Proceedings of the V4 Nuclear Training Course

V4 nuclear training course

Budapest University of Technology and Economics, Hungary
Czech Technical University in Prague, Czechia
Slovak University of Technology in Bratislava, Slovakia
University of Warsaw, Poland

The project is co-financed by the Governments of Czechia, Hungary, Poland and Slovakia through Visegrad Grants from International Visegrad Fund. The mission of the fund is to advance ideas for sustainable regional cooperation in Central Europe.

supported by

Visegrad Fund

Project objective

The common history and culture and close economic cooperation will help with sharing of knowledge and experience, providing training and education programs for students, researchers and professionals among V4 universities.

Moreover, Poland is the only one from V4 countries without nuclear power plants, but it is at the beginning on its way to introduce this source of energy to the energy mix. The crossborder cooperation is very important and necessary to ensure regional nuclear safety by providing support for Poland as a newcomer country. This approach will certainly raise the awareness of inhabitants of V4 countries of their regional solidarity related to nuclear energy development, and contribute to a more positive perception of cooperation in this regional format.

We would like to extend already established collaborations among all V4 countries in nuclear field and increase our expertise and range. This first common project will help to developed closer relationships in education and training and of course also in research and development field. The V4 nuclear training course was organized by Czech Technical University, Slovak University of Technology, University of Warsaw and Budapest University of Technology and Economics, leaders in education of staff for nuclear sector in V4 countries. This course, addressed to students and researchers involved in educational process, will be the first step to establish cooperation between these academic centers and establish professional relationships between the academic staff and students.

Although the training was online, during 5-day course lectures, experiments and technical visits were organised. The aim of the meeting is gaining knowledge on advanced issues of reactor physics, exchanging experiences, as well as a discussion about future joint projects, potential research, academic and student exchange between four universities. The specific goals of the project: develop the collaboration in nuclear education and training students, researchers and professionals; ensure the quality of nuclear education and training in all of the V4 countries; integration of scientific and academic community all of the V4 countries, specializing in nuclear energy development, sharing of academic resources and capabilities at the international level.

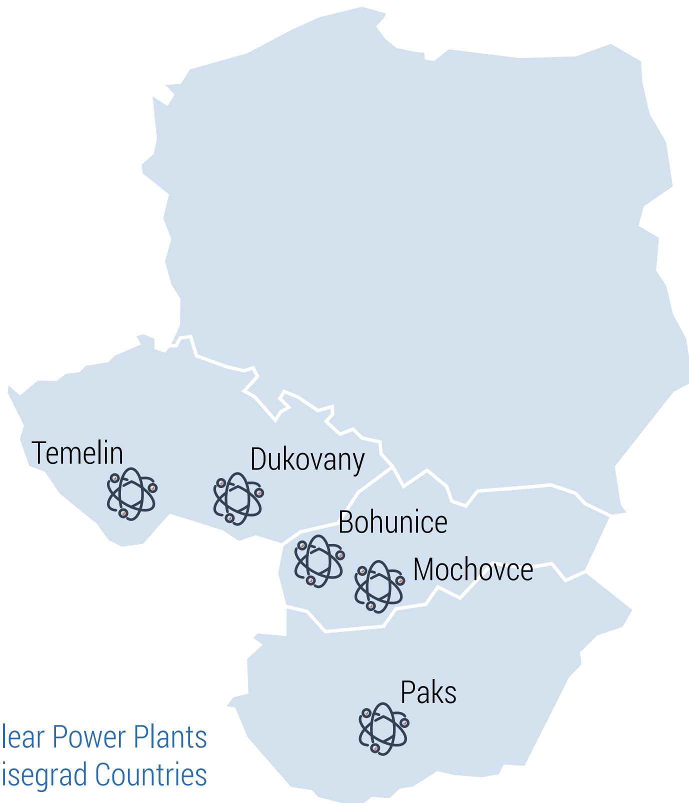
Find more at the project website http://v4nuclear.fuw.edu.pl/index.html.

Nuclear energy in Visegrad countries

All of the Visegrad countries either operate nuclear power plants or plan to do so. In Czech Republic about 30% of electricity is generated in nuclear power plants, in Hungary 50% and in Slovakia 55%. Poland, as a newcomer nuclear country, plans to introduce the nuclear power into to its energy mix till 2033, according to the government's policy.

The most important issue is education and training of people in various fields within nuclear industry (nuclear power plants operation, regulatory body, radiation protection, utilization of neutrons, etc.). Czech Republic, Slovakia and Hungary have been working on human resource development in this industry for nearly 50 years and there is already established collaboration in this field. Although Poland has lack of experts specializing in nuclear power plants licensing and operation, there is long experience in the operation of research reactors and their application to production of radiopharmaceuticals.

Developing of international cooperation between academic centers is very important since sharing of knowledge and experience in education and training will increase value of our graduates in the labor market. By organization of common training we want to give students chance to meet the best experts in this area and make new contacts that will be helpful in their future.



Nuclear Power Plants
in Visegrad Countries

Project partners

CTU
CZECH TECHNICAL UNIVERSITY IN PRAGUE

The Czech Technical University in Prague (CTU) is one of the biggest and oldest technical universities in Europe. CTU currently has eight faculties and about 18,000 students.

The Faculty of Nuclear Sciences and Physical Engineering (FNSPE) was established in 1955, and at that time its primary mission was to train new experts with a strong theoretical background for the emerging Czechoslovak nuclear programme. Gradually, however, its responsibilities were extended to cover a wider scope of fields and courses in mathematics, physics and chemistry so that, in keeping with its tradition, it can now offer excellent education with a personal approach to students' tuition. Faculty provides courses in the branch of nuclear engineering - focused on theoretical and experimental reactor physics, neutron applications, nuclear reactor operation and engineering. The Faculty operates Training reactor VR-1, lightwater zero power fission reactor, and also fusion reactor Golem.

more info: www.fjfi.cvut.cz

MŰEGYETEM 1782

The Budapest University of Technology and Economics is one of the leading V4 universities in the field of nuclear engineering and use of nuclear energy has long tradition in Hungary. The Faculty of Natural Sciences operates pool type water cooled reactor. The Training Reactor, which started operation in 1971 and has 100 kW nominal thermal power, is the scene of numerous reactor and radiation related exercises for undergraduate and graduate students and serves as a neutron and gamma radiation source for research.

The staff of the Faculty has different types of qualifications as Independent Technical Experts in the Nuclear Field from the Hungarian Chamber of Engineers, such as reactor physics, thermal-hydraulics, mechanical engineering, chemical engineering, radiation protection, proliferation resistance and transport of radioactive and nuclear material.

more info: reak.bme.hu

FACULTY OF PHYSICS
UNIVERSITY OF WARSAW

The University of Warsaw is the best university in Poland and one of the leading ones in this region of Europe, where app. 45,000 people study. The candidates are offered a very broad range of courses in the fields of humanities, social sciences and natural sciences.

The Faculty of Physics is a large research and teaching center. It consists of The Institutes of Theoretical Physics, Experimental Physics, Geophysics, The Astronomical Observatory and The Department of Mathematical Methods in Physics. The Faculty is regarded as one of the best in the country, recognized internationally for the high quality of research and education. Nuclear Energy and Chemistry is a field of study provided jointly by the Faculty of Physics and the Faculty of Chemistry since 2011. As a response to the government plan of construction of nuclear power plants in Poland, the main goal of this field of study was to educate specialists and scientists who will be participating in nuclear power development program. A new two-years master's program on Reactor Physics is curently being developed at the Faculty of Physics.

more info: www.fuw.edu.pl

STU FEI

SLOVAK UNIVERSITY OF TECHNOLOGY IN BRATISLAVA
FACULTY OF ELECTRICAL ENGINEERING AND INFORMATION TECHNOLOGY

The Slovak University of Technology in Bratislava (STU) is a modern educational and scientific institution. According to the Slovak higher education ranking scheme, STU has been the best university in chemicals technologies, computer and technical sciences in the long term. STU offers education in technical fields and involves students in research in natural sciences, computer sciences, construction, architecture, materials technologies, chemistry and food technologies. STU provides 3 level education (bachelor, master and PhD.) at all 7 faculties. Since its foundation in 1937 more than 145,000 students have graduated. In average, 17,000 students study at the STU every year.

The Institute of Nuclear and Physical Engineering (INPE) is one of the 10 institutes working as a part of the Faculty of Electrical Engineering and Information Technology (FEI). It is responsible for university education in the area of nuclear and physical engineering. Through education, scientific, research and development activities, INPE is active in the fields of: General Physics, Mathematical Physics, Physics of Condensed Matter and Acoustics, Nuclear and Sub-nuclear Physics, Material Science, Environmental Engineering, Electro-technology and Materials, Nuclear Power Engineering and Technology, Biomedical Engineering, and Physical Engineering.

more info: www.fei.stuba.sk

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Methods in neutron detection and spectroscopy

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Neutrons are elementary particles that are part of the atomic nucleus and have mass but no electrical charge. They can penetrate thick layers of certain materials without any interaction. Free moving neutrons in a material medium undergo elastic or inelastic scattering on the nuclei of the medium and may be absorbed by nuclear reactions which can produce charged particles and gamma quanta. If the medium in which the neutron moves is a fissile element, a fission reaction may occur, which is also a source of charged particles.

Introduction

A neutron detector can be practically any charged particle detector which makes use one of these types of nuclear reactions which produces electrons, protons, alpha particles or other types of ionizing radiation. Different types of neutron detectors are available:

- Gas ionisation detector (ionisation chambers, proportional counters, Geiger-Müller counters)
- Scintillation detector (counter)
- Semiconductor detector (counter).

The basic principle of operation of detectors (counters) is to generate electrical signals in their active volume in the form of current pulses. When current flows through the anode, (working) resistance in the counters circuit causes a short potential drop, i.e., a voltage pulse, on this resistance.

These detection systems consist of an active material for reaction with neutrons, a detector, and appropriate electronics, and can measure electric current intensity (current systems), amplitude of voltage pulses or their frequency count per unit time (impulse systems). The detection type and its efficiency strongly depend on the isotope used as a target and the energy of the neutrons we want to record. Therefore, when determining the nuclear reaction for use in a neutron detector, several requirements should be taken into account:

- The microscopic cross-section of a nuclear reaction should be as large as possible to build a very efficient detector with small dimensions.
- The energy of the charged particles emitted during the nuclear reaction should be as high as possible to obtain a large amplitude of pulses or a high intensity of current from the detector.

- The percentage of the isotope whose nuclei are involved in the nuclear reaction should be sufficiently large or the isotope should be easily separable.

In general, the methods of neutron detection can be divided into active (immediate reading) and passive (reading after a certain time, a technique often used in dosimetry). However, if we concentrate on measurement techniques related to electron energy, we can distinguish the following methods:

1. Neutron scattering and measurement of the rejected particle energy (Recoil Proportional Counter, Recoil Proton Telescope),

2. Measurement of the energy of charged particles released by a nuclear reaction involving neutrons (^{3}He Proportional Counter, Ionization Chambers, Semiconductor Crystals),

3. Time of Flight Method (ToF) neutron velocity measurements,

4. Threshold reactions involving only minimal neutron energy (foil activation technique),

5. Methods where the neutron energy distribution is studied with a set of detectors which have different sensitivities to various energies (Bonner Sphere Spectrometer),

6. Neutron diffraction,

7. Methods where the slowing down time of neutrons in an appropriate environment is measured.

Neutrons produced in a fission reaction have different kinetic energies; about 75 % of all neutrons have energy over 0.8 MeV. However, as a result of the scattering reaction, the energy spectrum (energy distribution) of the neutrons in a nuclear reactor is more complex. Neutrons are classified according to their kinetic energy, as shown in Table 1. Nuclear reactors contain neutrons ranging from cold to fast, and for their detection, appropriate methods should be used (different depending on the energy range), which are described in the following chapters.

Table 1 Classification of neutrons for their average kinetic energy

Name	Kinetic energy	Temperature
Cold	0.001 eV	11.6 K
Thermal	0.025 eV	293.15 K
Resonance*	1 - 1000 eV	Over 10 000 K
Intermediate energies	1 - 500 keV	
Fast	0.5 - 50 MeV	
High energies	> 50 MeV	

** Probability of some reactions with neutron changes rapidly producing resonance excitations.*

Methods of detecting thermal neutrons

The cross section (σ) for particle absorption (i.e., the probability of a reaction) depends on the target material and the neutron energy. In many cases, σ is inversely proportional to the neutron velocity (1/v), therefore most materials can be used for thermal neutron detection.

Depending on the isotope used and the type of nuclear reaction occurring in it, the following types of thermal detectors are used:

1) Proportional counter with boron trifluoride (BF3)

This is a standard proportional counter where the fill gas is doped with BF_3. The isotope ^{10}B, which is characterised by a large cross section for thermal neutron absorption (Table 2), in this case is the active target material. The reaction can occur through two channels:

$^{10}_{5}B + n \rightarrow {}^{7}_{3}Li + \alpha + 2.79$ MeV (6 % of total cases)

$^{10}_{5}B + n \rightarrow {}^{7}_{3}Li^{*} + \alpha + 2.31$ MeV (94 % of total cases)

$^{7}_{3}Li^{*} \rightarrow {}^{7}_{3}Li + \gamma$ (0.48 MeV)

The reaction products are α-particles (^4He), and the recoil nuclei ^7Li ionise the counter's gas atoms, generating an electric impulse in its circuit. De-excitation of the excited $^{7}_{3}Li^{*}$ nucleus to the ground state is accompanied by the emission of a γ quantum with the energy of 0.48 MeV. In the case of photon "escape" from the counter, the main peak is shifted by 0.48 MeV.

2) Helium counter

The isotope ^3He is added to the counter gas, which acts as a target material because of its large cross section for neutron absorption (Table 2), leading to the reaction:

$^{3}_{2}He + n \rightarrow {}^{3}_{1}H + p + 0.764$ MeV

The products of this reaction are protons (p) and ^3H rejection nuclei, which ionise the counter gas.

3) Lithium counter

The principle of the counter uses the reaction of thermal neutrons on the isotope ^6Li:

$^{6}_{3}Li + n \rightarrow {}^{3}_{1}H + \alpha + 4.786$ MeV

The counter gas in this case is ionised by both reaction products: ^3H and the α-particle, which have kinetic energy.

The cross sections of neutron reactions with selected isotopes (including ^{10}B, ^3He and ^6Li) are shown in Table 2.

Table 2 Integral data for cross-section of neutron conversion reactions [1]

Reaction	Average cross section (b) in the energy region		
	Thermal	Resonance	Fission
$^{10}B(n,\alpha)$	3841.7	1724.8	0.462
$^{7}Li(n,^{3}H)$	938.4	422.1	0.336
$^{3}He(n,p)$	5315.2	2380.2	0.814
$^{235}U(n,f)$	570.8	268.8	1.225
$^{238}U(n,f)$	< 0.0001	0.002	0.303
$^{157}Gd(n,\gamma)$	215590	759.6	0.109

4) Fission ionisation chamber

Under the influence of thermal neutrons, the ^{235}U nucleus splits into two fragments, which have a large electrical charge and strongly ionise the detector gas. This means that large signal amplitudes are generated which do not require gas amplification. Therefore, in this case, we use ionisation chambers and no need arises to use proportional counters. The fissile material, as a solid, is placed inside the chamber so that the reaction products can penetrate the active volume of the chamber.

5) Geiger-Müller counter with cadmium shielding

The isotope ^{113}Cd is characterised by an extremely large cross-section for radiative capture of thermal neutrons according to the reaction:

$$^{113}Cd + n \rightarrow {}^{114}Cd + \gamma$$

Therefore, an ordinary G-M counter shielded by a cadmium sheet with a thickness of about 1 mm, works perfectly as a neutron counter. The largest contribution to the emitted γ spectrum are photons with energies of 0.558 MeV (72.7%) and 0.651 MeV (13.9%), which are easily registered in a G-M counter.

6) Scintillation counter with ZnS(Ag) crystal

If an additional impurity of ^{10}B or ^{235}U is added to the ZnS(Ag) crystal, the counter can be used for neutron detection. The resulting products of the reaction with neutrons, i.e., fission fragments or α-particles, are the source of light scintillations which generate electric impulses after amplification in a photomultiplier. The advantage of using a ZnS(Ag) scintillator is a very low efficiency for the detection of γ-rays, which often accompanies reactions with neutrons and causes interference in particle detection.

Methods of fast neutron detection

An important reaction in the detection of fast neutrons is the elastic scattering of neutrons, mainly on hydrogen nuclei (protons). Polyethylene or paraffin, acting as a target material, is placed inside the gas detector. During elastic scattering, the neutron transfers some of its kinetic energy to the nucleus, which remains in the ground state and moves in a different direction with reduced energy. The kinetic energy transferred to the nucleus by the neutron causes it to recoil. The recoiled hydrogen nuclei (called recoil protons) ionizes the gas filling the detector. This type of fast neutron detector is a spectrometric detector, i.e., they also allow the measurement of neutron energy.

Another method of detecting fast neutrons is to use any thermal neutron detector covered with a layer of moderator, usually paraffin of appropriate thickness (~ cm). Fast neutrons slow down and thermalize on the paraffin hydrogen nuclei, and then as thermal neutrons they can be registered by the counter.

Activation method

Activation is the production of radioactive isotopes in nuclear reactions which do not occur naturally in nature. If we insert a target material into the neutron flux, it becomes radioactive as a result of the nuclear reaction. The probability of a reaction depends on the cross section, energy and flux of incident neutrons. Therefore, by using the same materials at the same activation time, it is possible to determine the spatial distribution of neutron flux around the sources or inside the reactor. The activated discs are usually very small in size (mm diameter discs), therefore we can treat them as point-like. Their selection depends on the value of the cross section for the reaction of interest, the half-life of the resulting radioactive isotope and the type of radiation emission. Commonly used materials are isotopes of silver, indium, cobalt or manganese.

On a silver target bombarded with fast neutrons, the reaction of detaching one neutron from the target nucleus and emission of two neutrons takes place. This produces the radioactive isotope ^{108}Ag, which de-excites to its ground state, and then undergoes β^- transformation with a half-life ($T_{1/2}$) of 2.3 minutes, accompanied by the emission of gamma quanta with energy of about 0.6 MeV.

$$^{109}Ag + n \rightarrow {}^{108}Ag^* + {}^{2}n$$

Radiative neutron capture produces the radioactive isotope ^{116}In, which undergoes β^- decay ($T_{1/2}$= 54 minutes) and emits gamma radiation of different energies in the range 0.14 - 2.1 MeV.

$$^{115}In + n \rightarrow {}^{115}In^* + \gamma$$

Using the activation method, the target material is often selected so that the neutron capture reaction can occur from a certain threshold value of the neutron energy (threshold detectors).

Due to the difficult physical conditions inside a reactor core and the limited space, the best method to control and study the conditions in these spaces (flux and energy distribution of neutrons) is the activation method using threshold reactions. Threshold reactions are nuclear reactions which start to occur at a well-defined minimum energy of particles which trigger them. An additional advantage of this method is the lack of sensitivity to other types of ionising radiation. Because activation detectors are small in size, they can be placed in hard-to-reach places. It should be noted that in many cases neutron flux measurements using activation detectors are tedious and labour-intensive, and there are also many factors which affect the measurement result. For example, when measuring the flux of thermal neutrons, the detector material is activated not only by the absorption of thermal neutrons but also epithermal and fast neutrons. The effect of epithermal and fast neutrons can be taken into account by additionally performing a special measurement. It should also be noted that placing the detector in a specific place changes the neutron flux in its vicinity. It is therefore necessary to make an appropriate correction to the measurement result.

Table 3 presents examples of target materials used in threshold detectors which use the (n,p) reaction. Higher threshold energies are generally observed for reactions of the type (n,2n). An example is the gold (Au) isotope, whose threshold energies for various reaction channels are shown in Table 4.

Table 3 Characteristics of sample isotopes used in threshold detectors based on (n,p) reactions [1]

Isotope	Threshold energy (MeV)	Half-life $T_{1/2}$
^{24}Mg	4.9	14.8 h
^{27}Al	1.96	10.2 min
^{31}P	0.97	170 min
^{32}S	1.0	14.3 d
^{49}Ti	1.1	57 min
^{52}Cr	2.8	3.9 min
^{56}Fe	2.1	2.6 h

Table 4 Selected threshold reactions for the isotope ^{197}Ag [1]

Isotope	Threshold energy (MeV)	Half-life $T_{1/2}$
^{197}Au (n,2n) ^{196}Au	8.1	6.183 d
^{197}Au (n,3n) ^{195}Au	14.8	186.1 d
^{197}Au (n,4n) ^{194}Au	23.2	38.02 h
^{197}Au (n,5n) ^{193}Au	30.2	17.65 h
^{197}Au (n,6n) ^{192}Au	38.9	4.94 h
^{197}Au (n,7n) ^{191}Au	45.7	3.18 h
^{197}Au (n,8n) ^{190}Au	54.5	42.8 min
^{197}Au (n,9n) ^{189}Au	61.8	28.7 min
^{197}Au (n,10n) ^{188}Au	70.9	8.84 min

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Nuclear Data – Measurement and Theory

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Objectives

The objectives are to:

- Define the term "nuclear data".
- Briefly mention the history of nuclear data.
- Demonstrate:
    - What type of experimental data are openly available.
    - Where the data can be obtained.
    - Introduce to nuclear model calculations.

Details of theoretical nuclear model calculations are beyond the scope of the present work. Likewise, descriptions of the multitude of experimental measurement techniques are very specific and cannot be accommodated in the limited time frame allocated to the subject.

Evaluated nuclear data and application libraries are discussed separately.

Overview of the Content

The term "Nuclear Data" refers to information which describes the internal structure of the nucleus (energy levels, decay constants, etc.), fission product yields, particle interaction reaction rates upon collision of the target nucleus with the projectile particle, emitted particle types, spectra and angular distributions. This information is required for simulating particle transport, activation, radiation dose, nuclear heating and radiation damage. These simulations are not only relevant for nuclear power but also analytical work, medical application such as therapeutic and diagnostic radionuclide production, computerized tomography, oil-well logging, demining of former war-zones, etc. Users in these fields generally apply codes which are supplied with customized data libraries, however the origin of the nuclear data in these libraries is often lost to the users. The notes here provide an introduction to nuclear data in general. Nuclear data evaluation and processing are described separately.

The term "nuclear data" denotes the following information:

- Internal structure of the nucleus (energy levels, decay constants, branching ratios, emitted radiation, etc.).

- Fission product yields (including energy-dependence for different fissile/fissible nuclides).

- Data which describes particle interaction reaction rates:

    - Separate for different incident particles (neutrons, photons, protons, other charged particles).

    - Cross sections, emitted particle types, their energy spectra and angular distributions, emitted photon spectra from direct transitions or the continuum).

    - Thermal scattering law data (taking into account atom binding in molecules and crystals).

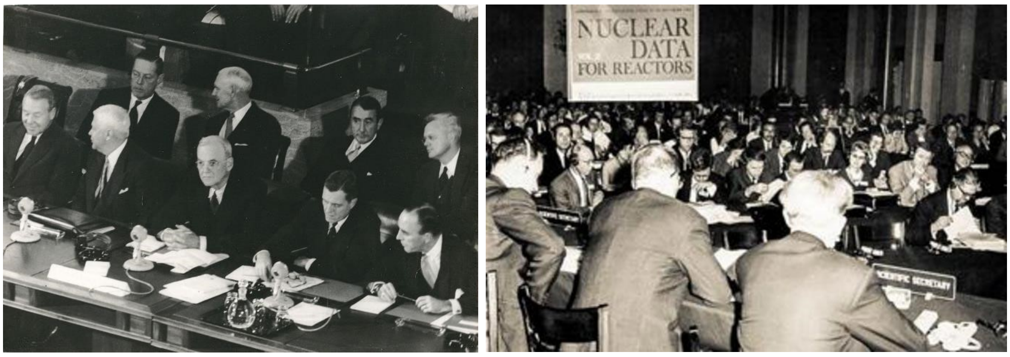

Figure 1 a) The 1955 Geneva Summit took place (left); (b) The IAEA Nuclear data Section was established in 1964 (right).

Brief Historical Overview of Nuclear Data

Nuclear data have been measured since the discovery of radioactivity and gained importance after the discovery of fission, particularly within the Manhattan Project during World War II. Scientists quickly realized that the volume of information required for the development of nuclear technology was enormous and that broad international collaboration would be required. Some important milestones are:

- The Geneva Summit (Figure 1a) in 1955 (note that this was during the cold war period).

- Declassification of nuclear data, allowing the free exchange of nuclear data worldwide while making provisions for the non-proliferation of materials and technology for military purposes.

- The International Atomic Energy Agency (IAEA) was created (Figure 1b) in 1957 to promote the peaceful uses of nuclear energy and safeguard nuclear materials for non-proliferation.

- The nuclear data section (IAEA-NDS) was created in 1964 to facilitate the exchange of nuclear data.

- Establishment of data centre networks (see https://www-nds.iaea.org/):

    - NRDC: Nuclear Reaction Data Centre Network.

- NSDD: Nuclear Structure & Decay Data Network.
- INDEN: International Network of Nuclear Data Evaluators.

The function of the data centres are:

- Compile and disseminate nuclear data.
- Exchange information (the same data is available online regardless of the data centre users connect to).

Several data centres are part of the Nuclear Reaction Data Network. The links to the web pages of each centre are given in the table below.

Table 1 Data centres

Country	Centre	Joined
Hungary	Nuclear Data Group (ATOMKI) Institute for Nuclear Research, Debrecen	1992
Russia	Centre for Photonuclear Experiments Data (CDFE) Moscow State University, Moscow	1982
Russia	Russian Nuclear Data Center (CJD) Institute of Physics and Power Engineering, Obninsk	1966
China	China Nuclear Data Center (CNDC) China Institute of Atomic Energy Beijing	1987
Russia	Center of Nuclear Physics Data (CNPD) All Russian Scientific Research Institute of Experimental Physics, Sarov	1997
Japan	Nuclear Data Center (JAEA/NDC) Japan Atomic Energy Agency, Tokai-mura, Naka-gun, Ibaraki	1991
Japan	Hokkaido University Nuclear Reaction Data Centre (JCPRG) Hokkaido University, Sapporo	1975
Korea	Korea Nuclear Data Center (KNDC) Korea Atomic Energy Research Institute (KAERI), Yuseong, Daejeon	2000
India	Nuclear Data Physics Centre of India (NDPCI) Bhabha Atomic Research Centre (BARC), Trombay, Mumbai	2008
IAEA	IAEA Nuclear Data Section (NDS) Vienna	1966
OECD	OECD NEA Data Bank (NEA DB) Boulogne-Billancourt	1966
USA	US National Nuclear Data Center (NNDC) Brookhaven National Laboratory, Upton, NY	1966
Ukraine	Ukrainian Nuclear Data Center (UkrNDC) Institute for Nuclear Research, Kyiv	1998

The data centres collect, compile and disseminate **bibliographical** information and databases of **numerical** information with measured data, evaluated data and some application libraries.

Nuclear data – what is the problem?

Reaction cross sections vary by many orders of magnitude and exhibit resonance structure. Due to the complexity, an enormous amount of information is needed. An example of the ^{235}U total cross section is shown below in Figure2.

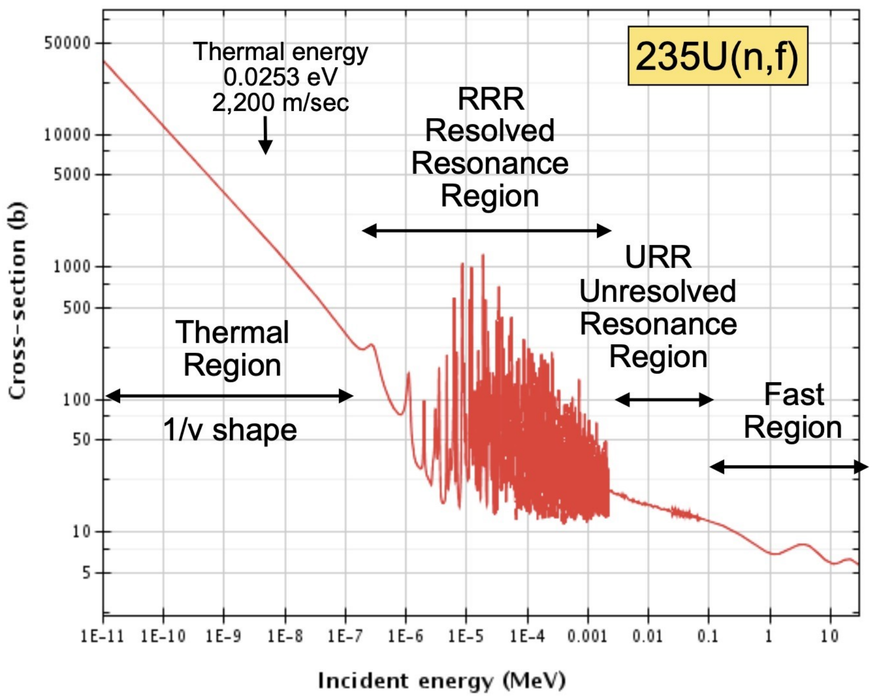

Figure 2 Total cross section of ^{235}U

Bibliographic Information

CINDA https://www-nds.iaea.org/exfor/cinda.htm

CINDA is a database of bibliographic references, namely documents reporting experimentally measured nuclear reaction data and fission product yields.

Originally, CINDA served as a quick index to data which needed to be compiled numerically. The current trend by evaluators is to compile the numerical data directly; CINDA database is a by-product.

NSR https://www.nndc.bnl.gov/nsr/

The NSR is database of bibliographic references, namely documents reporting experimentally measured nuclear structure and decay data.

The master file is maintained by the National Nuclear Data Center at the Brookhaven National Laboratory (NNDC/BNL).

Numerical Nuclear Data (Differential)

EXFOR database https://www-nds.iaea.org/exfor

- The data base contains numerical nuclear reaction data and fission product yields published in journals (and sometimes internal reports).
- The master copy is maintained at the International Atomic Energy Agency, Vienna (IAEA).

- Compilers at the data centres take responsibility to scan the literature published in their area (journals, lab reports, etc.) and enter them into the database in a strictly defined format which allows automated retrieval by utility codes. Data centres regularly exchange the compiled data, hence users receive exactly the same data regardless of the data centre they connect to.

- The online EXFOR retrieval system selected information to be downloaded in various formats for further processing by the user.

ENSDF database https://www.nndc.bnl.gov/ensdf

- The database contains compiled/evaluated nuclear structure and decay data.

- The data entries are evaluated and approved by NSDD members.

- The master file is maintained by the National Nuclear Data Center, Brookhaven (NNDC/BNL).

- Various additional utilities are available for data visualization and online retrieval (NuDat, Nuclear Wallet Cards, Live Chart of Nuclides, Isotope Browser for mobile phones).

XUNDL database https://www.nndc.bnl.gov/ensdf/ensdf/xundl.jsp

- The database contains experimental Un-evaluated Nuclear Data.

- The master file is maintained by the National Nuclear Data Center, Brookhaven (NNDC/BNL).

Readers can familiarize themselves with data retrieval by connecting to the web interface using the links above to any of the cited data centres.

Numerical databases contain measured values from many different types of experiment. The time-of-flight method (Figure3) is usually applied to measure cross sections in the resonance energy range.

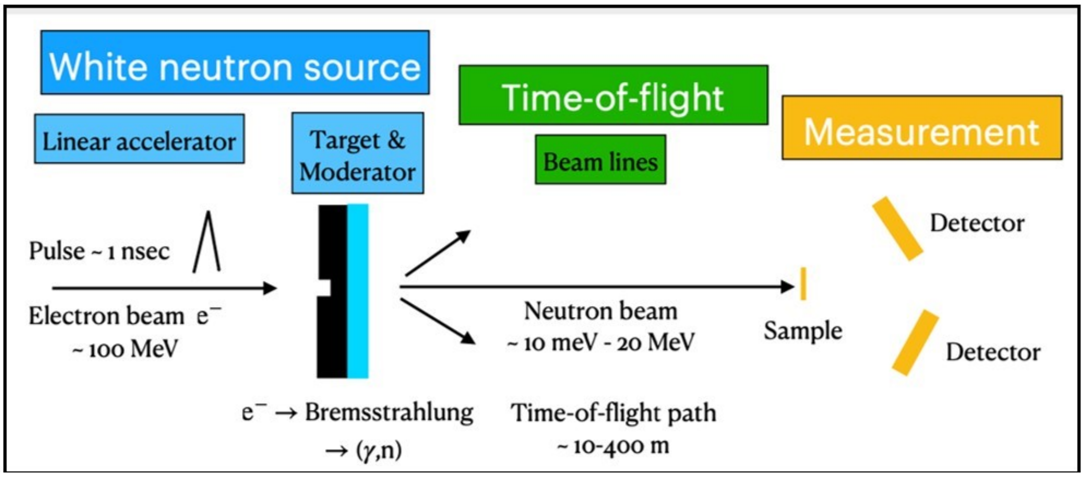

Figure 3 Schematic diagram of a time-of-flight measurement. Neutrons are produced either from the (γ,n) reaction, where gammas are produced by an electron beam on a target, or by spallation, where protons are incident on a target and "blow up" the nucleus. Energy-resolution of neutrons is achieved by the time they travel to the sample (low energy neutrons take longer).

Important experimental facilities in the USA and Europe are:

Table 2 Important experimental facilities in the USA and Europe

Name	Place	Energy	Beam	Pulse	Target	ToF	Neutron Energy
LINAC	RPI Troy, USA	60 Mev	e-	7 ns	Ta	10 - 250 m	Thermal - 20 MeV
GELINA	Geel, Europe	140 MeV	e-	1 ns	Pb	10 - 400 m	10 meV - 20 MeV
n_TOF	CERN, Europe	20 GeV	p	6 ns	Pb	20, 185 m	Thermal - 1 GeV
LANSCE	LANL, USA	800 MeV	p	0.2 ns	W	8 - 90 m	Fast region

Laboratories are equipped with sophisticated detector systems (Figure 4, 5). The requirements are:

- High-performance detectors: arrays of scintillators for capture and scattering.
- High-performance data acquisition: complete information for offline analysis.
- High-resolution experiments: suitable for resolved resonance regions.
- High-flux experiments: suitable for small radioactive samples.
- High demanding data analysis: time-consuming process.
- High regulatory requirements: may become real nightmare.

