Novak Zuber, Gary E Wilson, Mamoru Ishii, Wolfgang Wulff, B.E Boyack, A.E Dukler, P Griffith, J.M Healzer, R.E Henry, J.R Lehner, S Levy, F.J Moody, M Pilch, B.R Sehgal, B.W Spencer, T.G Theofanous, J Valente, An integrated structure and scaling methodology for severe accident technical issue resolution: Development of methodology, Nuclear Engineering and Design, Volume 186, Issues 1–2, 1 November 1998, Pages 1-21, ISSN 0029-5493, https://doi.org/10.1016/S0029-5493(98)00215-5.
(http://www.sciencedirect.com/science/article/pii/S0029549398002155)

Abstract:
Scaling has been identified as a particularly important element of the Severe Accident Research Program because of its relevance not only to experimentation, but also to analyses based on code calculations or special models. Recognizing the central importance of severe accident scaling issues, the United States Regulatory Commission implemented a Severe Accident Scaling Methodology (SASM) development program involving a lead laboratory contractor and a Technical Program Group to guide the development and to demonstrate its practicality via a challenging application. The Technical Program Group recognized that the Severe Accident Scaling Methodology was an integral part of a larger structure for technical issue resolution and, therefore, found the need to define and document this larger structure, the Integrated Structure for Technical Issue Resolution (ISTIR). The larger part of the efforts have been devoted to the development and demonstration of the Severe Accident Scaling Methodology, which is Component II of the ISTIR. The ISTIR and the SASM have been tested and demonstrated, by their application to a postulated direct containment heating scenario. The ISTIR objectives and process are summarized in this paper, as is its demonstration associated directly with the SASM. The objectives, processes and demonstration for the SASM are also summarized in the paper. The full body of work is referenced.

Gary E. Wilson, Brent E. Boyack, The role of the PIRT process in experiments, code development and code applications associated with

reactor safety analysis, Nuclear Engineering and Design, Volume 186, Issues 1–2, 1 November 1998, Pages 23-37, ISSN 0029-5493, https://doi.org/10.1016/S0029-5493(98)00216-7.
(http://www.sciencedirect.com/science/article/pii/S0029549398002167)

Abstract:
In September 1988, the United States Nuclear Regulatory Commission issued a revised emergency core cooling system rule for light water reactors that allows, as an option, the use of best estimate plus uncertainty methods in safety analysis. To support the 1988 licensing revision, the United States Nuclear Regulatory Commission and its contractors developed the code scaling, applicability and uncertainty evaluation methodology to demonstrate the feasibility of the best estimate plus uncertainty approach. The phenomena identification and ranking table (PIRT) process, Step 3 in the code scaling, applicability and uncertainty methodology, was originally formulated to support the best estimate plus uncertainty licensing option. Through further development and application, the PIRT process has shown additional utility as a robust means to establish safety analysis computer code phenomenological requirements in their order of importance to such analyses. The generic PIRT process, including typical and common illustrations from prior applications that promoted further development of the process, are described. Analysis of the results of the prior applications is also described. The analysis results provide information that can help guide future applications of the process in a graded approach based on phenomena relative importance.

M.Y. Young, S.M. Bajorek, M.E. Nissley, L.E. Hochreiter, Application of code scaling applicability and uncertainty methodology to the large break loss of coolant, Nuclear Engineering and Design, Volume 186, Issues 1–2, 1 November 1998, Pages 39-52, ISSN 0029-5493, https://doi.org/10.1016/S0029-5493(98)00217-9.
(http://www.sciencedirect.com/science/article/pii/S0029549398002179)

Abstract:

In the late 1980s, after completion of an extensive research program, the United States Nuclear Regulatory Commission (USNRC) amended its regulations (10CFR50.46) to allow the use of realistic physical models to analyze the loss of coolant accident (LOCA) in a light water reactors. Prior to this time, the evaluation of this accident was subject to a prescriptive set of rules (Appendix K of the regulations) requiring conservative models and assumptions to be applied simultaneously, leading to very pessimistic estimates of the impact of this accident on the reactor core. The rule change therefore promised to provide significant benefits to owners of power reactors, allowing them to increase output. In response to the rule change, a method called Code Scaling, Applicability and Uncertainty (CSAU) was developed to apply realistic methods, while properly taking into account data uncertainty, uncertainty in physical modeling and plant variability. The method was claimed to be structured, traceable, and practical, but was met with some criticism when first demonstrated. In 1996, the USNRC approved a methodology, based on CSAU, developed by a group led by Westinghouse. The lessons learned in this application of CSAU will be summarized. Some of the issues raised concerning the validity and completeness of the CSAU methodology will also be discussed.

