% and the subcooling is small (~1°C). Here, the reactivity release can be several dollars (\$) meaning that a prompt condition can take place with the potential of significant reactivity insertion. Due to the operating condition, both the Doppler and void reactivity feedbacks will directly mitigate the event shortly after the prompt criticality is reached. These two conditions, "cold" with high subcooling and "warm up" with low subcooling, define the possible outcome of the in-sequence CRD events, ranging from low control rod worth with a delay in the feedback to high control rod worth with prompt feedback.

The most important consequence of the two cracked rods and the observations of crack indications in the ongoing cycles is that the CRD scenarios described above change. The new information means that the old constraint -only control rods in the sequence where the control rod drive has been withdrawn can drop- is not valid anymore. Due to a complete break of the control rod shaft, *any* rod can drop at *any* time. However, note that due to geometrical constraints, the maximum drop can be one meter (from initially fully inserted). Also, it is important to mention that, the control rod radial position, within the inter-assembly space between the fuel boxes as well as the possible incline of the control rod at a condition of broken shaft, is not significantly altered compared to normal operation because: (a) the control blades are equipped with rollers that preserve at least a 2 mm radial accuracy, and (b) the guide tubes preserve the radial position below the core.

### V.A. Control Rod Worth – Cycle Specific Case

The control rod worth is defined as the reactivity that will be released, in a given situation, if one control rod is withdrawn (or drops out) from the core, as calculated by steady state codes like SIMULATE-3. In standard calculations, at "cold" conditions, the calculation is trivial.

If the control rod worth is below 1\$, the *transient* event will lead to a slow power increase based on the typical correlation between reactivity insertion and period. If the reactivity worth is above 1\$, the prompt condition will lead to a rapid power increase, where the maximum value depends on the reactivity that exceeds the fraction of delayed neutrons. At "cold" condition, the Doppler feedback is the mechanism that turns the power around. At "warm up" conditions, the fuel temperature, as well as the heat transferred to the coolant, will increase and subsequent voiding will take place. The Doppler and void reactivity feedbacks will counterbalance the inserted reactivity and a new steady state power level will be reached.

Typical results of a "cold" case, with a reactivity insertion just above the fraction of delayed neutrons, are shown in Fig. 6. This figure includes the total reactivity and its components (6a); fuel rod peak centerline, average and cladding temperatures (6b); the SRM/IRM detectors signals<sup>a</sup> (6c), and the peak radial averaged fuel enthalpy (6d). The total impact of a control rod withdrawal/insertion is defined in two components; "CRD" and "Flux". The "CRD" component represents the direct contribution from the change in cross sections. The "Flux" component represents the contribution from the change in 3D neutron flux distribution.

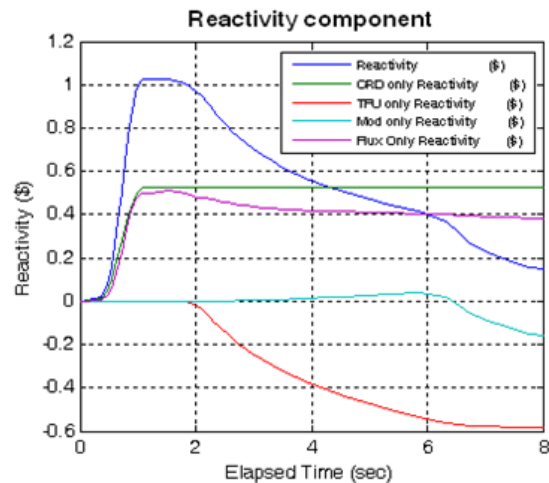


Fig. 6a. In-sequence - cold case: Reactivity components.

<sup>a</sup> SRM/IRM detectors are located in the top third of the core at different radial positions

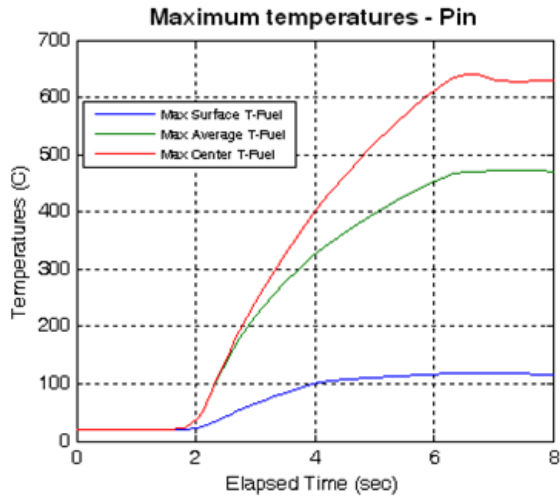


Fig. 6b. In-sequence - cold case: Peak fuel temperatures.

