APPLICATION OF GRS METHODOLOGY TO COOLANT VOID REACTIVITY UNCERTAINTY ANALYSIS IN THE ACR-1000®

A. Colton, A. Connolly, M. Ovanes
Atomic Energy of Canada Limited
Mississauga, Ontario
coltona@aecl.ca, connollya@aecl.ca, ovanesm@aecl.ca

ABSTRACT

The primary focus of this paper is to describe the methods used to calculate the uncertainty in the full-core, instantaneous, coolant void reactivity (CVR): this is defined as the change in reactivity from a fully cooled core at nominal conditions to a totally voided core, where the only change between the two states is the coolant density. This uncertainty analysis considers only equilibrium conditions. Therefore specific operational states (such as initial core, transition to reference core, and start-up after long shutdown) are outside the scope of the work. The ACR-1000 is designed to have a slightly-negative full-core CVR under nominal design-centre conditions so that if all the coolant is lost, core reactivity would still decrease.

The coolant void reactivity uncertainty analysis for ACR-1000 relies on an integrated approach based on the Best Estimate and Uncertainty (BEAU) analysis and Gesellschaft fuer Anglagen- und Reaktorsicherheit (GRS) method of uncertainty calculation, using the MCNP5[1] code. This analysis method identifies the sources of uncertainty that have an impact on the value of the safety margin parameter, determines their associated uncertainty range and distribution, and ranks them through a Phenomenon Identification and Ranking Table (PIRT) process. The sources of uncertainty considered are, for example, those associated with the boundary conditions in the analysis and core representation (modeling of the core). The identified uncertainties are then propagated through the analysis to provide an overall uncertainty assessment for the safety margin parameter, and the one-sided 95%/95% tolerance limit for the Figure of Merit (FOM) CVR is evaluated by the Ordered Statistics approach used in the GRS method.

I. INTRODUCTION

In the ACR-1000 design, the light-water coolant inside the fuel channel behaves both as an absorber and as a moderator, Figure 1 shows a picture of the ACR-1000 fuel channel. Upon coolant voiding, the spectral changes of the CANFLEX-ACR® lattice leads to a significant reduction in the thermal flux in the lattice. Figure 2 displays an ACR-1000 lattice cell. The 24 cm lattice pitch and the superior down scattering ability of the light-water compared to the traditional heavy-water used in natural uranium CANDU reactors

® ACR-1000 is a registered trademark of Atomic Energy of Canada Limited

® CANFLEX-ACR is a registered trademark of Atomic Energy of Canada Limited leads to more moderation in the coolant of the ACR-1000. Upon voiding of the coolant, this thermal-neutron source is reduced, resulting in a negative reactivity effect. The net reactivity change due to a postulated LOCA depends on the relative magnitudes of the absorption and moderation effects attributable to the H_2O coolant. The absorption effect depends primarily on the volume of H_2O in the fuel channel, whereas the moderation effect depends both on the volume of the H_2O and the moderator (D_2O) to fuel volume ratio. The ACR-1000 achieves a nominal negative full-core CVR by two distinct methods:

- Reducing the moderator/fuel volume ratio to enhance the moderation effect of the coolant in a given fuel channel design by reducing the lattice pitch, and
- Generating a negative reactivity component in a LOCA by using a large poison element in the center of the fuel bundle.

The ACR-1000 is designed such that the combined effect in the nominal configuration results in a slightly negative CVR when all of the fuel channels in the reactor are completely voided. Figure 3 shows a 2×2 array of channels in the voided state.

II. GRS METHODOLOGY

The GRS Method [2][3] was used in this case to estimate the one-sided tolerance limit on the coolant void reactivity for the ACR-1000. This approach is comprised of the following steps:

i. Identify the important parameters to model (with MCNP) when calculating the CVR, and for each parameter, determine the range of values this parameter may take.

ii. The "Ordered Statistics" methodology known as GRS is used to estimate the required tolerance limit for the margin parameter of interest, in this case, coolant void reactivity. Once a specific percentile for the limit is selected and a specific confidence level in the final statement is agreed upon, then the GRS methodology gives the minimum required number of code cases to run for the particular "order" of statistics chosen (e.g. "first-order" means "take the most extreme of the code predictions").

