Assessment of damage domains of the High-Temperature Engineering Test Reactor (HTTR)

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A B S T R A C T

This paper presents an assessment analysis of damage domains of the 30 MW_{th} prototype High-Temperature Engineering Test Reactor (HTTR) operated by the Japan Atomic Energy Agency (JAEA). For this purpose, an in-house deterministic risk assessment computational tool was developed based on the Theory of Stimulated Dynamics (TSD). To illustrate the methodology and applicability of the developed modelling approach, assessment results of a control rod (CR) withdrawal accident during subcritical conditions are presented and compared with those obtained by the JAEA.

Keywords:

HTTR

Dynamic reliability

Modelling

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Simulation

Damage domain

1. Introduction

The High-Temperature Engineering Test Reactor (HTTR) aims at demonstrating possible advantages of high-temperature helium gas cooled reactors, which in conjunction with the production of conventional thermal electricity are also suitable for process heat, hydrogen and/or hydrocarbon production. The concept of coupling of high-temperature gas to the heat utilisation process is considered as a key aspect for Generation-IV Very-High Temperature Reactors (GIF, 2002) and has been developed in detail during several projects, for example the RAPHAEL Euratom 6th Framework Programme project (Hittner et al., 2006).

As it is customary, a standard deterministic safety assessment approach, included in the safety analysis report, is applied for HTTR (Saito et al., 1994) to identify representative initiating events and sequences. These are then grouped appropriately into individual classes within the Design Basis Envelope (DBEP) in order to demonstrate that the implemented prevention and mitigation measures are adequate.

Given the complexity of interactions in-between protection and mitigation actions as well as different physical phenomena and dynamic processes involved, it is a complex task to verify that the acceptance criteria are indeed met. To avoid possible erroneous grouping of individual event and sequences a new approach to tackle this complexity (Sánchez et al., 2009), has been developed within the regulatory framework applied in the Nuclear Industry. This approach, as presented in this paper, is based on the Theory of Stimulated Dynamics (TSD) and includes a platform of simulation tools for its implementation (Izquierdo and Cañamón, 2008; Izquierdo and Cañamón, 2006; Izquierdo et al., 2008; Labeau and Izquierdo, 2005a,b; Sánchez et al., 2009). This methodology presents an extension of a classical Probabilistic Safety Assessment (PSA), aiming at the identification of the fraction of transients belonging to the same sequence, but with a different end state in the PSA Level 1 (i.e., discriminating between “success” and “core damage” transients in the same sequence).

Abbreviations: ACS, auxiliary cooling system; AGC, auxiliary gas circulator; AHX, auxiliary heat exchanger; AOO, anticipated operational occurrence; ATWS, anticipated transient without scram; CR, control rod; DBEP, design basis envelope; EOS, equation of state; HTTR, High Temperature Engineering Test Reactor; IHX, intermediate heat exchanger; JAEA, Japan Atomic Energy Agency; PCS, primary cooling system; PDF, probability distribution function; PGC, primary gas circulator; PPWC, primary pressurised water cooler; PRMS, power range monitor system; PWCS, pressurised water cooling system; SHCS, secondary helium cooling system; SPWC, secondary pressurised water cooler; TSD, theory of Stimulated Dynamics; VCS, vessel cooling system.

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Nomenclature

Symbol Description and unit

α_j	fraction of neutron power generated in node j
β	fraction of delayed neutrons
ϕ_{nf}	fission power, W
ϕ_{fv}	power fuel-cladding, W
ϕ_{vc}	power cladding-coolant, W
λ	Decay constant, s^{-1}
Λ	mean generation time, s
ρ	reactivity
τ	time to complete one loop, s
ω	pump rotation speed, min^{-1}
C	precursors power, W
C_G, C_p	Heat capacity at constant volume or pressure, J/K
C_{pf}	fuel heat capacity, J/K
G	flow rate, kg/s
h	Enthalpy, J
j	volumetric flow, m^3/s
M^{loop}	mass in the loop, kg
n	fission power, W
n_d	steady-state power prior to the transient, W
P_A	probability of the initiating event A
P_{E1}	probability of failure of the system E1
p_{inlet}	pressure at pump inlet, MPa
p_{outlet}	pressure at pump outlet, MPa

P_{act}	decay power of actinides, W
P_d	fission product decay power, W
q	neutron source power
R	random variable
R_f	fuel resistance, K/W
R_g	gap resistance, K/W
R_C	cladding resistance, K/W
ss	steady state
t_s	elapsed time after scram, s
T_{inlet}	temperature inlet, °C
T_w	temperature of sink water, °C
T_f	fuel temperature, °C
T_C	cladding temperature, °C
T_{He}	Helium temperature, °C
Z_{bc}	control rod position, m

Symbol Description and value

Q	average energy released per fission, 200 MeV
E_U	average energy released by decay of ^{239}U , 0.474 MeV
E_{Np}	average energy released by decay of ^{239}Np , 0.419 MeV
λ_{Np}	decay constant of ^{239}Np , $3.41 \times 10^{-6} s^{-1}$
λ_U	decay constant of ^{239}U , $4.91 \times 10^{-4} s^{-1}$
R	generation ratio of ^{239}U during reactor operation 0.636

The developed methodology and modelling approach are generic and thus applicable to any reactor technology and concept. For illustration purposes, the HTTR was selected because of the comprehensive amount of information and data available to the authors.

In Section 2, the process and methodology of damage domain and safety margin assessments are briefly recounted, while the individual calculation modules of the developed HTTR5+/GASTEMP computational analysis tool are presented in Section 3. Damage domain and safety margin assessment results related to the control rod withdrawal transient under subcritical conditions in HTTR are given in Section 4. Section 5 presents conclusions and future outlook of our work.

2. Damage domain and safety margin assessment

The methodology of Safety Margin Assessment, as proposed in the OECD/NEA's Safety Margin Action Plan (SMAP) (SMAP Task Group, 2007), is based on the comprehensive identification of the Risk Space and the application of uncertainty analysis methods with the objective to determine exceedance frequencies of specified safety limits. The Risk Space can be understood as an extension of the Probabilistic Safety Assessment (PSA) event trees aimed at including every possible sequence of events (Izquierdo et al., 2009), which might challenge any safety limit. Given a sequence, i.e., an ordered set of discrete events, there are several or many possible plant transients matching the specified sequence and differing in the timing of the events or in the initial and boundary conditions. From this point of view, a sequence can be considered as a group of plant transients with common events occurring in the same order. Note that, in the context of this paper, a plant transient (or, simply, a transient) is any non-steady dynamic history, ranging from normal operations to accidents resulting in core damage.