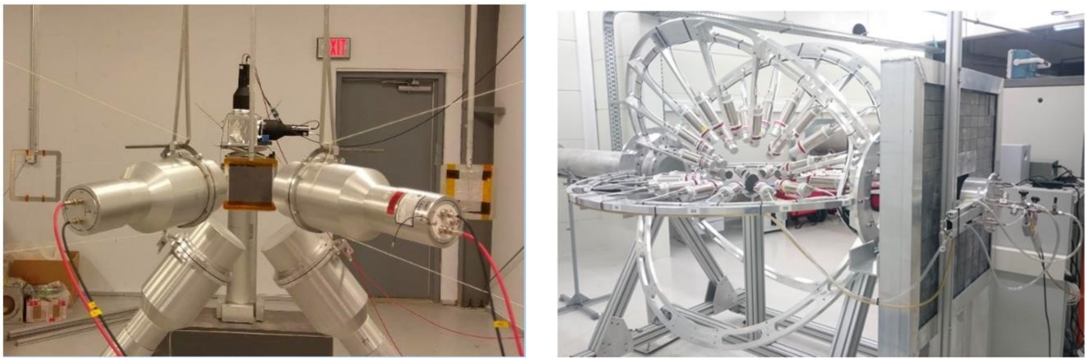

Figure 4 Examples of detector systems at two of the major facilities (left - RPI Troy 2018: Four C6D6 scintillators for gamma, right - GELINA 2018: Array of 32)

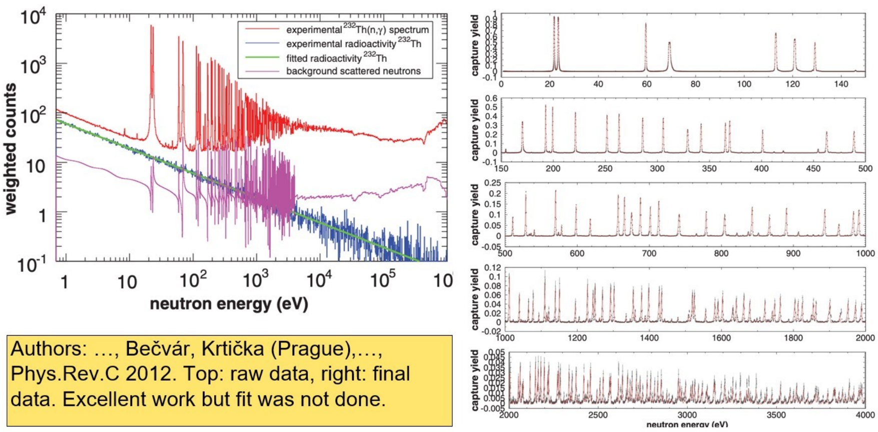

Figure 5 Example of $^{232}Th(n,\gamma)$ measurement

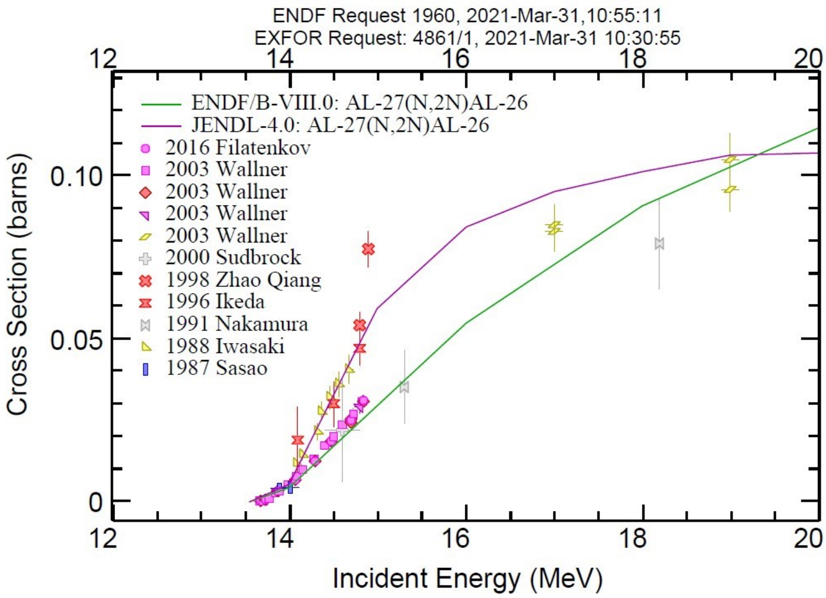

Figure 6 Example of the EXFOR/ENDF retrieval. Online comparison of cross sections with evaluated nuclear data is possible

Online retrieval engines available from the nuclear data centres (Figure 6) allow the retrieval of the data in numerical form in various formats, and high quality graphical representations.

Numerical Nuclear Data (Integral)

Average cross sections measured in different spectra are useful for data validation, if the spectrum is well-known (e.g., ^{252}Cf spontaneous fission spectrum). Some of these data are included in the EXFOR database.

In addition, several compilations are available, organized through the OECD/NEA Data Bank:

- ICSBEP handbook https://www.oecd-nea.org/jcms/pl_24498/international-criticality-safetybenchmark-evaluation-project-icsbep
    - Compilation of over 5000 critical assembly configurations with detailed descriptions, benchmark specifications and uncertainty analyses (Figure 7).
- IRPhE handbook https://www.oecd-nea.org/jcms/pl_20279/international-handbook-ofevaluated-reactor-physics-benchmark-experiments-irphe
    - Compilation of reactor experiments with detailed descriptions and benchmark specifications for various measured integral quantities (not limited to criticality).
- SINBAD compilation https://www.oecd-nea.org/jcms/pl_32139/shielding-integralbenchmark-archive-and-database-sinbad
    - Compilation of shielding experiments (reactors, fusion, accelerator applications).

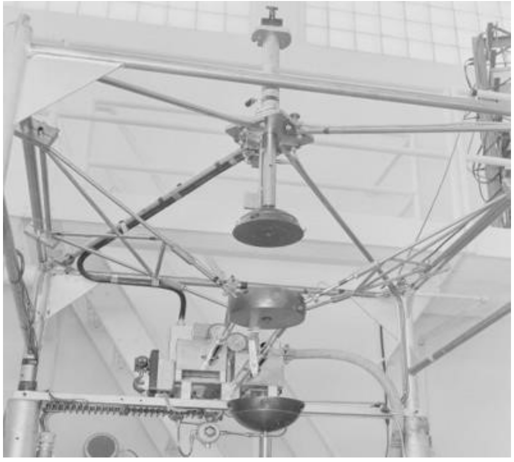 

Figure 7 a) Godiva was a well-known well-known critical assembly at Los Alamos. b) according to legend Lady Godiva rode naked, covered only by her hair, through Coventry in 11th century to gain a remission of oppressive tax imposed on tenants.

Derived Data Sources – the RIPL Database

Optical model parameters are an essential part of nuclear model calculations. In the past, authors published their model parameters, but in some cases, there were slight differences in definitions and the users who applied them would obtain different results. P. Obložinský (from the IAEA at that time) initiated a project to compile optical model parameters in a numerical database with a strict definition of parameters, thereby allowing a verification exercise so that all codes which use a certain set of optical model parameters obtain the same result. The current Reference Input Parameter Library (RIPL-3) was developed through a sequence of projects organized by the IAEA and is available from the IAEA website: https://www-nds.iaea.org/RIPL-3/
The documentation is available online:

https://www.sciencedirect.com/science/article/abs/pii/S0090375209000994?via%3Dihub

The library contains essential input parameters for nuclear reaction model calculations:

- Nuclear masses
- Energy levels of the nucleus
- Average spacings of neutron resonances
- Optical model potential parameters
- Level densities according to a variety of models, shell correction prescriptions
- Experimental giant dipole resonance parameters (experimental and theoretical), photon strength functions
- Fission barriers: empirical, Hartree-Fock-Bogoliubov method (HFB) and liquid-drop fission barriers, level densities at the saddle point

A number of codes for optical model calculations are also available via links from the RIPL-3 web page.

Atlas of Neutron Resonances

The resonance data in the Atlas were systematically compiled and analyzed for over 50 years by S. Mughabghab (1934-2018†). Six editions were published in the period 1966–2018. They are used by all evaluators around the world.

The Atlas contains resonance parameters, thermal cross sections (including isomers) and resonance integrals.

For example, ^{235}U includes 1500 resonances, with 6 parameters each, resulting in a total of around 9000 parameters.

Said F. Mughabghab is a nuclear data legend. Born in Lebanon, an immigrant to the USA, he worked his entire life on resonances. His devotion to the Atlas was absolute.

Nuclear Reaction Model Codes

Nuclear reaction

$n + {}^{A}_{Z}X \rightarrow {}^{A+1}_{Z}X^*$ (Compound Nucleus) → Emitted particle(s) + Residual Nucleus

Differential

(microscopic) cross sections σ

Macroscopic cross section $\Sigma = N.\ \sigma$

Major challenge

Vastly different nuclei from ^{1}H to ^{265}Fm

Vastly different energies (13 decades)

Importance

The data must be consistent with physics.

What is gained by using nuclear models?

- Consistency (energy balance, cross section unitarity: total cross section is the sum of the partials).
- Completeness (incident energies, emitted particles, energy spectra, angular distributions, recoils, etc.)

Basic neutron reaction models

- Optical
- Direct
- Pre-equilibrium
- Statistical

A schematic diagram of nuclear model classification is shown in Figure 8.

Model	Main feature	Which reaction	Comment
Optical	Imaginary V(r)	Total, elastic, CN formation	Important for all other models
Direct	Fast process	Inelastic	Most complicated physics
Preeq	Intermediate	nn', np, ng, ...	Last highly inspirative idea
Statistical	Slow process	All reactions	Compound nucleus mechanism

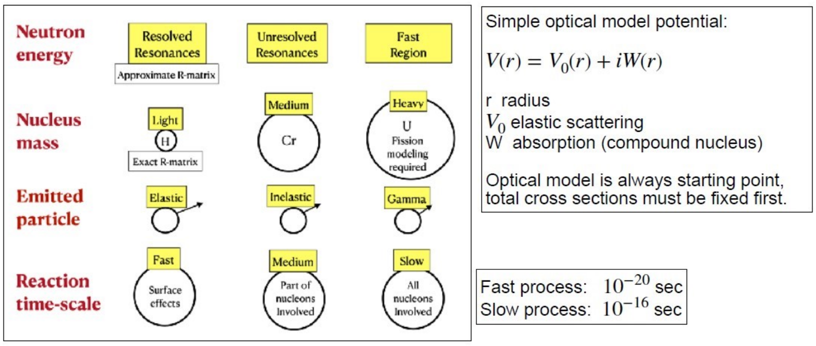

Figure 8 Schematic diagram of nuclear model classification

Fission is more than cross sections, of great importance are **fission observables**:

- Average number of neutrons per fission nu-bar (MF1, MT452)
- Prompt fission neutron spectra PFNS (MF5)
- Fission yields FY ⇒ radioactive, decay data needed (Decay Sub-library)

User's view: Fission produces 2-4 neutrons, several γ-rays, 2 fission products (radioactive) and releases ~200 MeV energy

Physicist's view: double-humped fission barrier, deformation, tunneling. Fission is extremely sensitive to height of barrier.

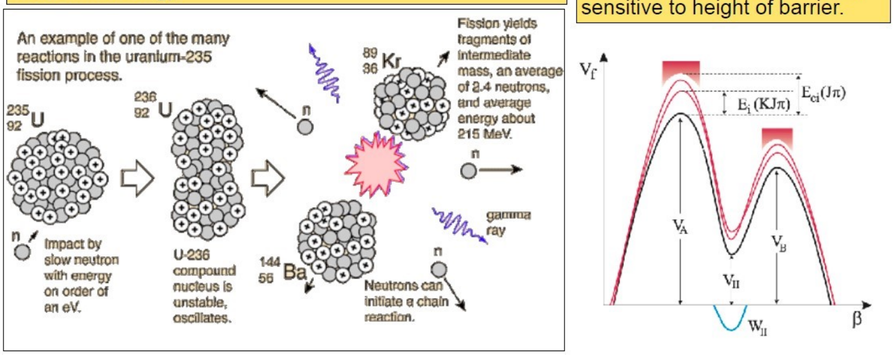

Figure 9 Fission cross section from different perspectives

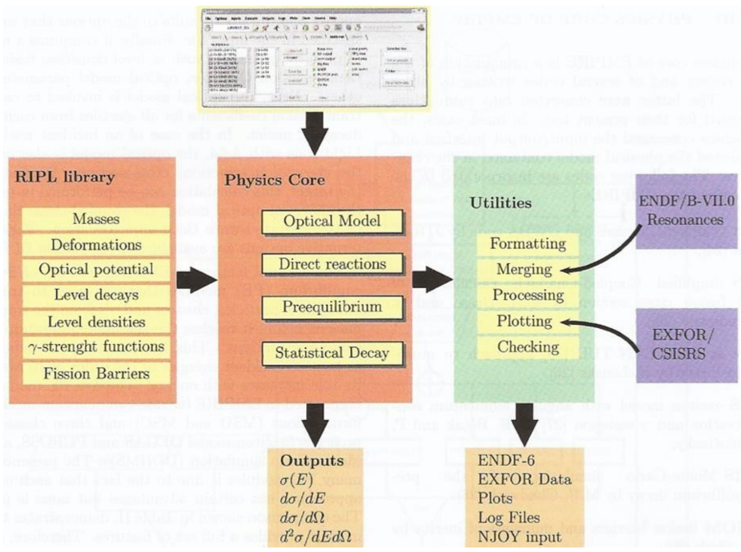

Figure 10 Schematic structure of the EMPIRE nuclear model code

Nuclear Fission

Fission is the most complicated reaction tin the model (Figure 9).

Nuclear Reaction Model Codes

EMPIRE https://www-nds.iaea.org/empire/index.html

EMPIRE is a modular system of nuclear reaction codes (Figure 10) for advanced modelling of nuclear reactions using various theoretical models. The system was developed through broad international collaboration and features a graphical user interface and integrated ENDF formatting and plotting with experimental data from the EXFOR database.

Original authors: Herman & Marcinkowski circa 1970 in Warsaw. Today, this is a top code for neutron cross section calculations, maintained by Capote, IAEA & Herman, LANL. EMPIRE puts together partial codes developed independently. It is coupled to libraries, equipped with scripts, interfaces and utility codes. It is a highly popular complete tool for nuclear data evaluation.

Advantages:

- Great flexibility for use with a variety of optical models and other parameters directly from the RIPL library.
- Advanced models for describing the fission reaction.
- Graphical user interface.
- Integration with the EXFOR database for quick comparison with experimental data.
- Formatting of results in ENDF-6 format.

Drawbacks:

- Less easy to install the most recent version, mostly because of the graphics interface, however "frozen" versions are available for Linux and Windows.
- Calculations are performed mostly one nuclide at a time.

TALYS https://tendl.web.psi.ch/tendl_2019/talys.html

TALYS is an open source software package (Figure 11) with a GPL license for simulating nuclear reactions. It is the heart of six TALYS-related codes for the creation of the ambitious TENDL Library, including random files by sampling model parameters for use in the so-called Total Monte Carlo technique.

- Evaluated data files and covariance matrices for an unprecedented number of nuclei.
- Uncertainty propagation from basic data to reactor calculation using the Total Monte Carlo approach.
- Feedback from integral data as a final step in improving the quality of nuclear data.
- Evaluated data files for various incident particles (neutrons, protons, deuterons, ^{3}He, alphas, protons).

TALYS is tuned to calculate cross sections for practically all known nuclei including isomers, around 3000 in total.

Advantages:

- Fortran source code is easy to install and very robust. The latest version also includes a module to produce results in ENDF-6 format.
- Detailed documentation.
- Bulk calculations can be performed – in fact, TENDL libraries are based heavily on TALYS.

Drawbacks:

- Less flexibility in the choice of theoretical models.
- No graphical user interface.
- Some of the high-fidelity major actinide evaluations in TENDL-2019 and some structural materials are taken from ENDF/B-VIII.0 (which are EMPIRE evaluations).

Neither of the two codes listed above would handle very light nuclei since statistical theory does not apply. In such cases, special treatment is required, which usually involves R-matrix fitting codes.

Other codes exist, but they are usually focused on specific reaction channels.

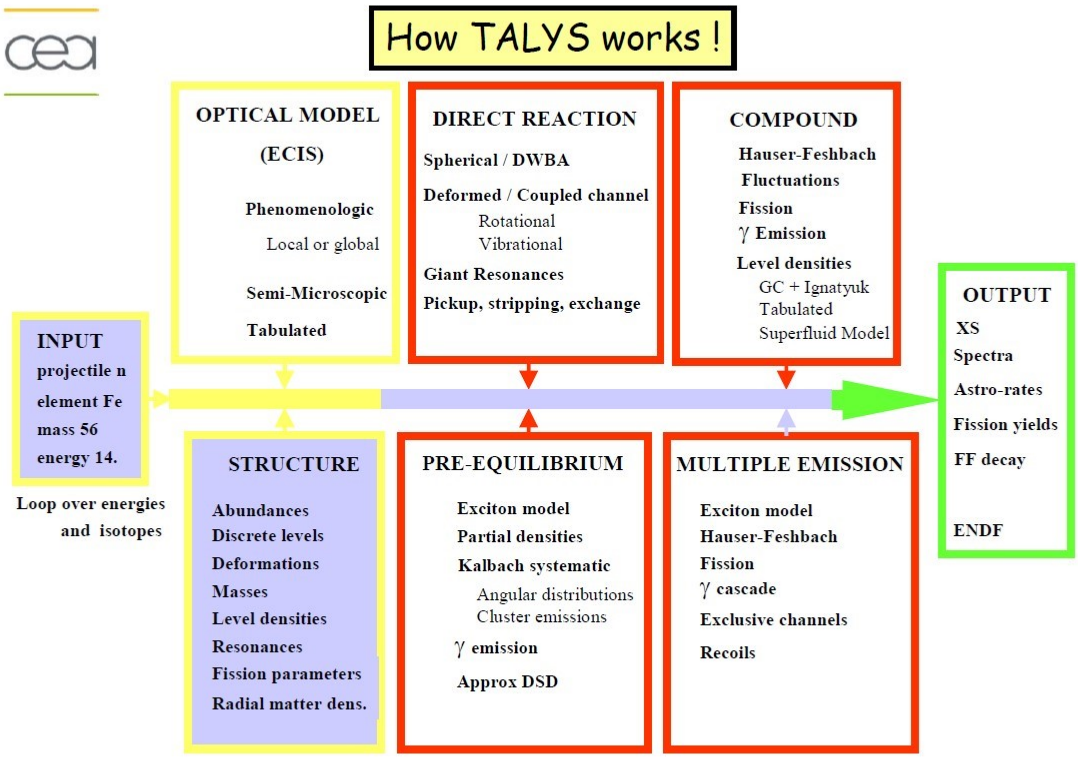

Figure 11 Schematic structure of the TALYS nuclear model code

Summary and Conclusions

An introduction to experimental nuclear data and theoretical nuclear reaction modelling was presented, with a focus on links to relevant information. More than 50 years of experimental and theoretical work has been outlined in some of the major databases and nuclear reaction modelling codes, namely:

- EXFOR and CINDA for nuclear reaction data. o ENSDF and NSR for nuclear structure and decay data
- RIPL as the derived data base combining information relevant for nuclear. model calculations
- Atlas of Neutron Resonances o ICSBEP, IRPhE and SINBAD for integral data
- EMPIRE and TALYS nuclear model codes

These are the foundations for nuclear reaction data evaluation. Nuclear data evaluation is the subject of a separate presentation.

Details on the theoretical basis for nuclear reaction modelling is beyond the scope of the present work. Interested readers should seek workshops specifically dedicated to the use of such codes.

Acknowledgment

Some of the figures and text were provided by P. Obložinský.

Nuclear Data – Evaluation Process and Evaluated Data Libraries

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Faculty of Mathematics and Physics, University of Ljubljana
Slovenia

Objectives and Overview of the Content

The objectives are to:

- Define the process of "nuclear data evaluation".
- Describe the basic principles of data evaluation with reference to:
    - Optimization of nuclear model calculations.
    - Selection and adjustment of experimental data.
    - Introduction of experimental data into the evaluation process.
    - Resonance parameter evaluation.
    - Data file assembly.
    - Data file verification.
    - Data file benchmarking and validation.
- Present the processing of evaluated data into application libraries.
- Discuss the whole-library verification and validation.

Introduction

Evaluated nuclear data are the starting point in generating application libraries for specific particle transport codes. They exist for different incident particles, such as neutrons, charged particles, photons and electrons. The data source for evaluation are theoretical nuclear models and experimentally measured values.

Nuclear data evaluation is a lengthy process, which may be subjective in some cases and only resemble the art in science, but if it was easy and unambiguous, the work would have been done a long time ago. Subjectivity in the process is the reason why different evaluation for the same nuclide in different evaluated data libraries differ, even though the evaluators have essentially the same experimental data at their disposal. Still, the performance of different libraries on certain benchmarks can be similar, because the adjustment to single integral parameters is not unique and the cancellation of errors occurs. Nevertheless, enormous progress has been made and convergence is observed, which is reflected in the overall improvements in the performance of new libraries.

What are "Evaluated Nuclear Reaction Data"?

Evaluated Nuclear Reaction Data are the most probable values of nuclear parameters (e.g., cross sections, emission spectra, etc.). They must be complete and defined at every point as a function of incident particle energy (i.e., interpolation between the points must be uniquely defined). They must also be computer-readable, which implies a strict format for data representation. In the past, several formats existed: KEDAK in Germany, UKNDL in the UK, SOKRATOR in the former Soviet Union, Livermore format and ENDF in the USA. Trough good documentation, the ENDF format has prevailed. Currently, we have:

- ENDF-6 format – widely accepted world-wide, supported by processing codes.

- GNDS format – new proposal, better adapted to modern IT technology, less experience and software support.

Major evaluated nuclear data libraries are usually supported by national projects. The most recent libraries are:

- ENDF/B-VIII.0 USA.
- JEFF-3.3 Europe.
- JENDL-4 Japan.
- CENDL-3.2 China.
- TENDL-2019 European endeavour (D. Rochman PSI, A. Koning IAEA) based heavily on TALYS calculations, but light elements, major actinides and some structural materials are taken from ENDF/B-VIII.0.

Other libraries are available (partial, special purpose, obsolete, etc.) For a more extensive list, see the ENDF retrieval web site at the IAEA https://www-nds.iaea.org/exfor/endf.htm.

Nuclear Model Calculations

In modern evaluations, one usually starts from a nuclear model calculation by applying a customized optical model potential and other embedded models to describe the different reaction cross sections, emission spectra and angular distributions. A library of optical model parameters can be found in the RIPL data-base, available from the IAEA. The model parameters in RIPL are usually generic, i.e., they apply to a range of nuclides in a certain mass range. When starting the evaluation process for a specific nuclide, a set of model parameters for the relevant mass range is usually a good starting point, but small adjustments are needed in some cases. One should also bear in mind that nuclear models are not perfect. So-called "tuning factors" are therefore available in the nuclear model codes to perform adjustments. The requirement for evaluation is that the calculated cross sections pass through the bulk of the consolidated experimental data (see next section) for all reactions so that subsequent adjustments when introducing "least-squares" fitting of experimental data do not introduce major changes. This is necessary to avoid the effects of non-linearities. Also, nuclear models are not applicable in the energy range where strong fluctuation in the cross sections are seen due to resonances. This has implications on delicate balance when deciding how high to go with energy in the resonance range. With an optimized set of nuclear model parameters, a complete set of cross sections, angular distributions, emitted particle energy/angle correlated spectra are obtained for incident particle energies above the resonance range. The data can be stored in the ENDF-6 or GNDS format, which are generally accepted for the exchange of nuclear data.

Open-source nuclear model codes which can be used for the calculations according to the above procedure are TALYS and EMPIRE. The use of such model codes fulfills physics constraints such as mass, conservation of energy and momentum, and unitarity (the total cross section being the sum of the partials).

Nuclear model parameters have associated uncertainties which are estimated from the systematics. By random sampling of the parameters (usually assuming a Gaussian distribution), a set of perturbed evaluated data files can be generated. A few hundred samples is usually sufficient for convergence of the cross-section mean values and their covariance matrix, which serves as the prior in a subsequent analysis in which experimental data are included.

$$Cov(p_i, p_j) = \sum_{k=1}^{N} \left\{ \frac{p_{i,j} \times p_{j,k}}{N} \right\} - \bar{p_i} \times \bar{p_j} \quad (1)$$

The covariance matrix from nuclear model calculations is usually very stiff, implying strong correlations between neighboring points. This would prevent any local adjustment in a shape which might be implied by experimental data. One way of "softening" the covariance matrix is to increase the diagonal elements of the matrix, but other approaches can also be found in the literature.

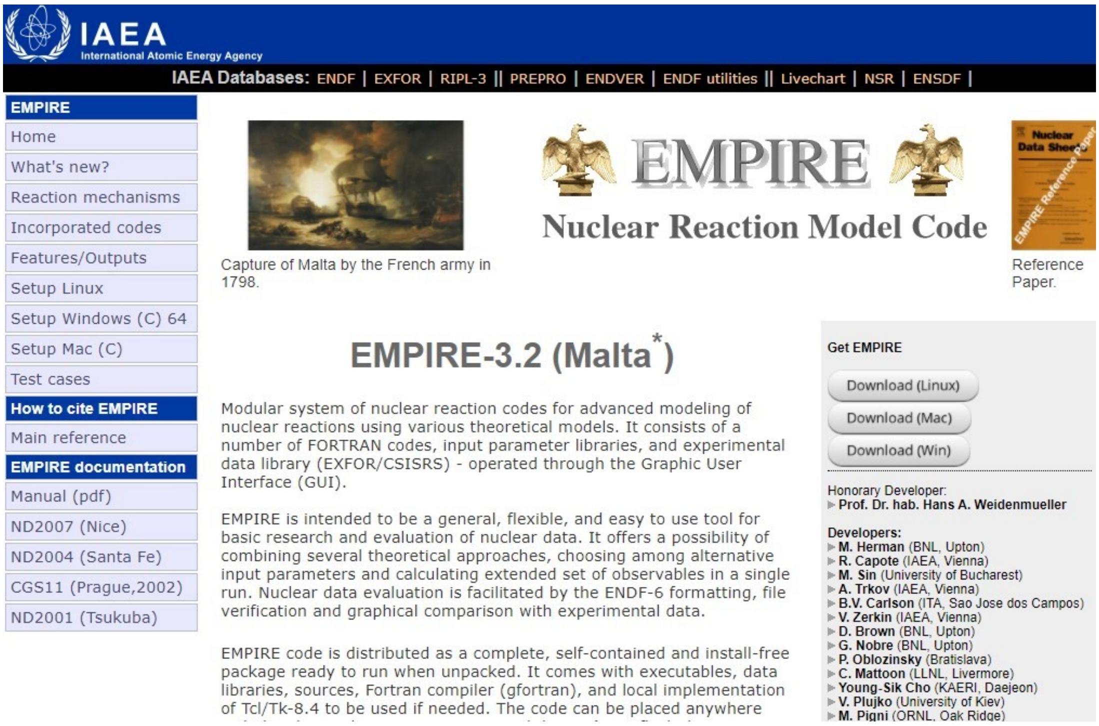
Web pages and links to two nuclear model codes are shown in Figure 1.

Experimental data

A comprehensive database of experimentally measured values is EXFOR. This database contains information in numerical form from papers published in scientific literature. It is very extensive for incident neutrons but is being extended with the data for other incident particles in recent years.

It is quite common that experimental data are discrepant either due to underestimated uncertainties, the use of obsolete standards or other unidentified errors in the measurement technique. The generalized least-squares method (GLS) is blind to such inconsistencies, therefore it is essential that the user removes the inconsistencies either by renormalizing the data to the most recent standards, adding contributions to uncertainties that were not accounted for, or by simply rejecting the data if the inconsistencies cannot be resolved. An example of the problem that the evaluator faces is shown in Figure 2. This is the main source of subjectivity in the evaluation process, which depends entirely on the evaluator.

https://www-nds.iaea.org/empire/index.html



https://tendl.web.psi.ch/tendl_2019/talys.html

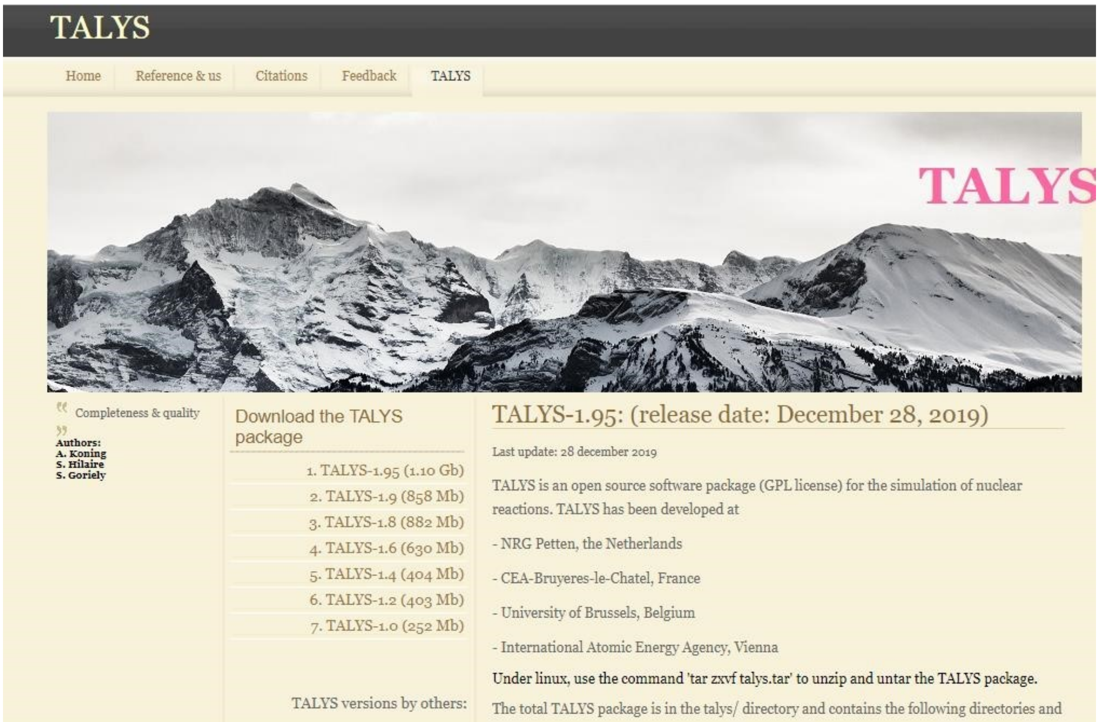


Figure 1 Web pages for the major nuclear model codes

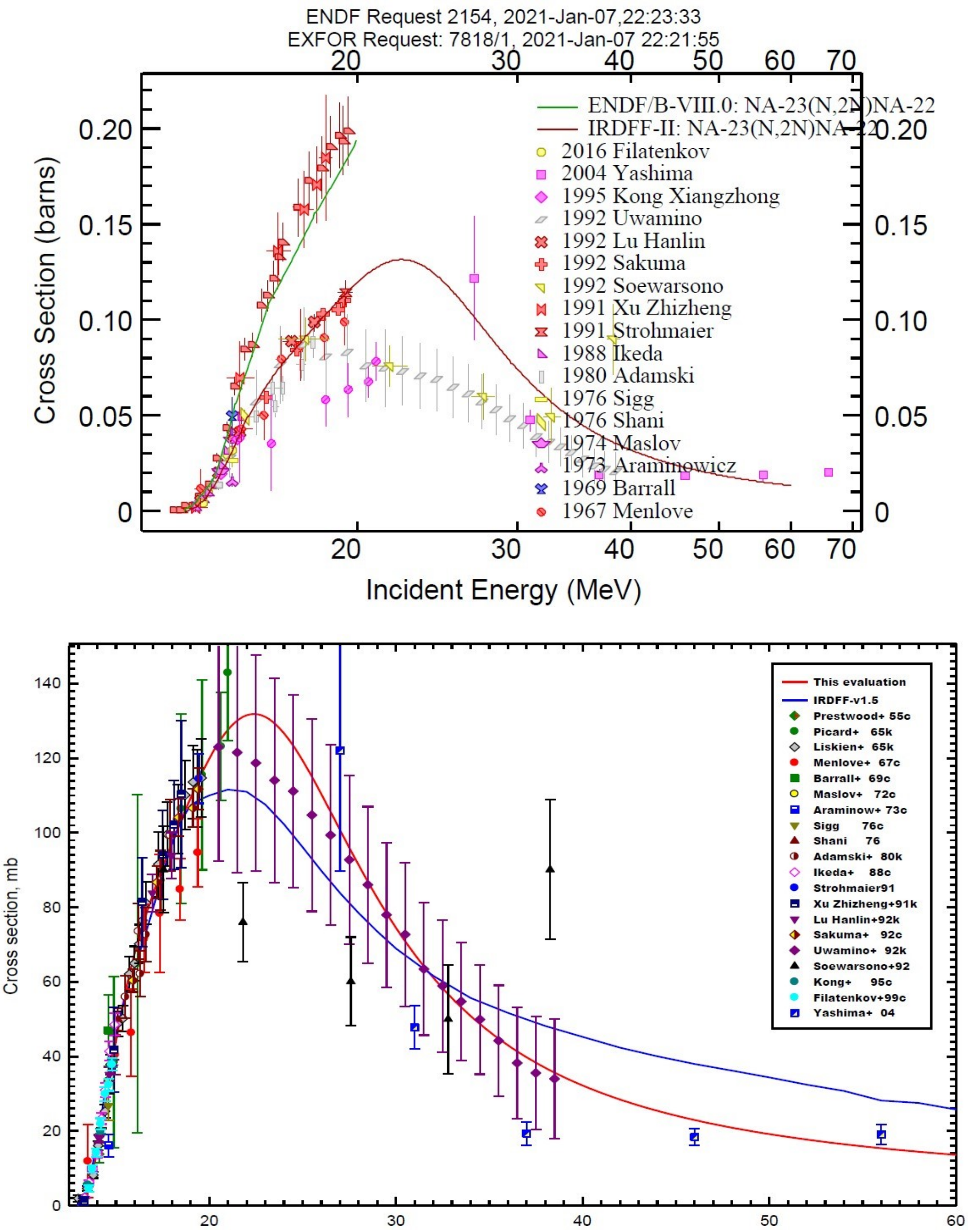


Figure 2 Raw EXFOR data for the 23Na(n,2n) reaction (top) and corrected data (bottom) from the evaluation work by K. Zolotarev for the IRDFF-II library evaluation

The GANDR system can be used to introduce experimental data into the evaluation. It makes use of the prior from nuclear model calculations and the set of selected consistent experimental data prepared by the user. The result is a final evaluated data file with covariance information for the energy region above the resonance range. The web page of GANDR is shown in Figure 3.

Resonance Range

Cross sections in the resonance region are a special problem. No theory can predict the position and strength of the resonances, therefore the resonance parameters are derived from capture, fission and transmission measurements using the time-of-flight method.

Figure 3 Web page of the GANDR System at the IAEA https://www-nds.iaea.org/gandr/

R-matrix theory is an elegant solution of the Schrodinger equation and scattering problem, illustrated in Figure 4.

The R-matrix theory is the legacy of two major Central European physicists: Erwin Schrodinger from Vienna and Eugene Wigner from Budapest. In honour of the theory, Schrodinger was depicted on a bank note of the former Austrian schilling, shown in Figure 5.

The main issue in the R-matrix theory is matching the wave function at the boundary of the nucleus. The theory is applied to define the rapid fluctuations in cross sections with a set of parameters, providing a major step forward in a sound theoretical basis for the resonance formalism compared to Breit-Wigner line shapes which were traditionally used in the past; the cross sections in the resonance peaks could be described empirically, but less well in the region between the resonances.

Resonance analysis of measured data provides the resonance parameters and covariance matrix of their uncertainties, which must be merged with the data evaluated above the resonance range. SAMMY, REFIT and CONRAD are the codes most often applied for resonance analysis. The evaluator selects the experimental data sets (transmission, capture, fission), usually measured with the time-of-flight method, considering:

- Reliability (reputation of the experimentalist and the facility).

- Resolution of the experiment (flight-path length; longer flight path provides better resolution but poorer statistics). Resolution limits the upper energy range above which too many resonances are lost.