The parameters of importance for calculating coolant void reactivity with the MCNP code were identified, and ranked.

The analysis was an integrated uncertainty analysis, in that the uncertain parameters were all varied simultaneously to assess their impact on the selected parameter, the coolant void reactivity. There are two ways in which this has been done in the nuclear industry, namely the Response Surface Method, and the Ordered Statistics Approach (GRS Method).

In the Response Surface Method [4], the response of the parameter of interest to variations in the inputs is represented by a surrogate to the code. Based on the results of a number of code runs such a code surrogate (Response Surface) is usually fitted by a polynomial expression. The set of code runs is designed to consider carefully the uncertainty space of the input parameters and the possible interaction between them. The resulting Response Surface can subsequently be used to generate the distribution of the output parameter by randomly sampling the input parameters from their distributions.

An alternative to the Response Surface technique has been proposed by the German organization Gesellschaft fuer Anglagen- und Reaktorsicherheit (GRS) [5]. This method is based on the Wilk's formula [6], several examples of which will now be given.

The one-sided tolerance limits are given by the formula:

$$1-\alpha^{n} \geq \beta$$

Where $\beta \times 100$ is the confidence level (%) that the maximum code result will not be exceeded with the probability $\alpha \times 100$ (%) (fractile) of the corresponding output distribution and n is the required minimum sample size (number of MCNP calculation runs in the present analysis). As an example, suppose that the "95%/ 95%" confidence limit for CVR is required. In other words, what is the CVR value for which 95% of randomly calculated CVR values (given the input parameter uncertainties) would be smaller than this value, and furthermore, that there is 95% confidence in this statement. For the one-sided 95%/95% tolerance limit α is 0.95 in the Wilk's formula, and the value of n is found to make the left hand side of the formula greater than or equal to 0.95. The solution for n is 59. The number of required simulations (59) for 95%/95% is independent of:

i. The number of input parameters with uncertainties included in the GRS matrix;
ii. The form of distribution of the calculated output parameter; in fact there might not be any form of distribution expected.

An alternative result of the Wilk's method, for the one-sided confidence limit example, is now presented. If 93 code simulations are completed rather than 59, the highest predicted value can be omitted and the 95% confidence level in the 95th percentile is set by the second largest of the 93 CVR values (this is termed "2nd Order" in the GRS approach), see Figure 5 for further data on number of code runs required for each order. The 2nd order methodology of running 93 cases was used in the CVR uncertainty analysis. While running 93 cases (and using 2nd order statistics), versus 59 (and using 1st order), does not improve the "quality" of the tolerance limit in any way, the additional cases provide more code output data with which to examine trends.

III. REACTOR PHYSICS CODES AND SUPPORTING DATA LIBRARY

The suite of reactor-physics computer codes that are used to model the ACR-1000 is based on the industry standard toolset (IST) of codes established for current CANDU reactors. Certain codes have been upgraded with extensive enhancements for ACR-1000 applications. The suite of codes consists of RFSP [7], WIMS-AECL [8] and DRAGON [9]. A third-party code, MCNP is utilized for full-core voiding uncertainty calculations.

In order to apply state-of-the-art reactor-physics modelling capabilities, MCNP version 5 was selected as the main code for the simulation of the neutronic properties of the ACR-1000. MCNP applies nuclear data in a continuous energy approach representing in detail all of the ENDF/B-VI nuclear data used in the work and implements a detailed and accurate representation of the ACR-1000 core, and so is considered to be the most rigorous analysis approach available for this problem.

A second code used in the analysis was RFSP. The role played by RFSP in this analysis was simply to provide a basis for constructing a burnup distribution in the full-core MCNP model. The burnup values used in MCNP are based on a 14 channel grouping that was done using RFSP. Similar channels were grouped together to simplify the analysis and reduce the number of compositions given to MCNP. It was found that the impact on CVR calculations using channel grouping or different compositions for every channel is very small, about 0.004 mk. Therefore, this approximation is considered to have a negligible effect.

The WIMS-AECL code was used to generate fuel isotopics for the MCNP input cases, as well as to create the fuel tables for use in the RFSP full-core model. The WIMS calculations were done for a 30 year aged core at mid-burnup fuel composition. Also, the DRAGON code was used to generate the incremental cross sections of reactivity devices required for full-core calculations with RFSP.