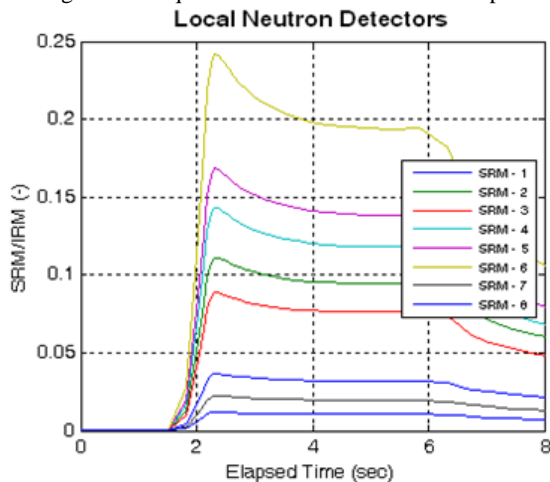


Fig. 6c. In-sequence - cold case: Neutron detectors.

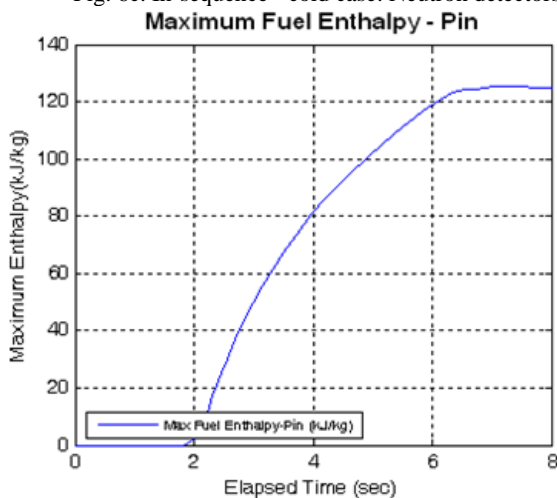


Fig. 6d. In-sequence - cold case: Peak radial avg. enthalpy.

The control rod drop takes place in the first second. The reactivity components (“CRD” and “Flux”) add up to the total reactivity (“Reactivity” in Fig. 6a). The inserted reactivity is a little above 1\$. The neutron flux starts to

increase, leading to the power increase, and becomes significant at 1.5 sec. Because of the power increase, the fuel temperature starts to increase (Fig. 6b) and the Doppler feedback (“TFU” in Fig. 6a) causes the net reactivity to decrease. The power is still high locally and the radial averaged fuel enthalpy (Fig. 6d) and the fuel rod temperature continue to increase. Finally, 6 s. into the simulation, the coolant voiding causes the reactivity to decrease further (see void component in Fig. 6a (“Mod”).

If the initial condition is more realistic, e.g., it resembles a typical reactor at “warm up,” the inserted reactivity may cause fuel temperature and void feedback shortly after prompt criticality is first achieved. This situation can lead to repeated prompt re-criticality conditions. Typical results of a CRD calculation at reactor “warm up,” with low subcooling are shown in Fig. 7.

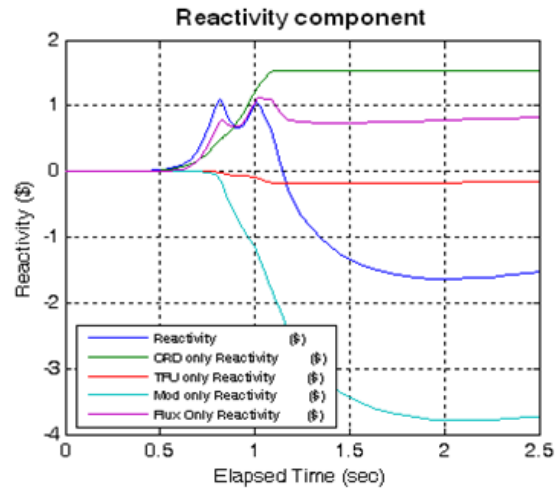


Fig. 7a. In-sequence - warm up case: Reactivity components.

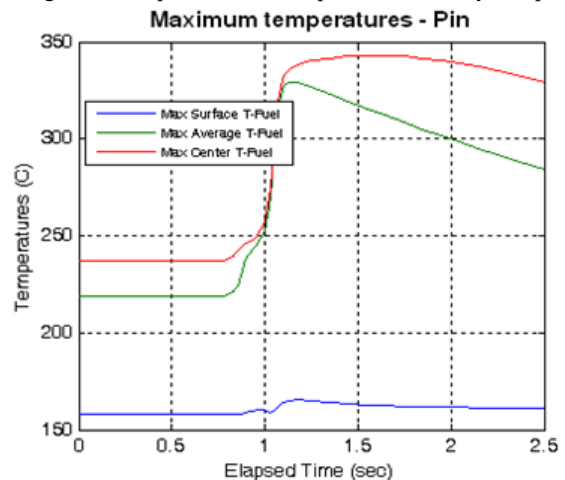


Fig. 7b. In-sequence - warm up case: Peak fuel temperatures.