- Energy range (characteristics of the facility, such as the neutron source spectrum, pulse width, etc., which cause limitations which imply that the same measurement accureacy cannot be maintained over the entire energy range of interest.

Examples of resonance fitting are shown in Figure 6 for the case of ^{235}U. Note that in transmission experiments, fitting is applied to the dips in the neutron spectrum due to resonances. In capture experiments, fitting is applied to the yields of the gamma rays produced from the capture reaction by neutrons of a certain energy. In Figure 6, the axis labels of the function are on the right; the residual (the difference between the fitted value and the measurement) is on the left. Application of the R-matrix theory to the data with the generalized least-squares method produces a set of resonance parameters and their covariances.

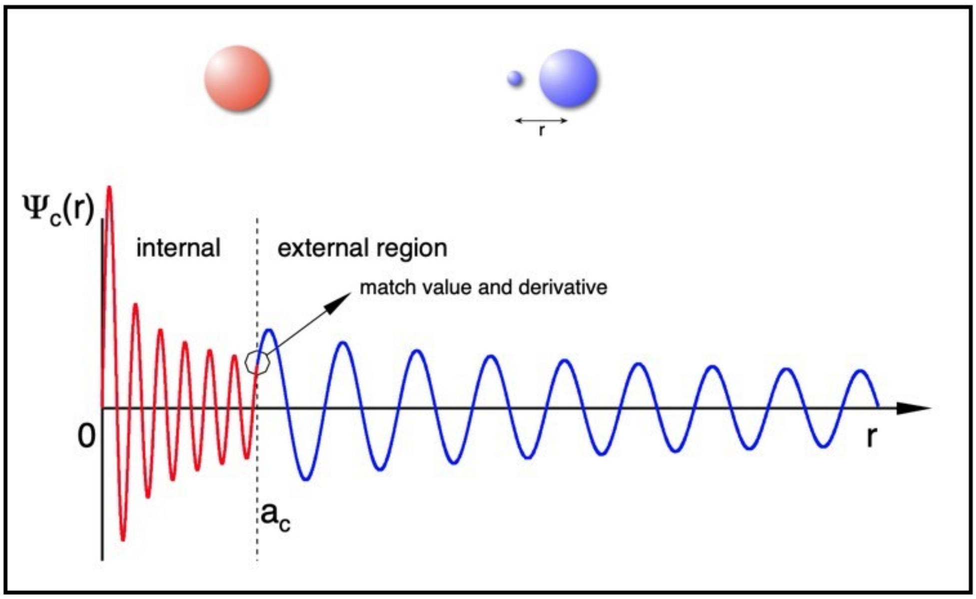


Figure 4 Illustration of matching the wave function at the boundary of the nucleus

Figure 5 Schrodinger depicted on the former Austrian bank notes

Evaluated Data File Assembly

Evaluation of the region above the resonance region must be combined with the resonance area consistently and strictly follow the ENDF-6 format rules. ENDF Utility Codes can be used to check the formal correctness and internal consistency of the file. These codes are CHECKR, FIZCON and PSYCHE, developed decades ago by C. Dunford at the Brookhaven National Laboratory.

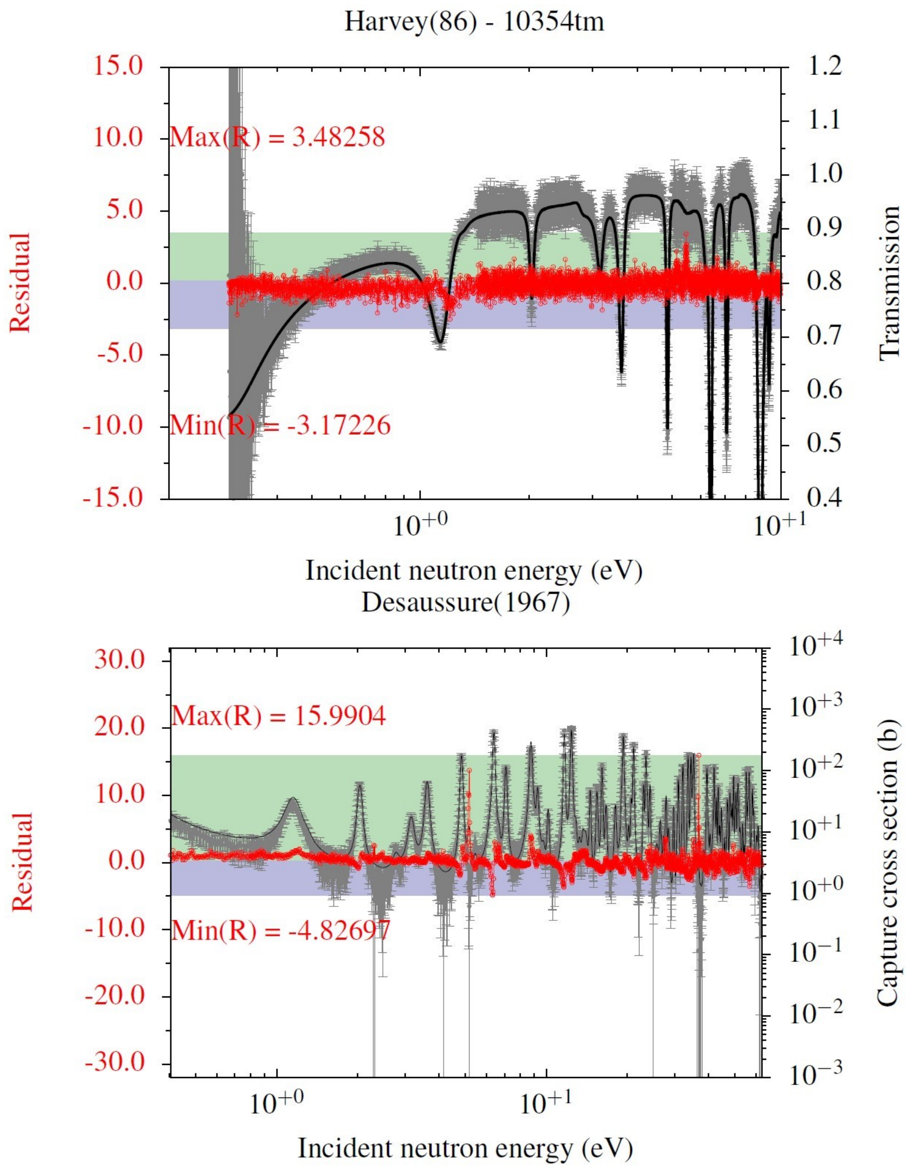


Figure 6 Example of the R-matrix fit of transmission and capture data.

The assembled file must also be processable. The most elementary codes which linearize the data (if logarithmic interpolation is prescribed), reconstruct cross sections from the resonance parameters and perform Doppler broadening in the resonance range are the LINEAR, RECENT and SIGMA1 codes from the PREPRO package (developed by D.E. Cullen), which also includes a number of other useful utilities. The PREPRO codes and documentation are available freely from the IAEA web site: https://www-nds.iaea.org/public/endf/prepro/. Messages from these codes should be inspected to trap possible errors or inconsistencies in the file.

With a formally corrected and processable file, the next step is to ensure that it truly represents the experimental data used in the evaluation. Although this seems trivial, it surprises are frequent, since the number of operations on the data is large and the data sets of modern evaluations are also large. The ENDVER package was designed for these types of comparison and is available from the IAEA https://www-nds.iaea.org/public/endf/endver/ . ENDVER components are included in the EMPIRE graphics interface and the on-line data retrieval software at the IAEA and the NNDC/BNL websites.

Nuclear Data Processing for Applications

It is said that "the proof of the cake is in eating it", therefore it is necessary to process the evaluated data file into a form which can be tested in transport calculations. Until recently, NJOY has been the only generally available code (with some restrictions) for producing application libraries for deterministic and Monte Carlo transport codes. Recently, other codes have become available, however NJOY remains the "golden standard" since it isan open-source code and freely available. Thus, a new evaluation is processed into a transport file and is ready for integral testing. The major data processing codes are:

- PREPRO suite of codes, which always been freely available from the IAEA. The codes are very robust, but they perform only basic operations and graphic data display https://wwwnds.iaea.org/public/endf/prepro/

- NJOY was the only code for producing application libraries for a long time. The code was not available to some users. This is no longer the case. NJOY is freely available from the LANL web site http://www.njoy21.io/

- Recently, more codes have become available:

  - FRENDY from Japan (K. Tada) https://jopss.jaea.go.jp/pdfdata/JAEA-Data-Code2018-014.pdf is a comprehensive data processing system, but mainly limited to incident neutrons.

  - GRUCON from Russia (V. Sinitsa) https://www-nds.iaea.org/grucon/ is a comprehensive code system with performance features similar to NJOY.

  - ACEMAKER from the IAEA (D.L. Aldama) https://wwwnds.iaea.org/publications/nds/iaea-nds-0223/ is a stand-alone module which complements PREPRO codes to generate ACE libraries.

Several other codes are under development, but not generally available. The codes involved in IAEA activity for the validation of codes which produce libraries in ACE format are mentioned in the report https://www-nds.iaea.org/publications/indc/indc-nds-0798.pdf

Nuclear Data Validation

Nuclear data validation is the final steps which determines the applicability of the evaluated data in practical applications. The simplest integral tests are thermal cross sections, resonance integrals and spectrum-averaged cross sections. Thermal cross sections and resonance integrals can be compared to the Atlas of Neutron Resonances, but very often the Atlas values have already been used during resonance analysis. Another practically independent source is the Kayzero library (http://www.kayzero.com/k0naa/k0naaorg/Nuclear_Data_SC/Entries/2019/4/15_Update_of_k0database_Ba-131.html) for neutron activation analysis according to the k_0 standardization method. The k_0 and Q_0 parameters can be converted into thermal cross sections and resonance integrals of capture cross sections for all nuclides which have radioactive capture products that emit distinctly measurable gamma photons.

Another useful test is the comparison of spectrum-averaged cross sections. The ^{252}Cf spontaneous fission (SFNS) spectrum is a standard, which means that its shape is very well known. Comparison of the average cross sections measured in the ^{252}Cf

SFNS with the calculated cross section provides an estimate of the relevant cross section in the fast energy range.

Measurements of spectrum-averaged cross sections in the ^{235}U prompt fission neutron spectrum (PFNS) are quite common, but one should be very careful in their interpretation since many sources of uncertainty exists and are more difficult to control. Experimental data can be found in the EXFOR data base.

Astrophysicists maintain a data-base of the Kadonis library of MACS cross sections, which are average cross sections in a Maxwellian spectrum with a temperature of 30 keV. However, one should be careful with this database. For a long time, the community used the gold cross section for normalization as standard, which was incorrect by a few percent. The error was eventually acknowledged, and the KADoNiS database is now under review (https://exp-astro.de/kadonis1.0/), but this is still work in progress.

Compilations of benchmark experiments such as ICSBEP and IRPhE contain detailed descriptions of integral experiments (geometry, material composition) and measured integral quantities (multiplication factor, reaction rate ratios, etc.). The ICSBEP compilation contains over 5000 cases, many of which are often quoted in publications on data validation. The experiments are simulated by Monte Carlo calculations using the evaluated data files. One should bear in mind that a transport calculation is done with cross sections of millions of numbers, and only one number exists for comparison, i.e., the multiplication factor. It is very easy to make adjustments to an evaluation to fit one integral benchmark, but the adjustment could have an adverse effect on other benchmarks. The adjustment is not unique, which explains why major libraries perform similarly well in certain groups of benchmarks despite significant differences in the cross sections. Finally, many of the benchmarks contain unreasonably small uncertainties and unidentified systematic errors. Reporting validation work on criticality benchmarks is tricky and requires great care.

Similarly, the SINBAD compilation contains shielding experiments. These are sensitive to different cross sections and in different energy ranges, therefore they are complementary to the criticality benchmarks.

The validation step is an integral part of the evaluation process. The evaluator must select as many reliable benchmarks as possible which are sensitive to the nuclide under evaluation. The performance in benchmark experiments together with cross section sensitivity profiles guides the evaluator in identifying the cross section and the energy range which is most likely responsible for degraded performance. This can help discriminate between discrepant differential data, but it should not be used to override indications from differential data.

The uncertainties are another area of vigorous argument. It is known that the uncertainties in the multiplication factors propagated from nuclear data are much larger than the reported uncertainties in the multiplication factors of criticality benchmarks. First, the uncertainties in many criticality benchmarks are under-estimated. Second, the average number of neutrons and the fission cross section in the differential data are practically uncorrelated because they are measured in completely different experiments, however a very strong anti-correlation exists between these two quantities in critical systems. One is tempted to include criticality problems (and integral data in general) in the GLS fitting process, but this is bad practice since the sensitivities of nuclear data to the measured quantities are generally incomplete and lead to biases and unrealistic uncertainties in problems which are not really similar to the integral experiments included in the evaluation. Resolving the issue of propagating uncertainties from nuclear data to the multiplication factor is the subject of in-depth investigation.

Stories of Success and Failure

The project WPEC-SG40 (CIELO) was initiated by the OECD/NEA Data Bank to engage the international community in improving the evaluated data files of the major nuclides in nuclear reactor technology. The project was successfully completed. Following the success of CIELO, the International Nuclear Data Evaluation Network (INDEN) was set up at the IAEA.

• Evaluations from the CIELO project for the most important fission reactor constituents received recognition by being incorporated into the ENDF/B-VIII.0 library.

• Benchmarking of the ENDF/B-VIII.0 library on criticality benchmarks significantly reduced Chi^2/DoF over 119 benchmarks, commonly used at LANL for data validation, compared to its predecessor ENDF/B-VII.1. This was a major success (see Figure 7).

• Unfortunately, ^{56}Fe performance in shielding degrades during thick penetration problems, e.g., leakage current from a 60 cm diameter sphere with a ^{252}Cf source (40 % under-prediction of the flux near 2 MeV, see Figure 8).

• The problem was in the newly measured inelastic cross sections used in the evaluation, which revealed to have a normalization problem.

• Improved 56Fe-56 evaluation is available at the INDEN web site https://wwwnds.iaea.org/INDEN/ .

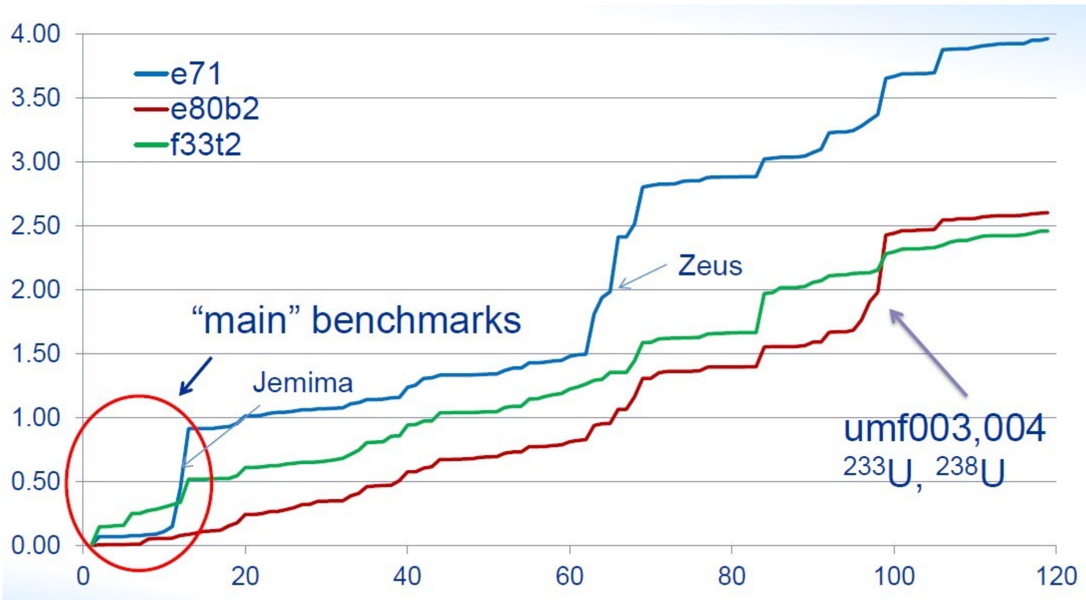


Figure 7 Reduction of Chi-squared per degree of freedom using CIELO evaluations for 235,238U contained in the ENDF/B-VIII.0 library (label "e80b2") compared to ENDF/B-VII.1 (label "e71")and JEFF-3.3 (label "f33t2")

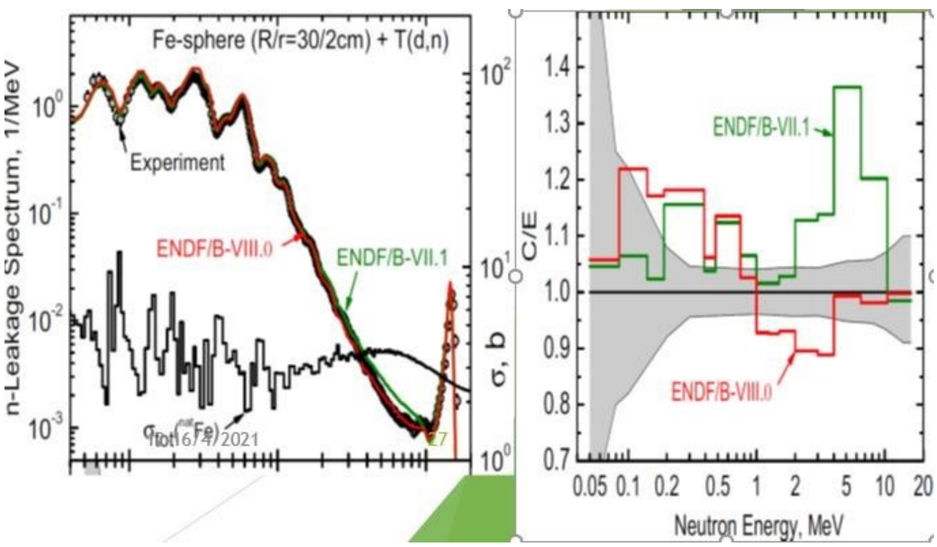


Figure 8 Validation study of ^{56}Fe on the leakage spectrum from a thick iron sphere with a ^{252}Cf source in the centre shows degraded performance in the energy range 1-4 MeV due to the inelastic cross section of ^{56}Fe in ENDF/B-VIII.0.

Conclusions

The basic steps in the nuclear data evaluation process were discussed, namely:

- Nuclear model calculations.
- Selection of experimental data.
- Special treatment in the resonance evaluation.
- Evaluated data file assembly and verification.
- Data file validation.

In modern evaluations, validation is an integral part of the evaluation process, but it requires great care to avoid introducing biases into the data. Nuclear data evaluation is a lengthy process (^{235}U evaluation took over three years to complete).

Evaluation is to some extent subjective (the evaluator often must make choices when faced with discrepant data). New measured data may not always be more accurate.

Extensive validation is necessary on different types of benchmarks, considering that integral benchmarks may also contain systematic errors.

Nevertheless, progress has been made and some degree of convergence has also been achieved. Uncertainty estimates are becoming more reliable. However. much work is still to be done!

Acknowledgment

Some of the figures and text were provided by P. Obložinský.

Fuel Cycle Reload Analysis of US LWRs

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Approximately one year before a single UO_2 fuel pellet is pressed and fresh fuel assemblies are manufactured for a fuel cycle reload, design calculations are performed to effectively guarantee the performance of a light water reactor (LWR) core for an upcoming fuel cycle. Energy requirements must be fulfilled in conjunction with all other reactivity, thermal, and operational limits. Furthermore, several months prior to a reactor startup, the proposed core must be licensed by the regulating authority. Therefore, all built-in design conservatisms (i.e., target margins) established in advance must satisfy all safety, operational, and regulatory constraints, later on, during operation (i.e., actual margins). The magnitude of target margins directly impacts cycle energy efficiency, which is why this design allowance is sometimes referred to as the "cost of margin" because it ultimately affects the cost of electricity generated by a LWR. This summary and presentation illustrate the role of nuclear fuel management software tools in a typical US LWR core reload design process, highlighting some of the history, current practices, and modern advancements in the field.

Introduction

The last quarter century of advancements of nuclear fuel management optimization have been considerable and widespread [1-3], noting that Reference [1] provides an extensive list of fuel management works from the 1980's into the 2000's. Therefore, it is not surprising that the design of today's nuclear fuel reloads can be a highly automated process that is often accompanied by sophisticated optimization software and graphical user interfaces to assist core designers.

Most typically, among other objectives, optimization software seeks to maximize the energy efficiency of a fuel cycle while satisfying a variety of safety, operational, and regulatory constraints. Concurrently, the general trend of the nuclear industry continues to be one of pursuing higher generating capacity (i.e., power up-rates) alongside cycle length extensions (i.e., more energy for a longer time!). Therefore, as these increasingly invaluable software tools and ambitious performance goals are pursued in unison, more aggressive core designs ultimately emerge that effectively minimize the margins to limits, and in some cases, may turn out less forgiving or accommodating to changes in underlying key assumptions. Hence, it is very important for nuclear designers to carefully understand the design, licensing, and operational management constraints involved with a commercial LWR, especially before attempting to employ the sophisticated analysis tools that are now readily available.

Reactivity, Thermal, and Mechanical Margins

Figure 1 illustrates graphically the difference between the hypothetical target and actual thermal margins. In essence, during the design and licensing stages of a core reload, target margins to limits are pre-established, and these margins must be large enough to accommodate expected changes or fluctuations in underlying design assumptions. Therefore, if all goes according to plan, when a particular design goes into operation, the actual margin to any limit will be positive and finite (i.e., limits are not violated). In fact, the allowance that exists between a target and an actual margin is sometimes referred to as "the cost of margin." This is because excessive conservatism in target margins could otherwise be converted into energy gains or cost savings. Thermal margins are often established by limiting transients, which can have exposure dependencies (e.g., scram worth can vary with axial power shape, which changes versus exposure).

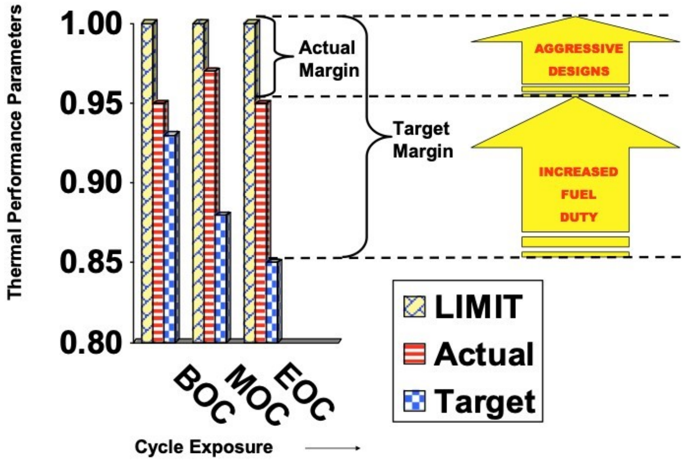

Figure 1 Illustration of Actual and Target Thermal Margins [1]

In some cases, the accuracy with which the design tools at hand can predict or simulate actual operation inherently determine a baseline margin allowance that must be absorbed. In other situations, however, the magnitude of expected changes in underlying design assumptions is proportional to the design's built-in flexibility and can be an overriding factor in how large or small the cost of margin is. Ideally and in principle, with perfect software tools and zero changes in underlying assumptions, the target margins would equal actual margins, and the cost of margin would simply vanish. Reality, however, dictates otherwise and experienced designers play a key role in properly establishing target margins to mitigate the cost of margin, but most importantly, avoid the need for re-designing core reloads, which can happen.

Figure 2 shows, more specifically, the relationship of the linear heat generation rate (LHGR) to the many constraining factors that establish fuel performance limits (base figure extracted from Ref. [NEA#7072, 2012]). On that figure, for illustrative purposes only, we have superimposed a hypothetical representation of what would correspond to each fuel rod's limiting operational power histories (red circles) to highlight the fact that the thermal mechanical design limit (the blue line) is, in fact, a bounding limit intended to contain design and actual operational conditions from approaching or crossing over into situations that lead to the limits of cladding strain, fuel melt, PCI failure, hydrogen-assisted cracking, and cladding liftoff.

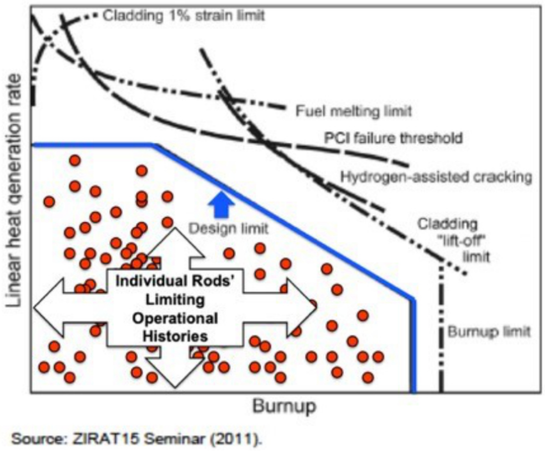

Figure 2 Relationship of LHGR to fuel performance limits (from ref. [NEA#7072, 2012])

Reload design process and constraints

Understanding limitations and the role of the designer

It is safe to say that a large fraction of LWR fuel vendors and utilities today have access to a wide variety of optimization tools from a wide variety of sources, ranging from commercially available products, to software from research and development centers, universities, or in-house developments. The usefulness of these tools in assisting and efficiently streamlining engineering efforts is certainly undeniable. However, regardless of the nature and origin of the software, for users and developers alike, it is crucial to understand the specific realm of applicability of each specific software tool, to carefully recognize its limitations, and to understand that there exist many "non-standard," though typical, constraints that can affect a real core design but which are often difficult, if not outright impossible, to implement into an automated setting. Thus, in a way, highlighting the unique role and irreplaceable nature of the experienced (human) designer in light of real-life limitations. Furthermore, it should be noted that the advent of fuel management optimization tools has, likely or in part, facilitated the creation of more aggressive core designs, particularly in response to newly emerging power uprate and cycle extension design requirements. With all else being equal, the end product is a reduction of excess design margin, which can imply a loss in design flexibility and lead to increased fuel duty, two features of a core design that can have highly undesirable implications; namely, a higher risk of re-design, and potentially reduced fuel reliability, respectively.

Consequently, a key aspect of future human intervention into the fuel management optimization process will be the ability effectively and carefully "invest" available margin into areas other than increasing energy efficiency. For example, to develop objectives and constraints that would render core designs more resilient to changes in underlying design assumptions, or perhaps even increase fuel reliability. Prior to understanding what is herein considered to be a standard versus a non-standard constraint, the "big picture" of core reload design must be presented, accordingly, Figure 3 provides a schematic description of a typical reload engineering process, as it applies to a Boiling Water Reactor (BWR). It should be noted that the timetable presented is only representative and approximate and can vary considerably in actual situations. Also, it is herein implied that an equivalent process applies to a PWR. The primary stages include design, licensing, and startup (SU), followed by an operational stage that extends from BOC to EOC. The comparison of actual (online) versus

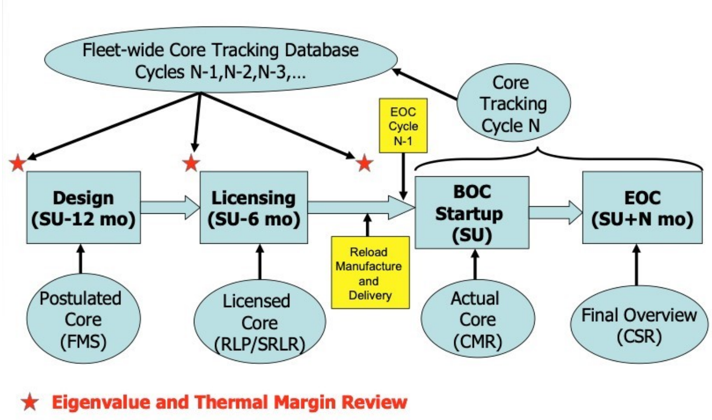

Figure 3 Overview of the BWR Reload Engineering Process (GE example from Ref. [1])

predicted (offline) thermal margins and eigenvalues is denoted as "core tracking." To supplement Figure 3, Table 1 below describes some of the standard acronyms employed in a typical General Electric BWR reload activity.

Table 1 Acronyms employed in a typical BWR reload process (GE example from Ref. [1])]

Acronym	Description
SU	Startup
FMS	Fuel Management Summary
RLP	Reference Loading Pattern
SRLR	Supplemental Reload Licensing Report
CMR	Cycle Management Report
CSR	Cycle Summary Report
BOC	Beginning of Cycle
EOC	End of Cycle
Cycle N	Present Cycle (or cycle being designed)
Cycle N-1	Prior Cycle (or currently operating cycle)

Potential multiple core designs for the same reload

From Figure 3, the primary study that determines the bundle design and batch size for a given reload (Cycle N) occurs roughly a year prior to the end of an operating cycle (Cycle N-1). That study is documented within the Fuel Management Summary (FMS, or equivalent document) and contains a thorough multi-cycle analysis that follows the newly designed fuel batch through several future cycles until the Cycle N feed fuel is either discharged from the core or an equilibrium cycle is reached.

Ideally, the underlying assumptions employed within the FMS do not change significantly such that the same core design becomes the Reference Loading Pattern (RLP) that is employed for licensing studies (e.g., transient analysis) and is also employed to define the safety and operating limits provided to the regulatory authorities in the Supplemental Reload Licensing Report (SRLR). Ultimately, if no significant changes occur in the underlying assumptions, the actual loaded core documented in the Cycle Management Report (CMR) can, in fact, be the same as the RLP, thus providing a fully consistent and seamless core design process from the FMS through the CSR. This situation, in fact, constitutes the ideal case during which "standard constraints" are sufficient and adequate to help accommodate the actual variations in the underlying assumptions. A typical source of underlying assumptions, for example, is the Energy Utilization Plan (EUP) that is provided to the fuel vendor by the utility in advance of the multi-cycle analysis that leads to the FMS.

In contrast to the fully-consistent core design described in the previous paragraph, 12 months is quite a long period of time for an operating nuclear power plant. Therefore, more often than not, several underlying assumptions can change significantly between issuing the FMS and the actual startup of the upcoming cycle. Therefore, alterations to the original core design reported in the FMS are not uncommon, and thus, revised core designs at each stage of the process may be required, with some adjustments being more significant than others. Clearly, changes can yield difficulties after the licensing studies have been completed. Sometimes, a relatively simple "license validation" study might be sufficient, but in extreme cases an entirely new licensing analysis might be required. Later in the process, the logistical difficulties associated with postmanufacturing changes can be extremely complex and costly. For the purpose of this summary, the root causes of these unpredictable changes are what are herein defined as "non-standard" constraints to a reload design. The sections that follow provide some examples of "standard" and "non-standard" constraints, both, as defined within the context of this article. Also, as previously noted, a BWR is herein employed as a common denominator for LWRs, for illustration purposes.

Standard Constraints

Some of the most common constraints are very well-known. In fact, given their nature, they can be more easily implemented as constraint and/or objective functions within nuclear fuel optimization software. Some typical examples include:

- Burnup/Exposure limits on a pin, assembly, or batch basis
- Thermal Margins (e.g., MFLCPR, MAPRAT, MFLPD)
- Cold Shutdown Margin (SDM)
- Hot Excess Reactivity Constraints on Reactivity Coefficient Values
- Number of fresh bundle types and number of lattice types per bundle (axial zones)
- Enrichment and burnable poison loading restrictions

Non-Standard Constraints

A "non-standard" constraint is more of an unpredictable moving target, therefore much more difficult to prevent or prepare for. These are typically triggered by changes in underlying assumptions of a magnitude large enough to promote a core change or redesign during one the various stages outlined in Figure 3. A few non-standard constraints are described below, while additional constraints and detailed real examples are discussed in Reference [1]. Needless to say, there are many more of these types of unpredictable changes in underlying assumptions that can effectively constain a core design.

Cycle N-1 Energy Uncertainty: An unexpected rescheduling of a refueling outage, for example, can change the end-of-cycle (EOC) date for Cycle N-1. Therefore, this can imply a smaller or larger actual requirement of energy during the prior cycle that can affect the performance of a future cycle design by carrying forward more or less reactive burnt bundles, respectively. Similar outcomes can be expected from significant variations in the actual versus projected capacity factor of a power plant. The primary direct impact is on the reactivity of the future cycle, lowering or raising it.

Eigenvalue Basis Drift: Due to the finite accuracy with which the actual trend of eigenvalues can be predicted to behave with the available core simulator models, deviations between actual and target eigenvalues can have a significant impact on the reliability of Cycle N predictions and also on the actual Cycle N-1 operation. For instance, a drift between the target and actual hot operation eigenvalue of ±0.001 at EOC can impact the cycle length by approximately ±100 MW-day/Short-Ton (MWd/ST). Likewise, deviations from the cold eigenvalue basis can impact important standard constraints, such as the cold shutdown margin of a core, a limit dictated by plant technical specifications.

Fuel Reliability Event: A fuel failure, often referred to as a "leaker," can considerably impact the entire reload design process. In mild situations, power suppression of a leaker cell using control blades constitutes an operational adjustment and change in operational strategy for Cycle N-1. In contrasting severe situations, a mid-cycle outage to remove affected bundles may entail a complete core redesign and re-licensing midway through the original cycle, which impacts Cycle N-1 and Cycle N.

Objective of the upcoming presentation and workshop

The presentation associated with this summary will include a description of the "big picture" herein provided, and an overview of the reload design process for a typical US LWR. Also, it will present some simple illustrative examples of typical calculations that are performed during the process to determine constraints, including examples of codes employed.

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Defence-in-Depth in Design of Existing and New Nuclear power Plants

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Introduction

The lecture presents key principles of defence in depth for nuclear power plants (NPPs) as a hierarchical deployment of different levels of equipment and procedures to protect the integrity of barriers against radioactive releases to the environment. The importance and means for complying with the principles of defence in depth are discussed. The principles are illustrated with descriptions of mechanisms which challenge these barriers, methods which ensure compliance with defence in depth, and examples of deviations from defence in depth principles. More information about the subjects covered in this lecture can be found in reference documents [1] to [10].

Safety and safety objectives for nuclear power plants

According to the IAEA Safety Fundamentals [1], safety means the protection of people and the environment against radiation risk. The fundamental safety objective is to protect people and the environment from the harmful effects of ionizing radiation. The safety objectives for NPPs apply to all stages of an NPP's lifetime, including the planning, siting, design, manufacturing, construction, commissioning, operation and decommissioning stages. Safety at NPPs under the IAEA Safety Standards relates to potential harm to people and the environment due to radiation.