IV. THE PIRT PROCESS

The CVR uncertainty analysis uses a methodology for estimating the one-sided tolerance limit on the coolant void reactivity for ACR-1000.

The parameters of importance for calculating coolant void reactivity with the MCNP code were identified, and ranked.

An expert panel was convened to identify sources of uncertainty that could impact the calculation of CVR by MCNP. Sources of uncertainty considered were those associated with the boundary conditions in the analysis, core representation (modeling of the core) and code modeling uncertainties.

Phenomena were ranked as follows according to their impact on the figure of merit, namely the CVR.

- High: Phenomenon has a controlling impact on the CVR
- Medium: Phenomenon has a moderate impact on the CVR
- Low: Phenomenon has a minimal impact on the CVR
- Inactive: Phenomenon has no impact on the CVR

Once the important phenomena were selected, specific parameters associated with these phenomena were identified. For each of the identified important parameters chosen for uncertainty propagation, an uncertainty range was assigned. Where possible, uncertainties were determined using existing relevant validation exercises or qualified previous assessments. When no such information was available, expert judgment was used to determine an appropriate uncertainty range and distribution based upon CANDU operational experience. Some parameters were correlated due to physical association with another parameter.

For example, the design-centre value for moderator purity in the ACR-1000 is 99.78 wt% D_2O. In current CANDU reactors the moderator purity is chemically controlled within +/- 0.05wt%. In this analysis, the chosen range encompasses +/-0.1wt% to account for additional uncertainties in the code calculations as well as special operational conditions not explicitly accounted for. This range of ±0.1 wt% was implemented in the uncertainty matrix as a uniform distribution.

Using the GRS methodology it was established that to obtain the one-sided 95%/95% tolerance limit (using 2^{nd} order statistics), 93 MCNP full-core cases were required for each configuration (cooled, voided). Each of the 93 cases corresponds to one row in the GRS matrix. The columns of the matrix correspond to those parameters included in the statistical analysis that were identified through the PIRT process and sensitivity studies. Also, for each configuration, an unperturbed or base case was run. The base case is an ACR-1000 full-core model with no variation of the parameters.

Calculations were performed with MCNP to find the k_{eff}, and its estimated standard deviation, σ. The CVR standard deviation related to sampling for each of the 93 cases and the reference case is defined as:

$$\sigma_{CVR}^2 = \frac{\sigma_{COOL}^2}{k_{COOL}^4} + \frac{\sigma_{VOID}^2}{k_{VOID}^4}$$

After using MCNP to calculate CVR for each case, a table with 93 values is assembled. Each CVR value has a standard deviation calculated using the definition given above.

The GRS methodology then guarantees that the second highest value of the 93 CVR cases has a 95% probability with a 95% confidence interval.

V. GENERATION OF THE GRS MATRIX

The parameters used for the reference core in the GRS method are described in Figure 4. In total, there are 15 parameters, whose values were varied throughout the cases run.

Given that up to 93 GRS code run cases could be performed, Microsoft Excel was used to set up 93 cases (that is, values for the 15 identified parameters spread over the 93 cases). Initially, the 93 × 15 block of random numbers were generated using the uniform distribution, in the range 0 to 1. This initial “block” of uniform distribution random variables (a block of 93x15 numbers between 0 and 1 in the current case) was checked to ensure that the correlation coefficient, R, (between the 15 parameters over the 93 cases) was reasonably small.

Following this, a simple transformation for each parameter mapped the {0, 1} uniform distribution value into a mapped value in the range {min max}, where “min” or ‘max” is the minimum or maximum value for the parameter in question. Several of the parameters are “discrete”, taking on the values {1, 2} or {1, 2, 3}, each possible value being equally likely. In the case where either “1” or “2” is required, the {0, 1} uniform distribution random variable (call this “*x*”) was mapped as follows:

- For $x < ½$ map to “1”

- For $x \geq ½$ map to "2"

A similar mapping was used for the {1, 2, 3} case (with 1/3, 2/3 and 1 being the boundaries used to determine which of the three cases is selected).