In the context of classical PSA Level 1, event tree sequences are classified as “success” or “core damage” depending on whether the safety limits representing sequence success criteria are expected to be exceeded or not. In classical PSA Level 1, it is assumed that all

the transients grouped into a sequence have the same outcome with respect to whether safety limit was exceeded or not, i.e., no sequence is assumed to group “success” and “core damage” transients together.

However, the resulting end state of a sequence can differ since it depends on which particular transient in the sequence is considered (cf. Fig. 1). Lines at the bottom of Fig. 1 represent different transients – success (blue) or damage (red) – that belong to the same sequence ($P_A \times P_{E1}$) in the event tree. The Risk Space approach therefore aims at distinguishing between “success” and “core damage” transients belonging to the same sequence.

Given a specific safety limit, the objective of the uncertainty analysis applied to the Risk Space sequences is to calculate the contribution of each sequence to the limit exceedance frequency. When all these contributions for each sequence are aggregated, the exceedance frequency of the safety limit is obtained, which can be interpreted as a collective frequency of all possible plant transients resulting in consequences beyond the specified safety limit. As such, this analysis identifies to which exceedance frequencies the transient contributes to.

Even though the description of the Risk Space can be given in a format similar to PSA sequences and event trees, there are alternative formats that are more useful for the purpose of exceedance frequency calculations. To this aim, the concept of a dynamic sequence is introduced.

A dynamic sequence is an ordered set of discrete events that changes the plant dynamics.

The assessment of the contribution of a dynamic sequence to the exceedance frequency of a given safety limit consists of a twofold task: (1) damage domain identification, i.e., determination of the set of all possible “core damage” transients, which are grouped into one sequence, and (2) computation of the exceedance frequency sensitivity to the safety limits, i.e., determination of safety margins. The damage domain of a sequence (with respect to a safety limit) is, therefore, the set of transients that result in the exceedance of the analysed safety limit.

Fig. 1. Graphical representation of a sequence containing “core damage” transients and “success” transients.

Fig. 2. Simplified process to identify the damage domain and calculate the exceedance frequency.

The Fig. 2 describes the whole process.

The damage domain of anticipated operational transients (AOOs) should be empty, so no AOOs should exceed design safety limits (acceptance criteria) for any of the transients included in its categories. However, the examination of the damage domains sensitivity to setting of the safety limits is of paramount interest to verify the robustness of the design and its safety margins.

To minimise a number of unnecessary simulations an appropriate methodology for the identification of the damage domain is of primary importance. For this purpose, the use of very detailed, powerful simulation models is not necessary. In other words, lower fidelity models can be first applied to reduce the very large amount of transients to be analysed, improving the model fidelity and detail only in successive iterations. In fact, the larger the domain the lower the need for details, provided it can be verified that no success path damage found with a lower fidelity model turns out to be a damage path if simulated with a higher fidelity one. In view

of the possible large dimensions of damage domains this approach is in fact inevitable.

In this paper, we focus on the development of an adequate model for the first iteration of the damage domain analysis with the above-stated adequacy criterion. The model is subsequently used to determine a damage domain for one of the AOO of HTTR, namely the control rod withdrawal accident under subcritical conditions.

3. Modules of the HTTR5+/GASTEMP Code

This section describes the main modules of our in-house computational tool, called HTTR5+/GASTEMP, developed for the damage domain identification. The model uses a linear dynamic approach.

Its constituent part, the GASTEMP code (Benikhlef et al., 2012; Izquierdo and Sanchez, 2007), has been developed in the past to calculate enthalpy and mass balances in system components that may include a mixture of non-condensable and condensable gases. The corresponding component volumes may be heated by an external heat source and/or by internal chemical reactions.

In our case, there is just a single, non-reacting, non-condensable gas and an external heat source, which is due to nuclear fission and radioactive decay heat, as described in Sections 3.4 and 3.6, respectively. The external heat source is computed by the HTTR5+ sub-code, as described below.

3.1. Point kinetics

A single group point-kinetics model is implemented in HTTR5+/GASTEMP, with the kinetic parameters (cf. Table 1) taken from the ACCORD code (Takeda et al., 2000)

$$\dot{n} = \frac{dn}{dt} = \left(\frac{\rho - \beta}{\Lambda} \right) n + \lambda C + q \quad (1)$$

$$\dot{C} = \frac{dC}{dt} = \frac{\beta}{\Lambda} n - \lambda C \quad (2)$$

3.2. Heat transfer

HTTR is a graphite-moderated, helium-cooled reactor with a thermal power of 30 MW and a reactor coolant outlet temperature of 850 °C (Shiozawa et al., 2004).

The reactor cooling system consists of a primary helium cooling system (PCS), a secondary helium cooling system (SHCS), a

Table 1

HTTR core kinetic parameters (Takeda et al., 2000).

Δ [s]	λ_i [s ⁻¹]	β_i	$\beta_i \lambda_i$	λ [s ⁻¹]
7.32E-04	3.88	1.74E-04	4.48E-05	7.8996E-02
	1.4	8.39E-04	5.99E-04	$\lambda = \frac{\beta}{\sum_{i=1}^6 \lambda_i}$
	0.311	2.65E-03	8.52E-03	
	0.116	1.22E-03	1.05E-02	
	0.0317	1.37E-03	4.32E-02	
	0.0127	2.46E-04	1.94E-02	
Total		6.499E-03	8.23E-02	

pressurised water cooling system (PWCS) and a vessel cooling system (VCS), as schematically shown in Fig. 3.

The reactor cooling system of HTTR has two nominal operational modes. One is the single loop operational mode using the primary pressurised water cooler (PPWC) only, while the other is the coupled loop operational mode using both the PPWC and the He-to-He gas intermediate heat exchanger (IHX) to cool primary helium gas. In the single loop operational mode, the PPWC removes 30 MW of heat, while in the coupled loop operational mode, the PPWC and the IHX remove 20 and 10 MW of heat, respectively (Furusawa et al., 2004).

Fig. 4 shows the schematic block diagram of the different cooling loops indicating the five main systems & components of the main reactor cooling system (outlined by the black box). For transient situations occurring during the nominal operational mode with the IHX out of service, as in the example analysed in this paper below (Section 4), only these five indicated systems and components are modelled. The heat removal is then accomplished through the PPWC only.

In the accidental operational mode when only the auxiliary helium exchanger (AHX), the auxiliary gas circulator and water pump are available to remove decay heat, the cooling loop involving the core and the AHX loop are of interest and can be therefore simulated in HTTR5+/GASTEMP. Additionally, at nominal power operation, the IHX secondary helium circuit with the secondary pressurised water cooler (SPWC), gas circulator and water pump can be also treated in HTTR5+/GASTEMP. Thus, the basic calculation model for our simulations consists of five systems/components.