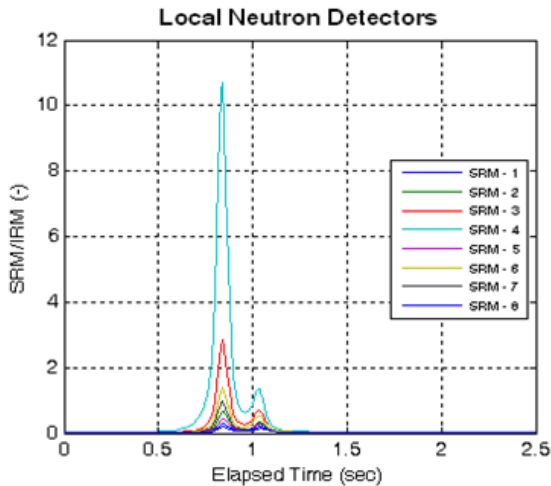


Fig. 7c. In-sequence - warm up case: Neutron detectors

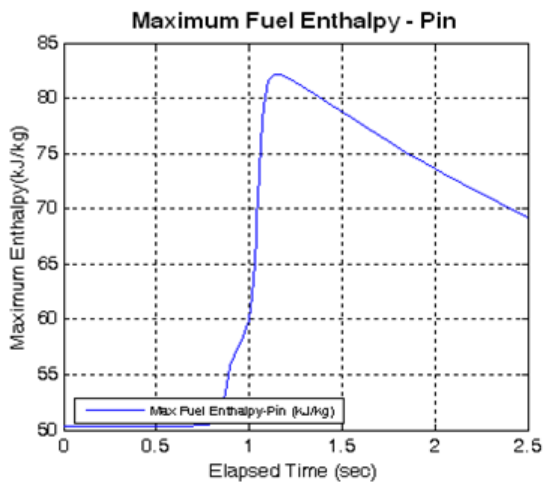


Fig. 7d. In-sequence - warm up case: Peak radial avg. enthalpy

The rod drops completely from full insertion, during the first second of simulation. The reactivity components (see “CRD” and “Flux” in Fig. 7a) add up to the total reactivity (“Reactivity”). The neutron flux starts to increase directly when the reactivity increase starts at around 0.5 sec. The power increase causes both the fuel temperature and the void fraction to increase. The fuel temperature increase (Fig. 7b) is moderate and the Doppler feedback (“TFU” in Fig. 7a) is not as important for the “warm up” case as for the “cold” one. Note that in “warm up” conditions the negative reactivity due to the void formation is the main feedback mechanism (see void component in Fig. 7a (“Mod”)). The continuation of the rod drop results in a second prompt criticality (see “Reactivity” in Fig. 7a and detector signals Fig. 7c).

In summary, the “warm up” case is characterized by a significantly higher rod worth compared to the “cold” case. However, the maximum radial averaged fuel enthalpy for the “warm up” case (~82 kJ/kg) is significantly lower compared with the “cold” case (~125 kJ/kg). Note that, in both cases, the radial averaged fuel enthalpy is below the

pin fragmentation limit 963 kJ/kg. The explanation is simple: in the “cold” case the Doppler reactivity is the only feedback mechanism so a significant fuel temperature increase is required to compensate the rod worth. In the “warm up” case, the void reactivity is the most important feedback which causes the event to be milder.

### V.B. Control Rod Worth – Out of Sequence

The two individual cracked control rods in Oskarshamn 3 and Forsmark 3 and the subsequent crack indications cause the rod worth scanning phase in the safety analysis to include all control rods, inserted or withdrawn. The maximum rod worth at each cycle depletion point is shown in Fig. 8. The maximum reactivity worth location in the core is shown in Fig. 9 (“-1” represents in-sequence control rods that have low reactivity worth). The limiting situation in the startup sequence is when half of the control rods are withdrawn.

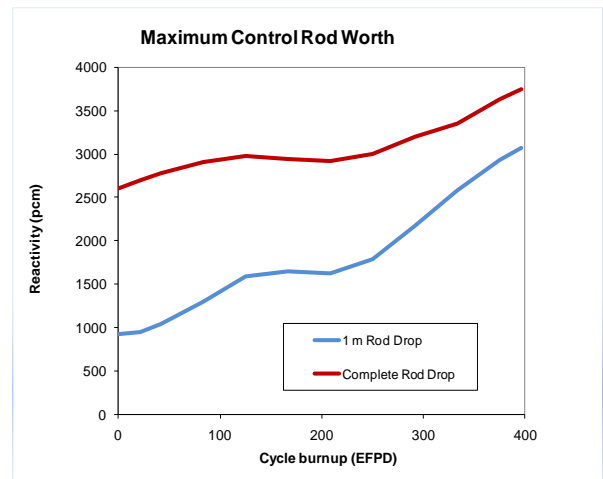


Fig. 8. Maximum rod worth in the cycle

The full stroke results in a maximum reactivity release of 3819 pcm which, at EOC, is between 6 and 7\$. The one-meter drop is just over 3000 pcm, which is approximately 5\$. The axial difference between the BOC and EOC reactivity worths is shown in Fig. 10.