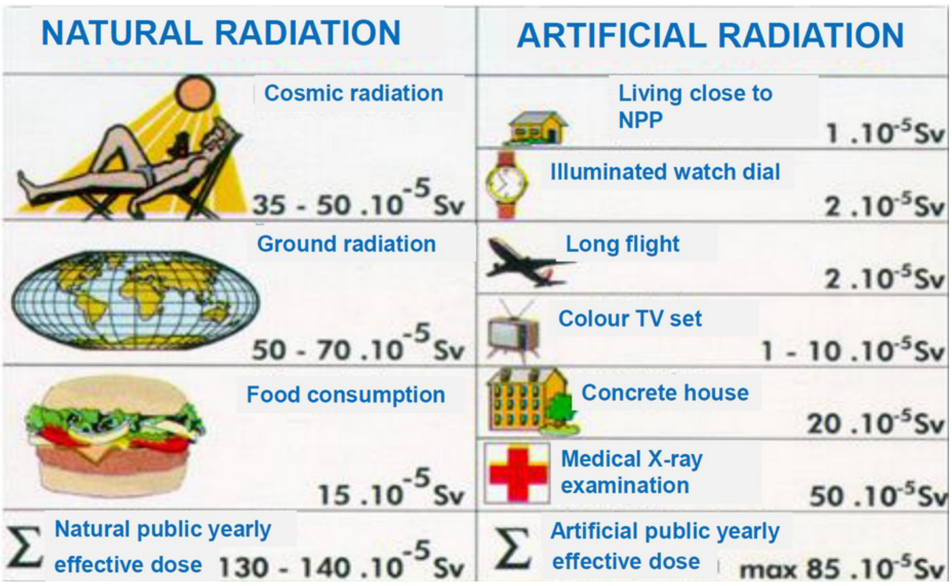


Figure 1 Sources of natural and artificial radiation.

Radiation is a natural phenomenon, and people in their normal life are continuously exposed to forms of radiation (Figure 1). Although the stochastic effects of any radiation dose cannot be fully excluded, non-negligible risk from NPPs is only associated with overexposure due to accidents (Table 1).

Table 1 Comparison of effective doses from different sources.

Effective dose [mSv]	Source of exposure
0.01	One dental X-ray examination or a colour TV
0.02	Nuclear weapon tests plus deposits after Chernobyl
0.1	One X-ray examination of lungs
0.4	Natural radioactive substances present in the body
1.5 - 7.5	Average annual dose from natural sources in Europe (UK to Finland)
12	Computerized axial tomography of stomach
1000	Symptoms of radiation sickness begin to appear if received in less than 24 hours
4000	Lethal radiation dose, the person can be saved with good care
6000	If received suddenly is likely to cause death
10000	Life can not be saved even with best care

Note: The gray, or Gy, is the derived unit of ionizing radiation dose, defined as the absorption of one joule of radiation energy per kilogram of matter. The sievert, or Sv, is a derived unit of ionizing radiation dose and replaced the older unit rem, where 1 Sv = 100 rem. Sv measures the health effect of low levels of ionizing radiation on the human body; it is applied to equivalent doses, effective doses, and committed doses. The biological effects of radiation depend on the type of radiation. For gamma and beta radiation, 1 Gy causes an effective dose of 1 Sv, for alpha radiation, 20 Sv, for neutrons the dose depends on their energy.

The safety measures implemented at NPPs are therefore designed to prevent accidents, minimize radioactive releases and mitigate the consequences of radioactive release if it occurs. Uncontrolled radioactive release from an NPP via its various exposure pathways would result in the spread of radioactive substances in the environment and the exposure of people above acceptable radiation limits.

Defence in depth is based on the concept of multiple barriers and variety of methods (provisions) to protect the barriers against radioactive releases. In this lecture, the concept is upheld as an essential strategy in ensuring safety at both existing and new NPPs [2], [3], [4]. The defence in depth strategy provides the rules for the application of provisions in preventing harm to people and the environment due to radioactive release.

Sources of radioactivity and inventory of fission products in nuclear power plants

The source of heat in a nuclear reactor is the fission of nuclei in fuel material (usually ^{235}U) by a neutron. The process produces fission products (FP1, FP2) and releases additional neutrons and energy according the equations:

$$^{235}\text{U} + \text{n} \rightarrow \text{FP1} + \text{FP2} + 2.42\text{n} + 215\ \text{MeV}\ (1\ \text{MeV}=1.6 \times 10^{-13}\ \text{J})$$

$$^{239}\text{Pu} + \text{n} \rightarrow \text{FP1} + \text{FP2} + 2.86\ \text{n} + 215\ \text{MeV}$$

$$^{233}\text{U} + \text{n} \rightarrow \text{FP1} + \text{FP2} + 2.48\ \text{n} + 215\ \text{MeV}$$

In addition to a large amount of energy (nuclear fission compared to chemical reactions produces 50,000,000 x more energy per reaction), a large amount of fission products are produced. Most of the fission products are unstable nuclei which emit radiation associated with the production of additional heat. The main source of radioactivity in an NPP is fission products. The inventory of fission products and other radionuclides in the reactor fuel and reactor coolant system (RCS) depends on a number of factors, particularly the quantity of fissile material, fuel power and burnup, neutron flux distribution in the core, operational power history, fuel management, and decay time after shutdown.

Figure 2 shows the yields of various fission products (characterized by the mass number of different isotopes) produced from the fission of fissile materials. One can observe that the most typical fission products resulting from U-235 fission have mass numbers of around 90 – 100 (e.g., isotopes of Kr, Sr, Ru) in one group and 130 – 140 (e.g., isotopes of Xe, Cs, Te, Ba) in another group. The most significant isotopes related to potential radiation exposure are listed in Table 2. The table also shows the differences in half-lives of the isotopes, varying from a few minutes up to several years. Apart from noble gases, the most typical isotopes released in a reactor accident are isotopes of iodine, especially I-131 (half-life 8.04 days), which can potentially cause thyroid cancer, and isotopes of caesium, especially Cs-137 (half-life 30.17 years), which can potentially cause long-term contamination of the ground and vegetation and late cancer. Table 3 lists examples of the total inventories of selected fission products in typical large reactors.

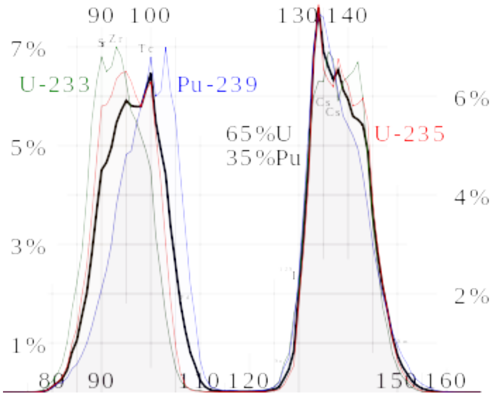


Figure 2 Fission product yield as a function of mass number from the fission of various fissile materials.

Table 2 Significant fission products in the analysis of radiological consequences

Fission product	Half life	Fission product	Half life	Fission product	Half life
Kr85	10.72 year	Te131m	30 hour	Xe133m	2.19 day
Kr85m	4.48 hour	Te132	78.2 hour	Xe135	9.11 hour
Kr87	76.3 min	Sb127	3.85 day	Xe138	14.13 min
Kr88	2.84 hour	Sb129	4.40 hour	Cs134	2.062 year
Sr89	50.55 day	I131	8.04 day	Cs136	13.16 day
Sr90	28.6 year	I132	2.30 hour	Cs137	30.17 year
Sr91	9.5 hour	I133	20.8 hour	Ba140	12.789 year
Y91	58.51 day	I134	52.6 min	La140	40.22 hour
Mo99	66.06 hour	I135	6.61 hour	Ce144	284.3 days
Ru103	39.35 day	Xe131m	11.84 day	Np239	2.355 days
Te129m	33.6 day	Xe133	5.245 day		

Table 3 Examples of fission product inventory for large reactors [TBq]

Isotope	PWR *3468 MWt* *3-year cycle* *EOC*	VVER *3012 MWt* *43 Mwd/kg*	VVER *3000 MWt* *all fuel* *60 Mwd/kg*	PWR *4900 MWt* *average burn-up* *43 Mwd/kg* *Maximum burn-up 63.6 Mwd/kg*	PWR *4063 MWt* *2-years refuelling* *57 Mwd/kg*
Xe133	7.03E6	7.03E6	6.44E6	9.7E6	9.04E6
I131	3.56E6	2.81E6	2.91E6	4.8E6	4.29E6
Cs137	4.18E5	2.89E5	5.41E5	6.4E5	7.59E5
Te131m	5.18E5	4.81E5	315E5	4.1E6	8.99E5
Sr90	3.07E5	2.66E5	3.58E5	4.7E5	5.23E5
Ru103	5.37E6	2.74E6	4.71E6	7.4E6	7.46E6
La140	6.73E6	5.92E6	5.41E6	9.4E6	8.32E6
Ce141	6.03E6	5.18E6	4.70E6	8.1E6	7.69E6
Ba140	6.33E6	5.92E6	5.03E6	8.9E6	8.30E6

The decay of unstable fission products is not only a source of radiation but a source of significant thermal power, called residual heat. During operation at power, the decay of fission products represents about 7 % of total power. After reactor shutdown, this decreases exponentially with time, as shown in Figure 3. Table 4 shows that despite its small percentage, residual heat without cooling can result in core melt and rapid destruction (within an hour) and the amount of coolant needed to remove residual heat, for example by evaporation, is large even after three days since reactor shutdown.

Table 4 Effects of residual heat (Plant data: PWR where P_{el} = 1300 MW; $P_{thermal}$ = 3750 MW; M_{UO_2} = 107 t, specific heat capacity cp_{UO_2} = 350 J/kg/K, Specific power (100%) P = 35 MW/kg, evaporation enthalpy h (1 MPa) = 2; h (7 MPa) = 1.5; h (18 MPa) = 0.75 MJ/kg).

Isotope	18 seconds	1 hour	10 hours	3 days
Decay heat relative to full power [%]	4.0	1.3	0.7	0.4
Adiabatic heat up rate of core [K/s]	4.0	1.3	0.7	0.4
Evaporation of water at 7 Mpa [kg/s]	100	32	17	10

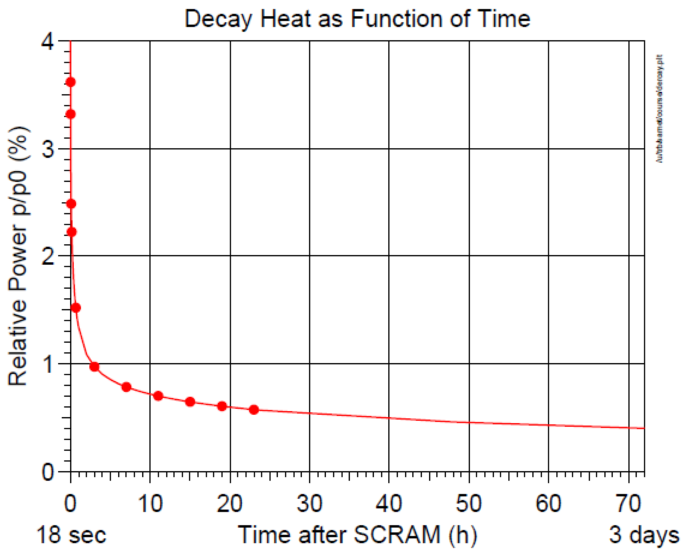


Figure 3 Decay heat produced in the core after reactor shutdown

In a comprehensive summary of sources of radiation, it should be noted that due to fission products leakage from fuel elements, activation and corrosion, some radioactive substances are also present in the RCS coolant. Activity in the primary coolant depends on the number of leaky fuel rods, the type and size of the leak, burnup and power level, the materials used in the RCS, the total amount and composition of RSC coolant, and the removal rate of fission products by RCS purification systems. Radioactive substances include activation products for the coolant and additives – C-14, O-15, H-3, N-16, Cu-164, K-42, Ar-41, Cl-38, Na-24, corrosion products – Co-60, Co-58, to a lesser extent isotopes of Fe, Ni, Mn, and volatile fission products—mainly isotopes of iodine, caesium, krypton and xenon.

The importance of fission product barriers and methods of protecting these barriers

Tables 2 and 3 clearly show that the core contains a very large number of highly active radionuclides (expressed in TBq, 1 Bq= 1 decay per second, 1 Curie = 3.7 x 10^{10} Bq). For example, the core of a large reactor contains around 300,000 to 800,000 TBq of Cs-137 and 3,000,000 to 5,000,000 TBq of I-131. However, the analysis of radiological consequences show that for a design basis accident (DBA) the acceptable value of release of Cs-137 into the environment is about 1 TBq, and in the case of severe accidents, about 100 TBq. For a DBA, the acceptable value of the release of I-131 is about 10 TBq, and in case of severe accidents, about 1,000 TBq.

This comparison underlines the importance of maintaining the integrity of barriers against radioactive release. The barriers should be placed between the fission products and the environment. In light water reactors, there are four physical barriers against potential releases of radioactive substances (Figure 4):

- Fuel matrix and its structure (1^{st} barrier)
- Fuel cladding (2^{nd} barrier)
- Reactor pressure vessel and primary circuit pressure boundary (3^{rd} barrier)
- Containment (4^{th} barrier)

The integrity of the barriers (at least the final barrier) is extremely important in all NPP states [5]. Deviation from the normal operation of a plant results in different plant states with increasing severity, as follows (Figure 5):

- Normal operation (NO)
- Anticipated operational occurrences (AOOs)
- Design basis accidents (DBAs)
- Design extension conditions (DECs), including sequences without significant fuel degradation and sequences with core melting

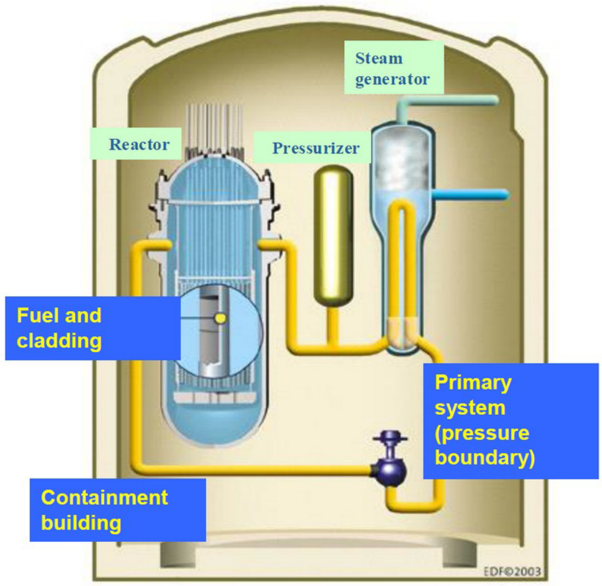


Figure 4 Multiple fission product barriers preventing the release of radioactive substances

Plant design envelope

Operational states		Accident conditions		
NO	AOO	DBAs	Design Extension Conditions	
			Without significant fuel degradation	With core melting (severe accidents)
Loads and conditions generated by External & Internal Hazards (for each plant state)				
Criteria for functionality, capability, margins, layout and reliability (for each plant state)				
Design basis of equipment for Operational states		Design Basis of Safety Systems including SSCs necessary to control DBAs and some AOOs	Design Basis of safety features for DECs including SSCs necessary to control DECs Features to prevent core melt	Features to mitigate core melt (Containment systems)

Figure 5 Design basis of SSCs

Deviation from normal operation is caused by various specific failures or sequences of failures [5], including (Figure 6):

- Failures of structures, systems and components of the plant (partial failure if relevant), including possible spurious actuation;
- Failures initiated by operator errors, which could range from faulty or incomplete maintenance operations to incorrect settings of control equipment limits or wrong operator actions;
- Failures of structures, systems and components of the plant arising from internal and external hazards.

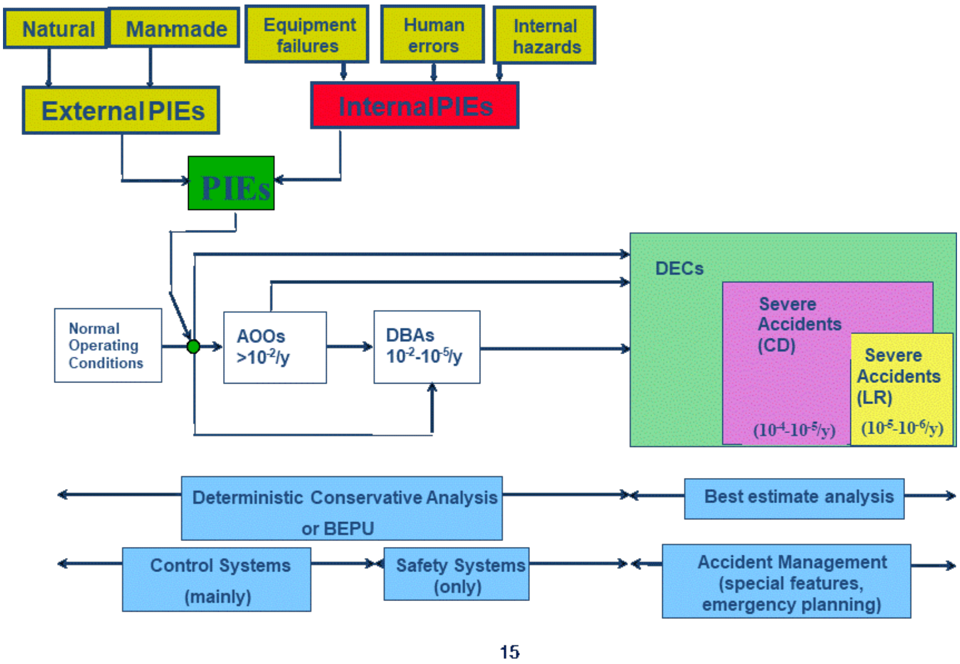


Figure 6 Transition from normal operation to different plant states (with indication of frequency)

To safeguard barrier integrity, the methods applied to protect the barriers should ensure that:

- The integrity of all barriers under NO and AOOs is maintained.
- Under accident conditions (DBAs and DECs), including selected severe accidents, at least the integrity of containment is maintained.

It is also important to see the location of radioactive substances in relation to the barriers (Figure 7). From the diagram in Figure 7 and a comparison with acceptable releases, it can be concluded that even the release of a very small portion of the radioactivity contained in the primary coolant is significant in potential consequences, and that the radiological effects of a reactor accident strongly increases with the scope of damage to other barriers. The largest consequences may result from accidents with a release of volatile fission products from the molten fuel, i.e., severe accidents. All the effort for safety in an NPP should therefore be devoted to the

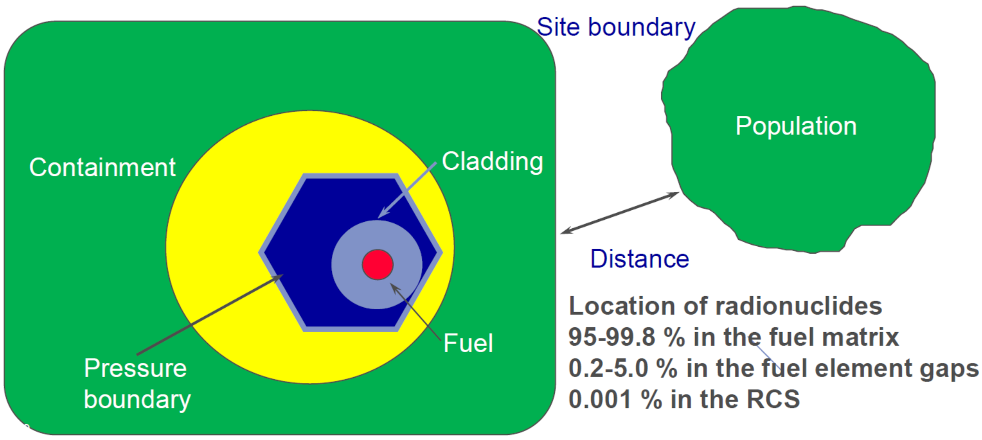


Figure 7 Distribution of radioactive nuclides in an NPP

prevention and mitigation of severe accidents. This is determined principally by maintaining the integrity (or preventing by-pass) of the final barrier, i.e., containment. The aim of defence in depth is therefore to protect the barriers against the release of radioactive material and mitigate the consequences of accidents if the barriers are damaged.

Safety functions and the provisions for applying safety functions

To avoid failure of the barriers against the release of radioactive materials and the need to mitigate the consequences of their failure, it is essential to maintain a balance of heat production and heat removal from the nuclear fuel. More generally, the following fundamental safety functions are necessary [4]:

1 Control of reactivity

2 Removal of heat from the reactor and fuel store

3 Confinement of radioactive materials, shielding against radiation, and control of planned radioactive releases, including limiting any accidental radioactive releases

Fundamental safety functions must be performed for all plant states, i.e., operational states (NO and AOO), during and following a DBA, and during and following the considered plant conditions under a more severe accident such as a DEC. The removal of heat means not only the heat from the reactor but also from the fuel removed from the core, which remains on site and is a potential source of radioactive release.

Figure 8 Defence in depth since the beginning of nuclear power

The concept of defence in depth and levels of defence

Defence in depth has been used to guarantee a high level of safety since the beginning of nuclear power (Figure 8). According to the IAEA Fundamental Safety Principles [1], defence in depth is the primary means of preventing and mitigating the consequences of an accident. Defence in depth is applied primarily by installing and preventing the damage of fission product barriers through a combination of consecutive and independent levels of protection which would all have to fail before any harmful effects could be caused to people or the environment [2], [3]. If one level of protection or barrier fails, the subsequent level or barrier is still available (Figure 9). When properly implemented, defence in depth ensures that no single technical, human or organizational failure could lead to harmful effects, and that the combination of failures which could result in significant harmful effects have a very low probability.

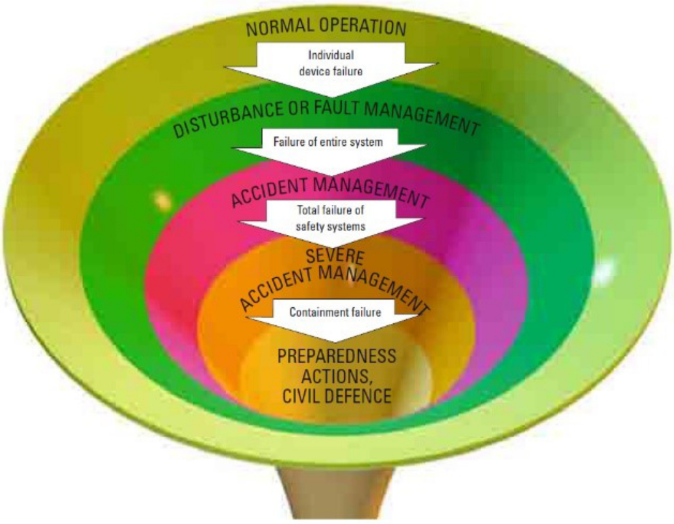


Figure 9 Several levels of protection applied in accordance with the defence in depth principle

Defence in depth is therefore a hierarchical deployment of different levels of equipment (design provisions) and procedures (operational provisions) designed to maintain the effectiveness of the physical fission product barriers placed between the radioactive material and the operators, the public and the environment under NO and AOO, and for some barriers, during an accident. Defence in depth ensures that the fundamental safety functions are reliably performed with sufficient margins to compensate for equipment failure and human error. Defence in depth encompasses all safety activities, including the siting, design, manufacturing, construction, commissioning, operation and decommissioning stages of NPPs.

Under the IAEA Safety Standards for NPPs, five levels of defence in depth are implemented [4]. There are different methods of illustrating the concept of defence in depth. For example, Figure 10 shows the relationships between the levels of defence, barriers and systems of protection for the barriers. Figure 11 shows propagation of severity of an initiating event through the levels of protection. Figure 12 is a simplified flow chart showing the logic of defence in depth. Success is defined for each level of defence in depth. According to the principles of defence in depth, if the provisions of a given level of defence fail to control the evolution of a sequence, the subsequent level will come into play.

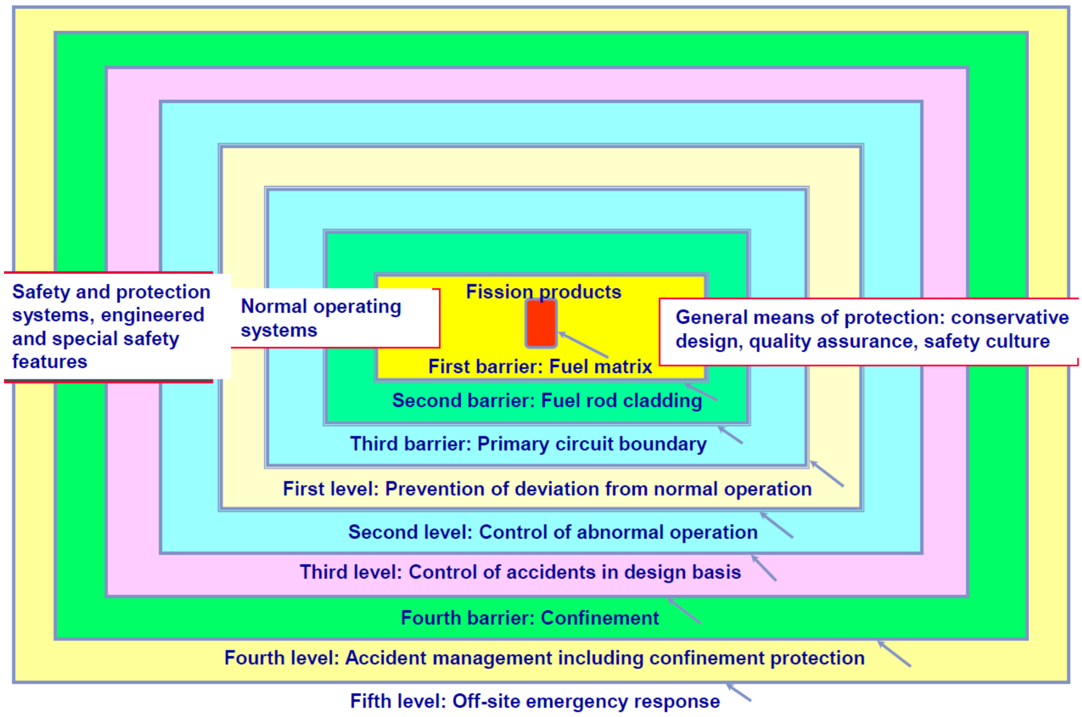


Figure 10 The relationship between the levels of defence, barriers and systems

More comprehensively, the characteristics of the levels of defence together with their specific safety objectives, associated plant states, design and operational means (provisions) to protect barrier integrity and expected barrier status is summarized in Table 5 [6]. The objective of Level 1 of defence is prevention of abnormal operation and system failures. If a failure occurs at this level, an initiating event takes place. This can happen either if the defence in depth provisions at Level 1 were not effective enough or if a certain mechanism was not adequately considered in establishing the provisions at Level 1. Level 2 will detect these failures, to avoid or control abnormal operation. If Level 2 fails, Level 3 ensures that fundamental safety functions will be

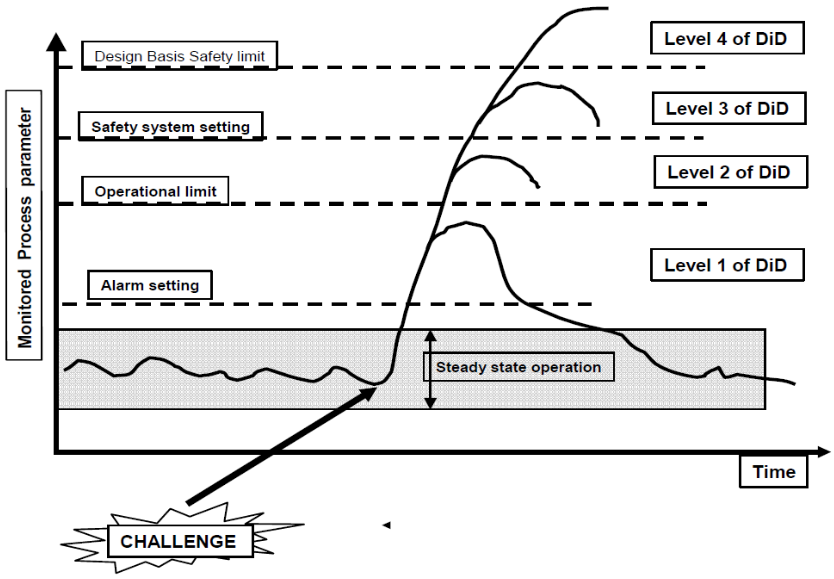


Figure 11 Design limits and defence in depth

activated mainly specific safety systems which attempt to limit the possible consequences of a DBA. If Level 3 fails, Level 4 limits further progression of an accident through safety features for DECs and accident management provisions in order to prevent or mitigate severe accident conditions involving an external release of radioactive materials. The objective of Level 5 is mitigation of the radiological consequences of significant external releases through on-site and off-site emergency responses.

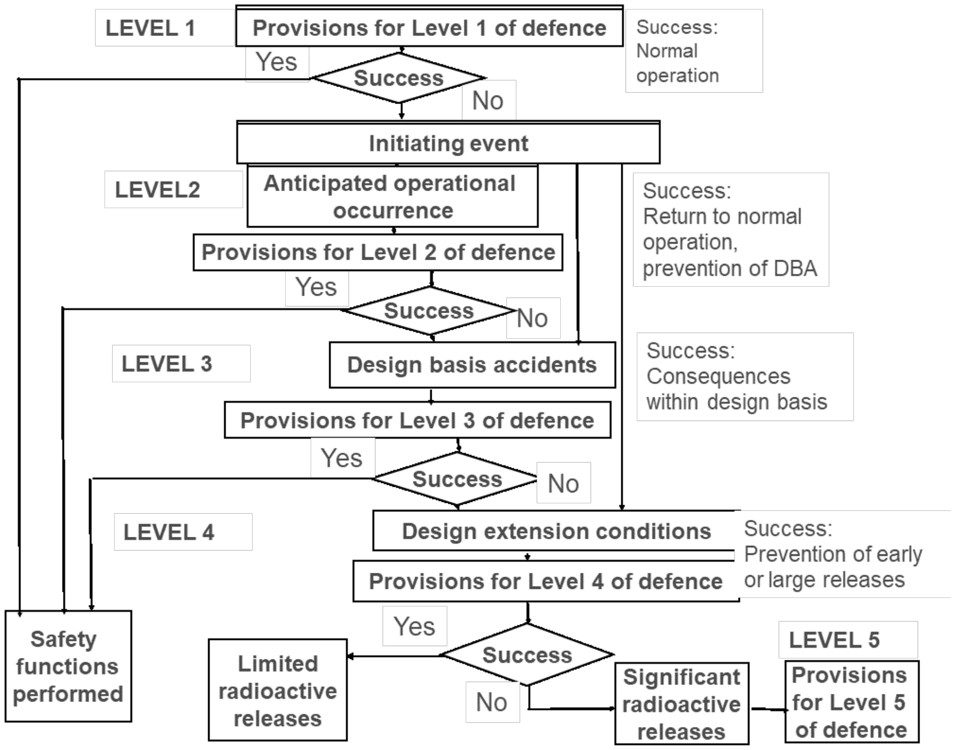


Figure 12 Logic flow diagram of defence in depth

Key attributes of defence in depth

In summary, the key attributes of defence in depth are the availability of several levels, robustness of each of the levels and reasonably practicable independence between the levels. The IAEA document [7] describes the basis for defence in depth, as follows:

- Safety must be ensured by implementing safety provisions at all five levels of defence in depth.
- Each level should be individually robust; this is usually ensured through the redundancy of several safety chains, further supported by separation and diversity. High strength in a certain level of defence should not be misused as justification for the weaknesses of other levels.
- Provisions implemented at different levels of defence (in particular levels 3 and 4) should be reasonably independent to prevent propagation of a failure through several levels of defence. This is primarily ensured through a diversity of provisions (usually from different physical principles applied to perform the function).

Figure 13 illustrates the principles of redundancy, diversity and separation.

Table 5 Characteristics of levels of defence in depth

Level of defence	Objective	Associated plant state	Design means	Operational means	Integrity of barriers
Level1	Prevention of abnormal operation and failures	Normal operation	Conservative design, high quality in construction of normal operation systems, including monitoring and control systems	Operational rules and normal operating procedures	No failure of any of the physical barriers except minor operational leakages
Level 2	Control of abnormal operation and detection of failures	Anticipated operational occurrence	Limitation and protection systems and other surveillance features	Abnormal operating procedures/ emergency operating procedures	No failure of any of the physical barriers except minor operational leakages
Level 3	Control of design basis accidents	Design basis accident	Engineered safety features (safety systems)	Emergency operating procedures	No consequential damage of the RCS, maintaining containment integrity, limited damage of fuel
Level 4a DEC-A	Control of DEC to preven significant fuel degradation	DEC without significant fuel degradation	Safety features for DEC without significant fuel degradation	Emergency operating procedures	No consequential damage of the RCS maintaining containment integrity, limited damage of fuel.
Level 4b DEC-B	Control of DEC to mitigate the consequences of severe accidents	DEC with core melt (severe accident)	Safety features for DEC with core melt. Technical Support Centre	Complementary EOPs / severe accident management guidelines	Maintaining containment integrity both in an early as well as late phase, and practical elimination of fuel melt when the containment is disabled or by-passed
Level 5	Mitigation of radiological consequences of significant releases of radioactive materials	Accidents with releases requiring implementation of emergency plans	On-site and offsite emergency response facilities	On-site and off-site emergency plans	Containment integrity severely impacted, or containment disabled or bypassed

Identification of what may impact the performance of a fundamental safety function and the variety of options available to avoid this impact for each level of defence is an essential task in the development of the framework for creating an inventory of the defence in depth capabilities of a plant. In developing the framework, it is useful to summarize the sequence of considerations in implementing the principles of defence in depth for NPPs [7]:

- The fundamental safety objective is to protect people and the environment from the harmful effects of ionizing radiation.
- The safety objective is achieved through the deployment of five consecutive levels of defence in depth. If one level of protection fails, the subsequent level comes into play.
- For each level of defence, additional safety objectives derived from the fundamental safety objective can be defined to prevent the progression of any initiating event to more severe conditions.
- Progression to more severe conditions is prevented by maintaining the integrity of the physical barriers which correspond to individual levels of defence.

- To maintain the integrity of the barriers, the safety functions should be in place.

- Safety functions may be challenged by various mechanisms and affect their performance, potentially leading to damage to the barriers.

- For each mechanism, it is possible to identify different measures (provisions) which could be applied in order to prevent the mechanism from affecting the safety functions. The provisions may have different natures, including inherent safety characteristics, safety margins, active and passive systems, procedures, operator actions and organizational measures, which include human behaviour in accordance with safety culture principles. The adequacy of the provisions must be evaluated through safety assessments which consist of safety analyses (deterministic and probabilistic) and the evaluation of other factors important in safety (verification of compliance with various safety regulations).

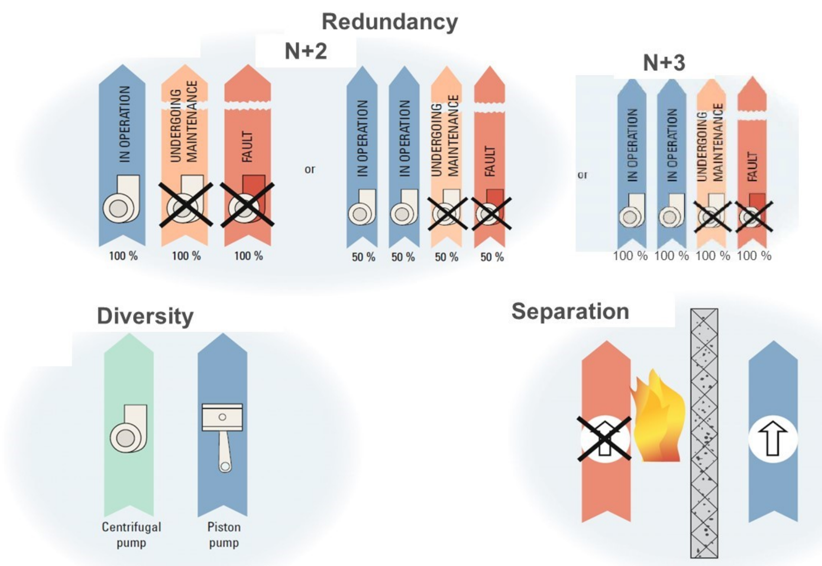


Figure 13 Design principles of NPP systems: redundancy, diversity and separation

The sequence of considerations described above may be graphically depicted in an objective tree [7], as shown in Figure 14. The top of the tree shows the level of defence in depth, followed by the objectives which it should achieve, including the barriers which should protect against the release of radioactive materials. Below this is a list of safety functions which must be maintained to achieve both the objectives and the protection of the barriers under the level of defence concerned. Any potential challenges to the safety functions are dealt with by the defence in depth provisions established at the given level of defence. All mechanisms which can challenge the performance of the FSFs should be first identified for each level of defence. These mechanisms are used to determine the set of initiating events which can lead to deviation (initiation or worsening) from normal operation. A set of provisions must be implemented to prevent each of the mechanisms from occurring.