3.2.1. Core heat transfer: heat transfer to the primary He coolant

Schematic depictions of fuel compact, rod and assembly of HTTR are shown in Fig. 5. A coated fuel particle (CFP) consists of a micro-sphere of low-enriched (6wt% avg. U²³⁵) UO₂ with the TRISO (TRi-structural ISOtropic) coating. The CFPs are incorporated within a graphite matrix into fuel compacts. The fuel rod, which is composed of fuel compacts and a graphite sleeve (cladding), is inserted into vertical holes in the graphite block of the fuel assembly. Helium gas coolant flows through gaps between the graphite block and the rods (Nakagawa et al., 2004).

To model heat transfer from the fuel rod to coolant, we chose a classical, simplified model accounting for different thermal resistances of each component node, as shown in Fig. 6-A for the fuel rod and Fig. 6-B for a heat transfer tube of the heat exchangers.

The power transferred between two consecutive layers, for instance cladding, *c*, and fluid, He, is

$$\phi_{n-f} \propto n(t) \quad (3)$$

$$\phi_{c-He} = \frac{T_c - T_{He}}{R_c} \quad (4)$$

With residual decay heat treated separately, cf. Section 3.6, energy balance equations then yield

$$\dot{T}_f = -\frac{1}{C_f(R_f + R_g)}T_f + \frac{1}{C_f(R_f + R_g)}T_c + \frac{1}{C_f}n \quad (5)$$

$$\dot{T}_c = \frac{1}{C_c(R_f + R_g)}T_f - \frac{R_c + R_f + R_g}{C_c R_c(R_f + R_g)}T_c + \frac{1}{C_c R_c}T_{He} \quad (6)$$

with the gap conductance R_g incorporated in the equations. The fuel-clad gap is relatively small and contains gas, which compared to fuel and cladding have negligible heat capacity. This is therefore not taken into account, consistently with other simulation models to evaluate the high temperature gas-cooled reactor performance (IAEA, 2004).

If the core is divided axially into *n*-core nodes then

$$n_j(t) = \alpha_j n(t) \quad \sum_{j=1, n \text{ core}} \alpha_j = 1 \quad (7)$$

with α_j being the fraction of fission power generated in node *j*.

Fig. 3. Schematic depiction of the main systems and components of HTTR (Saito et al., 1994).

Fig. 4. Schematic diagram of the reactor cooling system of HTTR.

Fig. 5. HTTR fuel compact, rod and assembly (Nakagawa et al., 2004).

Fig. 6. Cross-section of a fuel element and heat transfer tube in the IHX and water coolers. *Scales: Usually oxide layers of corrosion products formed on inner and outer surfaces of heat exchanger tubes.

Table 2 shows the specification of fuel compact, graphite and coolant used in the model, other detailed values were taken from (Fujimoto et al., 2004; Furusawa et al., 2004; Takeda et al., 2000).

3.3. Main reactor cooling loop global mass and enthalpy balances

Without the loss of overall generality of our model we assumed nominal operational and transient modes in which the loop mass remains constant (i.e., loss of coolant situations are not considered, for example). In this case, the loop global mass- and enthalpy-balance equations read.

$$M^{loop} = G\tau \frac{dh^{loop}}{dt} = \phi_{c-He} + \phi_{He-w} \quad (8)$$

where ϕ_{c-He} and ϕ_{He-w} represent the power transferred from the fuel cladding to the He coolant and from the coolant to the water heat sink, respectively

$$\phi_{c-He} = \frac{T_c - T_{He}}{R_c} \quad (9)$$

$$\phi_{He-w} = -U(T_{He} - T_w) \quad U \equiv \frac{1}{R_{He-t}} + \frac{1}{R_{t-w}} \quad (10)$$

Table 2

Specification of fuel compact, graphite, fuel block and coolant (Fujimoto et al., 2004; Furusawa et al., 2004).

Fuel compact	
Type	Hollow cylinder
Materials	Coated fuel particles
Dimensions	
Outer/inner diameter	26/10 mm
Height of a compact	39 mm
Effective length of a fuel rod	546 mm (14 fuel compacts)
Graphite sleeve	
Type	Cylinder
Materials	Graphite
Dimensions	
Outer diameter	34 mm
Thickness	3.75 mm
Length	580 mm
Gap width between fuel compact and graphite sleeve	0.25 mm
Fuel block	
Type	Pin-in-block
Configuration	Hexagonal
Dimensions	
Width of across flats	360 mm
Height	580 mm
Fuel hole diameter	41 mm
Material	Graphite
Number of fuel rods in a block	31
Coolant	
Material	Helium gas
Pressure	4 MPa
Flow rate	12.4 kg/s
Inventory	2000 kg

As a boundary condition, water heat sink temperature T_w is assumed to be constant (473 K), considering a very large flow and available mass of water. In our modelling approach, transients are therefore assumed to be initiated on the primary loop side only.

The global power balance can be then determined as

$$\dot{Q}^{loop}(t) = \phi_{c-He} + \phi_{He-w} \quad (11)$$

and the system of equations composed of Eqs. (1), (2), (5), (6), and (11) then represented as

$$\begin{bmatrix} \dot{n} \\ \dot{C} \\ \dot{T}_f \\ \dot{T}_c \\ \dot{T}_{He} \end{bmatrix} = \begin{bmatrix} \frac{\rho-\beta}{\lambda} & \lambda & 0 & 0 & 0 \\ \frac{\beta}{\Lambda} & -\lambda & 0 & 0 & 0 \\ \frac{1}{C_f} & 0 & -\frac{1}{C_f(R_f+R_g)} & \frac{1}{C_f(R_f+R_g)} & 0 \\ 0 & 0 & \frac{1}{C_c(R_f+R_g)} & -\frac{R_c+R_f+R_g}{C_c R_c(R_f+R_g)} & \frac{1}{C_c R_c} \\ 0 & 0 & 0 & \frac{1/R_v}{G C_p \tau} & -\frac{(1/R_c+UA)}{G C_p \tau} \end{bmatrix} \begin{bmatrix} n \\ C \\ T_f \\ T_c \\ T_{He} \end{bmatrix} + \begin{bmatrix} 1 & 0 & 0 & 0 & 0 \\ 0 & 0 & 0 & 0 & 0 \\ 0 & 0 & 0 & 0 & 0 \\ 0 & 0 & 0 & 0 & 0 \\ 0 & 0 & 0 & 0 & \frac{UA}{G C_p \tau} \end{bmatrix} \begin{bmatrix} q \\ 0 \\ 0 \\ 0 \\ T_w \end{bmatrix} \quad (12)$$

which can be subsequently written in matrix notation as

$$\dot{X} = AX(t) + BC \quad (13)$$

3.4. Reactivity feedback

The reactivity feedback module takes into account reactivity coefficients of the HTTR, these being the fuel Doppler reactivity coefficient ($\frac{\partial \rho}{\partial T_f}$) and the moderator temperature reactivity coefficient ($\frac{\partial \rho}{\partial T_c}$) (Saito et al., 1994; Takeda et al., 2000). The model is