Due to typical BWR cycle design of a one-year cycle, (bottom-peaked axial power shapes) the reactivity in the top third of the core is relatively high at EOC, which results in the observed axial reactivity worth distribution. In the EOC case more than 90% of the reactivity is released when the rod drops the first meter.

-1	0						
1578	-1	0	-1	0			
-1	2048	-1	2631	-1	0		
211	-1	594	-1	2868	-1	0	
-1	429	-1	1165	-1	3758	-1	
48	-1	66	-1	698	-1	0	
-1	29	-1	755	-1	2027	-1	0
3	-1	15	-1	112	-1	1554	-1
-1	121	-1	469	-1	1813	-1	0
53	-1	150	-1	509	-1	0	
-1	514	-1	1378	-1	2166	-1	
220	-1	594	-1	3525	-1	0	
-1	2424	-1	3849	-1	0		
1587	-1	0	-1	0			
-1	0						

Fig. 9. Maximum rod worth location during the cycle.

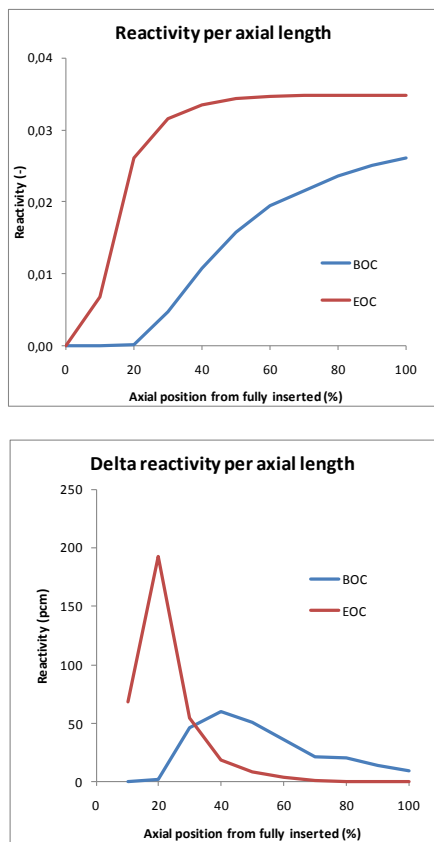


Fig. 10. Axial total and incremental rod worth.

V.C. Rod Drop Analysis – Out of sequence

The selected control rod, the time in the cycle when the analysis will be performed and the position in the startup sequence, are based on the control worth search presented in section V.B. Two operating conditions in the start up, as defined in Table III, were analyzed. The results presented in this section were obtained with S3K

standalone. Section V.D will discuss the S3K/RELAP5 results.

TABLE III

Conditions for the out of sequence CRD cases

Case	Pressure (MPa)	Subcooling (K)	Initial power (%)
Cold	0.3	40	1
Warm up	7.0	2	1

The “cold” case results are summarized in Fig. 11. The control rod drop takes place in the first 0.6 sec. The reactivity increase (“Reactivity,” “CRD,” and “Flux” in Fig. 11a) is similar to the in-sequence “cold” case (Fig. 6a). However the total inserted reactivity is much higher (~2\$). The neutron flux, power, and subsequent fuel temperature start to increase at 0.4 sec. The Doppler feedback (“TFU” in Fig. 11a) becomes important and mitigates the continuation of the control rod reactivity insertion. The significant power increase, and more specifically, the direct power deposited in the active flow, causes voiding at around 0.5 sec, which also contributes to the mitigation of the reactivity release (see void component in Fig. 11a (“Mod”)).

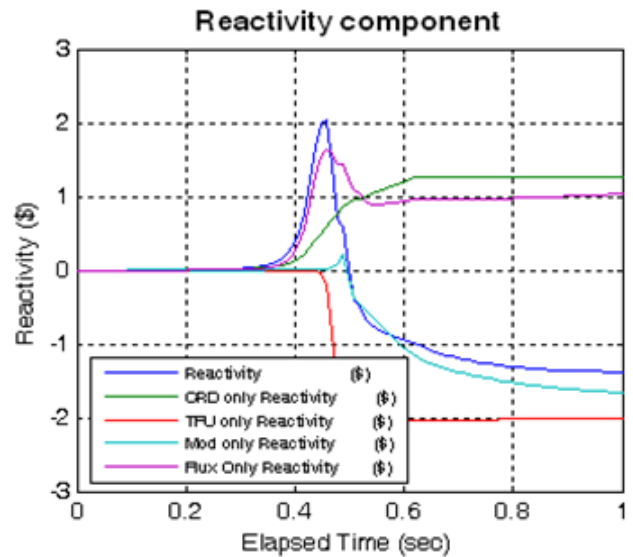


Fig. 11a. Out of sequence - cold case: Reactivity components.