Independence of levels of defence in depth

The independent effectiveness of each level is a necessary component of defence in depth aiming to ensure that the failure of one level should not cause the failure of subsequent levels. This objective can be achieved by incorporating specific design features such as redundancy, separation and diversity to prevent common cause

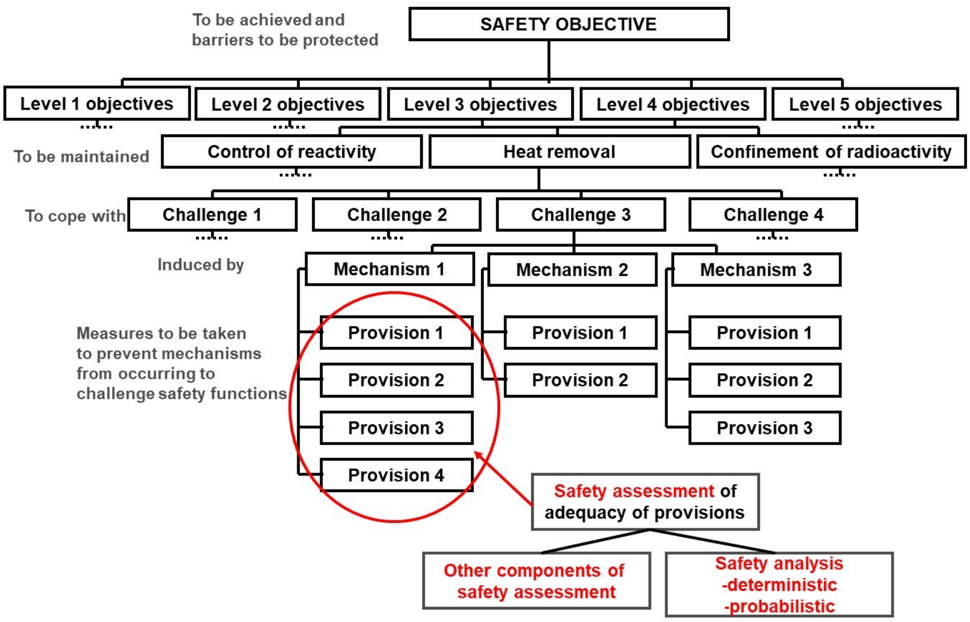


Figure 14 Structure for defence in depth provisions at each level of defence

failures. Independence should apply to all the components of safety systems and safety features for DECs (protection (I&C), actuation, support systems).

It is important to note that the levels of defence are not and cannot be fully independent because some SSCs in the plant (e.g., control rooms, containment, control rods, the operators, and protection against external hazards) are shared. Independence of the levels of defence must be understood as a "degree of independence", which should be the highest possible.

Independence between levels 3 and 4 is essential in preventing the transition to the consequences and mitigation of severe accidents, and a necessary precondition for the practical elimination of early or large releases, as described in section Practical elimination of early or large releases below.

Examples of independent means for levels 3 and 4 are illustrated in Figure 15 and 16.

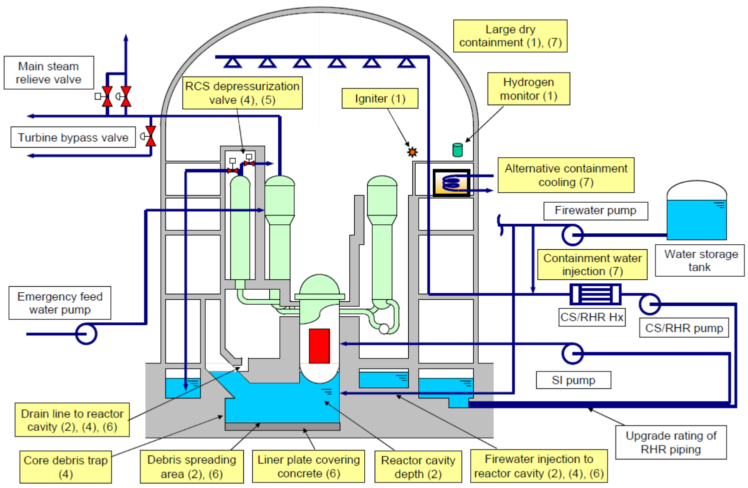


Figure 15 Examples of systems for level 4 of defence

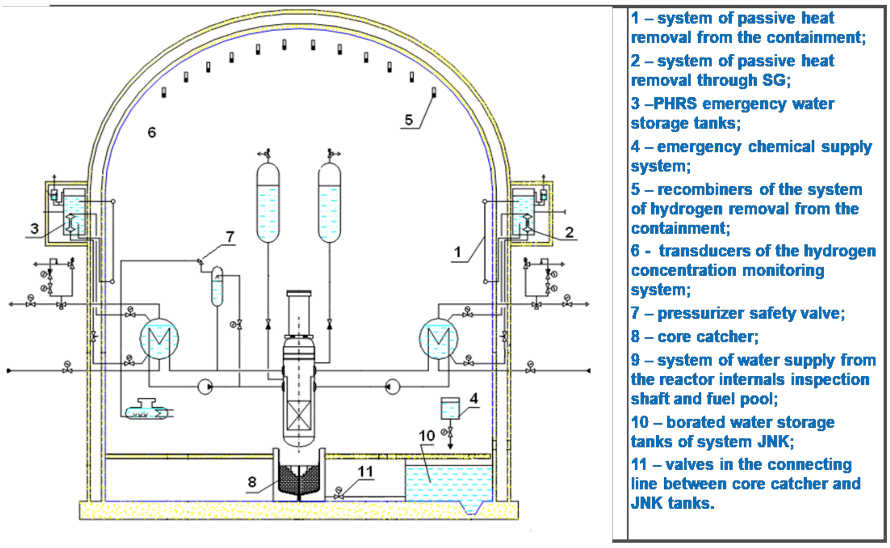


Figure 16 Examples of systems for level 4 of defence

Examples of inadequate independence of levels of defence can be as follows:

- Normal operational systems used for spent fuel pool cooling, yet also performing emergency injection function in case of accidents.
- Use of the pressurizer relief or safety valves for both design basis accidents and severe accidents.
- Absence of a dedicated containment heat removal system for severe accidents (e.g., containment spray system).
- Use of the same equipment for residual heat removal to the ultimate heat sink (e.g., essential service water cooling system) from the safety systems and the safety features for DECs.
- Use of the same sensors for the initiation of safety system actions and the safety features for DECs.

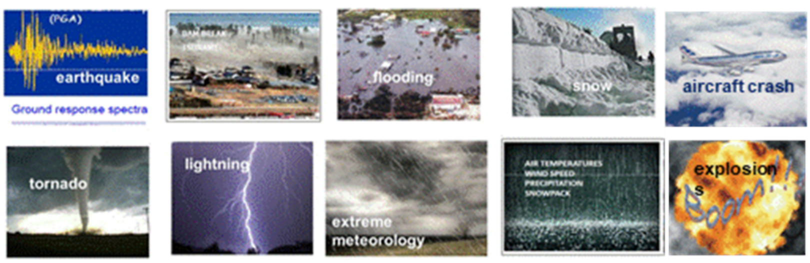


Figure 17 External hazards considered in the design of NPPs

Relationship between hazards and defence in depth

Defence in depth should closely consider hazards (fires, explosions, flooding, earthquakes, meteorological conditions, aircraft, crashes, etc), especially external hazards (Figure 17) which may have the potential to adversely affect more than one barrier simultaneously.

External hazards are beyond the control of the operating organization, and it is difficult to make an upper estimate of their intensity and frequency, which are associated with large uncertainties. According to the IAEA [5], hazards are not considered plant states but loads which potentially could (but not necessarily) cause deviation from normal operation, including accident conditions [4], [5]. Hazards exceeding the design basis of safety systems are not considered DEC and therefore are not included in the current definition of DECs; the term "Beyond Design Basis External Event" (BDBEE) should be used.

The IAEA Safety Standard SSR-2/1 [4] imposes more demanding requirements on the robustness against external natural hazards on equipment ultimately necessary to prevent early or large releases (such as the containment barrier, heat transport systems to the ultimate heat sink, active systems for power supply, control rooms). The design of these items is expected to be particularly robust and include margins to withstand loads and conditions generated by natural external hazards exceeding those derived from the site evaluation; this implies that cliff edge effects (see explanation of the term below) should not occur with small variations but also significant variations in load and conditions.

Design options for natural external hazards which exceed the design basis could be as follows:

1 Adopt a higher value of the design basis event for these SSCs,

2 Demonstrate, following a best estimate approach, with high level of confidence that the values of the parameters for which cliff edge effects would occur are not reached because the design margins are adequate.

A cliff edge effect [6] implies high consequences following a small deviation in a certain determining parameter; the worst case being a large release (Figure 18). A typical cliff edge effect is the failure of a physical barrier. A fission product barrier could fail if the safety functions which protect the barrier fail as a result of a change in the input parameter. If a large difference occurs between the calculated parameter and a safety limit, there is no special importance to demonstrate the prevention of cliff-edge effects. If the calculated parameter is near the safety limit, avoidance of the cliff edge effects should be demonstrated by a sensitivity analysis for the given parameter.

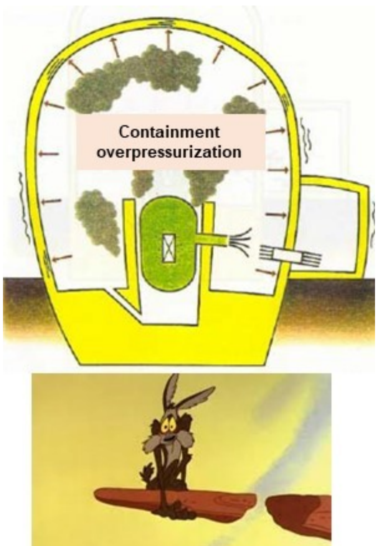


Figure 18 Cliff-edge effects

Practical elimination of early or large releases

IAEA Safety Requirements on NPP design [4] as well as other international safety rules (e.g. WENRA Safety Objectives for New NPPs: Objective O3, Accidents with core melt [8] or EU Council Directive 2014/87/EURATOM of 8 July 2014 on nuclear safety [9]) require compliance with practical elimination of early or large releases stating that the NPP design shall be such that the possibility of conditions arising that could lead to an early radioactive release or a large radioactive release is 'practically eliminated.

IAEA Safety Standards such as SSG-2 (Rev.1) [5] or other IAEA guidance documents such as TECDOC-1791 [6] indicate the steps to demonstrate practical elimination as follows:

- 1st step: identification of the conditions (challenges) for practical elimination.
- 2nd step: whenever possible, demonstrate practical elimination based on physical impossibility (e.g., insufficient hydrogen/oxygen concentration, intrinsic safety coefficients, etc.).
- 3rd step: identification and implementation of design provisions to prevent challenges.
- 4th step: identification and implementation of operational provisions (procedures) to prevent challenges.
- 5th step: deterministic safety analysis and engineering assessment of the effectiveness of provisions.
- 6th step: whenever appropriate, a probabilistic safety analysis indicating very low probability of failure of provisions.

Although not specifically given in the standards, a frequency value of about $1x\ 10^{-7}$ per reactor year for each of the conditions identified appears to be acceptably low for the concept of defence in depth.

For light water reactors, accident sequences which potentially lead to releases typically include:

(a) Events which could lead to prompt reactor core damage and consequent early containment failure, such as:

(i) Failure of a large pressure-retaining component in the reactor coolant system;

(ii) Fast reactivity insertion accidents.

(b) Severe accident sequences which could lead to early containment failure, such as:

(i) Highly energetic direct containment heating;

(ii) Large steam explosion;

(iii) Explosion of combustible gases, including hydrogen and carbon monoxide.

(c) Severe accident sequences which could lead to late containment failure, such as:

(i) Basemat penetration or containment bypass during molten core concrete interaction;

(ii) Long term loss of containment heat removal;

(iii) Explosion of combustible gases, including hydrogen and carbon monoxide.

(d) Severe accident with containment bypass, such as:

(i) Loss of coolant accident with the potential to drive leakage outside the containment via supporting systems (interface system-LOCAs).

(ii) Containment bypass consequential to severe accident progression (e.g., induced steam generator tube rupture);

(iii) Severe accident in which the containment is open (e.g., shutdown state).

(e) Significant fuel degradation in a fuel storage pool and uncontrolled release.

Conclusion

There are many non-nuclear sectors with application of defence in depth, including the military, chemical, aviation and car industries. However, nuclear power is special due to:

- Systematic implementation of provisions at 5 levels
- Robustness of barriers and provisions for the protection of barriers
- International harmonization of standards
- Periodic safety reviews with enhancement of existing NPPs to current standards
- International peer reviews

If the rules and approaches for nuclear power are so strict, why accidents (Figure 19) still happen? The lessons learned from accidents (as another approach used in the nuclear sector) show that accidents occur not because the defence in depth rules are incorrect or not stringent enough, but because the rules are violated and risk was either not addressed or the magnitude of risk was underestimated.

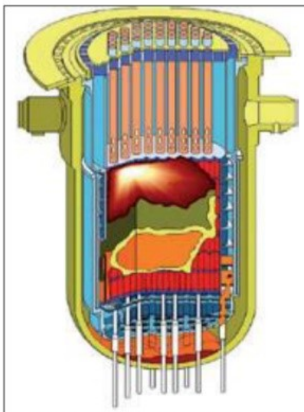
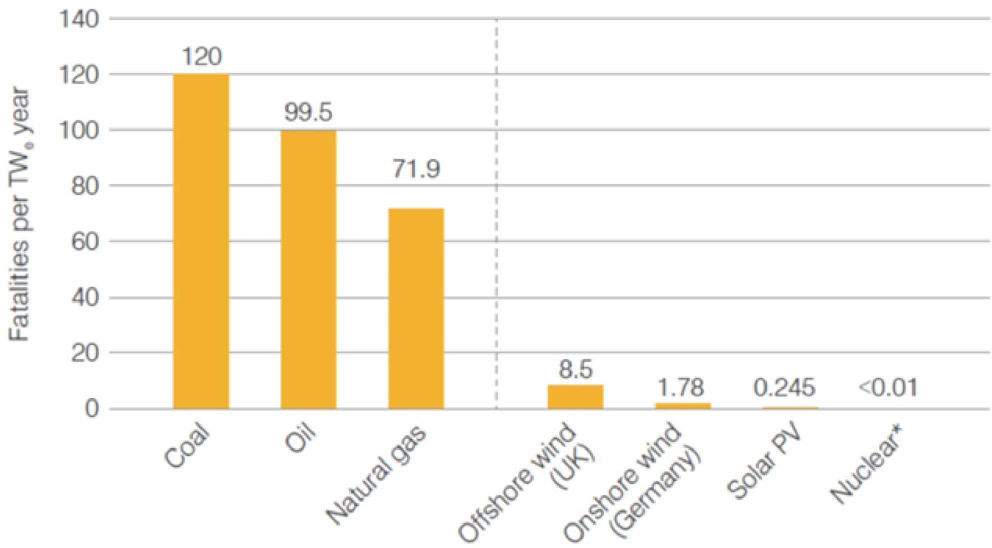
 **Accident of Unit 2 of Three Mile Island NPP** • Complex scenario with large scale core melting • Containment functioning reliably • Small radiological consequences • Important lessons learned for safety of PWRs	 **Accident of Unit 4 of Tchernobyl NPP** • Design very different from PWRs • Violation of the principles of Nuclear safety • Nuclear explosions and fire • Complete destruction of the core • Practically no containment • Large scope contamination Lessons learned for management of emergencies • Warning regarding reactivity accidents	 **Accident of 4 units of NPP Fukushima Daiichi** • Combination of two extreme natural hazards • Wrong evaluation of the risk, inadequate safety upgrading in the operation • Accident developing in line with the expectation, but taking place in parallel of countrywide catastrophe and local chaos • Lessons learned: More focus on the role of nuclear regulator and international supervision

Figure 19 Lessons learned from reactor accidents

On the other hand, it should be acknowledged that any source of electric power is associated with risk and an accident may occur in any area of electricity production (Figure 20). All such accidents are associated with fatalities (Figure 21), and in this comparison, nuclear power is one of the safest ways of producing electricity.

Nevertheless, it needs to be taken into account that the public is more sensitive to nuclear power than any other source (renewable power sources, in particular). It is therefore imperative to continue raising the level of NPP safety. Under these efforts, defence in depth is and should remain the safety concept for both existing and future reactors, and compliance with the standards should be systematically reassessed and improved. Independence, diversity and segregation of levels of defence should be further strengthened to eliminate common mode failures. Risk from an NPP should be sufficiently low to be acceptable to the public.

Figure 20 Examples of accidents in energy infrastructure (source: Hirschberg et all, PSI)



Source: ** The Silent Giant. The need for nuclear in a clean energy system, WNA, https://www.world-nuclear.org/getattachment/Our-Association/Publications/Position-statements/the-silent-giant/the-silent-giant.pdf.aspx

Figure 21 Comparison of number of fatalities due to electricity generation (deaths per TW-year)

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[10] Council Directive 2014/87/EURATOM of 8 July 2014 on nuclear safety,

Nuclear Forensics

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The objective of our short program on nuclear forensics is to introduce the basic aspects of the subject. For the past five years I have taught courses and workshops in nuclear forensics to policy students in a master's program at the Middlebury Institute. This paper will provide an overview of the concepts of nuclear forensics. It will hopefully provide a basic level of understanding of how nuclear forensics analyzes the nuclear and radioactive materials essential for making either a nuclear yield producing device (for example an Improvised Nuclear Device (IND)) or a Radiological Dispersal Device (RDD). Program participants should also come away with an understanding of the ability and limits of nuclear forensics to attribute an RDD or IND to a particular state or nonstate actor.

Historical Background

During the Cold War all the nuclear weapon states developed the ability to diagnose their weapons tests to determine, for example, the yield of the device. When testing was atmospheric many of the determinations were relatively easy to make, but when testing went underground many determinations had to be made by post-detonation analysis of device residue and/or the effects of the device on the surrounding media.

When the Cold War and weapons testing ended in the 1990s, a new generation of concerns began to arise about the possibility of terrorist or rogue state use of nuclear weapons or radiological devices. With the collapse of the Soviet Union illegal sales and marketing of nuclear and radioactive materials became a growing international problem. It was a natural progression to apply many of the techniques used to diagnose nuclear weapons and passively assess radiation sources to this new problem. Recognizing that the investigation and later prosecution of perpetrators of these activities would require combining conventional legal techniques and scientific work, the science of nuclear forensics was developed.

The subject of nuclear forensics can be parsed in several different ways. Nuclear forensics deals with both nuclear and other radioactive materials the "nuclear" in "nuclear forensics" and with the "forensics" the legal aspects of dealing with these materials in a forensic/criminal context. Another parsing can be to look at pre-detonation and post-detonation examination of materials used for or resulting from an IND or RDD.

From a forensics viewpoint, it is important to understand that nuclear forensics should be capable of dealing with both nuclear and other radioactive material to obtain forensic evidence and also be able to perform conventional forensics on radioactive leak contaminated items. Most, if not all, conventional police forensics laboratories have no ability to deal with radioactive material and therefore obtaining conventional forensics such as DNA samples, fingerprints, etc. from radioactive or contaminated devices must be done in specific nuclear forensics laboratories that are properly configured to handle radioactive materials.

A worldwide organization of nuclear forensics laboratories has developed and has been fostered by the Nuclear Forensics International Technical Working Group (ITWG). ITWG is an organization of nuclear forensics practitioners that identifies and promotes best practices in nuclear forensics. It does this by conducting coordinated exercises, holding informational exchange meetings, and assisting in the preparation of guides for radiological crime scene management and subsequent analysis of evidence. Experts from more than 50 countries have participated in ITWG activities[1].

A fully functional nuclear forensics laboratory is extremely expensive to maintain. As will be discussed below, large amounts of expensive equipment and numerous technologies are employed in performing a thorough nuclear forensic analysis. Therefore, it is probably cost prohibitive for most countries to develop their own nuclear forensics laboratories. However, development of educational programs in both nuclear forensics technologies and policies should be encouraged along with the development of the ability to perform first stage analytics such as radionuclide identification.

Due to the prohibitive costs of developing domestic capabilities in most countries, most states need to develop a plan to get international assistance if an incident occurs and they have need for nuclear forensics analysis. A thorough understanding of how an international system would interact with the state's domestic legal program and an understanding of how items such as chain of custody for evidence, admissibility of expert opinions, etc. are essential for each state to develop.

The International Atomic Energy Agency (IAEA) does not provide nuclear forensics analysis capability for its member states. Although its Safeguards Laboratory at Seibersdorf may use some of the same techniques that are used in nuclear forensics analysis, nuclear forensics has not been a task of the IAEA. The IAEA probably will not engage in nuclear forensics in the future since this type of analysis could be perceived as pitting member states against each other. Although it does not provide nuclear forensics analysis services, the IAEA does provide training in nuclear crime scene management and in the principles of nuclear forensics and provides guidance documents on these issues through its Department of Nuclear Security[2].

Pre-detonation and Post-Detonation Analysis

Pre-detonation analysis refers to the analysis of nuclear material or other radioactive material before a nuclear device has been detonated. Seized uranium and plutonium are examples of nuclear materials that would be examined in a pre-detonation scenario. For other radioactive materials that are not part of a nuclear yield device the description of analysis pre-and post-detonation may similar, but there are additional complications that arise in post-detonation analysis[3]. To date, all nuclear forensics

1 Curry, M., Mayer, K., Girard, P., Thompson, P., & Kin, P. (2017). The Nuclear Forensics International Technical Working Group – Being Different and Making a Difference in Nuclear Security. International Atomic Energy Agency (IAEA): IAEA. Available via: https://inis.iaea.org/search/search.aspx?orig_q=RN:50017440 . See, also B.C. Garrett, et. al, The Nuclear Forensics International Technical Working Group (ITWG) An Overview. Available at: https://www-pub.iaea.org/MTCD/Publications/PDF/SupplementaryMaterials/P1706/Opening_Session.pdf

2 See the IAEA's description of its nuclear security series available at: https://www.iaea.org/resources/nuclear-security-series and note particularly Implementing Guide No. 2-G (Rev. 1), Nuclear Forensics in Support of Investigations.

3 Note that in most circumstances other radioactive materials will be associated with some type of RDD and will probably retain their physical form and basic radioactive characteristics even if the RDD device has been detonated, or, if explosives or burning have been employed, the material has been spread.

analysis has been done pre-detonation. Conversely, post-detonation analysis refers to analysis of the residue of a nuclear yield device or perhaps can be extended to analysis of other radioactive material from an RDD which has employed some type of disruptive technology such as explosion or fire to disperse the radioactive material. Whereas pre-detonation analysis deals with nuclear materials that are only slightly radioactive, post-detonation analysis must deal with highly radioactive material and its associated problems.

Both pre-and post-detonation analysis have the same goal of attribution trying to determine the origin of the nuclear or other radioactive material. Post-detonation analysis, however, would obviously put the analysis under far greater time pressure to produce results[4]. A theatrical example of post-detonation analysis can be found in the movie version of Tom Clancy's Sum of All Fears, starring Ben Affleck and Morgan Freeman. Although the movie portrayal of the nuclear forensics examination of bomb debris is far too simple, it is interesting that nuclear forensics was scripted to determine the origin of a fictional device detonated in Maryland.

Legal Aspects of Nuclear Forensics

For most nuclear forensics' scenarios, the goal is ultimately to discover who is responsible for a nuclear or radiological incident and to be able to successfully prosecute them under a country's domestic laws. Nuclear forensics scientists need to be developed and trained in the arts of handling evidence and presenting expert opinions in legal proceedings.

For many scientists this does not come naturally, and unfortunately, some nuclear forensics scientists make extremely poor witnesses in legal proceedings because they cannot effectively communicate their scientific knowledge, such as the results of tests they are performed, to the non-scientifically educated judges or juries that are involved in the conviction of criminal perpetrators of nuclear and radiological incidents. Training of scientists who work in nuclear forensics early in their careers about how to communicate to non-technical people can be an important factor in developing competent nuclear forensics scientists.

In addition to training scientists in nuclear forensics, it is important to understand the laws regarding nuclear and radiological crime in each state. Although United Nations Security Council Resolution 1540 (UNSCR 1540) requires that a state's legal system deals appropriately with Weapons of Mass Destruction (WMD), many states' penal codes do not adequately cover all aspects of nuclear and radiological crime.[5]

Asking the right questions of nuclear forensic scientists in order to meet the standards required to convict an alleged criminal under a state's laws can be difficult. Prosecutors and judges often need training in elementary concepts and terminology associated with nuclear and other radioactive materials.

4 It is important to understand that many nuclear forensics techniques may take days, weeks, or even months to perform. This can create problems in the urgency associated with a post-detonation analysis or even in a high threat scenario associated with a pre-detonation analysis.

5 For example, the state's domestic legal structure may not penalize fraudulent sales of nuclear materials or other radioactive materials.

Scientific Analysis in Nuclear Forensics

For the scientists involved in nuclear forensics analysis, a foundational text is by Moody, Grant and Hutcheon.[6] Moody is considered the gold standard in the field. A free e-book from the Stockholm International Peace Research Institute (SIPRI) titled The New Nuclear Forensics: Analysis of Nuclear Materials for Security Purposes edited by Vitaly Fedchenko can be downloaded and gives a good overview of the combination of sciences such as radiochemistry, spectrometric analysis, etc. that are applied in nuclear forensics.[7] Nuclear Forensics by M. J. Kristo also provides an up-to-date picture of nuclear forensics analysis techniques.[8]

Without going into detail, it is interesting to note that in one of the actual nuclear forensics investigations discussed in the Moody book the following analysis techniques were applied: optical microscopy; scanning and transmission electron microscopy, both with energy-dispersive x-ray analysis; x-ray and electron diffraction; radiochemistry followed by α- and γ-spectrometry and mass spectrometry; optical emission spectrometry; ion-, gas-, and gel-permeation chromatography; gas chromatography mass spectrometry; infrared spectrometry; x-ray photoelectron spectroscopy; x-ray fluorescence spectroscopy; and metallurgical analysis. Having a laboratory equipped to perform all these tasks is a very expensive proposition.

The material under analysis in the Moody example referred to above was a small quantity of Highly Enriched Uranium (HEU) that was seized at a border crossing in Bulgaria in 1999. Analysis of the material by scientists at Lawrence Livermore National Laboratory (LLNL) determined that the sample was probably HEU reactor fuel that had an original pre-burnup enrichment level of 90%. It was determined that the material was chemically reprocessed on November 1, 1993, with an uncertainty of about one month. All indications were that the material was of non-US or Western European origin.[9]

The technique used to determine the "age"[10] of the Bulgarian material is a standard technique in nuclear forensics analysis of nuclear material called chronometry. Because nuclear materials are radioactive and decay by alpha particle emission to a series of alpha and beta particle emitters, they are part of four distinct decay chains. The alpha decay occurs when a helium nucleus (alpha particle) consisting of two protons and two neutrons is ejected, thus changing the decaying radionuclide to a new element with two fewer protons and an atomic number which is four units lower than the decaying radionuclide. The four decay chains for heavy elements are depicted in Appendix I attached to this paper.

Starting from a clean separation (and this may not always be the case) in which the radioactive daughters of a radionuclide of interest have been removed, the radioactive

6 Moody, Kenton J., Patrick M. Grant, and Ian D. Hutcheon. Nuclear forensic analysis. 2nd edition, CRC Press, 2014.

7 Fedchenko, Vitaly, ed. The new nuclear forensics: analysis of nuclear materials for security purposes. Oxford university press, 2015. Available for download at: https://www.sipri.org/publications/2015/sipri-monographs/new-nuclear-forensics

8 Kristo, M. J., Nuclear Forensics, LLNL-BOOK-756300, 2018. Available at: https://www.osti.gov/servlets/purl/1603873

9 See Chapter 20 of the Moody text fn 4 that describes the Bulgarian seizure and the subsequent analysis of the seized material.

10 Age of material in nuclear forensics refers to the time since the material was last processed or enriched in a manner that separated uranium or plutonium from its decay products.

daughters, granddaughters, etc. in these four chains will grow back into the sample at well-known and relatively easily calculated rates. Measuring the ratio of activities of members of the chain can allow back calculation to determine the point in time at which the repopulation of the chain started (and therefore the "age" of the material). Age determination along with other factors is a significant first step in determining where the material was produced. Note, however, that this may give no indication as to the party that was in control of the material after its separation.

Although the calculation of the ratios of the various radionuclides in the chain is straightforward, these ratios may be extremely difficult to measure. Interference from other materials and very small ratios may require that extremely precise radio chemical separations be performed to isolate and observe these ratios. Decay measurements may require long counting intervals. Frequently, a number of ratios, perhaps four or five, are determined from a sample and they should in theory produce the same age of the material or give an indication of the uncertainty of the age calculation.

Relative abundances of uranium isotopes enabled the analysts of the Bulgarian material to match computer calculations of reactor burnup to determine the pre-burnup enrichment of the HEU sample. Once again, this enrichment level was inconsistent with reactor fuels found in the US and Western Europe. The morphology determination of the HEU material showed that the grain size of the particles of HEU powder was also inconsistent with known processes in the US and Western Europe.

In the Bulgarian seizure, in addition to determination of the age of the material and the various radionuclides and their abundances in the sample, the LLNL analysts made significant non-radioactive determinations about the packaging in which the HEU sample was discovered. These determinations were, once again, consistent with the source of the HEU being in Eastern Europe or the Russian Federation.

The techniques which have now been used in nuclear forensics analysis have received sufficient scientific review and corroboration that they can be used to support expert opinion testimony in legal processes. However, often the questions asked of nuclear forensics are far simpler than the complex age and attribution issues. It may be legally sufficient for a nuclear forensics expert to only opine that a particular seized sample of radioactive material is in fact radioactive material of the specific type that should be under control and for which the alleged criminal had no right to possess.

Finally, although a general rule of nuclear forensics is to proceed from passive analysis to destructive analysis if necessary, there is no "cookbook" solution that can be generally applied to all analysis. The infrequent actual events require creative interpretation and constant review and refining of the analysis procedures. Despite all the developments in nuclear security, open questions still remain about how the legal system could treat a number of defense objections to nuclear forensics procedures.

Analysis of Other Radioactive Material

The ability of nuclear forensics to analyze other radioactive material is significantly different from its ability to deal with nuclear material. Whereas nuclear material may provide significant clues as to where it was produced, enriched, or reprocessed, the same is not true for other radioactive material. Cs-137 or Co-60 will look identical regardless of where they are produced. Therefore, in the analysis of other radioactive materials, looking for trace chemical or radioactive residue that might indicate the manufacturer or the manufacturing process may be very important.

It is generally considered far less likely that nuclear forensics will be able to successfully attribute radioactive material in comparison to its ability to attribute nuclear material to a source. However, nuclear forensics still has a significant role to play in dealing with other radioactive material and evidence that may be contaminated by these materials.

Conclusion

Nuclear forensics capability is a modern necessity. There are significant capital costs for the equipment and facilities in the scientific analysis aspect of nuclear forensics. There are also significant training and maintenance costs to keep equipment and trained personnel at a high level of readiness for the infrequent actual need for the analysis. However, when the ability is needed, it must be readily available to quickly respond to an incident. International cooperation is essential to effectively create nuclear security systems to minimize the need for nuclear forensics analysis and to aid analysis when needed. International sharing of nuclear forensics techniques and information about nuclear and radioactive materials produced in various states significantly aids the ability of the system to function properly.

This paper has provided some basic background information on nuclear forensics and provides further resources to allow further exploration of the topic.

Appendix I

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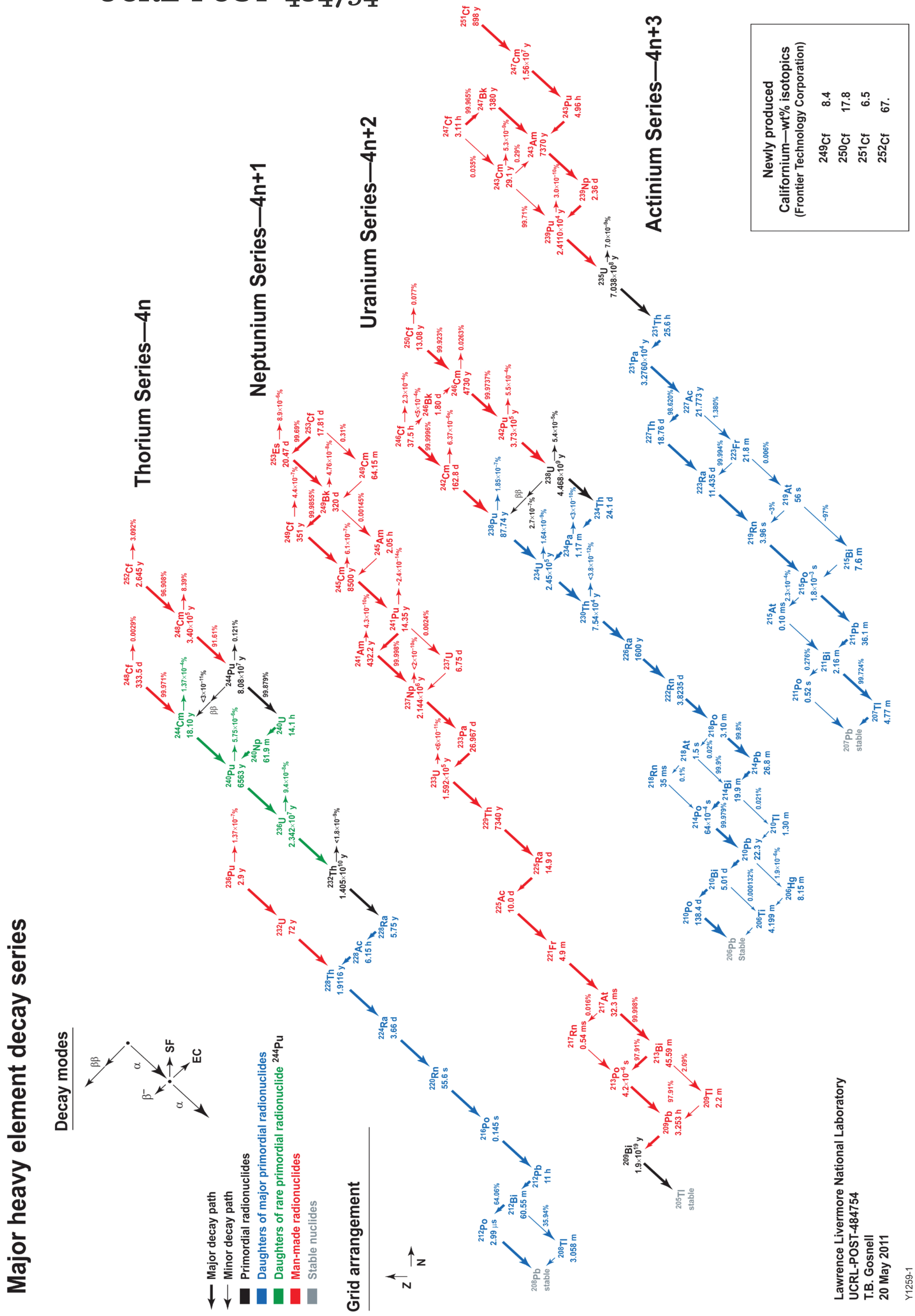


Nuclear Fuels for Light Water Reactors

Martin ŠEVEČEK

ALVEL s.r.o.
Czechia

This lecture provides a short overview of fuel system design and construction with a focus on PWR and VVER fuels. The response of a nuclear fuel system during accidental scenarios is the main limiting factor for nuclear reactor design and construction. Industrial and R&D state-of-the-art such as advanced or accident tolerant fuels will be introduced.