Jose N. Reyes Jr., Lawrence Hochreiter, Scaling analysis for the OSU AP600 test facility (APEX), Nuclear Engineering and Design, Volume 186, Issues 1–2, 1 November 1998, Pages 53-109, ISSN 0029-5493, https://doi.org/10.1016/S0029-5493(98)00218-0.
(http://www.sciencedirect.com/science/article/pii/S0029549398002180)

Abstract:
In this paper, the authors summarize the key aspects of a state-of-the-art scaling analysis (Reyes et al., 1995. Westinghouse Electric Corporation, WCAP-14270) performed to establish the facility design and test conditions for the Advanced Plant Experiment (APEX) at Oregon State University (OSU). This scaling analysis represents the first, and most comprehensive, application of the Hierarchical Two-Tiered Scaling (H2TS) Methodology (Zuber,

1991. US Nuclear Regulatory Commission, Washington DC, NUREG/CR-5809) in the design of an integral system test facility. The APEX test facility, designed and constructed on the basis of this scaling analysis, is the most accurate geometric representation of a Westinghouse AP600 nuclear steam supply system. The OSU APEX test facility has served to develop an essential component of the integral system database used to assess the AP600 thermal hydraulic safety analysis computer codes.

S. Banerjee, M.G. Ortiz, T.K. Larson, D.L. Reeder, Scaling in the safety of next generation reactors, Nuclear Engineering and Design, Volume 186, Issues 1–2, 1 November 1998, Pages 111-133, ISSN 0029-5493, https://doi.org/10.1016/S0029-5493(98)00219-2.
(http://www.sciencedirect.com/science/article/pii/S0029549398002192)

Abstract:
A technique was developed to evaluate the applicability of data from small scale facilities for validation of codes for analysis of nuclear safety with emphasis on the next generation of reactors. The technique first divides an accident into phases based on the components that come into play as the accident evolves. Conservation equations, resolved to the component level and their interconnections, are derived for the active components in each phase. The equations are then nondimensionalized and reference parameters are selected such that the dependent variables, other than the system response of interest, are of order 1. Order of magnitude analysis is then performed for each equation and then between equations, based on the numerical values of the nondimensional coefficients for each term, with only the large order terms being retained. The resulting equations then contain terms whose impact on key system responses (e.g. reactor vessel level) are ordered in terms of the magnitude of the nondimensional groups multiplying the O[1] dependent variables. The reduced set of equations and nondimensional groups are validated with experimental data where possible. The validation process is meant to demonstrate that the important terms have been retained and enhance confidence

in the system of equations used to capture the main processes occurring in each phase. The methodology was demonstrated by evaluating the applicability of small-scale facility data for next generation reactor SBLOCA. Based on the nondimensional equations, the dominant nondimensional groups, and hence the dominant physical mechanisms and their dependence on geometric and operational parameters, were identified for a particular scenario, an AP600 cold leg break, starting from the initiating event through long term cooling. The important parameters entering the groups included elevation differences between the reactor vessel and other components, PRHR heat transfer rates, fluid thermophysical properties, liquid levels in tanks, flow resistances in the CMT lines and IRWST lines, flow resistance in the pressurizer surge line, and pressurizer drain rate. It was also shown that, after the beginning of CMT draining and accumulator injection, the dominant processes do not depend on break size provided they are small. The dominant processes were dependent on plant geometry and the operation of engineered safety features, such as the automatic depressurization system. The same transient events were evaluated for three experimental facilities and the same nondimensional groups, and hence mechanisms, were shown to be important. It was found that these nondimensional groups covered the range expected in the AP600, indicating that while there may be some distortions in scaling for a particular facility, between them, the important phenomena were captured and the small-scale facility data appear applicable for SBLOCA in the AP600 system. In more general terms, the methodology appears suitable for assessing scaling of various facilities for other postulated accidents and for other reactor concepts.

V.H Ransom, W Wang, M Ishii, Use of an ideal scaled model for scaling evaluation, Nuclear Engineering and Design, Volume 186, Issues 1–2, 1 November 1998, Pages 135-148, ISSN 0029-5493, https://doi.org/10.1016/S0029-5493(98)00220-9.
(http://www.sciencedirect.com/science/article/pii/S0029549398002209)

Abstract:
In this paper a method is described for using RELAP5 models to corroborate the scaling methodology that has been used for design of the Purdue University multidimensional Test Apparatus. This facility was built for the U.S. NRC to obtain data on the performance of the passive safety systems of the General Electric Company simplified boiling water reactor. Similarity between the prototype system and the scaled test facility is investigated for a main steam line break accident.