Fig. 7. Fuel Doppler reactivity coefficient.

valid for fuel and moderator temperatures between 25 and 1800 °C, the data were taken from (Saito et al., 1994; Takeda et al., 2000) and were fit to polynomial curves (Figs. 7 and 8). Also taken into account is the reactivity change due to a change in the control rod (CR) position, ($\frac{\partial \rho}{\partial Z_{cr}}$). The total reactivity change can be then determined with the linear dynamic approach as

$$\dot{\rho} = \sum \frac{\partial \rho}{\partial x} \dot{x} = \frac{\partial \rho}{\partial T_f} \frac{\partial T_f}{\partial t} + \frac{\partial \rho}{\partial T_c} \frac{\partial T_c}{\partial t} + \frac{\partial \rho}{\partial Z_{cr}} \frac{\partial Z_{cr}}{\partial t} \quad (14)$$

In comparison to the above-considered reactivity feedbacks, the He coolant reactivity feedback is small and therefore not taken into account in our model.

Figs. 7 and 8 show the fuel Doppler and graphite moderator temperature reactivity coefficients, respectively. The latter includes all graphite in the core geometry, i.e., in the fuel compacts, rod sleeves, as well as in the graphite block of the fuel assembly.

As specific heat of fuel compact is in itself a function of fuel temperature, its value is also updated accordingly during each time step.

In this respect, an inconsistency was found in the expression of the specific heat of the HTTR fuel compact as given in (Takeda et al., 2000). Eq. (15) shows the correct equation for the specific heat of the fuel compact

$$Cp_f = 4.187 \times 10^3 \left(\begin{aligned} &1.15689 \times 10^{-1} \\ &+ 5.38719 \times 10^{-4} (T - 273.15) \\ &- 6.27977 \times 10^{-7} (T - 273.15)^2 \\ &+ 3.82411 \times 10^{-10} (T - 273.15)^3 \\ &- 1.12556 \times 10^{-13} (T - 273.15)^4 \\ &+ 1.27705 \times 10^{-17} (T - 273.15)^5 \end{aligned} \right) \quad (15)$$

Fig. 8. Graphite moderator temperature reactivity coefficient.

Fig. 9. Horizontal arrangement of the HTTR core (Tachibana et al., 2004).

3.5. Reactivity insertion

In the analysis of abnormal events, a scram reactivity of 4.0% $\Delta k/k$ (4000 pcm) is used. The scram is performed by CRs positioned in the region of the replaceable reflector, cf. Fig. 9. Scram reactivity providing thus adequate shutdown margin was determined on the basis of the assumption that a pair of the second ring CRs in the reflector region having the maximum reactivity worth, is stuck and cannot be inserted. During operation, reactivity is controlled by the control rod system, which consists of 32 control rods (16 pairs, i.e., two rods moving always together) and 16 control rod drive mechanisms except for the case when the centre control rods are removed to perform an irradiation test.

The insertion curve of the scram reactivity as used in the safety analysis and the results of the CR insertion test are shown in Fig. 10. Position of the CRs at the onset of a scram is assumed to be such that it corresponds to the end-of-life (EOL) conditions (660 days), i.e., the CRs are in their highest withdrawn positions.

3.6. Decay heat

When the reactor scrams, the corresponding power is replaced by decay heat power source ($P_d + P_{act}$), which is implemented in the HTTR5+/GASTEMP code following the approach of the ACCORD for fission products and actinide decay heat as follows:

$$\frac{P_d}{n_d} + \frac{P_{act}}{n_d} = \frac{1}{Q} \sum_j A_j t_s^{-B_j} \theta_j + \frac{RE_U}{Q} \exp(-\lambda_u t_s) + \frac{RE_{Pu}}{Q} \times \frac{\lambda_u \exp(-\lambda_u t_s) - \lambda_{pu} \exp(-\lambda_{pu} t_s)}{\lambda_u - \lambda_{pu}} \quad (16)$$

Fig. 10. Insertion characteristics of scram reactivity used in analysis (Saito et al., 1994).

Table 3

Decay heat parameters in Shure's formula (Takeda et al., 2000).

t_s (s)	A	B
$0.1 \leq t_s < 10$	12.05	0.0639
$10 \leq t_s < 150$	15.31	0.1807
$150 \leq t_s < 4 \times 10^6$	26.02	0.2834
$4 \times 10^6 \leq t_s < \infty$	53.18	0.335

Parameters A and B depend on the elapsed time after reactor scram, t_s , and are shown in Table 3. The parameter θ_j is 1 if t_s is in interval of $t_{j-1} < t_s < t_j$, while it is zero otherwise.

3.7. Volumetric flows

To close the system of equations it is necessary to compute volumetric flows $j_k^{in}(t)$. This calculation is performed by the GASTEMP code (Benikhlef et al., 2012). The classical volume conservation equation has been given by Herrero and Izquierdo (2011) as

$$j_k^{out}(t) - j_k^{in}(t) = \pi_k \dot{Q}_k^{reg}(t) + \gamma_k \frac{dp_k}{dt} + \frac{dV_k}{dt} \quad (17)$$

$$\rho_{k+1} j_{k+1}^{in} = \rho_k j_k^{out} \quad \rho_k = \frac{m_k}{V_k} \quad \pi_k \equiv \left[\frac{\partial(1/\rho)}{\partial h} \right]_p \quad \gamma_k \equiv \left[\frac{\partial \ln \rho}{\partial \ln p} \right]_s \quad (18)$$

Helium equation-of-state (EOS) provides the pressure as a function of enthalpy and density. In (Takeda et al., 2000) a second order Virial series describe helium as a real gas. This Virial format is the format for EOS chosen in the GASTEMP code.

$$p = \frac{RT\rho}{Mol} [1 + B(T)\rho] \quad B(T) = 4.5 \times 10^{-4} \left[\frac{m^3}{kg} \right] + \frac{5.42}{1890 + T} \left[\frac{m^3}{kg} \right] \quad (19)$$

Coefficients π_k and γ_k are also determined from the EOS, the latter being well approximated by a constant.