VI. RESULTS AND ANALYSIS

The GRS methodology was applied to obtain the amount of MCNP runs required to predict the 95%/95% tolerance limits that there is a 95% confidence level that 95% of randomly generated cases are below the second highest value of CVR. The MCNP input cases were prepared using 93 randomly sampled sets of the uncertain parameters; 15 parameters were included in the present analysis.

The 93 sets of the parameters used to generate the 93 MCNP input cases (for cooled and voided configurations) were used to predict the 95th percentile values of the output parameters for the 93×3 cases within a 95% confidence interval. This 95%/95% tolerance limit is represented by the second highest value of CVR, derived from the k_{eff} values obtained from the 93 sets of MCNP runs.

A multivariable linear fit was used to analyze trends occurring as a result of varying the 15 PIRT parameters; this approximation is of the following form:

$$y = ax_1 + bx_2 + cx_3 ... + \varepsilon$$

Where {x_1, x_2, …, x_{15}} are the 15 PIRT parameters

{a, b, c, …, o} are the 15 corresponding slopes for each of the 15 parameters

{ε} is the intercept

{y} is the approximation of CVR as a function of the 15 PIRT parameters

A linear multivariate fit was chosen based on the small applicable range of each of the parameters. To determine which parameters have the greatest effect on coolant void reactivity, they were then rated based on the absolute value of the product of their standard deviation and their slope (and then normalized to 1). Figure 6 contains the sensitivity for each parameter. Note that the following parameters are the ones with the largest effect on CVR, and will be discussed in order from largest effect to smallest.

The moderator purity has the largest effect on CVR in this analysis. When deuterium isotopic is increased, it causes a significant reduction in the coolant void reactivity on the order of -7.6 mk/at% D_20. The operational range for the moderator purity is about +/- 0.111 at% D_20. The pressure tube inner diameter increases CVR by 19 mk per cm, note that the actual tolerance of the diameter is about +0.08/-0.0 cm. However, the modelling assumption is that the pressure tube inner diameter is the same everywhere in the reactor, while in reality each pressure tube would have a slightly different inner radius and facilitate a smaller coolant void reactivity effect due to averaging across the core.

Moderator temperature has a negative effect on CVR as it is increased. This effect is quantified in -0.20 mk/°C, and the predicted operational range of moderator temperature is +/-3°C about the nominal average operational temperature. The calandria tube outer diameter manufacturing tolerance has a range of +0.12/-0.00 cm and the resultant coolant void reactivity effect is -6.3 mk/cm. Pressure tube diametrical creep has an estimated range of +/-5%, (95%-105% of the nominal dimensions of the pressure tube), with an effect on CVR of +0.056 mk/%. Lastly, the central element material has a reactivity effect of -0.21 mk/wt% (Gd + Dy oxides), where the manufacturing tolerance is about +/-1 wt%.

REFERENCES

[1] T.E. Booth et al. (The X-5 Monte Carlo Team), “MCNP – A General Monte Carlo N-Particle Transport Code Version 5”, Los Alamos National Laboratory, LA-UR-03-1987, 2003.

[2] H. Glaeser, E. Hofer, B. Krzykacz-Hausmann, “GRS-Method for Uncertainty Evaluations of Thermal-Hydraulic Code Results”, RMPS Project, Contract No. FIKS-CT-2000-00073, Project No. FIS5-1999-00250, April 2002.

[3] H. Glaeser, E. Hofer, M. Kloos and T. Skorek, “GRS Analyses for CSNI Uncertainty Methods Study (UMS)”, in Report on the Uncertainty Methods Study, OECD Nuclear Energy Agency, REP/NEA-CSNI-R(97) 35 Volume 2, (1998)

[4] A.I. Khuri and J.A. Cornell, “Response surfaces: design and analyses”, New York: Marcel and Dekker, 1996.

[5] E. Hofer, “Probabilistische Unsicherheitsanalyse von Ergebnissen umfangreicher Rechenmodelle GRS-A-2002", 1993.

[6] S. Wilks, “Statistical Prediction with Special Reference to the Problem of Tolerance Limits”, The Annals of Mathematical Statistics, Vol. 13, No. 4 (Dec., 1942), pp. 400-409

[7] P. Schwanke, “RFSP-IST Version REL_3-04: Users’ Manual”, SQAD-06-5054, December, 2006.

[8] G. Jonkmans, “WIMS-AECL Version 3.1 User’s Manual”, ISTP-05-5115, August, 2006.