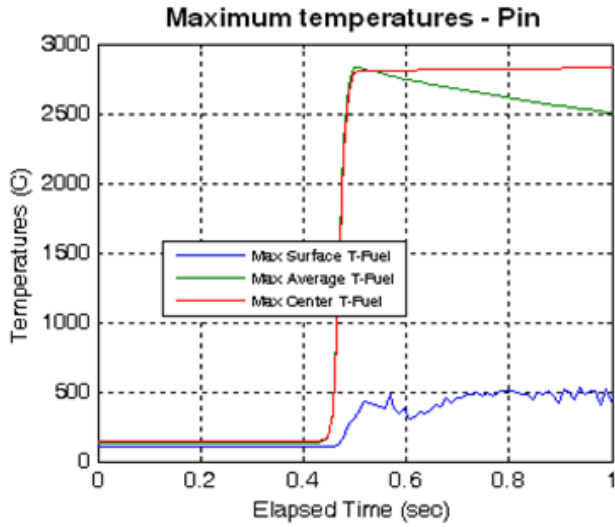


Fig. 11b. Out of sequence - cold case: Peak fuel temperatures.

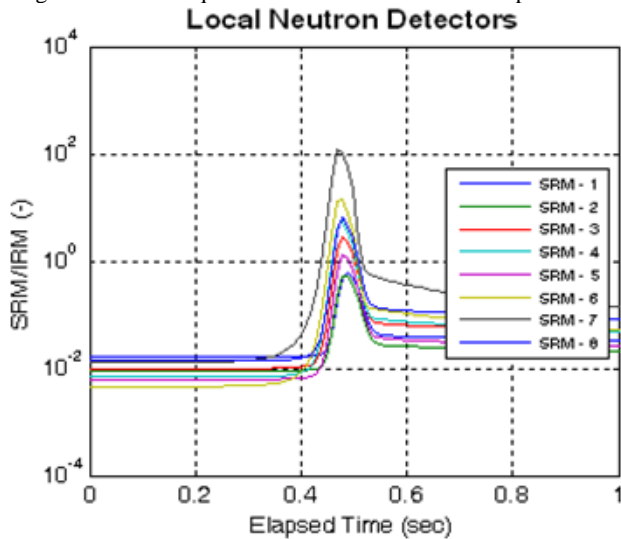


Fig. 11c. Out of sequence - cold case: Neutron detectors.

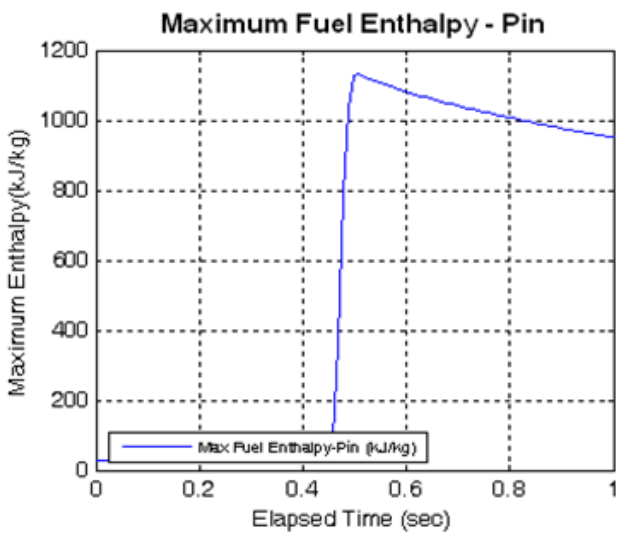
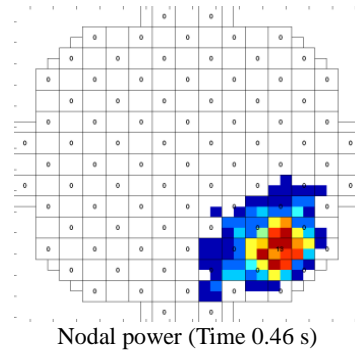
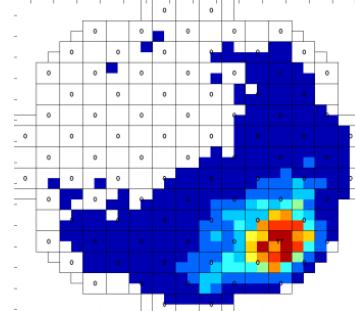


Fig. 11d. Out of sequence - cold case: Peak radial avg. enthalpy.

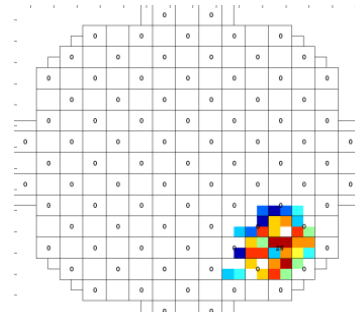
The radial distributions of important parameters during the transient, namely: nodal power, maximum peak pin enthalpy, pin cladding temperature, and peak centerline temperature, are shown in Fig. 12 (values below 10% of the maximum increase are marked white).