3.7.1. Modelling of helium circulators

The above equations allow finding volumetric flows at all regions except at the circulator inlet and outlet, where the circulator characteristics need to be provided. According to (Takeda et al., 2000), the circulator characteristics can be described as

$$\frac{p_{outlet}}{p_{inlet}} = a + b \left(\frac{G \sqrt{T_{inlet}}}{p_{inlet}} \right)^3 + c \frac{\omega^2}{T_{inlet}} \quad (20)$$

When combined with the EOS and mass conservation at the circulator inlet and outlet, respectively, the above equation becomes (see Table 4)

Table 4
Characteristic parameters of the HTTR helium circulators.

Parameter	Value
a	0.9935665
b	$-2.094E^{-4} \left[\frac{s}{K^{1/2} m^2} \right]^3$
c	$6.09E^{-4} [Ks^2]$

$$\frac{p_{outlet}}{p_{inlet}} = \frac{T_{outlet} \rho_{outlet} (1 + B_{outlet} \rho_{outlet})}{T_{inlet} \rho_{inlet} (1 + B_{inlet} \rho_{inlet})} = \frac{T_{outlet} j_{inlet} (j_{outlet} + B_{outlet} G)}{T_{inlet} j_{outlet} (j_{inlet} + B_{inlet} G)}, \quad (21)$$

which provides the relation between the circulator volumetric flows at inlet and outlet for a given circulator mass flow rate (G) and angular speed (B) that are given as part of the transient scenario description.

Fig. 11 shows a reasonable fit (using Eq. (20)) to characteristic curve of the primary and secondary helium circulators displaying relationship between the invariant flow rate and relative pressures between inlet & outlet (Takeda et al., 2000).

3.8. Steady-state modelling

In our modelling methodology, “success” sequence transients start and finish at steady state conditions (which might, however, differ from each other). “damage” sequence transients will start at a steady state and finish when any damage indicator variable reaches its safety limit.

For any critical reactor at steady state, $\rho = 0$. Hence, solving Eq. (13) for the initial steady state gives

$$A(0)x_{ss}^{ini} + BU = 0 \quad x_{ss}^{ini} = A^{-1}(-BU) \quad (22)$$

4. Code application

4.1. Transient scenario

The control rod withdrawal transient is modelled by HTTR5+/GASTEMP, according to the scenario described in the safety analysis report of HTTR (Saito et al., 1994). As such, it is assumed that a pair of CRs is abnormally withdrawn due to an erroneous operation from a sub-critical condition (with a sub-criticality of $-4.14E-7 = -0.0414$ pcm) during a reactor start-up. As a consequence, reactor power will rise with accompanying increase of, among others, coolant temperatures and neutron flux, which

Fig. 11. Fit of Eq. (20) to the characteristic curve of primary and secondary helium circulators.

can be detected and the transient consequences appropriately mitigated. In case the reactor is subsequently scrammed, the decay heat shall be removed by the auxiliary cooling system (ACS). In the HTTR safety analysis report, this initiating event is considered as the anticipated operational occurrence (AOO).

4.2. Initiation and boundary conditions

For comparison purposes, the same initial and boundary conditions are assumed in our study as in Saito et al. (1994).

“The reactor power is $10^{-7}\%$ (0.03 W) of the rated power (30 MW). This is equivalent to a power level generated by a neutron source. The primary coolant flow rate is assumed to be corresponding to the high temperature test operation mode, 37 t/h (Furusawa et al., 2004). The initial temperatures of the fuel, moderator and primary coolant are all assumed to be 200 °C as the most conservative value in the state of low power operation from the view-point of the temperature feedback effect, since there is a region with positive reactivity feedback (see Fig. 8).

The reactivity coefficients used in the safety analysis are assumed to have the maximum value with respect to the temperatures of fuel and moderator, so that the maximum peak core power might be attained by assuming the most conservative temperature reactivity feedback effect.

The reactivity insertion rate initiated by the CR withdrawal is assumed to be 1.2×10^{-6} $\Delta k/k/s$ (0.12 pcm/s) with a CR speed of 1 mm/s, which is the boundary reactivity between situations with and without automatic scram, and which gives the maximum fuel temperature (Saito et al., 1994)”.

The list of other HTTR reactor scram signals is given in Table 5, but consistently with modelling approach chosen for our studies, only the scram signal based on the increase of the reactor power above 105.5% of the nominal (i.e., above 31.65 MW) is considered in our further analysis.

4.3. Acceptance criteria

Transient consequences are characterised by the damage that may occur, identified through process variables (damage indicators) in case they exceed pre-specified allowed limits (hereafter safety limits) during the transient. In Table 6 the safety limits (acceptance criteria) of concern for HTTR AOO are specified.

4.4. Transient modelling

Each transient simulation starts from steady state conditions and is terminated when one of three bellow-listed conditions is reached:

1. The fuel damage is predicted to occur (i.e., fuel temperature ≥ 1600 °C, cf. also Table 6);
2. Final steady state is reached (i.e., transient is a “success”/ safety transient);
3. The maximum time of simulation is exceeded (the solution does not converge).

The final (resulting) steady state is reached when these two conditions are met:

$$\rho(\infty) = 0 \text{ and } X_{n-1} \cong X_n \quad (23)$$

This means that all the variables (n , C , T_f , T_c and T_{He}) are calculated at the time t_n and their values are compared with the previous time step t_{n-1} . If the difference between X_n and X_{n-1} is equal or less than epsilon (which value user specifies in the input deck) and, at the

Table 5
Signals for reactor scram (Saito et al., 1994).

Signal for reactor scram	Value	
Reactor power at PRMS	High	105.5%
Primary coolant flow rate of IHX	Low	92%
Primary coolant flow rate of PPWC	Low	93%
IHX outlet temperature	High	410 °C
Reactor outlet coolant temperature	High	967 °C
Core differential pressure	Low	78%
Pressurised water flow rate PPWC	Low	87%
Differential pressure between PCS and PWCS	High	8.5 kg/cm ²
Differential pressure between PCS and PWCS	Low	1.5 kg/cm ²
Differential pressure between PCS and SHCS	Large	1.8 kg/cm ²
Secondary helium flow rate	Low	88%
Manual scram	-----	-----

same time, the total reactivity is less than delta, where delta is small, close to 0, then the convergence criteria will be met.

4.5. Code adequacy – single transient test results

The transient described in Section 4.2 was analysed and Fig. 12 shows a transient behaviour of the reactor power and maximum fuel temperature during abnormal control rod withdrawal in HTTR at sub-critical conditions, comparing results obtained by the HTTR5+/GASTEMP code and by JAEA using the ACCORD code (Takeda et al., 1996).