Tay-Jian Liu, Chien-Hsiung Lee, Chien-Yeh Chang, Power-operated relief valve stuck-open accident and recovery scenarios in the Institute of Nuclear Energy Research integral system test facility, Nuclear Engineering and Design, Volume 186, Issues 1–2, 1 November 1998, Pages 149-176, ISSN 0029-5493, https://doi.org/10.1016/S0029-5493(98)00221-0.
(http://www.sciencedirect.com/science/article/pii/S0029549398002210)

Abstract:
Four scaled small break loss-of-coolant accident (LOCA) tests simulating the pressurizer power-operated relief valves (PORVs) stuck-open accidents and the recovery actions in a pressurized water reactor (PWR) were performed at the Institute of Nuclear Energy Research (INER) integral system test (IIST) facility. The objectives of this study are to verify the effectiveness of emergency operating procedure (EOP) and emergency core cooling system (ECCS) on reactor safety. The break sizes were volumetrically scaled down based on one and all three fully-opened PORVs which is equivalent to 0.23% and 0.69% hot leg flow area of the reference plant. The experimental results indicate that in case of high pressure injection (HPI) system failure, the rapid depressurization of the steam generators is proved to be an effective way in the depressurization of the reactor coolant system and the core cooling. In contrast, if only one HPI charging pump operates normally, which injected half (or minimum) flow rate of normal cooling water, the core cooling can be adequately provided without operating the secondary bleeding during

PORV stuck-open transient. This paper also presents the scaling methods for the reduced-height, reduced-pressure (RHRP) IIST facility and the test conditions. The validity of the present scaling methodology is confirmed by the results from previous IIST counterpart tests and comparison of the present results with those of the tests performed at the full-height, full-pressure(FHFP) stuck-open tests.

M. Ishii, S.T. Revankar, T. Leonardi, R. Dowlati, M.L. Bertodano, I. Babelli, W. Wang, H. Pokharna, V.H. Ransom, R. Viskanta, J.T. Han, The three-level scaling approach with application to the Purdue University Multi-Dimensional Integral Test Assembly (PUMA), Nuclear Engineering and Design, Volume 186, Issues 1–2, 1 November 1998, Pages 177-211, ISSN 0029-5493, https://doi.org/10.1016/S0029-5493(98)00222-2.
(http://www.sciencedirect.com/science/article/pii/S0029549398002222)

Abstract:
The three-level scaling approach was developed for the scientific design of an integral test facility and then it was applied to the design of the scaled facility known as the Purdue University Multi-Dimensional Integral Test Assembly (PUMA). The NRC Technical Program Group for severe accident scaling developed the conceptual framework for this scaling methodology. The present scaling method consists of the integral system scaling, whose components comprise the first two levels, and the phenomenological scaling constitutes the third level of scaling. More specifically, the scaling is considered as follows: (1) the integral response function scaling, (2) control volume and boundary flow scaling, and (3) local phenomena scaling. The first two levels are termed the top-down approach while the third level is the bottom-up approach. This scheme provides a scaling methodology that is practical and yields technically justifiable results. It ensures that both the steady state and dynamic conditions are simulated within each component, and also scales the inter-component mass and energy flows as well as the mass and energy inventories within each component.

P.F. Peterson, V.E. Schrock, R. Greif, Scaling for integral simulation of mixing in large, stratified volumes, Nuclear Engineering and Design, Volume 186, Issues 1–2, 1 November 1998, Pages 213-224, ISSN 0029-5493, https://doi.org/10.1016/S0029-5493(98)00223-4.
(http://www.sciencedirect.com/science/article/pii/S0029549398002234)

Abstract:
In this paper we develop scaling relationships for mixing in large stratified volumes, both for steam/nitrogen mixtures in containment compartments and for water in suppression pools. The results apply to scaling for integral tests of passive reactor containment systems. Buoyant jets from injected fluids and buoyant wall jets generated by hot and cold surfaces provide the primary mixing in these passive systems. The buoyant jets entrain and transport the stratified fluid, mixing the fluid and reducing the vertical temperature and concentration gradients. We show that scaling for mixing can be satisfied simultaneously with scaling for two-phase natural circulation. The scaling requires a reduced height, accelerated time facility. Accelerated time scaling is advantageous for studying long-term behavior of interest in passive systems, while reduced height improves long-term heat loss, decreases power requirements, and makes simulation of blow-down mixing feasible.