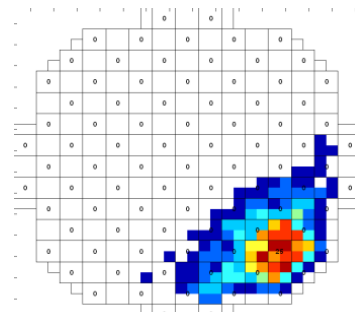
Nodal power (Time 0.46 s)



Peak pin radial averaged fuel enthalpy (Time 0.51 s)



Peak pin cladding temperature (Time 0.60 s)



Peak pin centerline temperature (Time 0.51 s)

Fig. 12. Out of sequence - cold case.  
 Radial distribution of key parameter.



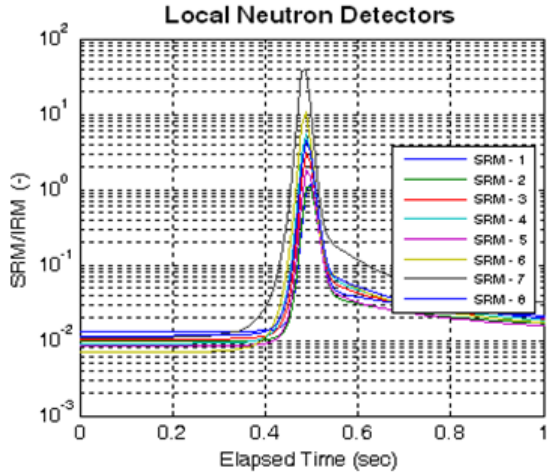


Fig. 15c. Out-of sequence - warm up case: Neutron detectors.

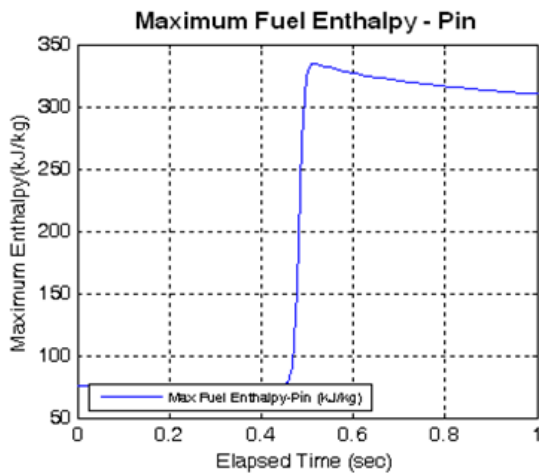
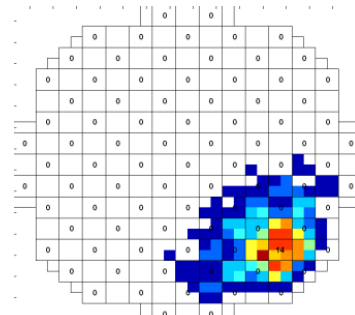


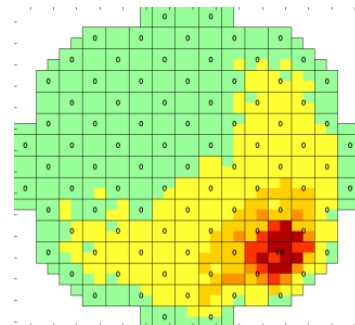
Fig. 15d. Out-of sequence - warm up case: Peak radial avg. enthalpy.

Note that the void feedback occurs early in the transient due to the low core inlet subcooling. Void feedback is the main reactivity mechanism that compensates the control rod reactivity insertion. The radial distribution of important parameters during the transient is shown in Fig. 16 (values 10% of the increase are white).

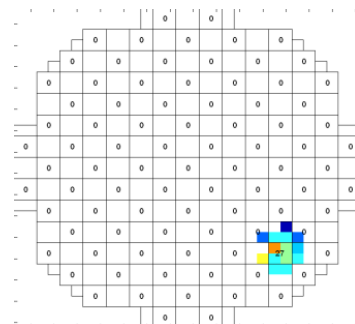
Nodal power, peak pin temperatures, and peak fuel enthalpy present a behavior similar to the “cold” condition. However, the peak radial averaged fuel enthalpy is higher initially, due to the higher water temperature, and the radial distribution becomes less peaked. The peak pin temperatures and enthalpies are about one third of the values for the “cold” case.