Despite the simplicity of the implemented model in HTTR5+/GASTEMP, our results appear to be acceptable, specifically with the view to use the code to preliminary identify HTTR damage domains. Shapes of transient curves obtained by HTTR5+/GASTEMP are very similar to the JAEA results and the steady state characteristics are also similar. In case of the HTTR5+/GASTEMP code, the calculated reactor power is 14.7 MW, while fuel temperature is predicted to be ca. 1080 °C. The corresponding JAEA values are 14.5 MW (for power) and 965 °C (for fuel temperature). The HTTR5+/GASTEMP code simulates this transient in 9 s, on a Pentium 4 personal computer @ 3.2 GHz with 3 GB RAM memory.

The difference between the fuel temperature obtained by HTTR5+/GASTEMP and the results of JAEA especially at 1000 s can be ascribed to the use of the simplified model in the HTTR5+/GASTEMP code. These simplifications include modelling in 1D, while using average heat transfers, heat capacities and heat resistances as shown in the Eqs. (5), (6), and (11). However, it is expected that the damage domain calculated with the HTTR5+/GASTEMP code includes the real damage domain. Our approach is thus a useful starting point to delimit the real damage domain, which can be later determined more accurately using best-estimate codes.

4.6. Simulating several transients of the same sequence

Using the set of initial and boundary conditions as given in Section 4.2, outcomes of five transients with different assumed control rod withdrawal velocities were first calculated, both without and with automatic scram protection assumed to be operational.

The results obtained for the reactor power are shown in Fig. 13, while fuel temperature is displayed in Fig. 14 and total reactivity in Fig. 15. As already discussed above, the fuel damage is assumed to occur when fuel temperature exceeds 1600 °C.

Fig. 13 shows the reactor power without (a) and with (b) automatic scram protection for different assumed control rod withdrawal velocities.

The Fig. 13(a) also shows a direct relation between the control rod withdrawal velocity and the time to reach the reactor power peak and asymptotic values.

The Fig. 13(b) shows the reactor power without automatic scram protection. For the transients with control rod withdrawal velocities 10, 7.8 and 5.6 mm/s the scram signal (increase of the reactor power above 105.5% of the nominal, i.e., above 31.65 MW) is reached and the scram is triggered.

Fig. 14(a) displays the fuel temperature for different values of control rod withdrawal velocities with two of the transients (VCR = 7.8 mm/s and VCR = 10 mm/s) reaching the damage value (fuel temperature ≥ 1600 °C). Note that although the transient with control rod withdrawal velocity VCR = 5.6 mm/s reach the scram signal (cf. Fig. 13(b)) this transient does not reach the damage value. This means that an adequate threshold value for the scram trigger was chosen. The minimum control rod withdrawal velocity to reach damage is, in this case, 6.8 mm/s (cf. Fig. 16). In other words, a safety margin exists between 5.6 mm/s when the scram is triggered and 6.8 mm/s when the fuel damage occurs.

Fig. 14(b) shows the fuel temperature for different values of control rod withdrawal velocities with the automatic scram protection available. Again, the trigger of the scram for the control rod withdrawal velocities 10, 7.8 and 5.6 mm/s can be observed. The fuel temperature remains below 800 °C, while the corresponding rate of increase is slow compared to the reactivity due to the effect of thermal inertia.

Fig. 15(a) shows the total reactivity inserted during the transient. Note that the maximum peak of reactivity is attained for 7.8 mm/s and not for the maximum control rod withdrawal velocity of 10 mm/s. This is due to the fact that at VCR = 7.8 mm/s graphite temperature remains most of the time during the transient in the temperature between 200 and 600 °C, where the graphite moderator temperature reactivity coefficient is positive (cf. Fig. 8) yielding the maximum observed increase of the total reactivity.

The Fig. 15(b) shows the total reactivity inserted but in this case with the automatic scram protection available. A moment at which the scram was triggered for three of the transients can be clearly seen, as evidenced by the strong insertion of the negative reactivity.

As next step, all transients in the assumed sequence were simulated. Fig. 16 shows the corresponding results, with each dot representing one of the analysed “damage” transients. The damage is predicted to occur when control rod speed exceeds 6.8 mm/s, i.e., the corresponding reactivity insertion rate exceeds $1.2 \times 10^{-6} \Delta k/k/s$ (0.12 pcm/s). For reactivity insertion speeds not exceeding 6.8 mm/s, the system is always predicted to reach “safe” steady state conditions.

4.7. Evaluation of the damage domain for the control rod withdrawal transient under subcritical conditions

Once we have established the adequacy of the model implemented in the HTTR5+/GASTEMP, the code can be used for its primary purpose, namely finding the damage domains.

In Figs. 17 and 18, we present results of the evaluation of the damage domain for the control rod withdrawal transient under subcritical conditions. The results were obtained with the postulated failure of the reactor scram system, either because the scram set-point was not reached, or the scram system failed, preventing control rod drop to the reactor core (ATWS, anticipated transient without scram).

As already discussed in Section 4.2, we only consider the high neutron flux (power) reactor scram set-point in our simulations. Furthermore, we only include the control rod withdrawal speed and the overall inserted reactivity worth as modelling parameters. The control rod bank is assumed to be initially totally inserted, which corresponds to the low power test start-up operation.

Table 6

Acceptance criteria for anticipated operational occurrence (AOO) for the HTTR (Kunitomi et al., 2004).

- The peak fuel temperature shall be less than 1600 °C
- Pressure on reactor pressure boundary shall be less than 1.1 times of maximum pressure in service (i.e., <5.28 MPa)
- Maximum temperatures of reactor pressure boundary are:
 - Less than 500 °C for the reactor pressure vessel (material: 9/4Cr-1Mo steel)
 - Less than 600 °C for heat transfer tubes in pressurised water coolers (material: austenitic stainless steel)
 - Less than 980 °C for heat transfer tubes in the intermediate heat exchanger (material: Hastelloy XR)

Fig. 12. Transient behaviour during abnormal control rod withdrawal at sub-critical condition (JAEA (Saito et al., 1994) vs. HTTR5+/GASTEMP).

Our results provide information about the sensitivity of the reactor protection system to the neutron flux (reactor power) scram settings and the associated safety margins. Two methods are used to determine the damage domain: random sampling method and parametric sweep method.