Fuel system

Fuels (Fuel pellet)

Fuel is the very heart of a nuclear reactor. It contains fissionable isotopes; fission takes place in the fuel, resulting in fission product accumulation. Fuel also acts as the first barrier between radioactive isotopes and the environment in the defence-in-depth safety fundamental. It signifies the retention of gaseous fission products up to high burnup and almost complete retention of solid fission products.

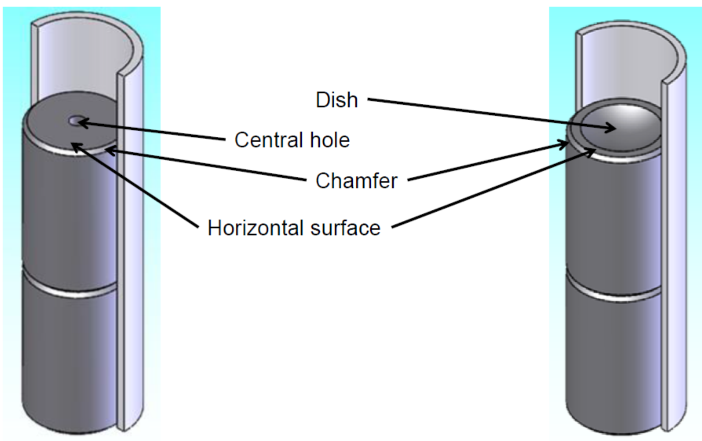


Figure 1 Fuel pellet nomenclature [1]

All light water reactors use as fuel ceramic UO_2 or UO_2-based pellets slightly enriched with ^{235}U.[1] The geometry, porosity, chemical composition (e.g., O/M ratio, burnable absorbers) depend on the fuel vendor, reactor type, operational requirements, and specific fuel design. Figure 1 shows typical pellet geometry. VVER fuels typically have central holes (similarly to fast reactor fuels) whereas PWR pellets have dishes to compensate for fuel swelling. Both types have chamfers to limit sharp edges which might cause failures after fuel contact with cladding.

1 MOX fuels, doped fuels, and fast reactor fuels are not covered here.

Figure 2 "Green pellets" in a sintering boat [2]

The simplified direct fabrication process is:

a. UO_2 powder preparation – dry/wet route

b. Conditioning – homogenization and additives

c. Compression into green pellets (Figure 2)

d. Sintering at 1700–1900 °C in a reduction atmosphere (Ar-H)

e. Dimension tuning, grinding, and polishing

f. Quality control

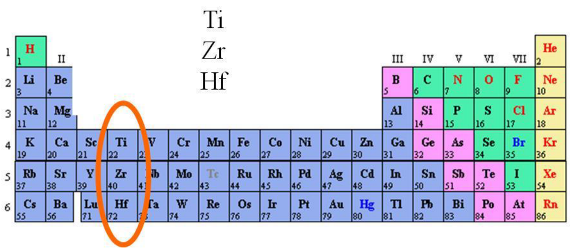
After several quality control steps, pellets are inserted into cladding tubes.

Cladding

Cladding is the second barrier in the defence-in-depth concept. Its performance is crucial for safety evaluation. Even though fuels act as the first barrier, they crack, fragment, or release fission products during reactor operation. These need to be confined by cladding. Cladding performance, therefore, indirectly limits operational parameters such as fuel burnup (i.e., hydrogen content in the cladding alloy) or determines the design of safety systems such as emergency core cooling system (e.g., equivalent cladding reacted during a postulated accident).

The main requirements for fuel cladding are:

- Low neutron absorption/activation/transmutation (Figure 3)
- Irradiation resistance
- Corrosion resistance in an aggressive LWR environment
- Good mechanical properties, machinability, cost, etc.



Ti, Hf, Zr – Chemically similar:

- Hf – needs to be removed – Zr then becomes a dual-use item
- Ti – processing methods used, duplicated for Zr

Stainless steel – up to 20x more absorption

- Fuel cycle economy
- Transmutation + other radiation defects

Is Zr "transparent" to neutrons?

- No! – scattering reactions with fast neutrons – irradiation damage

Pure Zr – poor corrosion resistance

Alloys – adding alloying elements such as Nb, Sn, Fe, Cr, etc.

Element	Σ (cm^{-1})	σ (barn)
0	0,0000000	0,000270
H	0,0000173	0,332
N	0.000099	1.85
C	0,000273	0.034
Be	0,0011	0,0092
Mg	0,0027	0,063
Zirconium	**0,0079**	**0,185**
Al	0,0139	0,230
Sn	0,021	0.63
Nb	0,064	1.15
Mo	0,170	2.65
Fe	0,216	2.55
Cr	0.26	3,1
Cu	0,320	3.79
Ti	0.35	6'1
V	0,355	5.04
Ni	0,404	4.43
Ta	0.57	10,4
w	1.17	18,5
Li	3.28	70,7
Co	3.38	37,2
Ag	3.73	63,6
Hafnium	**4.6**	**102**
In	7.41	193.5
B	100	759
Dy	29	930
Cd	113,4	2450
Eu	96	4600
Sm	175	5800
Gd	1490	49000

Figure 3 Neutronic performance of cladding materials [3]

Two main historical groups – (1) Zircaloys (Zr+Sn) and (2) Zr-Nb alloys, resulting in hundreds of combinations with various processing parameters and additions (Figure 4). Extensive testing and the development of new alloys is ongoing (e.g., E110M, HiFi, etc.) [4], [5]

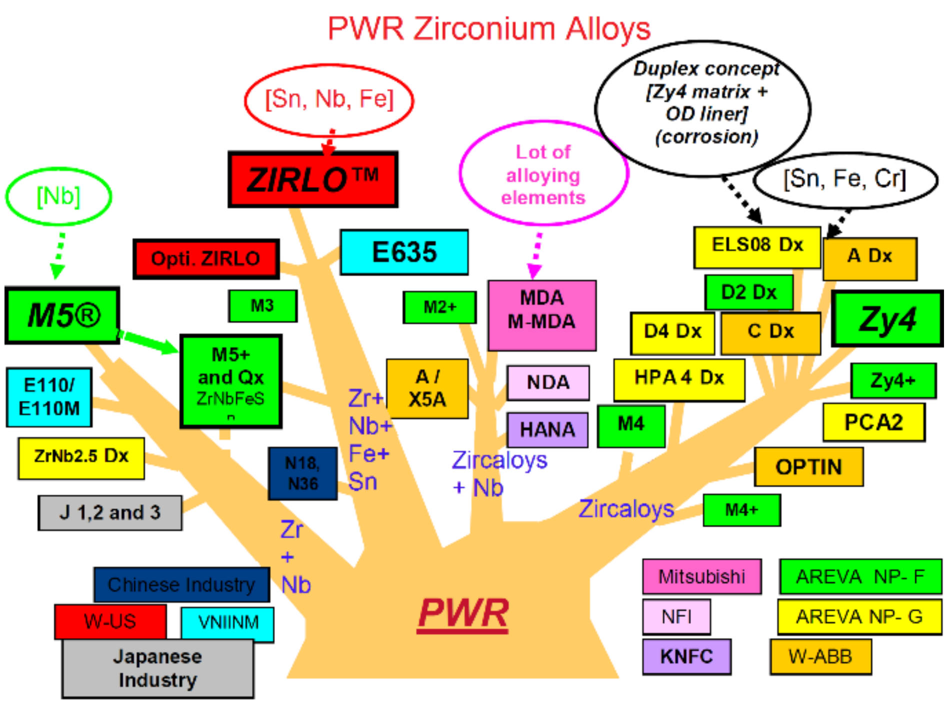


Figure 4 Zirconium-based cladding alloys used in PWRs [6]

Improved cladding properties lead to improved reactor performance, such as higher burnup or extended fuel cycles while minimizing fuel leakage during nominal operation. The historical trends in achieved cycle periods and burnup are shown in Figure 5.

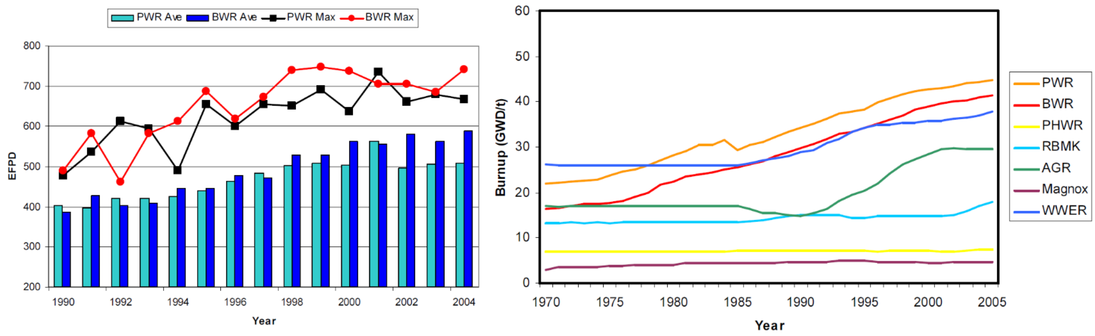


Figure 5 The continual increase of operational cycles and burnup facilitated by improved cladding performance [7], [8]

Fuel rod/pin/element

Zr-based tubes are first welded to a bottom end cap. Fuel pellets are then inserted with a plenum spring at the top, pressurized with He up to the required pressure, and sealed with a top-end cap. VVER and PWR fuel rods are schematically shown in Figure 6. End caps, the welding technique and filling pressure are usually proprietary. Different methods are used by different fuel vendors.

The operational regimes of reactors influence the pin design. Fuel vendors optimize many interlinked parameters, such as cladding-fuel gap closure time, end-of-life pressure, load-follow operation leading to PCMI, etc.

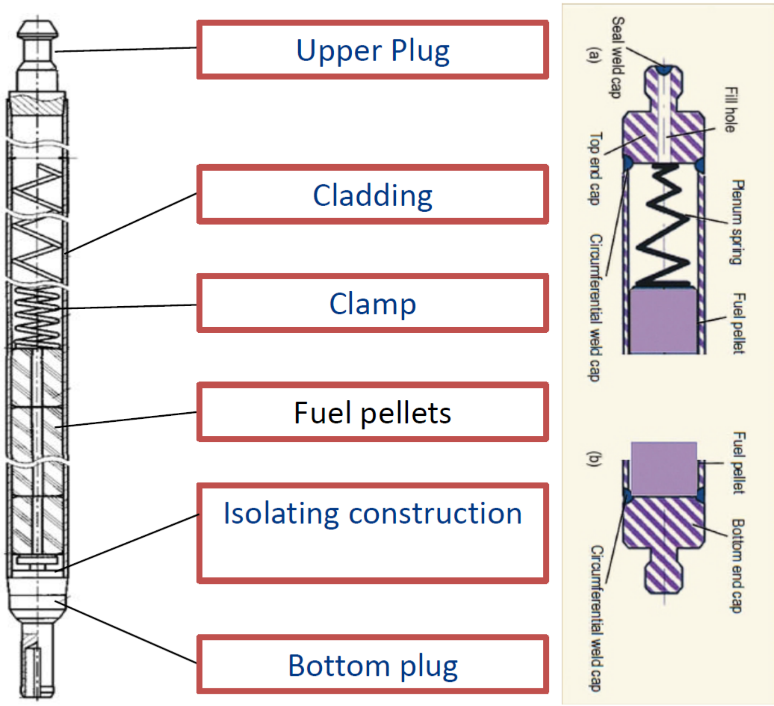


Figure 6 Fuel rod parts

Thermal analysis

Radial power profile in a fuel rod changes during operation, which also affects the radial temperature profile. A typical temperature profile with fresh fuel is shown in Figure 7. UO_2 is a ceramic material with low thermal conductivity that further

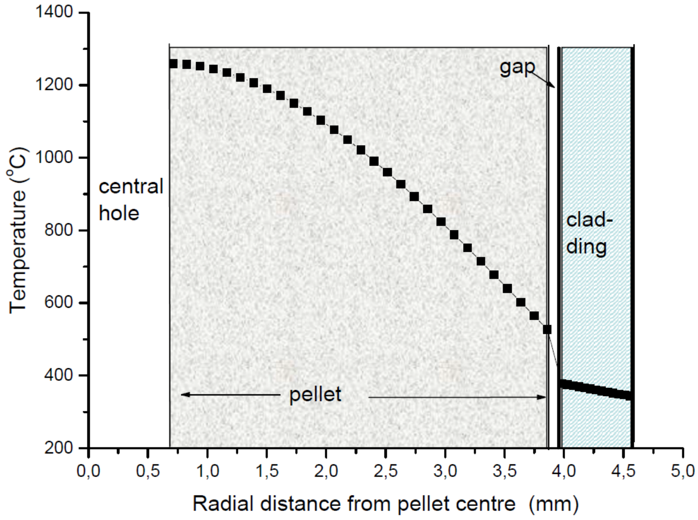


Figure 7 Temperature distribution in fuel pellet[2]

decreases with radiation defects = burnup. Fuel-cladding gap conductivity changes according to size, gas composition or surface roughness. Cladding and fuel change structure during irradiation and accidental scenarios make thermal analysis of fuel pins a complex and multidisciplinary challenge.

Mechanical analysis (internal pressure, gap closure, pellet-cladding interaction, etc.)

With the exception of complex thermal analysis, the mechanical conditions of a fuel rod develop continuously. There are many interlinked phenomena which affect other processes, as shown in Figure 8.

For example, a continuous increase of internal pressure exists due to cladding creepdown plus the release of gaseous fission products and fuel swelling. However, the internal pressure should be below coolant pressure to avoid lift-off effect—internal pressure opens the gap, which leads to a temperature increase, in turn leading to higher FGR (Xe and Kr mainly) and further positive feedback which must be avoided .

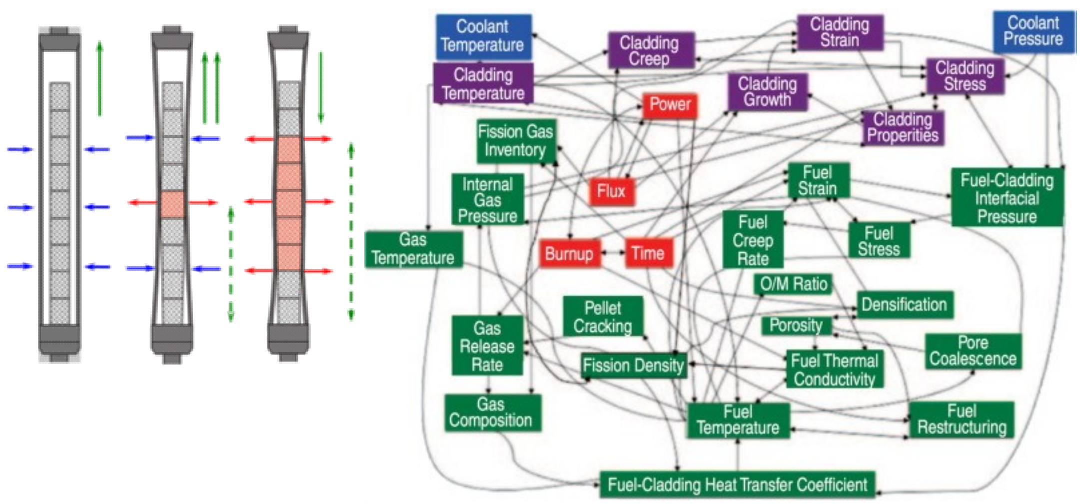


Figure 8 Schematics of fuel pin performance and links between separate phenomena considered in an analysis [10], [11]

Fuel assembly

Guide tubes – non-fuelled tubes made of Zr alloy are not pressurized or exposed to contact with fuel and should provide greater stiffness. Tubes are also used to lead control rods, place burnable poisons or insert in-core measurement and other devices. Different alloys can be used, for example E635 in VVERs, which better serve the specific purpose.

Spacer grids – the main purpose of these grids is regular spacing of fuel rods in a fuel assembly under the irradiation conditions:

- Vibrations throughout the coolant flow structures
- Deformations of the FA
- Changes od outer diameter of fuel rods
- Movement of fuel rods in the axial direction

Other construction components might be introduced to increase mixing of the coolant and improve heat transfer. The contribution of the spacer grids to hydraulic resistance should be minimal.

Top and bottom nozzle – compatibility with the reactor design and permitted manipulations. The lower nozzle contains an anti-debris filter which prevents any foreign objects entering the core.

The construction parts of VVER and PWR fuel assemblies with a skeleton design and inserted fuel rods are shown in Figure 9.

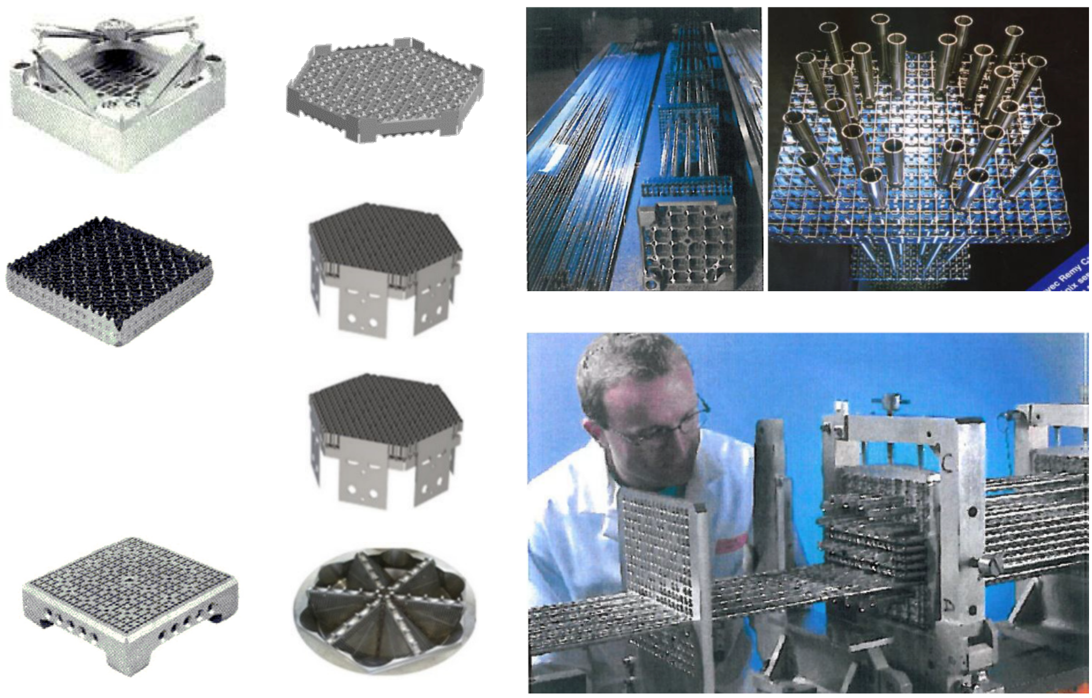


Figure 9 PWR and VVER fuel assembly construction parts (spacers and mixing grids, upper and lower nozzle and debris filters); fabrication of an assembly skeleton made of grids and guide tubes with fuel rods inserted into the skeleton [12]-[14]

Fuel system behaviour

The objectives of safety analyses are to study cladding and fuel failures. The loss of integrity leads to activity release into the primary coolant and beyond. The failure and related fuel failure mechanisms depend on the actual fuel rod state, which changes according to power, neutron fluence, fabrication parameters, etc.

Nominal/long-term operation

The sequences of fuel states during reactor operation are as follows (Figure 10):

1. Cold state fuel – stresses in the cladding due to internal and external pressure difference.
2. High-temperature reached during start-up – thermal expansion and crack formation.
3. High-temperature operation – densification – pellet volume/diameter reduction and cladding creep
4. Swelling of pellet – accumulation of fission products and continued cladding creep.
5. Gap closure – mechanical and chemical interaction between cladding and fuel.
6. Fission gas release – high pressure; pellet changes due to high burnup, and fuel/pellet interaction

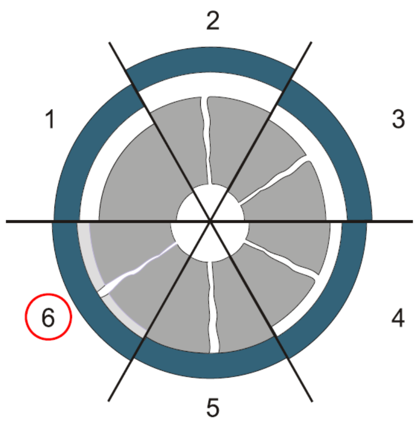


Figure 10 Schematics of fuel rod changes during normal operation [2]

Design basis accidents

Two limiting postulated scenarios are considered for fuel systems – Reactivity Insertion Accident (RIA) and Loss of Coolant Accident (LOCA). These conservative scenarios cover less severe events and transients. The bounding cases define limiting parameters in a form of safety/operational and other criteria.

RIA

RIA event – control rod ejection and prompt change of boric acid concentration. Scenario – very fast power increase, high power peak, and power decrease by Doppler effect (0.1-0.3 s). Several fuel rod failure mechanisms – (1) pellet thermal expansion with strong mechanical contact with cladding; (2) fuel rod overpressure with high temperature creep; (3) melting (Figure 11) [7] -[9].

To avoid this type of failure, limits are in place which depend on fuel rod conditions, typically burnup. The core and its control rod worth should be designed to stay below a limiting value [10].

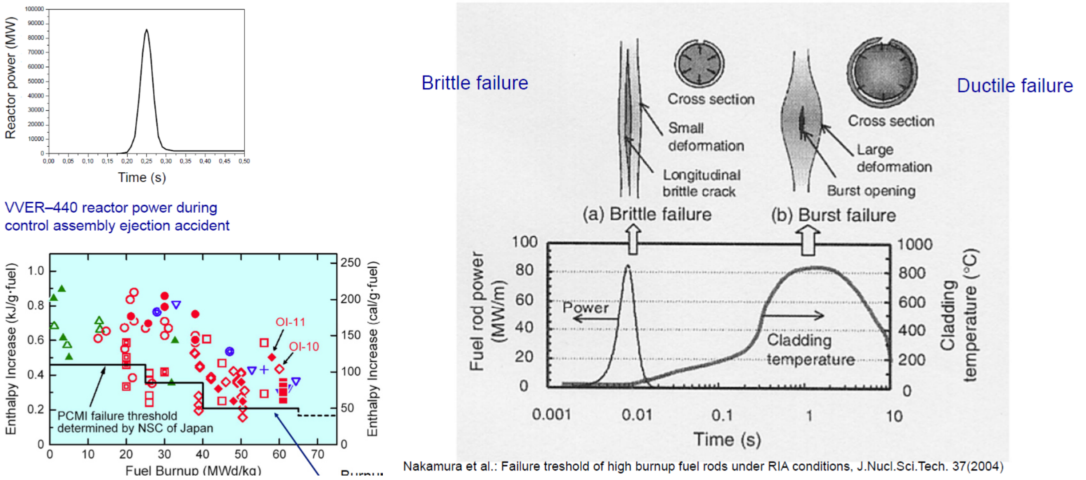


Figure 11 RIA power and temperature scenario with failure mechanisms. To avoid PCMI failure in the first stage, limits ensure the ductility of the cladding tube [19], [20]

LOCA

Guillotine rupture of a cold leg, leading to loss of coolant in the core. The first temperature peak leads to cladding creep, ballooning, and burst. High-pressure injection reduces the temperature for a short time. The temperature rises and cladding is exposed to high-temperature steam for a longer period. The core is then quenched by low-pressure pumps. ECCSs should be designed to handle these events and avoid the uncontrolled development of an accident and formation of non-coolable geometry [21]-[23]

As with RIA, several safety criteria are defined. The equivalent cladding reacted (ECR) degree of oxidation is limited to about 17 % (depending on the country) and ensures that the cladding is still ductile and does not fracture during the final quenching phase. The temperature is also limited to about 1200°C to avoid breakaway oxidation which would result in an uncontrollable scenario. Zr high-temperature steam oxidation is strongly exothermic and produces large amounts of hydrogen. The total volume of produced hydrogen is also limited during the potential scenario [18], [25].

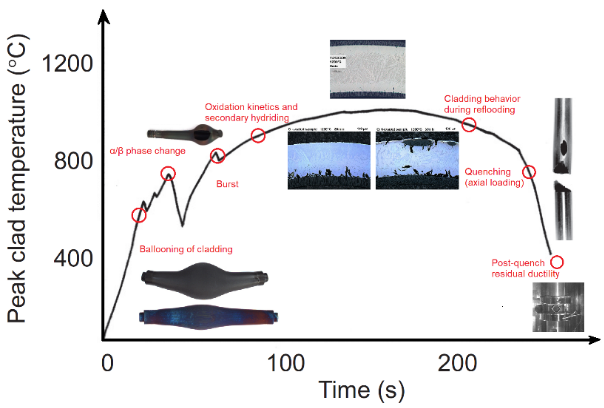


Figure 12 LOCA scenario with a set of expected phenomena which occurs during an accident [23]

Innovative fuels

History and current innovations

The fuel system based on the combination of cladding made of Zr alloys + ceramic UO_2 fuels has been in use for decades. It was first used in US nuclear submarines and later adopted by industry. Alternative fuels and cladding materials were studied but only incremental improvements were eventually adopted by the nuclear industry. Most of the improvements are linked to burnup increases, power uprates, longer operation cycles, higher operation flexibility due to load-follow requirements, fuel reliability, and economy [9], [26], [27].

Innovations implemented over the last 20 years:

- Modern cladding alloys with reduced corrosion resistance, lower H-pickup, better irradiation resistance
- Burnable absorbers – many different forms and methods to implement them
- Fuel dopants – improved UO_2 properties, such as grain size, retention of fission products, plasticity and thermal conductivity
- Assembly designs – new debris filters, new grid designs, higher stiffness, lower pressure losses, better neutron economy

Accident Tolerant/Advanced Technology Fuels

After the events at Fukushima-Daiichi, the US DOE initiated activities towards the development of safer fuels which would help operators in the case of similar accidents [28]-[30]. nterestingly, postulated RIA or LOCA scenarios have never occurred, yet several severe accidents have (Three Mile Island, Fukushima-Daiichi). Theoretical accident progression is shown in Figure 13. Fuel vendors and many research groups around the world have been developing new types of materials and designs since 2011 to further improve safety, with the first lead test rods being inserted into commercial reactors in 2019.

The proposed new fuel designs range from modification of current materials (protective coatings), replacements for Zr alloys (FeCrAl, SiC/SiC), new fuel materials (USi, UN, U-Mo) and accident tolerant non-fuel components (BWR channels, control rods).

The following are the most promising designs[2] currently being pursued by fuel vendors:

- FRAMATOME/Westinhouse/TVEL – Cr PVD coated Zr cladding and doped UO_2 pellet (Figure 14)
- GNF – FeCrAl alloy; ARMOR coating on Zr (Figure 16)
- TVEL – high-density fuels; Cr-Ni alloy (Figure 17)
- KAERI – microcell UO_2 fuel pellets + CrAl coated Zr (Figure 15)

2 There is much PR in this field, but physics still prevails. Any material will eventually fail if there is no heat removal. New designs will in theory give operators more time to react and manage an accident before any cliff edge. This is called coping time and estimates how much time a new fuel design provides [32] [22].

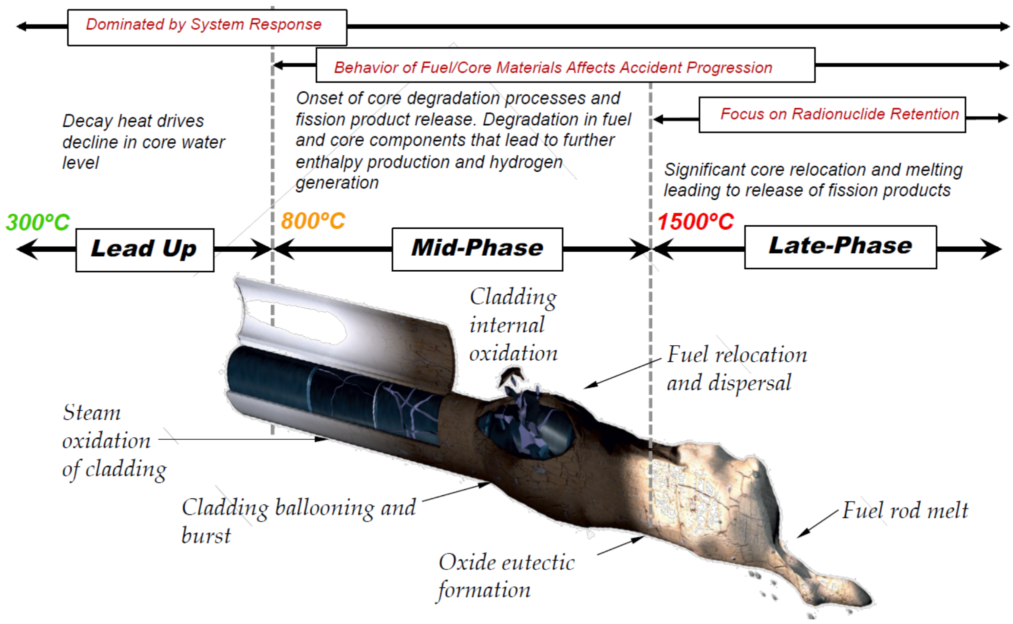


Figure 13 State of a fuel rod during accident progression up to fuel melting [31]

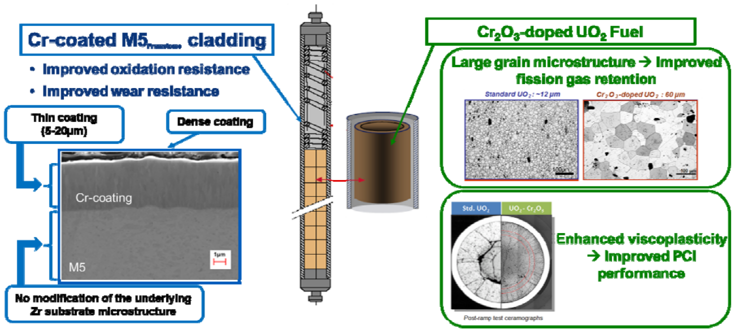


Figure 14 Framatome's ATF fuel concept [33]

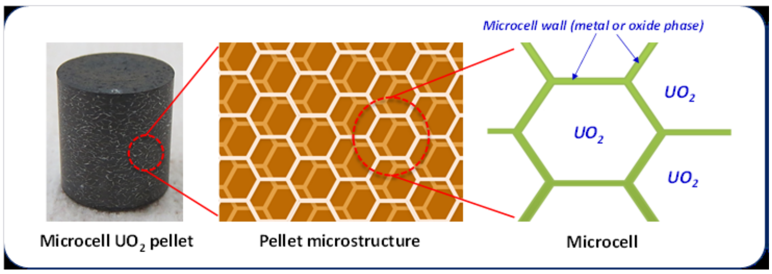


Figure 15 KAERI's microcell UO2 fuel concept [34]

Figure 16 FeCrAl cladding, commercialized by GNF [35]

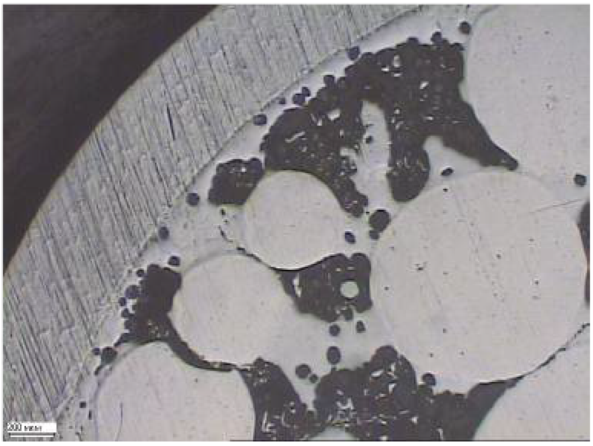


Figure 17 TVEL's concept of U-Mo fuel inside a matrix enclosed in a Cr-Ni alloy (similar concept used in Akademik Lomonosov) [36]

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Nuclear Weapons

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The objective of this short paper on nuclear weapons is to give some background for the lecture to be presented at the Czech Technical University. In such a short paper, it is only possible to cover highlights of the history of nuclear weapons development, some elementary concepts of weapons design and effects, some of the issues surrounding nonproliferation of nuclear weapons, and some relevant modern issues with nuclear weapons, such as Safeguards and the Conference of Test Ban Treaty and the Treaty to Ban Nuclear Weapons. Along the way, some references will be provided for those participants who want to read further on various subjects.

Historical Background:

After fission was discovered in the 1930s, it was soon realized that if the large energy releases associated with fission could be harnessed, a powerful explosive could be developed. In the United States, this resulted in the Manhattan Project, and in the Axis powers, both Germany and Japan explored the potential of developing a nuclear weapon.[1]

Of interest during the Manhattan project was the fact that the original goal was to use a gun assembled system (discussed further below) which would employ plutonium produced in a nuclear reactor as the fissile material. This concept was called Thin Man, and the gun design for a Highly Enriched Uranium (HEU) system[2] was later called Little Boy.

Figure 1 Thin Man casings - available at: https://en.wikipedia.org/wiki/Thin_Man_(nuclear_bomb)

1 Those who are further interested in the subject might consider reading *The Making of the Atomic Bomb* by Richard Rhodes and his later book *Dark Sun: The Making of the Hydrogen Bomb*. Rhodes, Richard. See, *The Making of the Atomic Bomb*. Simon and Schuster, 1986 and *Dark Sun: The Making of the Hydrogen Bomb*. Simon and Schuster, 1995

2 Highly Enriched Uranium (HEU) is defined as uranium enriched to 20% or more Uranium 235. Weapons grade HEU is generally considered to be enriched at about 90% or greater.

It came as a shock to the scientists of the Manhattan Project that reactor produced plutonium, for which they had anticipated a plentiful supply, contained a significant amount of plutonium 240, which made a plutonium gun assembly impractical for reasons we will discuss during the lecture.