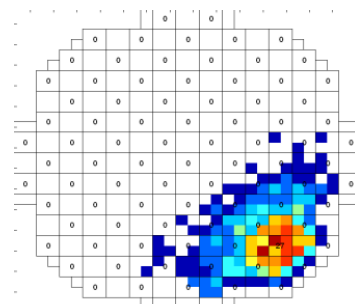
Nodal power (Time 0.46 s)



Peak pin radial averaged fuel enthalpy (Time 0.51 s)



Peak pin cladding temperature (Time 0.70 s)



Peak pin centerline temperature (Time 0.66 s)

Fig. 16. Out of sequence – warm up case.  
 Radial distribution of key parameter.

Fig. 17 compares the peak pin radial averaged fuel enthalpy for all the nodes in the core against the burnup dependent fuel fragmentation failure limit (limit that applies in Sweden).

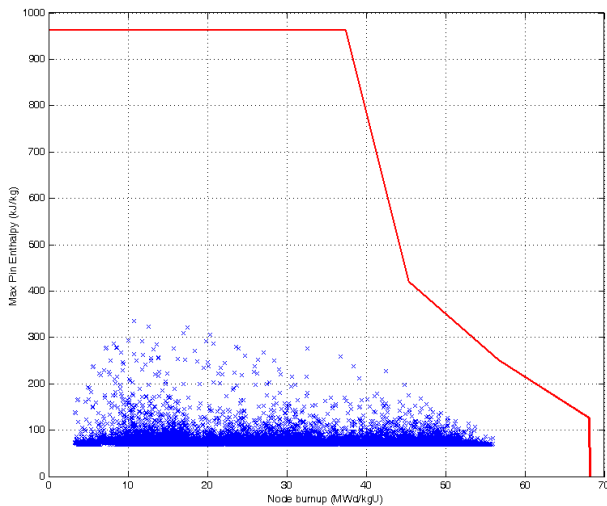


Fig. 17. Out of sequence – warm up case.  
 Peak pin radial average enthalpy vs. burnup.

Despite the fact that the “cold” case is, as expected, the most limiting scenario, the “warm up” case is still of interest because the majority of the startup time takes place under “warm up” conditions.

*V.D. Rod Drop Analysis – Local Pressure Evaluation*

For high rod worth RIA events, the amount of direct heat<sup>b</sup> is substantial and determines the initial part of the event. The direct power deposition is followed by heat transferred through the cladding. The direct heating causes the water to warm up and local voiding to take place. The local voiding is important for the evolution of the event. Moreover, the heat production may cause a local pressure increase that could limit the void formation, reduce the feedback mechanism and thereby increase the severity of the event.

Many codes used for standard RIA evaluations, including S3K, evaluate steam/water properties at one system pressure. To assess the influence of the local pressure buildup in the void formation and hence the void reactivity mechanism, the high reactivity cases, evaluated in the section V.C, were repeated with the coupled S3K/RELAP5 system.

The results for the “cold” case computed by S3K standalone and S3K/RELAP5 are compared in Fig. 18. The local relative power is plotted as a function of time for different axial nodes in the critical bundle in the core. It is important to remember that the three-dimensional core model is the same in both calculations. Only the thermal-hydraulic models differ.

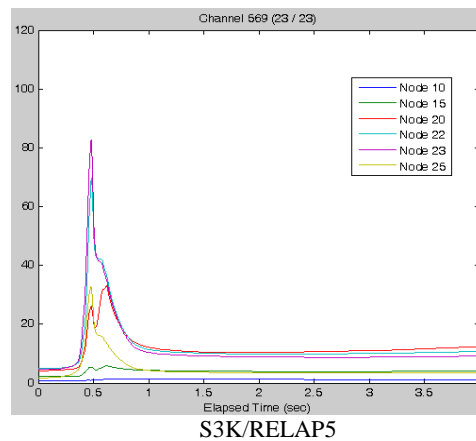
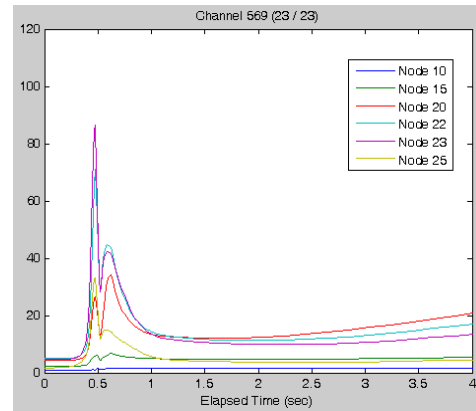


Fig. 18. Out of sequence - cold case.  
 Nodal relative powers in the critical bundle.

Based on Fig. 18 alone, it is obvious that the two different thermal-hydraulic models predict about the same power pulse with about the same integrated energy. The pressure evolution in node 23 of the critical bundle is shown in Fig. 19.