4.7.1. Random sampling method

The objective of the random sampling was to obtain information about the shape of the damage domain, for which purpose an initial random sampling of the two parameters of interest was performed. As such, pairs of parameter values were selected randomly from the pre-specified range for each parameter, namely, from 1 to 70 mm/s for the control rod speed (which correspond to the physical minimum and maximum withdrawal speed), and from $4.14\text{E}-6$ $\Delta k/k/mm$ to $1.242\text{E}-5$ $\Delta k/k/mm$ for the differential control rod bank reactivity worth. As for the latter, the maximum value is chosen to be an order of magnitude greater than the value used in Section 4.2, which was selected in order to demonstrate the robustness of the implemented methodology. The minimum value is selected such that no damage is reached during the transient.

The sampling function for the control rod withdrawal speed was obtained from the uniform probability distribution function (pdf), cf. Eq. (24)

$$V_{bc} = 69R + 1 \quad (24)$$

where R is a uniformly distributed random number between 0 and 1.

The sampling function for the inserted reactivity worth was obtained as the probability distribution function, cf. Eq. (25)

$$\rho_{bc} = 8.28\text{E} - 6R + 4.14\text{E} - 6 \quad (25)$$

Fig. 17 shows the damage domain using the random sampling method for a population of 1000 transients. Each dot in the figure

Fig. 13. Time response of the reactor power during the control rod withdrawal transient under subcritical condition for different velocities of control rod withdrawal (a) without scram; (b) with scram.

represents a damage transient. In total 413 transients finish in the damage state for this simulation.

Different simulations were performed by the random sampling method for six different transient population sizes. Table 7 summarises the results obtained and Fig. 18 shows the expected precision for each transient population size.

Fig. 18 displays the precision for each transient population size. There is a direct relation between the size of the population and the precision. The population size of 1000 transients was chosen as an adequate compromise between precision and size, for which set of simulations will be both reasonably fast and precise.

The population size of 1000 transients was subsequently chosen to perform 10 simulations. Table 8 summarises these results, out of which an average probability of exceedance (i.e., of damage) of $p = 0.3986 \pm 5.2\text{E}-4$ could be obtained.

In other words, this means that around 40% of transients belonging to the same sequence are “damage” transients; while other 60% are “success” transients.

In the classical probabilistic safety assessment the frequency of the damage occurrence can thus be overestimated as it is considered that all the transients belonging to a sequence have the same end state in terms of damage or no damage to the core. This example shows that only a fraction of transients resulting in core damage shall in fact be considered in the frequency of occurrence, for which it is necessary to multiply the frequency of sequence occurrence with the above-determined probability of exceedance, giving then a better estimate of frequency of safety limit exceedance.

Fig. 14. Behaviour of the fuel temperature during the control rod withdrawal transient under subcritical condition for different velocities of control rod withdrawal (a) without scram; (b) with scram.

4.7.2. Parametric sweep method

As a next step, using the data from the random sampling method, an average probability of exceedance was determined by the parametric sweep method. For each of the two parameters (i.e., CR bank speed and reactivity worth), maximum, minimum and user specified step gradient values were provided to the HTR5+/GASTEMP code. The simulation started by calculating the transient with both parameters at their maximum values. Then, the control bank worth was decreased in steps while maintaining the value of CR bank speed. This simulation continued until a “no-damage” transient was obtained (i.e., the boundary of the damage domain was reached) or the minimum CR bank reactivity worth was reached. The CR bank speed was then decreased by a delta step, kept constant and the bank worth sweeping was restarted again from its maximum value. The overall simulation process ended when additional damage transients were not expected or when both parameters reached their minimum values (Hortal, 2012).

For the parametric sweep method, the maximum and minimum values of each of the two parameters and the step gradient, i.e., the value by which the parameter is decreasing in each simulation step, need to be selected. A small step gradient parameter gives a finer mesh, but requires a larger number of simulation steps.

Figs 19–21 show different simulations with different step parameters.

Fig. 19 shows the damage domain obtained using the parametric sweep method, each dot in the figure representing a damage transient. For this group of simulations a reactivity step of $5\text{E-}7$ and velocity step of 3 mm/s were chosen.

Fig. 15. Time evolution of the total reactivity during the control rod withdrawal transient under subcritical condition for different velocities of control rod withdrawal (a) without scram; (b) with scram.

Fig. 16. Damage domain for the control rod withdrawal transient under subcritical conditions.

In order to reduce the grid of the mesh the velocity step was reduced from 3 to 2 mm/s. The corresponding results are shown in Fig. 20.

Finally, the grid of the mesh was also reduced using the reactivity step of $1\text{E-}7$ and velocity step of 1 mm/s, cf. Fig. 21.

The entire process can be summarised into a flow chart (see Fig. 22) as follows:

Fig. 17. Evaluation of the damage domain for the control rod withdrawal transient using the random sampling method.

Fig. 18. Expected precision for each transient population size for the control rod withdrawal transient under subcritical conditions using the random sampling method.

1. The method of the random sampling is used to determine the rough shape of the damage domain with possible damage islands;
2. Using the information obtained during step 1, parameter values corresponding to a transient with the smallest safety margin (i.e., the most penalising parameters) are determined;
3. The value of the pivot parameter (CR bank speed) is fixed and the sweep with the other parameter (reactivity worth) is started;
4. If a “damage” transient is predicted, this is recorded and the simulation continues with the decreased CR reactivity worth;
5. When a “no-damage” transient (i.e., “safety” transient) is predicted, the value of the pivot parameter (i.e., CR reactivity worth) is decreased by one step and the simulation continues to step 3;

Table 7
Summary of results for different transient population sizes.

Transient population size	Number of transients with damage	Ratio	Variance	Standard deviation (σ)	Error ($\pm 3\sigma$)	Simulation time
500	196	3.92E-01	4.79E-04	6.57E-02	$\pm 6.57E-02$	36 m 35 s
750	298	3.97E-01	3.20E-04	1.79E-02	$\pm 5.36E-02$	56 m 1 s
900	353	3.92E-01	2.66E-04	1.63E-02	$\pm 4.90E-02$	1 h 3 m 49 s
1000	406	4.06E-01	2.40E-04	1.55E-02	$\pm 4.64E-02$	1 h 13 m 53 s
10000	4013	4.01E-01	2.10E-05	4.90E-3	$\pm 1.47E-2$	12 h 26 m 8 s
50000	20011	4.00E-01	4.79E-06	2.19E-3	$\pm 6.57E-03$	2 d 3 h 30 m 3 s

Table 8
Summary of results for 10 simulations of with a population size of 1000 transients.

Simulation	No. of transients with damage	Probability of damage	Simulation time
1	422	0.422	1 h 17 m
2	391	0.391	1 h 15 m
3	356	0.356	1 h 12 m
4	429	0.429	1 h 13 m
5	400	0.400	1 h 19 m
6	395	0.395	1 h 13 m
7	412	0.412	1 h 15 m
8	364	0.364	1 h 13 m
9	397	0.397	1 h 10 m
10	420	0.42	1 h 12 m

Fig. 19. Evaluation of the damage domain for the control rod withdrawal transient using the parametric sweeping method. Reactivity step of $5E-7$ and speed step of 3 mm/s were used.