The HEU gun design was simple and straightforward, however producing enriched uranium for the device was a slow and laborious process. Due to the uncompressed nature of the gun design, more fissile material was required for an uncompressed gun system than for the later developed plutonium implosion systems. The gun assembly was so simple and straightforward that no testing was thought to be needed. However, the implosion system required testing. The first test in July 1945 at Trinity, New Mexico, proved that the implosion system would function as designed.

Two nuclear weapons were dropped on Japan, the Little Boy design on Hiroshima followed by the Fat Man design on Nagasaki. Unresolvable questions remain about how necessary these bombs were to ending World War II in the Pacific, and about the morality of using these weapons without first making a demonstration to Japanese officials.

In the postwar period, the newly created United Nations attempted to grapple with the threat that nuclear weapons posed to humanity. The United Nations Atomic Energy Commission (UNAEC) sought to bring about a system of control and nonproliferation. However, the nuclear arms race was already underway (the Soviet Union tested its first device in 1949) and the UNAEC was dysfunctional and finally disbanded in 1952. U.S. President Eisenhauer's Atoms for Peace speech to the UN in 1953 helped prompt the need for international cooperation in the nuclear arena. The result of these efforts was the current International Atomic Energy Agency (IAEA), which came into being in 1957 and is arguably a successor to the ill-fated UNAEC, albeit an independent international organization.[3]

One of the first functions of the IAEA was to establish a system to safeguard nuclear material. The first safeguards efforts were voluntary, but when The Treaty on the Non-Proliferation of Nuclear Weapons (NPT) went into effect in 1970, it required all Non-Nuclear Weapons States (NNWSs) to establish an agreement with the IAEA to safeguard all nuclear materials in the state. The declared Nuclear Weapons States of the NPT (United States, United Kingdom, Soviet Union, France, and China) have now been joined by India, Pakistan, and North Korea as states possessing nuclear weapons.

Testing of nuclear weapons during the Cold War, particularly testing in the atmosphere, caused significant worldwide concern. Agreements between the Soviet Union and the United States and later with others moved testing underground and reduced the atmospheric burden of radiation released from tests. Ultimately in 1996, the Conference of the Nuclear-Test Ban Treaty (CTBT) led to the virtual cessation of all nuclear weapons testing. However, non-signatory North Korea has continued to test. Sadly, the CTBT is not yet in force since some of the signatories, notably China and the United States, have failed to ratify the treaty although they have ceased testing.

3 The Statute of the IAEA which established the functions of the Agency can be found at: https://www.iaea.org/about/statute.

Physics of nuclear weapons

Nuclear weapons have two broad types: fission only weapons and thermonuclear weapons. Historically, fission only weapons are described as "atomic weapons", and thermonuclear weapons are described as "hydrogen weapons." Generically these may be bombs dropped from aircraft or warheads placed in weapon systems. Examples are ICBMs, SLBMs, and missiles of all types, including anti-air, antisubmarine, air-to-air and nuclear artillery and shorter-range missile systems such as the earlier generation of weapons of Soviet SCUD and FROG systems and the US Honest John and Lance.

The actual design of nuclear weapons is considered highly classified by most states. However, some general principles have been declassified and can be discussed openly. In this brief paper, we will discuss the major types of designs and will compare the fission process in nuclear weapons with the process which occurs in nuclear reactors.

Fission Gun Designs

As noted above, the simplest type of nuclear weapon is a gun design. In a gun design, two subcritical masses are pushed together in some manner (Figure 2). The process is similar to firing a gun where the bullet (one subcritical mass) is fired at the target (the other subcritical mass). When these two subcritical pieces come together a critical assembly is produced, and a runaway chain reaction is developed once a sufficient neutron source is present to start the chain. To ensure that neutrons are present at the desired time for the device to explode, a neutron source called an initiator is used. Early gun assembled systems used an initiator that contained an alpha emitting radionuclide and beryllium separated by a thin layer of gold. When the initiator was crushed by the two pieces coming together, a flood of neutrons was released from the ensuing (α,n) reaction in the beryllium and the device detonated at the point of full assembly.

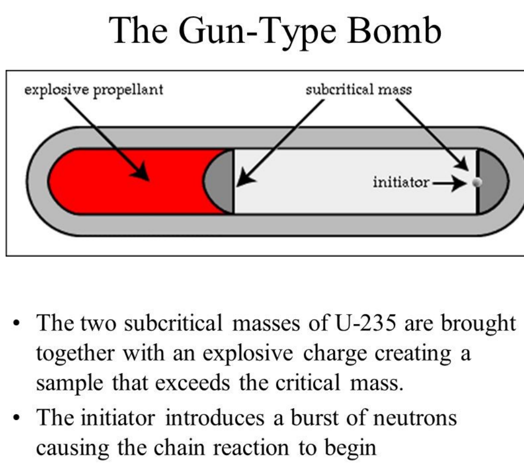

Figure 2 Generic Gun Design Concept (available at: https://people.wou.edu/~courtna/ch371/lecture/lecture5/sld014.htm)

If the two pieces of a gun design represented only one critical mass when fully assembled, there would be almost no yield and the initial reactions would quickly make a subcritical system. Thus, the pieces in total need to be more than a critical mass. Because the two pieces are more than a critical mass when fully assembled, a point called "first critical" occurs as the pieces come together. Although a gap exists between the pieces at the first critical point, the entire assembly at first critical is a critical mass and will detonate (albeit with a negligible yield) if sufficient neutrons are present at that time.

Detonation at first critical is what prevents a plutonium gun assembled system from producing any significant yield. Plutonium produced in a reactor, even the material considered weapons grade, contains so much plutonium 240 that the high spontaneous fission rate of the plutonium 240 will release enough neutrons per unit time that a plutonium gun assembly will always begin to explode at first critical and therefore produce negligible yield. Fizzle yield is the term used to describe the negligible production of yield in this or a failed system assembly.

Implosion Designs

When it was recognized that a plutonium gun assembly would produce only a fizzle yield and that insufficient uranium was available for the production of an HEU gun design weapon in a short time, the Manhattan Project turned to the development of a plutonium implosion weapon, recognizing that implosion starting with a subcritical mass of material and compressing it with high explosives to make it supercritical was the only way forward to having more than a few weapons. The concept of the implosion system is shown in Figure 3.

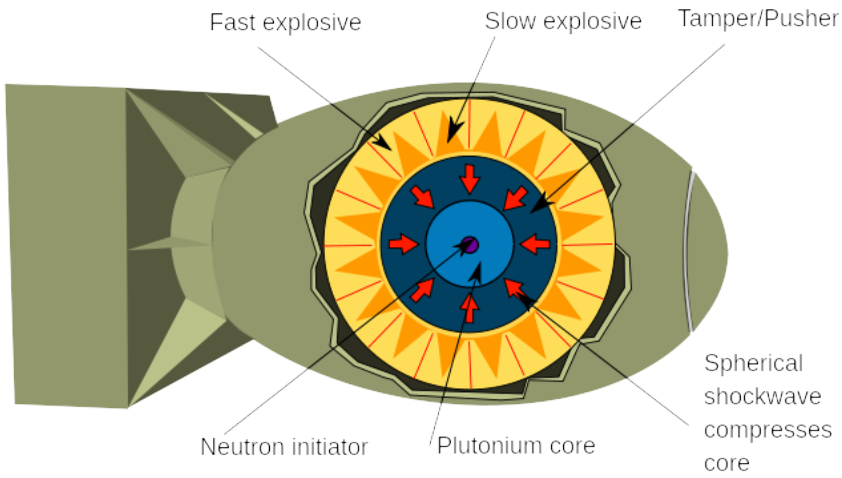

Figure 3 Basic Implosion System (available at https://www.quora.com/Is-the-amount-of-plutonium-in-an-implosion-based-atomic-bomb-relative-to-the-pressure-the-implosion-generates)

In an implosion design, plutonium or HEU[4] or perhaps combination of these starts in an uncompressed subcritical form. High explosives are used to compress the material until it is supercritical, at which time, as with the gun assembled system, a neutron initiator produces a flood of neutrons to trigger a runaway chain reaction.

4 Plutonium used in weapons by the USA and other major countries has typically been what is known as "weapons grade" plutonium. Weapons grade plutonium is produced by removing fuel from a reactor and reprocessing at a time when the plutonium 240 content of the fuel is approximately 6% or less. Typical nuclear power plant fuel if reprocessed after normal removal would be in the order of 24% plutonium 240. This is described as "reactor grade plutonium" and is a subject of some controversy regarding its usefulness in a nuclear weapon if reprocessed. Also note that the concept of HEU does not apply to uranium 233, which can be extracted from the thorium fuel cycle without enrichment. Plutonium and uranium 233 have much lower critical masses than weapons grade HEU and therefore are "better" for use in a nuclear weapon if they are available.

Boosted Fission Weapons

In an implosion device, a small amount of deuterium and tritium gas is heated by the implosion shock to the point where fusion occurs. A significant increase in yield (the "boost") can be produced by the additional fission caused by the 14 MeV fusion neutrons released in the deuterium and tritium fusion.

Thermonuclear Weapons

Thermonuclear or hydrogen weapons are two-stage devices. A fission device is the primary stage, and a fusion system is the secondary stage. Whereas the largest fission system was in the order of 500 kt of yield, thermonuclear systems can produce tens to hundreds of megatons of yield.

Safety of Nuclear Weapons Designs and Systems

A number of safety aspects relate to nuclear weapons. Obviously, some safety aspects regard the decision whether to use them in the first place. Here, however, in discussing safety, we will discuss how designs prevent accidental yield or intentional unauthorized use of the weapon.

US weapons are designed to be "one-point safe" so that they produce a negligible yield if the high explosive system of the weapon is accidentally or intentionally detonated at its most critical point. For example, if one were to fire a rifle or pistol accidentally or intentionally at the device and hit the most critical point at which the device could be struck, the weapon would only produce a trivial yield at most. Other safety concepts employed in US nuclear weapons are the use of Permissive Action Link (PAL) systems and environmental sensors. Separately, or in combination with environmental sensors in the weapon system, the PAL system prevents the weapon from being used by an unauthorized user.

Comparison to Nuclear Reactors

Although a supercritical nuclear reactor (critical on prompt neutrons only) can generate enough energy to disassemble itself, it is hardly thought of as a bomb. In a nuclear reactor, control is achieved by using control rods and taking advantage of delayed neutrons so that the reactor is critical on a combination of prompt and delayed neutrons. Furthermore, most reactors take advantage of the almost 1000-fold higher cross-section for thermal neutron induced fission and employ moderation to slow neutrons from the MeV energies at which they are born to a fraction of an eV at which they hopefully induce fission.

Nuclear weapons, on the other hand, function only on prompt fast neutrons. The timing in nuclear weapons is such that the explosion phase is over in a few microseconds.[5] There is no time for moderation or the generation of delayed neutrons. Uranium 238 fission is typically more important in weapons than in a reactor.

5 See Glasstone and Dolan, *The Effects of Nuclear Weapons*, U.S Department of Defense 1970, at page 17. Available at: https://www.dtra.mil/Portals/61/Documents/NTPR/4-Rad_Exp_Rpts/36_The_Effects_of_Nuclear_Weapons.pdf

Radioactivity of Nuclear Weapons

In contrast to what the public believes, nuclear weapons themselves are not highly radioactive. If they were, they would present a health hazard to potential users. Unfortunately for nuclear security considerations, this makes the detection of nuclear materials or nuclear weapons out of control difficult to achieve with passive detection systems.

Weapons Effects

The principal effects of nuclear weapons are distinctly different from those of conventional explosives. Nuclear weapons effects are shock/blast effects, thermal effects, and effects of radioactivity. One of the best sources to describe the effects of nuclear weapons is a US government publication by Samuel Glasstone and Philip Dolan.

To a first-order, weapons effects are scalable based on the yield. Therefore, if one knows the effects of, for example, a 10 kt weapon in terms of blast overpressure at a distance, one can then use a yield relationship to predict the effects of a 10 Mt weapon. Fundamental to the scaling is an understanding of the concept of cube-root scaling. In the popular press, thermonuclear devices are often compared to the Hiroshima and Nagasaki devices, which had yields in the order of 20 kt. Noting that modern airdrop bombs exist that can produce 20 Mt, such articles describe this as 1000 times more powerful, leaving the impression with the public that, for example, such a device would destroy 1000 times the area.

One way to understand the yield/effects comparisons of different yields is to imagine that a device is set off on the ground and that the energy release is isotropic. Thus, the volume above the surface that is “filled” with energy is a hemisphere. If the edge of the bowl represents an effect of interest (e.g., a particular overpressure) then we can ask how far away a bigger device would produce the same effect. The volume of the bigger hemisphere produced by the larger yield in this case relates directly to the ratios of the yields, because in order to produce the same effect the hemispheres have to be equally filled with energy. Since the volume of the hemisphere ratios scales as the cube of the radius, the effect therefore scales as the cube root of the yield ratio. As an example, if the volume of the hemisphere of the second yield is 1000 times greater than the yield of the first yield, the effect radius of the second yield compared to the first is the cube root of the factor of 1000 or a factor of 10. Thus a 20 Mt will expand the radial destructive power for a given effect by a factor of 10 compared to 20 kt device, not by a factor of 1000.

Conclusion

Hopefully, this short paper will trigger some ideas and provide some guidance as to where to learn more about nuclear weapons and their effects.

Legislative for Reactor Operation

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Historical Development

After about a quarter of a century of independent national development of nuclear reactors in a few countries (1950–1975), the need and usefulness of considering this new technology at the international level was felt and led to corresponding action. The following illustrates the development:

The strong need for international co-operation resulted in the creation of the IAEA in 1956. The objectives and functions of the IAEA are presented in the Statute of the IAEA. Article II states its essence: "The Agency shall seek to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world. It shall ensure, so far as it is able, that assistance provided by it or at its request or under its supervision or control is not used in such a way as to further any military purpose." Article III lists the main functions of the IAEA, which include"fostering the exchange of scientific and technical information", "encouraging the exchange and training of scientists and experts" and "establishing standards of safety for protection of health and minimization of danger to life and property, and providing for the application of these standards to its own operations as well as to operations making use of IAEA materials, services and information".

Commencement of the IAEA NUSS programme for nuclear power plants in 1974 after 10 years of good international co-operation followed with the publication of five codes of practice and about 60 safety guides in the IAEA Safety Series. On the basis of experience and new developments at both the technological and "philosophical" levels, revision of these documents was decided and began at the end of 1980s. The work for a complete, revised set of nuclear safety standards (now Safety Standards Series), including Safety Fundamentals, Requirements and Guides, is still ongoing. The International Nuclear Safety Advisory Group (INSAG) has produced useful basic philosophical reports such as expression of the basic safety principles which are reflected in the IAEA Safety Fundamentals and development of concepts, for example defence in depth and safety culture. In addition to nuclear power plant safety, other safety areas are being considered. The management of radioactive waste and transport of nuclear materials are among the most important of these areas. The future role of nuclear energy depends on a consistent, demonstrated record of safety in all applications.

Although the IAEA is not an international regulatory body, its nuclear safety efforts are directed towards creating multilateral, legally binding agreements, which are an increasingly important mechanism for improving nuclear safety, radiation safety and waste safety around the world. This is accomplished through international conventions (e.g., nuclear safety, civil liability, early notification of nuclear accidents and radiological emergencies, mutual assistance in case of nuclear accidents and radiological emergencies, radioactive waste management, physical protection). International conventions are binding legal instruments for the countries which sign and ratify them. The Convention on Nuclear Safety (for nuclear power plants) became effective on October 24, 1996. A "sister" convention on the safety of radioactive waste management became effective on June 18,2001.

IAEA Nuclear Safety Requirements and Guides

The development of nuclear and radiation safety standards is a statutory function of the IAEA, which is unique in the United Nations system. The IAEA statute expressly authorizes the agency "to establish standards of safety" and "to provide for the application of these standards".

In 1996, a new uniform preparation and review process was introduced, covering all areas in which the IAEA establishes safety standards. As a result, the IAEA's Safety Series was replaced with two new series of safety-related publications, namely:

- The Safety Standards Series (IAEA safety standards);
- The Safety Reports Series.

The purpose is to separate those IAEA Safety Standards publications which specify safety objectives, concepts, principles, requirements and guidelines—as a basis for national regulations or an indication of how various safety requirements may be met—from those publications which are issued for the purpose of fostering information exchange about safety.

The Safety Standards Series, whose structure was approved in 2008, contains the following types of documentation:

- Safety Fundamentals
- Safety Requirements
- Safety Guides

The series covers nuclear safety, radiation safety, waste safety, and transport safety. It also covers general topics (such as governmental organization, quality assurance, and emergency preparedness) relevant to all four of those fields, dealt with under a separate category of general safety documents.

The **Safety Fundamentals** are the policy documents of the IAEA Safety Standards Series. They state the basic objectives, concepts and principles which govern protection and safety in the development and application of atomic energy for peaceful purposes. They state, without providing technical details, and, as a rule, without detailing the application of principles, the rationale for action necessary to meet safety requirements. The **Safety Requirements** deal with the basic requirements which must be met to ensure the safety of certain activities. These requirements are governed by the basic objectives, concepts and principles presented in the safety fundamentals. The written style (with "shall" statements) is that of regulatory documents so that states may adopt the Safety Requirements at their own discretion, as national regulations. The **Safety Guides** documents contain recommendations (with "should" statements), based on international experience, regarding measures to ensure that the safety requirements are met. But unless alternative equivalent measures are implemented, the “should” statements become “shall” requirements. The IAEA **Safety Standards** have been developed on the basis of international consensus and as such they reflect very widely accepted safety levels.

Among others, there are three sets in the Safety Requirements which provide a good basis for the safety of nuclear power plants (reactors), as follows:

Design: The requirements give the basic safety requirements which must be incorporated into the concept and the detailed design in order to produce a safe plant. Following general practice, the requirements present the concept of defence in depth, for example, successive barriers to prevent the escape of radioactive material. In case of the failure of a barrier, design provisions are made available to mitigate the consequences of such failures.

Operation: The prime responsibility for the safety of the plant rests with the operating organization. This is the basic concept underpinning the requirements for operation. The requirements deal with safety related aspects of operation, including: operating limits and conditions, commissioning, structure of the operating organization, operating instructions and procedures, maintenance, testing, inspection, core management and fuel handling, review of operation and feedback of experience, emergency preparedness, radiation protection and decommissioning.

Siting: The requirements specified in the siting requirements deal with the evaluation of site-related factors to be taken into account to ensure that the plant-site combination does not constitute an unacceptable risk during the lifetime of the plant. This includes evaluation of the potential effect on the site of natural and other phenomena which might affect the area (i.e., earthquakes, floods, aircraft crashes, chemical explosions), evaluation of the effects of the plant itself on the site (i.e., dispersion of effluents in air and water), and consideration of population distribution and emergency planning.

Responsibilities of the Government, Regulatory Body and Operator

Looking in greater detail at the roles of these three organizations, we identify the main characteristics of their duties and responsibilities and the interrelationships at the implementation level.

The **government** is responsible for establishing the necessary legislative framework. The government, which is the executive that must implement the state’s duties and activities within the framework established by the legislative, fulfils the following global tasks:

- Establish and maintain the conditions necessary for controlling, from a safety perspective, implementation of the “nuclear energy programme” at all its stages.
- Establish and maintain the dedicated state’s organs (regulatory body) to implement the state’s surveillance and control of nuclear energy use within the legislative and regulatory framework.
- Protect the population against the risk associated with the use of nuclear energy; develop and establish the regulatory framework to govern effectively the state’s surveillance and control of all stages of the nuclear energy programme.

- Establish the regulatory framework for the radiological protection of persons of the population and workers in public health from all sources of ionising radiation and establish the corresponding surveillance body within the governmental organization.

It is clear that the **operating organization** has an essential and central role, and therefore, bears an important responsibility, a basic principle being: "The operating organization bears the prime (or overall) responsibility for safety". Because this prime responsibility cannot be delegated, the operating organization assumes globally the sum of "partial responsibilities" attributed to designers, constructors, suppliers, etc. during the realisation of the project (or programme).

The objective of the **Global Nuclear Safety and Security Framework** is to achieve and maintain a high level of safety and security at nuclear facilities and activities around the world. The IAEA plays a central role in strengthening this framework by assisting member states in building sustainable national competences and capabilities. It also promotes, through dedicated knowledge networks, the transfer of knowledge from countries with mature nuclear energy programmes to countries which have only just started to embark on such programmes.

International Organizations

OECD/ NEA

The European Nuclear Energy Agency was created in 1958, and its name was changed in 1972 to reflect its growing membership. It is a specialized agency within the Organization for Economic Cooperation and Development. The NEA facilitates cooperation between countries with advanced nuclear technology infrastructures.

Several committees and bodies have been established in the framework of the OECD/ NEA:

CSNI – Committee on the Safety of Nuclear Installations

CNRA – Committee on Nuclear Regulatory Activities

MDEP – Technical Secretariat Functions for the Multinational Design Evaluation Programme

WANO

WANO was created after the 1986 Chernobyl accident by the international nuclear industry. Every organization in the world which operates a nuclear power plant is also a member of WANO. WANO assists its members in achieving the highest practicable levels of operational safety by providing access to a wealth of world-wide nuclear operating experience. WANO is a non-profit organization with no commercial ties. Additionally, WANO is not a regulatory body and has no direct association with governments. WANO has no interests other than nuclear safety.

Major WANO programmes:

- Peer review (helps members compare their operational performance)

- Analysis and feedback of operating experience

- Technical support through the development of guidelines and sharing good practices. The programme also covers assistance for professional and technical development. This is achieved through workshops, conferences, seminars and expert meetings.

International Standards Organization (ISO)

American Society of Mechanical Engineers (ASME)

Institute of Electrical and Electronics Engineers (IEEE)

International Electrotechnical Commission (IEC)

Institutions of the European Union

ENSREG

European Nuclear Safety Regulators Group (formerly known as: European High-Level Group on Nuclear Safety and Waste Management) Council Directive 2009/71/Euratom of 25 June 2009 establishing a community framework for the nuclear safety of nuclear installations defines the basic obligations and general principles of safety for nuclear installations in the EU while enhancing the role of national regulatory bodies. ENSREG will become the focal point of cooperation between regulators and contribute to the continuous improvement of nuclear safety requirements, especially with respect to new reactors.

ENEF

European Nuclear Energy Forum is a platform to promote broad discussion between relevant stakeholders on the opportunities and risks of nuclear energy.

International Bodies

Several international expert bodies issue authoritative findings and recommendations on safety related topics. The advice provided by these bodies is an important input to the development of the IAEA Safety Standards and other international standards; it is frequently incorporated in national safety-related laws and regulations. Some of these bodies are listed below:

UNSCEAR – UN Scientific Committee on the Effects of Atomic Radiation

ICRP – International Commission on Radiological Protection

ICRU – International Commission on Radiation Units and Measurements

The IAEA offers several **review and appraisal** services to Member States.

Integrated Regulatory Review Service

The IAEA's Integrated Regulatory Review Service (IRRS) was established to advise member states on methods of strengthening and enhancing the effectiveness of national regulatory frameworks for nuclear, radiation, radioactive waste and transport safety while recognizing the ultimate responsibility of each state to ensure safety in these areas. The IRRS process sets out to accomplish this purpose by enabling structured peer review of national regulatory technical and policy approaches against IAEA safety standards and the sharing of relevant good practices.

In fact, the IRRS combines the common regulatory infrastructure elements of the various safety review services offered by the IAEA, resulting in a cross-cutting regulatory review of all facilities and activities which make use of radiation technologies in the receiving state.

The main features of the service are as follows:

- Integrated approach – review of legal and governmental framework and regulatory infrastructure
- Comparisons against IAEA safety standards, and where appropriate, good practices elsewhere
- Peer review in any state regardless of the level of development of its activities and practices involving ionizing radiation or nuclear programmes
- Three stages: pre-mission or preparatory stage (including information meetings, preparatory meetings and self-assessment)
- A peer review mission for two weeks (including completion and dissemination of mission report)
- A follow-up mission

Operational Safety Review Team

The IAEA's OSART programme assists member states in strengthening the safety of their nuclear power plants during commissioning and operation, comparing actual practices with IAEA safety standards.

The OSART review is based on the documents which describe the plant and its structures, systems and components; organization, training and qualification of plant personnel; written procedures applicable to the operation of the plant; interviews and discussions with plant personnel; observations of plant material conditions, operating practices and work in the field; and the records and reports of its operating history. The review focuses on performance in various areas important to safety, managerial aspects of policy implementation, control of activities, verification and correction, and document control. An OSART review may also take place during a nuclear power project at the critical commissioning stage when many decisions are taken which will affect operational safety throughout the lifetime of the plant (pre-operational OSART). Some key characteristic of the OSART are as follows:

- Peer review by international experts
- Focus on identifying gaps between plant operations and the requirements outlined in the IAEA Safety Standards

- Technical focus, but also identifying safety culture and organizational issues
- Scope of the review agreed between the IAEA and the host
- OSART missions commenced at the request of a member state
- Held over three weeks – ends with the preparation of a draft report for plant management to review – approved report issued within three months (encouraged to be made publicly available) – follow up mission (18 months later) to evaluate progress.

Further services offered by the IAEA:

- Integrated review service for radioactive waste and spent fuel management, decommissioning and remediation programmes (ARTEMIS)
- International Physical Protection Advisory Service (IPPAS)
- Integrated Nuclear Infrastructure Review (INIR)
- Occupational Radiation Protection Appraisal (ORPAS)
- Emergency Preparedness Review Service (EPREV)
- Safe Long-Term Operation (SALTO)
- Education and Training Review Service (ETRES)

IAEA Publications

In addition to the Safety Standards Series, the IAEA has many other series dedicated to different areas. Some of them are listed below:

- Nuclear Energy Series
- Nuclear Security Series
- International Law Series
- Technical Reports Series
- Safety Reports Series
- Emergency Preparedness and Response
- INSAG Series
- TECDOC Series
- Services Series

Note: The text has been prepared mainly as a compilation of relevant parts of the IAEA, Regulatory control of nuclear power plants, Part A (Textbook), Training Course Series 15, Vienna (2002).

Safe Operation of Nuclear Power Plants

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Safety Objectives and Safety Criteria for Nuclear Power Plants

Safety Objectives

Establishing and maintaining safety is the main purpose for establishing an adequate framework of surveillance and control of all activities associated with nuclear installations. For the sake of clarity for all parties involved, it is therefore imperative to provide the structure under which they can or are forced to act. The essential part of this structure is a coherent set of safety objectives which indicates what must be attained but does not impose or prescribe any method to obtain it. The essence of the IAEA requirements on nuclear safety published in the nuclear safety standards documents was formulated according to three overall safety objectives. These three overall safety objectives are as follows:

General nuclear safety objective

To protect individuals, society and the environment from harm by establishing and maintaining effective defences against radiological hazards in nuclear installations.

Radiation protection objective

To ensure that in all operational states, radiation exposure within the installation or due to any planned release of radioactive material from the installation is kept below the prescribed nine limits and as low as reasonably achievable, and to ensure mitigation of the radiological consequences of any accidents.

Technical safety objective

To take all reasonably practicable measures to prevent accidents in nuclear installations and to mitigate their consequences should they occur; to ensure with a high level of confidence that, for all possible accidents taken into account in the design of the installation, including those of very low probability, any radiological consequences would be minor and below prescribed limits; and to ensure that the likelihood of accidents with serious radiological consequences is extremely low.

All other principles and criteria relevant to nuclear safety and radiation protection are derived from these three overall safety objectives. In its report “Basic Safety Principles for Nuclear Power Plants”, the International Nuclear Safety Advisory Group (INSAG) formulated a number of these derived principles and proposed one possible way of presenting them graphically in a hierarchical presentation, and since the objectives are not independent of each other, showing their interrelationship. As they are the immediate sources of corresponding safety criteria, they will be considered together with such criteria.

Basic Safety Principles

It is useful to see the type of safety principles presented for the operation of nuclear power plants in the safety fundamentals document. A summary of the basic safety principles follows. These principles should form the basis for national safety criteria. The following is an extract of the Safety Fundamentals, presenting the safety principles for the operation of nuclear power plants:

Management of Safety

- Organizations engaged in activities important to safety shall establish policies which give safety matters the highest priority and ensure that these policies are applied under a management structure with clear divisions of responsibility and clear lines of communication.

- Organizations engaged in activities important to safety shall establish and apply appropriate quality assurance programmes which extend throughout the lifetime of the installation, from the siting and design stages through to decommissioning.

- Organizations engaged in activities important to safety shall ensure that sufficient numbers of adequately trained and authorized staff work in accordance with approved and validated procedures.

- The capabilities and limitations of human performance shall be taken into account at all stages in the lifetime of the installation.

- Emergency plans for accident situations shall be prepared and appropriately exercised by all organizations concerned. The capability to execute emergency plans shall be in place before an installation commences operation.

Siting

- Site selection shall take into account relevant features which might affect the safety of the installation, or be affected by the installation, and the feasibility of executing emergency plans. All aspects shall be evaluated for the projected lifetime of the installation and re-evaluated as necessary to ensure the continued acceptable levels of safety in site related factors.

Design and Construction

- The design shall ensure that the nuclear installation is suited for reliable, stable and easily manageable operation. The primary aim is the prevention of accidents.
- The design shall include the appropriate application of the defence in depth principle so that several levels of protection and multiple barriers are in effect to prevent the release of radioactive materials and to ensure that failures or combinations of failures which might lead to significant radiological consequences are of very low probability.
- Technologies incorporated in any design shall be proven or qualified by experience or testing or both.
- The systematic consideration of the man-machine interface and human factors shall be included in all stages of design and in the associated development of operational requirements.

- The exposure to radiation of site personnel and releases of radioactive materials to the environment shall be made by design as low as reasonably achievable. Comprehensive safety assessment and independent verification shall be conducted to confirm that the design of the installation will fulfil the safety objectives and requirements before the operating organization completes its submission to the regulatory body.

Commissioning

- Specific approval by the regulatory body shall be required before commencement of normal operation on the basis of an appropriate safety analysis and a commissioning programme. The commissioning programme shall provide evidence that the installation as constructed is consistent with design and safety requirements. Operating procedures shall be validated to the extent practicable as part of the commissioning programme, with the participation of the future operating staff.

Operation and Maintenance

- A set of operational limits and conditions derived from the safety analysis, tests and subsequent operational experience shall be defined to identify safe limits for operation. The safety analysis, operating limits and procedures shall be revised as necessary if the installation is modified.

- Operation, inspection, testing and maintenance and supporting functions shall be conducted by sufficient numbers of adequately trained and authorized personnel in accordance with approved procedures.

- Engineering and technical support with competence in all disciplines important for safety shall be available throughout the lifetime of the installation.

- The operating organization shall establish documented and approved procedures as a basis for operator response to anticipated operational occurrences and accidents.

- The operating organization shall report incidents significant to safety to the regulatory body. The operating organization and the regulatory body shall establish complementary analysis programmes to ensure that lessons are learned from operating experience and acted upon. Such experience shall be shared with relevant national and international bodies.

Radioactive waste management and decommissioning

- The generation of radioactive waste, in both activity and volume, shall be kept to the minimum practicable by appropriate design measures and operating practices. Waste treatment and interim storage shall be strictly controlled in a manner consistent with the requirements for safe final disposal.

- The design of an installation and the decommissioning programme shall take into account the need to limit exposure during decommissioning to as low as is reasonably achievable. Before the initiation of decommissioning activities, the decommissioning programme shall be approved by the regulatory body.

Verification of Safety

- The operating organization shall verify by analysis, surveillance, testing and inspection that the physical state of the installation and its operation continue in accordance with operational limits and conditions, safety requirements and the safety analysis.

- Systematic safety reassessments of the installation in accordance with the regulatory requirements shall be performed throughout its operational lifetime, with account taken of operating experience and significant new safety information from all relevant sources.

Safety Criteria for Nuclear Power Plants

Safety criteria are a means of assisting the implementation of safety principles and requirements. Safety criteria indicate the method (or methods) of satisfying a principle or a requirement. The nature of the safety criteria may be technical, administrative, organizational, etc., and can be qualitative or quantitative. It may be relevant to engineering, radiological protection, the man machine-interface (human factors), or physical protection, etc.

Safety criteria may be established either by the regulatory body or the applicant/ licensee:

- In the non-prescriptive approach, the applicant/licensee proposes a set of safety criteria by defining them and using them in its application; these safety criteria are eventually approved, modified or rejected by the regulatory body after review and assessment;

- In the prescriptive approach, safety criteria are established by the regulatory body; they can be established as regulations (they are then mandatory) or as guidelines (they indicate, in this case, how the regulatory body intends to conduct the review and assessment process); they must be available early enough to be considered by the applicant/licensee and its suppliers in preparing the application.

The regulatory body is responsible for ensuring that an adequate and complete set of safety criteria is available, and that each applicable criterion is or will be satisfied. Safety criteria are necessary for, and applied during, each stage of the licensing process, namely: siting, design, construction, operation, and decommissioning, as appropriate. Safety criteria should not only be compatible with internationally agreed basic safety objectives but should also express how to implement them and their supporting fundamental safety principles.

A systematic approach in establishing a coherent set of safety criteria may be to consider all the fundamental safety principles stated in the safety fundamentals presented above or the derived principles presented by INSAG.

Another approach may be based on the set of safety criteria in force in the country of origin of the reactor and a corresponding check against the above-mentioned safety principles. Each principle, or respectively, each requirement is the source of at least one criterion, but mainly of several complementary safety criteria, usually to be considered at the different stages of the licensing process (siting, design, construction, commissioning, operation, decommissioning).

Examples of Safety Criteria

The siting and design requirements are presented by the IAEA in its requirements documents on siting and design. The most well-known national example of safety criteria is given by the US NRC in the Code of Federal Regulation (CFR), in particular in title 10 “Atomic Energy”, Part 50 “Licensing of Production and Utilisation Facilities” and its Appendix A “General Design Criteria for Nuclear Power Plants” (64 criteria).

3 S Concept

The objective of the state regulatory authority is to ensure that the use of nuclear energy is applied in compliance with nuclear safety, security and safeguards. While nuclear safety measures aim to ensure normal safe operations, a low probability of accident and effective emergency preparedness, nuclear security and safeguards approach the joint fundamental objective from another angle by combatting unlawful and other intentional unauthorized acts. These objectives apply not only to operating a power plant but also to planning, designing, constructing and commissioning new nuclear installations and nuclear waste facilities and decommissioning old facilities. The coordination of safety, security, safeguards and their interfaces, synergies, and conflicts is essential in achieving the objectives.

While the synergy of 3S remains a challenge, safety and security interfaces are being actively solved.

The IAEA Safety Fundamentals, SF-1, stipulates that safety measures and security measures have in common the aim of protecting human life and health and the environment. The safety principles concern the security of facilities and activities to the extent that they apply to measures which contribute to both safety and security, such as:

- Appropriate provisions in the design and construction of nuclear installations and other facilities;
- Controls on access to nuclear installations and other facilities to prevent the loss of or unauthorized removal, possession, transfer and use of radioactive material;
- Arrangements for mitigating the consequences of accidents and failures, which also facilitate measures for dealing with breaches in security which give rise to radiation risks;
- Measures for the security of the management of radioactive sources and radioactive material.

Safety measures and security measures must be designed and implemented in an integrated manner so that security measures do not compromise safety and safety measures do not compromise security.