K. Takeuchi, M.Y. Young, Assessment of flooding in a best estimate thermal hydraulic code (WCOBRA/TRAC), Nuclear Engineering and Design, Volume 186, Issues 1–2, 1 November 1998, Pages 225-255, ISSN 0029-5493, https://doi.org/10.1016/S0029-5493(98)00224-6.
(http://www.sciencedirect.com/science/article/pii/S0029549398002246)

Abstract:
The performance of WCOBRA/TRAC code in predicting the flooding, the counter-current flow limit, is evaluated in three geometries important to nuclear reactor loss-of-coolant accident evaluation; a

vertical pipe, a perforated plate, and a downcomer annulus. These flow limits are computationally evaluated through transient conditions. The flooding in the vertical pipe is compared with the classical Wallis flooding limit. The flooding on the perforated plate is compared with the Northwestern flooding data correlation. The downcomer flooding in 1/15th and 1/5th scale model is compared with the Creare data. Finally, full scale downcomer flooding is compared with the UPTF test data. The prediction capability of the code for the flooding is found to be very good.

K. Takeuchi, M.E. Nissley, J.S. Spaargaren, S.I. Dederer, Scaling effects predicted by WCOBRA/TRAC for UPI plant best estimate LOCA, Nuclear Engineering and Design, Volume 186, Issues 1–2, 1 November 1998, Pages 257-278, ISSN 0029-5493, https://doi.org/10.1016/S0029-5493(98)00236-2.
(http://www.sciencedirect.com/science/article/pii/S0029549398002362)

Abstract:
The WCOBRA/TRAC-MOD7A, Rev. 1 code is currently licensed for best estimate LOCA analyses of 3 and 4 loop PWRs. As part of a licensing effort to extend the code application to plants equipped with upper plenum injection (UPI), scaling effects predicted by the code are investigated. The scaling effects of UPI tests were obtained through data analyses and summarized in Damerell and Simons (1993). The scaling subjects are: breakthrough flow area, downflow rate into the core, hot leg water carryover, and liquid level in upper plenum. The test facilities that supplied the data include UPTF and CCTF. In this report, the scaling trend is obtained from WCOBRA/TRAC analyses of CCTF Run 72 and Run 76 (scaling factor 0.091), UPTF Tests 20A, 20B, and 20C (scaling factor 2.1), and a typical UPI plant (scaling factor 1.0). The predicted scaling trend is found to agree well with the test data.

Jinzhao Zhang, S.M. Bajorek, R.M. Kemper, M.E. Nissley, N. Petkov, L.E. Hochreiter, Application of the WCOBRA/TRAC best-estimate

methodology to the AP600 large-break LOCA analysis, Nuclear Engineering and Design, Volume 186, Issues 1–2, 1 November 1998, Pages 279-301, ISSN 0029-5493, https://doi.org/10.1016/S0029-5493(98)00279-9.
(http://www.sciencedirect.com/science/article/pii/S0029549398002799)

Abstract:
The AP600 is a simplified advanced pressurized water reactor (PWR) design incorporating passive safety systems that perform the same function as the active emergency core cooling systems (ECCSs) on the current reactors. In order to verify the effectiveness of the AP600 design features for mitigation of a postulated large-break loss-of-coolant accident (LOCA), the recently United States Nuclear Regulatory Commission (USNRC)-approved best-estimate LOCA methodology (BELOCA) was applied to perform the AP600 standard safety analysis report large-break LOCA analysis. The applicability of the WCOBRA/TRAC code to model the AP600 unique features was validated against cylindrical core test facility (CCTF) and upper plenum test facility (UPTF) downcomer injection tests, the blowdown and reflood cooling heat transfer uncertainties were re-assessed for the AP600 large-break LOCA conditions and a conservative minimum film boiling temperature was applied as a bounded parameter for blowdown cooling. The BELOCA methodology was simplified to quantify the code uncertainties due to local and global models, as well as the statistical approximation methods, with the other uncertainties being bounded by limiting assumptions on the initial and boundary conditions. The final 95th percentile peak cladding temperature (PCT95%) was 1186 K, which meets the 10CFR50.46 criteria with a considerable margin. It is therefore concluded that the AP600 design is effective in mitigation of a postulated large-break LOCA.