Fig. 20. Evaluation of the damage domain for the control rod withdrawal transient using the parametric sweeping method. Reactivity step of $5E-7$ and speed step of 2 mm/s were used.

Fig. 21. Evaluation of the damage domain for the control rod withdrawal transient using the parametric sweeping method. Reactivity step of $1E-7$ and speed step of 1 mm/s were used.

6. The entire search for the damage domain is stopped when the minimum value of the pivot parameter is reached (i.e., the whole domain was sampled) or when there are no more “damage” transients predicted.

Although the outlined process might seem slow, in fact it is not, because when the parametric sweep method is used, the simulation of each transient stops when the damage is reached. In case of the control rod withdrawal simulated in the present study, this

often occurs relatively early into the transient. On the other hand, the random sampling method simulates both “safety” as well as “damage” transients and is therefore slower.

In our case, the following data were used:

- maximum reactivity worth of $1.242E-5 \Delta k/k$;
- maximum speed of control rod withdrawal of 70 mm/s ;
- step parameter for the reactivity worth of $1E-7 \Delta k/k$;
- step parameter for the CR bank speed of 0.5 mm/s .

The search of the damage domain takes around 1 h and 40 min simulating in total 4604 transients, on a Pentium 4 personal computer @ 3.2 GHz with 3 GB RAM memory.

Fig. 23 shows the damage domain obtained from the parametric sweep method for the control rod withdrawal transient under sub-critical conditions.

As outlined above, the occurrence of damage is determined on the basis of exceedance of pre-selected acceptance criteria (fuel temperature, in our case) over the entire domain of studied physical parameters. In order to determine the probability of having a damage transient (i.e., exceedance frequency), the volume integral of the probability (frequency) density over the damage domain needs to be calculated. For this purpose, the lower border of the damage domain was fitted by a polynomial curve, Eq. (26)

$$f(x) = -9.407E - 14x^5 + 1.824E - 11x^4 - 1.322E - 9x^3 + 4.242E - 8x^2 - 4.841E - 7x + 9.009E - 6 \quad (26)$$

Fig. 22. Flow chart for the identification of the damage domain.

Fig. 23. Evaluation of the damage domain for the control rod withdrawal transient using the parametric sweeping method. Reactivity step of $1E-7$ and speed step of 0.5 mm/s were used.

Fig. 24. Border area of the damage domain for the control rod withdrawal transient using the parametric sweeping method.

Fig. 24 displays the entire border area of the calculated damage domain.

As already discussed above, our assumption is that all the transients have the same probability of occurrence, so the probability density functions are:

$$pdf(x) = \frac{1}{69} \quad (27)$$

for the speed of control rod withdrawal, and

$$pdf(y) = \frac{1}{8.28E-6} \quad (28)$$

for the CR reactivity worth.

The exceedance frequency can be then determined as:

$$pr = \int_1^{70} pdf(x) \left[\int_{f(x)}^{1.242E-5} pdf(y) dy \right] dx \quad (29)$$

which yields **$pr = 0.384$** .

Consistently with the results obtained by the method of random sampling, only around 40% of all transients that belong to the sequence are predicted to be “damage” transients while the remainder 60% are “safety/no-damage” transients.

5. Conclusions

In the scope of this work, a deterministic dynamic risk assessment code HTTR5+/GASTEMP was developed based on the Theory of Stimulated Dynamics. The code, capable of analysing damage

domains for the typical design basis accident initiators, was subsequently applied to the control rod withdrawal transient sequence in the HTTR reactor under subcritical conditions. The results of the transients calculated by the HTTR5+/GASTEMP code compare well with those obtained by JAEA showing viability of our approach, which at the same time is fast and efficient enough to sample the damage domain of the transient. For this purpose, two different sampling methods (random sampling method and parametric sweep method) were used to perform the search for the domain. From these results intended to first delimit the damage domain, the following conclusions can be drawn:

1. Consistency between the results obtained using the two different sampling methods is confirmed;
2. In case the reactor scram is not actuated when required, the damage is predicted to occur when axial differential bank reactivity worth equals or exceeds $7.014E-6$;
3. However, no transient path is found in the damage domain, which would not lead to the high neutron flux/power reactor scram signal (105.5% of the nominal). The damage domain is therefore expected to have a low, beyond design basis frequency as it requires failure of the reactor scram system;
4. The most unfavourable combination of axial bank differential reactivity worth and bank speed is $\frac{\partial \rho}{\partial z_{bc}} = 7.014E-6 [\text{mm}^{-1}]$ and $\frac{\partial z_{bc}}{\partial t} = 10 [\frac{\text{mm}}{\text{s}}]$, as a result of the combination of individual reactivity effects;
5. The probability/frequency of exceedance is predicted to be $pr = 0.3986 \pm 5.2E-4$ for the method of random sampling and $pr = 0.384$ for the parametric sweep method;
6. Calculation time for the random sampling method is longer than for the parametric sweep method, because the former treats “damage” as well as “safety/no-damage” transients;
7. In our case, only around 40% of all transients that belong to the control rod withdrawal under subcritical condition sequence in HTTR are predicted to be “damage” transients while the remainder 60% are “safety/no-damage” transients.

As such, our model shows that frequency of damage occurrence can be indeed overestimated in the classical probabilistic safety assessment and in fact only a fraction of sequence transients, which are resulting in core damage, shall be considered in the corresponding frequency of damage occurrence.

For the reasons of code efficiency and speed a simplified modelling approach has been implemented in the HTTR5+/GASTEMP code to identify the damage domain for the sequence of control rod withdrawal under subcritical conditions, first for several different control rod velocities alone and thereafter, for both different control rod velocities and reactivities inserted.

In future, more precise models shall be implemented in the code to determine specifically the border region of the domain, which is of particular interest for assessments of safety margins. Also, the code shall be equipped to search for more complex multidimensional damage domains with more than two dimensions (parameters). It is important to note that such task will, however, become more difficult for each dimension added, with new variables and their combinations to be taken into account.

More detailed models and best-estimate codes can be used to refine the border of the damage domain to obtain higher accuracy of the results.

With appropriately adapted tools, the presented systematic methodology to identify damage domains could be applied to virtually any kind of design and/or safety assessment of nuclear reac-

tor systems. Its application will assist in increasing accuracy of predicting safety margins and likelihood of anticipated and beyond-design basis conditions.

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