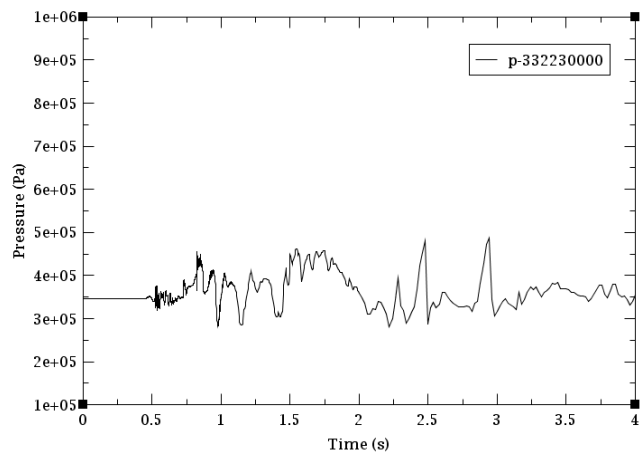


Fig. 19. Out of sequence - cold case.  
 RELAP5 pressure in node 23 in critical bundle.

<sup>b</sup> Roughly 2% of the generated power is deposited directly in the active coolant.

The rapid oscillations in local pressure are below 1 bar in amplitude. The impact on the void production is negligible.

The calculations were also repeated for the “warm up” case. Local relative powers for different axial nodes in the critical bundle in the core are compared in Fig. 20.

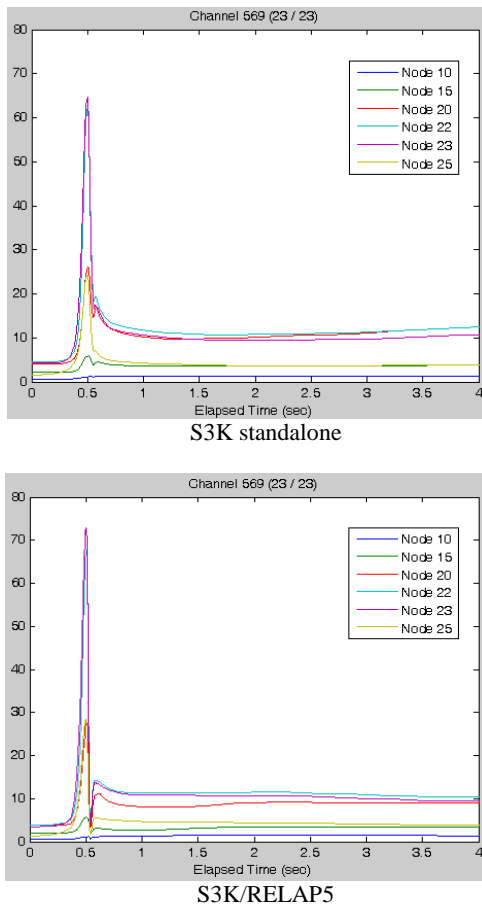


Fig. 20. Out of sequence – warm up case.  
 Nodal relative powers in the critical bundle.

Again, the two different thermal-hydraulic models predict essentially the same power pulse with about the same integrated energy. The pressure evolution in node 23 in the critical bundle is shown in Fig. 21.

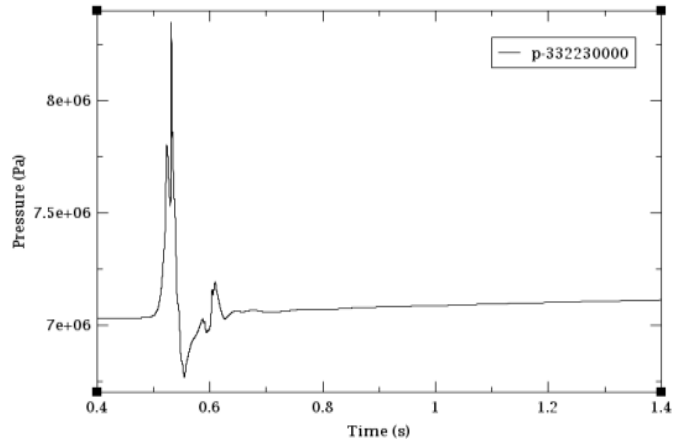


Fig. 21. Out of sequence – warm up case.  
 RELAP5 pressure in node 23 in critical bundle.

The power burst causes a rapid local pressure peak that was not present in the “cold” case. The pressure increase delays and reduces the voiding somewhat. The delay is small and the general power variations are almost unchanged. The integrated energy is about the same as well.

## VI. CONCLUSIONS

Observations of cracks in two control rods in Oskarshamn 3 and Forsmark 3 led to extended analysis of rod drop accidents, with the important requirement that, **any** rod can drop at **anytime**. A series of severe rod drop evaluations was undertaken and typical “cold” and reactor “warm up” scenarios were evaluated with S3K as well as with the coupled S3K/RELAP5 system.

The result demonstrates the applicability of S3K, either standalone or coupled with the safety analysis code RELAP5, to the class of severe reactivity insertion accidents. The detailed tracking of individual fuel pins, using the S3K Explicit Fuel Pin model was made to calculate the number of fuel pins that violates user-specified fuel enthalpy and temperature acceptance limits.

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