[9] G.Marleau, “A User Guide for DRAGON: Version DRAGON_000331, Release 3.04”, IGE-174 Rev 5, 2000.

FIGURES

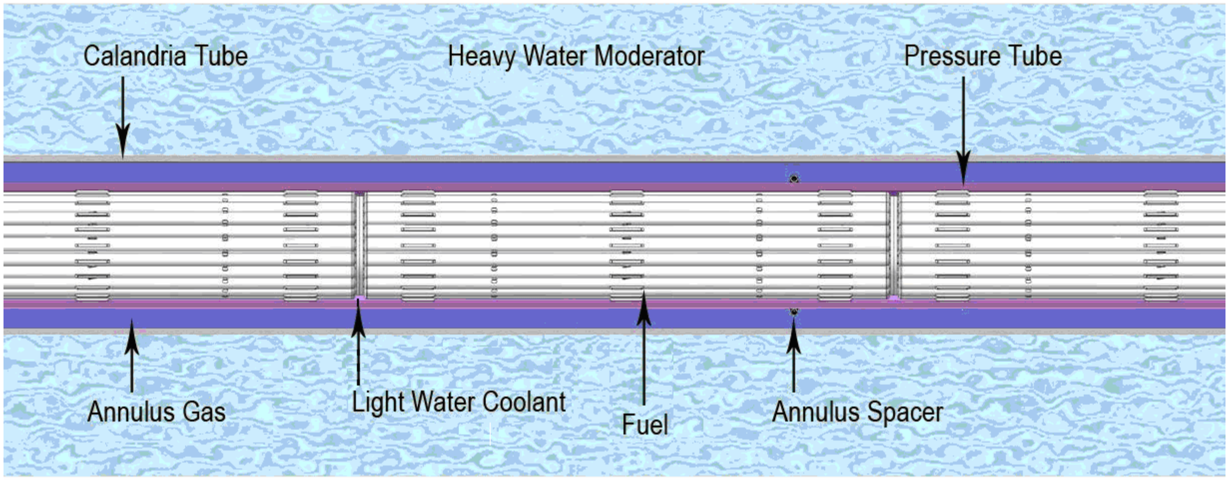

Figure 1 ACR-1000 Fuel Channel

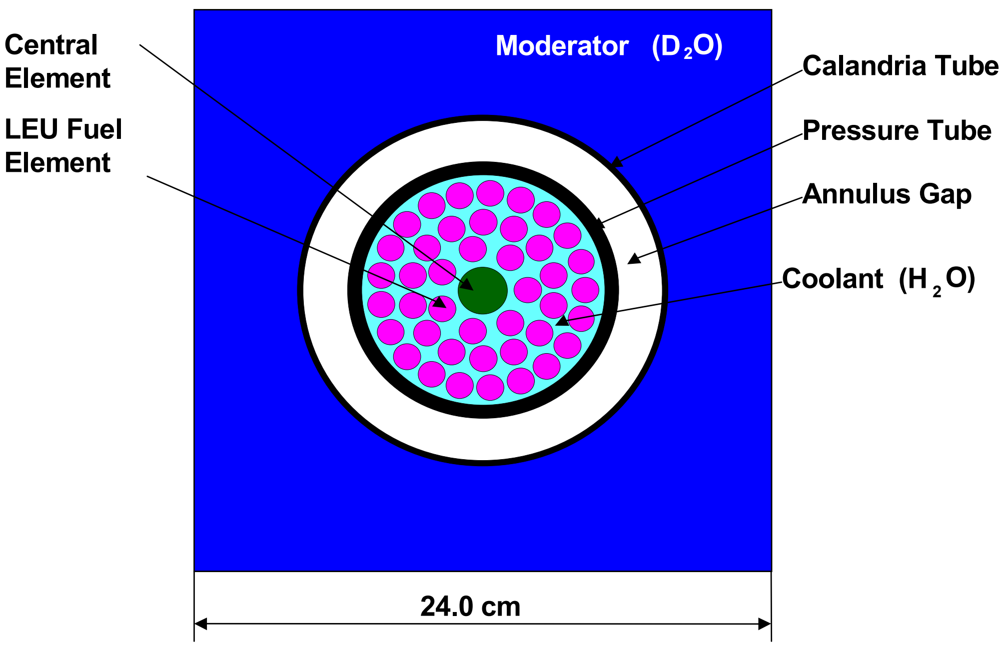

Figure 2 ACR-1000 Lattice Cell

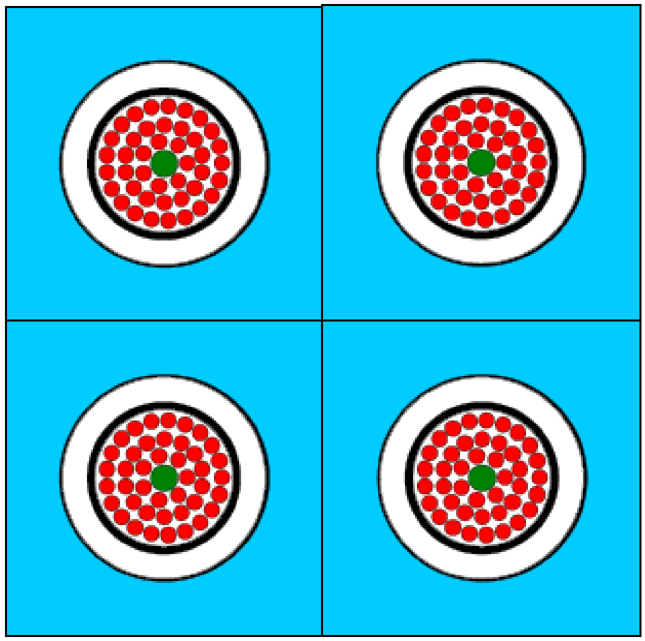

Figure 3 ACR-1000 Full-Core Void

Parameter Number	Parameter Name	Type	Units
1	Moderator Purity	Uniform	At%
2	Moderator Poison	Uniform	Gd ppm
3	Coolant Conditions	3 Distributions	%FP
4	Fuel Temperature	3 Distributions	%FP
5	PT Diametral Creep (+ Sag)	Uniform	% creep
6	PT Tolerance on ID	Uniform	cm
7	PT Material	Uniform	Nb wt%
8	CT Material	Uniform	Sn wt%
9	Fuel Sheath Clad Material	Uniform	Sn wt%
10	Central Element Total Poison	Uniform	Gd +Dy wt%
11	LEU level	Uniform	$^{235}U/U$ wt%
12	Gd Mass in Outer 42 Elements	Uniform	g
13	CT Tolerance on OD	Uniform	cm
14	Moderator Temperature	Uniform	°C
15	Central Element Temperature	3 Distributions	%FP

Figure 4 Parameters varied in coolant void reactivity uncertainty analysis

Total Number of Runs	Order	Runs Excluded from One Side
59	1	0
93	2	1
124	3	2
153	4	3
181	5	4
208	6	5

Figure 5 Minimum Number of Computer Code Runs Required for Various Ordered Statistics –One-Sided 95/95 Tolerance Limits

#	Parameter	Units	Multivariate CVR Sensitivity	Slope Units
1	Moderator Purity	at%	-7.66	mk/at%
2	Moderator Poison	Gd ppm	0.64	mk/ppm
3	Coolant Density & Temp	%FP	-0.03	mk/%FP
4	Fuel Temperature	%FP	-0.01	mk/%FP
5	PT Diametral Creep, PT/CT Sag	% creep	0.56	mk/% creep
6	PT Inner Diameter (Tolerance)	cm	19.23	mk/cm
7	PT Material	Nb wt%	0.09	mk/wt%
8	CT Material	Sn wt%	0.00	mk/wt%
9	Fuel Sheath Clad Material	Sn wt%	-0.04	mk/wt%
10	Central Element Material	Gd +Dy wt%	-0.21	mk/wt%
11	LEU Enrichment	$^{235}U/U$ wt%	0.15	mk/wt%
12	Gd mass in 42 Fuel Elements	g	-0.14	mk/g
13	CT Outer Diameter (Tolerance)	cm	-6.28	mk/cm
14	Moderator Temperature	°C	-0.20	mk/°C
15	Central Element Temperature	%FP	7.88e(-04)	mk/%FP

Figure 6 CVR Multivariate Sensitivity Results