The requirements for the interface between safety and security are explicitly included for nuclear power plants (NPP) in the “IAEA Specific Safety Requirement for Design, SSR-2/1(Rev.1)” and “Specific Safety Requirement for Commissioning and Operation, SSR-2/2(Rev.1)”. The “IAEA GSR Part 2, Leadership and Management for Safety” requires that the management system shall take into account the safety and security interfaces without compromising safety requirements and security guidelines.

Similarly, the "IAEA Nuclear Security Fundamentals" stipulates that nuclear security and nuclear safety have in common the aim of protecting persons, property, society and the environment. Security measures and safety measures must be designed and applied in an integrated manner so that security measures do not compromise safety and safety measures do not compromise security.

Security Fundamentals also recommend that site selection and design should take physical protection into account as early as possible and also address the interface between physical protection and safety to avoid any conflict and to ensure that all three elements support each other. Sabotage targets should include safety related equipment and devices based on safety analysis. Nuclear security systems and measures should take advantage of safety provisions and procedures.

Stages in the lifetime of a nuclear installation

- Siting and site evaluation
- Design
- Construction
- Commissioning
- Operation
- Decommissioning
- Release from regulatory control

Specific Safety Requirements: Design of Nuclear Power Plants (SSR – 2.1)

Requirement 21: Physical separation and independence of safety systems

Interference between safety systems or between redundant elements of a system shall be prevented by means such as physical separation, electrical isolation, functional independence and independence of communication (data transfer), as appropriate.

Requirement 24: Common cause failures.

The design of equipment shall take due account of the potential for common cause failures of items important to safety, to determine how the concepts of diversity, redundancy, physical separation and functional independence should be applied to achieve the necessary reliability.

Diversity

The presence of two or more independent (redundant) systems or components to perform an identified function, where the different systems or components have different attributes to reduce the possibility of common cause failure, including common mode failure.

Examples of such attributes are: different operating conditions, different working principles or different design teams (providing functional diversity), different sizes of equipment, different manufacturers, and types of equipment (providing equipment diversity) which use different physical methods (providing physical diversity).

Functional diversity: The application of diversity at the level of functions in process engineering applications (e.g., for the actuation of a trigger on both a pressure limit and a temperature limit).

Redundancy

The provision of alternative (identical or diverse) structures, systems and components so that any single structure, system or component can perform the required function regardless of the state of operation or failure of any other.

Physical separation

Separation by geometry (distance, orientation, etc.) through appropriate barriers or a combination of factors.

INSAG-10 Defence in Depth in Nuclear Safety

INSAG-10 presented a detailed description of the concept of defence in depth, including a table with objectives and the essential means of each level of defence.

Table 1 Characteristics of levels of defence in depth

Level of defence	Objective	Essential means
Level1	Prevention of abnormal operation and failures	Conservative design and high quality in construction and operation
Level 2	Control of abnormal operation and detection of failures	Control, limiting and protection systems and other surveillance features
Level 3	Control of accidents within the design basis	Engineered safety features and accident procedures
Level 4	Control of severe plant conditions, including prevention of accident progression and mitigation of the consequences of severe accidents	Complementary measures and accident management
Level 5	Mitigation of radiological consequences of significant releases of radioactive materials	Off-site emergency response

Convention on nuclear safety

Introduction

Before the adoption of the Convention on Nuclear Safety (CNS) , the control and regulation of nuclear energy for peaceful purposes was governed almost exclusively by the domestic national laws of states using nuclear technology. An important result of the Convention was to bring the subject of nuclear safety within the ambit of international law for the first time. When a state adheres to an international treaty or convention such as the CNS, this action has both internal and external legal consequences. Adopting an international instrument requires a state to conform its internal laws and regulations to the terms of that instrument. However, by adopting the instrument, a state also incurs obligations to all other states which are party to the instrument. This means that a state's activities regarding nuclear safety are properly subject to review and assessment by other states through the processes and procedures contained in the CNS. Under this legal regime, states now have a right (indeed, an obligation) to make judgements about how other states are conducting their nuclear safety activities, and whether they are complying with their obligations under the convention.

Historical and Political Background

Origins of the Convention on Nuclear Safety

As stated, from the beginning of the nuclear age, regulation of the safety of nuclear facilities was deemed a matter of strictly national jurisdiction. However, the major reactor accident at Chernobyl in the USSR (now Ukraine) in 1986 fundamentally changed the thinking of both the public and governments on this approach. Because of the transboundary impact of the accident, many governments urged that an international legal instrument be adopted to codify basic measures which states should follow to ensure an appropriate level of safety at their nuclear installations. Immediately following the accident, a number of member states of the IAEA called for the negotiation of a nuclear safety convention. However, at that time, there was insufficient political will to go forward, and the initiative languished for several years.

Negotiation of the CNS

In September 1991, the General Conference of the IAEA adopted a resolution requesting the Director General to establish an informal open-ended working group to develop the text of a safety convention. The terms “informal” and “open-ended” meant that the convention text would be developed by a body comprised of safety experts rather than governmental representatives with firm political instructions, and that the body would be open to all interested IAEA member states. The work of the expert group was not a formal diplomatic negotiation, but an extended technical and legal process conducted in some nine meetings over a three-year period. This approach permitted consultations on the text to be quite flexible; less shaped by political considerations than the technical and managerial principles of good practice on nuclear safety.

The working document for the CNS was the IAEA Safety Fundamentals document and reflected a consensus of technical experts over the previous years. The fundamental task of the working group was to convert the principles in this non-binding guidance document into provisions which states would be willing to accept as binding under international nuclear law. This process obviously involved many compromises and reformulations. For this reason, the CNS text differs in some respects from the underlining safety fundamentals documents.

After the open- ended working group produced a basic text, a more formal phase of the negotiations was needed to transform the informal document into an instrument which could be codified into international law. In June 1994, a diplomatic conference was convened to enable accredited government representatives to produce such an instrument. The month-long diplomatic conference considered a wide range of controversial issues and was able to adopt a consensus text. The Convention was opened for signature by states at the September 1994 IAEA General Conference. However, even after acquiring a number of signatures, a convention is not legally effective until the required number of states have completed their domestic procedures to formally approve it. By 1996 the required number of countries (in this case, 27) had formally completed their internal reviews and expressed approval of the text. Thus, the CNS entered into force as binding on its parties in October 1996. Some countries (including the United States of America) delayed approval because of complex internal procedures or policy reasons. The CNS has now been adopted by substantially all countries operating nuclear power reactors and several that do not. There is only one country which has a nuclear power installation and is not a CNS Party.

Basic Character of the Convention

The basic character of the Convention is an important issue. International instruments come in different forms, and the CNS could have been much different in its fundamental approach to improving nuclear safety worldwide.

One type of instrument could be characterized as a "Regulatory Convention". Such a convention would have established reasonably concrete rules for states which would be subject to supervisory measures implemented by an international secretariat. An example of such an instrument is the Nuclear Non-Proliferation Treaty (NPT). It establishes an obligation for a State party to accept the application of IAEA safeguards to certain nuclear activities under its jurisdiction. The IAEA has also established and maintains a professional Department of Safeguards to conduct inspections and other procedures in individual countries. During negotiation of the CNS, it was clear that few countries wanted a regulatory convention for nuclear safety. They were willing to accept a number of obligations under international law but unwilling to have those obligations monitored or enforced by an international regulatory body. The IAEA role in CNS implementation is, thus, quite limited—unlike the NPT. The IAEA has promulgated important safety guidance documents which assist in the application of the Convention's substantive obligations. The IAEA also conducts safety missions at the request of its member states to help demonstrate compliance with a nation's obligations under the CNS. However, these missions are not inspections, and their results do not amount to a regulatory system.

A second type of instrument under international law could be called a "Sanctions Convention". Such conventions or instruments establish clear obligations which, if violated, can lead to stringent penalties or enforcement measures by other parties. Many such instruments cover commercial or trade relationships, where violations can result in financial penalties or the withdrawal of economic benefits. During negotiations of the CNS, it became clear that involved experts and delegations were not interested in a sanctions regime where a state's parties would be subject to specific penalties for lack of compliance.

The rejection of regulatory and sanctions approaches led the negotiators to focus on a third alternative. For lack of a better term, this came to be known as an "Incentive Convention". An Incentive Convention is an instrument which contains a set of international obligations and an implementation process which produces political pressure on a state to comply with its obligations conscientiously and rigorously. In the case of the CNS, implementation is grounded in a so-called "peer review process" in which states prepare national reports demonstrating their compliance with the CNS and other countries are given the opportunity to review and comment on those reports at periodic meetings of the parties. This peer review process was judged most likely to encourage conscientious application of the CNS, without the disadvantages of a regulatory or sanctions approach.

Vienna declaration on nuclear safety

A diplomatic conference was convened in Vienna, Austria, on 9 February 2015, to consider a proposal by Switzerland to amend the CNS. During the conference, contracting parties unanimously adopted the Vienna Declaration on Nuclear Safety (the "Vienna Declaration"), which includes principles for the implementation of the objective of the Convention to prevent accidents with radiological consequences and mitigate such consequences if they occur.

The contracting parties decided that the principles contained in the Vienna Declaration should be reflected in their actions, in particular during the preparation of their national reports, starting with the National Reports for the 7th Review Meeting. Furthermore, the contracting parties committed to ensuring that the safety objectives set out in the Vienna Declaration formed an integral part of considerations during future review meetings and would be used as a reference to aid strengthening the peer review process of the CNS.

Note: The text has been prepared mainly as a compilation of relevant parts of the IAEA, Regulatory control of nuclear power plants, Part A (Textbook), Training Course Series 15, Vienna (2002).

Severe accidents, phenomenology and source term evaluation

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Introduction to severe accident phenomenology, fission product release and transport. Methods for source term evaluation.

Introduction

Controlled fission chain reactions were used for the first time during the second world war for military purposes. In this initial period, the main safety challenge was to effectively control the chain reaction. The war was followed by the Cold War for most of the second half of last century. As a symbol of technological superiority, the peaceful use of nuclear power was accelerated through political will. Technology industrialized on a large scale was selected in haste from the options which were available. However, at a time of cheap coal and oil, there was no genuine economic need for nuclear power [1].

LWR technology was the predominant winner in the initial selection as a civil nuclear power source, mainly as a result of previous successful military application [1, 2]. Relative simplicity and chain reaction control through negative feedback were significant advantages. However, (large) LWR has a drawback. Due to periodic batch core reloading, a large amount of fission products are accumulated in the fuel. After reactor shutdown, the reactor core must be adequately cooled by an active cooling system to remove residual heat. A loss of cooling can lead to core meltdown and the release of fission products from the fuel, i.e., a severe accident.

Murphy's Law states: *"Anything that can go wrong will go wrong"*.

Although simple in principle, LWR is complicated technology and many factors can fail. The question is not whether the next severe accident will occur, but when. Consider core damage frequency 10^{-5} / reactor-year and the number of operational reactors in the world. Initially, it was thought that the LWR era would not last long and that safer and more efficient technology would be created to replace it.

Containment buildings were devised to prevent fission product release into the environment during core meltdown. Containment structures should be strong and resilient enough to provide leak tightness under severe accident conditions for a sufficient time. The minimum time requirement for containment leak tightness has at least two objectives: to allow emergency actions in the vicinity of the damaged reactor and to limit the deposition of airborne contamination inside the containment.

At this point, we can define the task of source term evaluation. For postulated initiating events and additional failures, we simulate core heat-up after loss of cooling. We then evaluate the core degradation process, fission product release from the fuel, transport in the cooling circuit and escape to the containment.

The next step is evaluation of the behaviour of fission products released into the containment atmosphere. Deposition due to both natural phenomena and engineering safety functions should be considered. This result is sometimes called in-containment source term, or source term for historical reasons. This term represents the amount of fission products which would be released in the event of containment failure. The terminology is somewhat ambiguous here.

To calculate release into the environment, we continue with simultaneous evaluation of the containment state. Containment leak tightness and integrity are challenged by severe accident conditions. Assuming the timing and scale of the containment damage or complete failure, we estimate the media flow from the containment volume and the amount of radiation carried with it. The output of this analysis is the amount and timing of radiation release into the environment, i.e., the source term.

Even from this brief introduction, one may guess that some of the mentioned processes are stochastic in nature. Nevertheless, a deterministic evaluation approach is currently considered the state-of-the-art in source term evaluation. Note also that simulated reactor accident conditions are "postulated" as known boundary conditions. Prediction of the source term in a real emergency situation is another matter.

Accident Progression

Core Heat Up

Two typical core degradation scenarios are usually distinguished:

- Wet core – characterized by water boil-off from the core in which hot fuel rods in the uncovered upper part of the core are exposed to the steam produced by the coolant boiling in the lower part, which causes cladding oxidation:

  $Zr + 2H_2O \rightarrow ZrO_2 + 2H_2 + \text{Heat}$

  The reaction rate is usually limited by the availability of steam ("steam starvation" conditions).

- Dry core – typical for BWR short-term SBO with cooling circuit depressurization by the operator. After depressurization, only small amount of water remains in the reactor pressure vessel during core degradation. It means low steam production and much lower Zr oxidation than in the wet core scenario.

Core Melt and Relocation

Control rod material produces the first melt due to low melting temperature (i.e., Ag-In-Cd or B_4C+steel eutectic). It is followed by molten metallic Zr and some UO_2 liquefied from reaction with molten Zr. These metallic melts move downwards. Melting (or better said, liquefaction at temperatures much lower than the melting point) of the remaining UO_2 follows. The resulting mixture—corium—has a significantly lower liquidus temperature than the UO_2+ZrO_2 eutectic.

In the wet core scenario, the relocating metallic melt freezes above the water level in the core. It can create a crust which supports a ceramic molten corium pool (TMI-2).

With a decrease of water level, this structure may steadily progress down to the core support plate. When the crust/core support plate ruptures, a large amount of ceramic melt may quickly relocate downwards to the lower head.

In the dry core scenario, the core degradation process is expected to be more global and the melt flows directly to the lower head without crusting.

Lower Head Failure

Corium which relocates to the lower head would be partially quenched by water remaining in the lower head (wet scenario). After a significant amount of corium relocates to the lower head, it is assumed that failure of the lower head cannot be prevented by in-vessel coolant injection. Consider that at TMI-2, the vessel is already full of coolant at the time of crust failure and corium relocation to the lower head and the relocated mass is limited to 20-30 t. Nevertheless, the lower head is on the verge of failure.

The failure of the vessel lower head is a significant event. At this stage, the overall damage state to the plant deteriorates qualitatively. A strategy was devised (IVR) to prevent lower head failure by flooding the reactor cavity and cooling the vessel wall from outside. IVR has been already implemented as a design extension feature for several types of LWR.

Molten Core Interaction with Concrete

After lower head failure, the corium can flow to the reactor cavity and attack the concrete containment basemat. At high temperature, concrete is decomposed. Decomposition products are mixed into the corium or escape as gas into the containment. This reaction (MCCI) produces a large quantity of hydrogen, and depending on the type of concrete, flammable CO and non-condensable CO_2.

It is currently considered that corium present in the cavity may not be coolable by top flooding. The issue is under study through the ongoing and extensive international experimental project OECD/ROSAU.

At least limited MCCI occurred in the damaged units of Fukushima-Daiichi. It is presumed that in each case, corium relocation occurred into the previously flooded cavity. Relocated corium froze on the structures below the vessel's lower head (specific BWR feature).

Containment Challenges

During a severe accident, containment is the ultimate final barrier against the release of fission products into the environment. Leak tightness and the integrity of the containment is challenged by:

- Static over-pressurization

  Static overpressure is caused by water evaporation due to decay heat and the production of non-condensable from MCCI. To prevent containment failure, heat removal from the containment should be restored or the containment should be depressurized by venting.

- Dynamic energetic events

  - Hydrogen/CO burn or detonation

    Hydrogen burn occurred in the containment of TMI-2 during the accident. Containment leak tightness was not compromised by the burn. Hydrogen is not challenge for the integrity of BWR containments because the atmosphere is usually inertized during normal operation. However, hydrogen detonation destroyed the reactor hall of units 1, 3 and 4 of Fukushima-Daiichi [4]. These events had significant effect on the progress of the accident. The accomplishments of previous recovery actions were compromised, and further actions were delayed.

    The risk of hydrogen on the integrity of PWR containments is usually solved by the installation of passive autocatalytic recombiners (PAR) and igniters. However, mitigation of the risk of hydrogen in surrounding rooms is uncommon.

  - Steam explosion

    A steam explosion can occur when the molten corium comes into contact with water (FCI). Detailed analyses have shown that FCI caused by a steam explosion is not able to damage an intact RPV (note that this is not the case of an RIA initiated steam explosion, which is a completely different scenario). The prevention of a steam explosion in the containment is simple: it is only necessary to keep the reactor cavity dry. However, this is in direct contradiction with SAM strategies such as IVR or ex-vessel corium retention by cavity flooding [3]. In these cases, the effects of a steam explosion on containment integrity should be evaluated.

  - Direct containment heating (DCH)

    The risk of DCH was considered for certain containment designs with an open passage from the cavity to the bulk volume of the containment. It was hypothesized that in the case of lower head failure at high vessel pressure, the melt ejected from the vessel might be dispersed to fine particles in the reactor cavity. With direct open passage to the containment, particles might be dispersed in a large containment volume. Rapid cooling of particles and coincident rapid oxidation of the remaining metallic portion in air was considered. The resulting pressure spike would destroy the containment.

    The conclusion is that vessel failure at high pressure should be excluded. Additionally, most containment designs do not have an open passage from the cavity. Nevertheless, analysis of the DCH issue has led to the development of the risk assessment methodology ROAAM, which can be modified and successfully applied to other safety problems.

- Containment basemat melt-through

  It has been already mentioned that the contact of molten corium with concrete leads to MCCI. Driven by decay heat in the corium, this interaction can continue for long time and penetrate a thick layer of concrete, both in vertical and horizontal directions (recall the (in)famous “China Syndrome”).

Fission Product Release

Release from Fuel

Fission products are continuously generated during normal reactor operation as fission fragments with high kinetic energy. Most of them dissipate their energy in the UO_2 fuel matrix and remain trapped in the fuel pellet. Depending on their chemical nature, some become fixed in oxidic or metallic inclusions (low volatiles), some remain in vacancies of the UO_2 matrix (volatiles) and are able to slowly diffuse towards porosities. Diffusion is relatively slow in normal LWR fuel operation conditions, yet a small fraction of volatiles may escape from pellets to the fuel rod gas space.

During an accident, fuel temperature increases. Usually, at the onset of rapid cladding oxidation, the cladding ruptures. Volatiles accumulated in the gas space during normal operation are then free for release from the fuel rod.

As the fuel temperature increases its morphology changes (also depending on local fuel burn-up). The interconnection of closed porosities leads to the release of volatiles from porosities. Further increases in temperature increase the diffusion of volatile atoms in the fuel grain, and as these reach the grain surface, they are released through interconnected porosities. Consider that volatiles are (practically) insoluble in the fuel matrix, hence equilibrium concentration in the matrix is zero. At the onset of fuel melt, most of the volatiles are gone. This fact is well supported by experimental evidence.

The release of low volatiles is more complicated. Depending on the chemical composition, some possess a more volatile oxide form, some a more volatile metallic form. Hence, the release rate of low volatiles depends strongly on the local oxidation state of the fuel, i.e., on x in $UO_{2\pm x}$, which depends on the local composition of gas in the vessel, i.e., on the ratio of steam and hydrogen.

Specific conditions can occur in the case of accident in the SFP. For certain designs, SFP is outside the containment. During a loss of coolant accident, overheated fuel is exposed to air. Significant release of Ru oxide and volatilization of the fuel itself can be expected.

Transport in the Cooling Circuit

Volatiles released from the fuel are presumed to be in vapour form. As they move to colder regions of the cooling circuit, they condense on aerosols. Exceptions are noble gases (Xe, Kr) and volatile species of iodine: I_2, HI, CH_3I. The fraction of volatile iodine in the cooling circuit is usually considered negligible, however certain experimental data indicate that their creation may be enhanced by the presence of a B_4C neutron absorber in the degrading core.

Aerosols transported in the cooling circuit are subject to all the usual aerosol deposition phenomena. Chemisorption on steel surfaces can occur. In the case of local temperature increase (also due to decay heat in the deposit), re-evaporation may occur. In the long term, mechanical resuspension or the leaching of deposited fission products are possible.

It should be noted that fuel particles can also be transported through the cooling circuit. It is probably linked to disruptive events during core degradation, such as collapse of fuel rods or quenching of overheated fuel. During TMI-2 decommissioning, about 300 kg of UO_2 were recovered from the main coolant pumps in form of µm sized particles.

Containment Deposition

Fission products released into the containment atmosphere as aerosols are subject to deposition processes. Natural deposition processes can be significantly enhanced by systems such as chillers, coolers or containment sprays.

Only noble gas cannot be removed from the containment atmosphere. Another peculiar species is iodine. It is important, because in the case of early release from containment, iodine isotopes can be major contributors to the committed emergency dose. The chemistry of iodine is complicated. The equilibrium vapour pressure of I_2 above water with dissolved iodine is dependent on the water's acidity. It is higher for water with higher acidity, hence, iodine dissolved in acid water has a tendency to escape into the atmosphere as volatile I_2. To prevent this, buffer agents are added to the coolant in the sump on certain plants. However, the application of these buffers should be considered with care. The precipitation of buffers can cause failure of the safety core cooling systems.

Another interesting behaviour of iodine is its interaction with organic paints. Iodine deposited on a painted surface can be released again into atmosphere in HI or CH_3 volatile form. Currently, there is no consensus on the importance of various iodine behavioural phenomena among leading research groups.

Release from Containment

The character of release from containment depends on the containment design. Consider typical PWR full pressure and a large volume containment made from pre-stressed concrete. Design leakage is usually very small, and accidental release from intact containment is almost negligible. With a static pressure increase significantly above the design pressure, concrete will begin to crack. Consider also that a severe accident is always associated with something unexpected which may already compromise containment leak tightness at the initiating event, e.g., an earthquake.

Release through containment cracks may provide the retention of aerosols along the leak path. At this point, it is important to consider the purpose of the source term analysis. For a conservative estimation, retention in cracks is usually not considered. For a realistic estimation, some estimations should be performed.

When the static overpressure exceeds the strength of the tendons, global containment follows. Quick depressurization causes the mechanical re-suspension of deposits and bulk boiling and flashing of sump water with dissolved and suspended contamination. Consider water saturated at ~5 atm, suddenly exposed to atmospheric pressure. To cooldown 1 kg of water to a new saturation temperature, we need to evaporate 0.1 kg water. A fraction of the liquid droplets will also be entrained by escaping steam.

Melt-through of the containment basemat by MCCI will surely compromise containment leak tightnes. However, for most PWR containment designs, the leak would be directed to the ground (and well below ground surface elevation). In this case, direct release to the atmosphere would not occur, but long-term ground and mainly ground water contamination would be a very undesired consequence. For certain designs, the leak path may be directed to the atmosphere at above-ground elevation. This leads to a large late release, but the event would not be as dramatic as global containment failure by overpressure.

Another design is subatmospheric containment. It relies on a pressure suppression by steam condensation. Containment design pressure is usually lower, and for some older plants (e.g., VVER-440), it also has a quite higher design leakage rate. In this case, even release from an intact "leak-tight" containment may be high. To mitigate release, containment pressure should be decreased to approximately atmosphere pressure, or better below. Filtered venting is not beneficial due to the comparable pressure loss on filters and internal overpressure. Non-filtered venting may be necessary accident management action. It was done at Fukushima-Daiichi [4].

Numerical Simulations

Deterministic Integral Calculations

Evaluation of a source term for a postulated accident is usually performed using "integral codes" such as MELCOR, MAAP, ASTEC etc. The simulated object is the entire plant: core, cooling circuit(s), containment, and safety systems. Therefore, the simulation model should be significantly simplified. A lumped control volume approach is usually used. Physical models are mostly only parametric.

The algorithm used in integral codes are currently only deterministic, i.e., everything should follow cause-consequence rules. Recently, a method of using statistical methods to develop parametric correlations was published [5]. Nevertheless, the final algorithm for the integral code is still deterministic.

With the advent of less expensive computing power, attempts have been made to evaluate the uncertainty of source term evaluation using statistical assessment of series of integral calculations with random sampling of uncertain input parameters and expert assessment [6].

Detailed Mechanistic Calculation

Detailed models are developed mainly to understand basic phenomena and to interpret experimental data. Detailed models are also used to set up correlations for use in parametric models for integral codes. Detailed models are sometimes called mechanistic, the intention being to capture the physical nature of simulated processes.

Let us examine the model for fission product release from fuel as an example. Integral codes usually apply variations of CORSOR (correlation based on the Arrhenius equation) or the Booth model (based on the diffusion of volatile fission product atoms in a spherical fuel grain) [7]. By contrast, MFPR code [8] can be taken as an example of a detailed mechanistic simulation code.

MFPR simulates the evolution of $UO_{2\pm x}$ fuel morphology during irradiation or ramp temperature increase and fuel non-stoichiometry (x) depending on the boundary gas composition. Fission product transport in the fuel takes into account the chemical properties of each element [8].

Probabilistic Safety Assessment

With deterministic analyses, it is possible to evaluate a limited number of severe accident scenarios and corresponding source terms. However, it is not possible to evaluate the risk associated with plant operation using only deterministic analyses. This is the task for probabilistic safety assessment (PSA). Three levels of PSA are traditionally distinguished:

1. Core damage frequency (CDF) evaluation

2. Containment failure and source term evaluation

3. Off-site consequence evaluation

PSA level 1 is very well established and widely used. Mainly, PSA 1 results can be used to identify plant weak points which contribute to CDF. The results of PSA 1 are also used to identify important accident scenarios which are evaluated in detail through deterministic severe accident analyses.

PSA level 2 was originally intended to evaluate containment failure frequency. Currently, it is used to evaluate the frequency of predefined source term classes. Even without PSA 3, PSA 2 itself can be used to evaluate surrogate risk quantities: LRF and LERF. Large release is understood as release exceeding the acceptance criteria for off-site long-term contamination. Large early release is understood as release which causes unacceptable off-site health consequences with such a short warning time, that it is not possible to protect the public in the vicinity of the damaged unit through emergency preparedness actions.

PSA level 3 is still not consensually accepted due to a supposed high uncertainty. Nevertheless, the methodology is well established (see, for example, [9]). New advanced plant projects are generally provided with PSA level 3. The results of PSA 3 can be used to compare risks—inevitably associated with any plant operation—with a societally acceptable risk level.

Traditionally, PSA 3 applies input source terms provided by deterministic source term analyses. Alternatively, simplified fast-running source term evaluation code is used. The availability of inexpensive computing power and development of better models allows the convergence of both approaches. This is called IDPSA or DPSA, and it represents the latest development in safety assessment methods.

Conclusions

Each of the three well-known severe nuclear power plant accidents (TMI-2, Chernobyl, Fukushima-Daiichi) caused a serious blow to the nuclear industry. Almost no environmental impact by TMI-2, the physical impossibility of the Chernobyl criticality accident for an LWR, a natural disaster of unbelievable scale: none of these arguments are heard by anti-nuclear movements.

Large LWRs, both currently in operation or under construction, still need active safety systems to remove decay heat after shutdown. With non-zero core damage frequency, the next core meltdown accident will sooner or later occur—statistics have no mercy. It will be another serious blow for nuclear industry, regardless of the source term to the environment.

It should be noted that even a limited core damage event creates a very large economic loss due to the cost of decommissioning.

Finally, one important takeaway from the TMI-2 and Fukushima-Daiichi experiences should be highlighted: alternative heat removal systems for severe accidents should preferably be designed as closed-circuit heat exchanges using a clean cooling medium. Using open cooling circuits with contaminated water generates a large amount of contaminated water which is very difficult and costly to handle.

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SAMGs and Severe Accident Mitigation

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The keynote and this accompanying text examine the origins, basic concepts, principles and implementation of severe accident (SA) management guidelines (SAMGs) for light water reactors (LWR). An example of generic (Westinghouse-developed) set of SAMGs is presented and explained. Finally, the overview is complemented with illustrative examples of the application of some of the most common SA management strategies using the results of numerical SA simulations from the Czech nuclear power plant Temelin (2x VVER-1000), performed with the integral code for SA analysis MELCOR.

Introduction

According to the defence in depth (DiD) concept, which was developed after the Chernobyl accident as a fundamental and overarching principle of nuclear safety to prevent accidents and mitigate their consequences, the progression of a severe accident (SA), formerly referred to as beyond design basis accidents (BDBA) with severe core damage, will be limited if it occurs, and its effects, including potential radiological consequences, will be mitigated by complementary and severe accident management (SAM) measures. SAs constitute level 4 of DiD (Table 1) which is the penultimate level of DiD (level 5 of DiD already deals with radioactivity releases into the environment and mitigation with an off-site emergency response). The safety objective in the case of a severe accident at level 4 of DiD is:

1. Only protective actions which are limited in (i) length of time and (ii) area of application are necessary.

2. Event sequences which would lead to (i) early radioactive release or (ii) large radioactive release are required to be “practically eliminated”.[1]

For this purpose, severe accident management guidelines (SAMG) are developed, potentially together with complementary emergency operating procedures (EOP).

An SA is characterized by severe fuel damage or core melt and is listed in design extension conditions (DEC) (Table 2). The expected frequency of occurrence of an SA is less than 10^{-6} 1/(reactor-year).

Origins of SAMG

In the 1970s, nuclear power plants (NPP) were designed to withstand design basis events (e.g., a seismic event) and design basis accidents (DBA) (e.g., a large-break loss

1 Certain conditions arising may be considered “practically eliminated” if it is physically impossible for these conditions to arise or if these conditions could be considered with a high level of confidence to be extremely unlikely to arise.

Table 1 Levels of DiD [1]

Level of defence	Objective	Essential means
Level1	Prevention of abnormal operation and failures	Conservative design and high quality in construction and operation
Level 2	Control of abnormal operation and detection of failures	Control, limiting and protection systems and other surveillance features
Level 3	Control of accidents within the design basis	Engineered safety features and accident procedures
Level 4	Control of severe plant conditions, including prevention of accident progression and mitigation of the consequences of severe accidents	Complementary measures and accident management
Level 5	Mitigation of radiological consequences of significant releases of radioactive materials	Off-site emergency response

of coolant accident (LB LOCA) or steam generator tube rupture (SGTR)). These types of event would be responded to with emergency operating procedures (EOP). A breakthrough was described in the "Rasmussen Report" (1975), which describes in detail a probabilistic approach (PR(S)A) to assess the risk of occurrence of a BDBA. Although the risk of a substantial core melt was very low, the Three Mile Island Unit 2 (TMI-2) accident (1979) alarmed the scientific world, which launched a thorough investigation of BDBAs. Although neither the reactor pressure vessel (RPV) nor the containment (CTMT) were breached during the accident, a few years later, a major SA occurred at the Chernobyl NPP (1986). Although the reactor was of a different type compared to western PWRs/BWRs (a channel-type RBMK reactor not equipped with a CTMT), it motivated the first attempts to create SAMGs.

Table 2 Nuclear plant states [1]

Operational states		Accident conditions		
			Design extension conditions (DEC)	
Normal operation (NO)	Anticipated operational occurences (AOO)	Design basis accidents (DBA)	Without significant fuel degradation	With core melt

The first NPPs to commit to implementing SAMGs were in the US, the due date being the year 1998. SAMGs were developed by different owner groups, such as Westinghouse (WOG), Combustion Engineering (CEOG), Babcock & Wilcox (B&WOG), BWRs (BWROG) and PWRs (PWROG). The SAMGs were initially based on the use of existing equipment and the technology being developed by Electric power research institute (EPRI). In Europe, the approach to the development of SAMGs was not coordinated nor limited to the exploitation of existing hardware—measures for the mitigation of severe accidents started being implemented and included filtered CTMT venting systems (FCVS), passive autocatalytic recombiners (PAR), strategies for in-vessel melt retention (IVMR) by external reactor vessel cooling (ERVC), additional diesel generators (DG), diverse and flexible mitigation (FLEX) equipment, etc).

FCVS

To prevent over-pressurization of the CTMT, an FCVS is a solution which permits relief pressure via a venting stack equipped with robust aerosol and iodine filters. Pressurization of the CTMT might come from non-condensable gases produced, for example, from molten core concrete interaction (MCCI) or water vapor production (either from the IVMR application or due to corium cooling in the CTMT with application of CTMT flooding). In principle, two options are possible for an FCVS:

1. Dry filter method (DFM, Figure 1): a solution based on two-stage filtering, consisting of metal fibre filters for aerosols and usually Ag-doped zeolite filters for the retention of gaseous iodine, both elemental and organic.

2. Wet solution (Figure 2): a venturi/water scrubber not only acts as a filtration unit for aerosols and gaseous fission products (FP) but also a heat sink. Appropriate chemistry is added to the pool scrubber to achieve basic pH and thereby avoid the formation of elemental iodine.

The usual parameters of both types of filtration systems are similar and typically present a filtration effectivity (expressed as the decontamination factor (DF)) of > 104 for aerosol particles and > 100 for iodine. A typical CTMT response to the application of an FCV strategy is depicted in Figure 3. The pressure inside a CTMT during an LB LOCA scenario rises due to water evaporation while cooling the corium in the ex-vessel phase of an SA. Opening the FCVS in two stages prevents over-pressurization in the CTMT. The code used to simulate the sequence of an SA is the integral, lumped parameter (LP)-type code MELCOR.

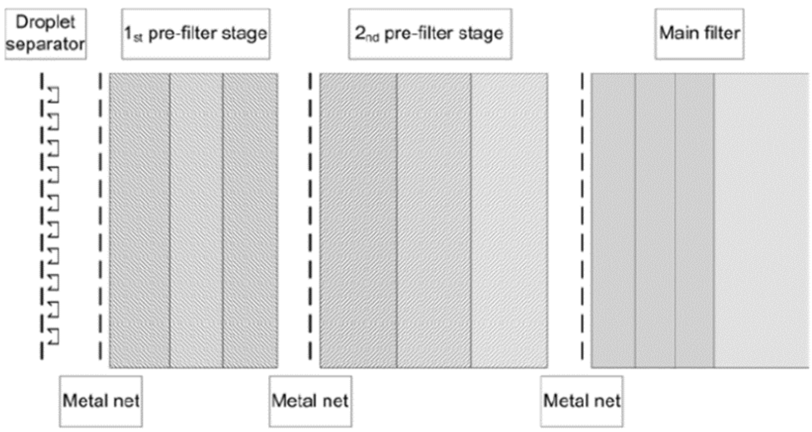


Figure 1 Dry filter unit solution [2]

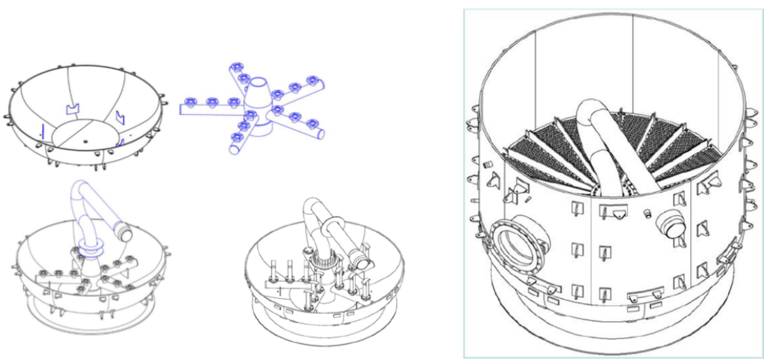


Figure 2 Wet-type filter unit employing pool scrubbing [3]

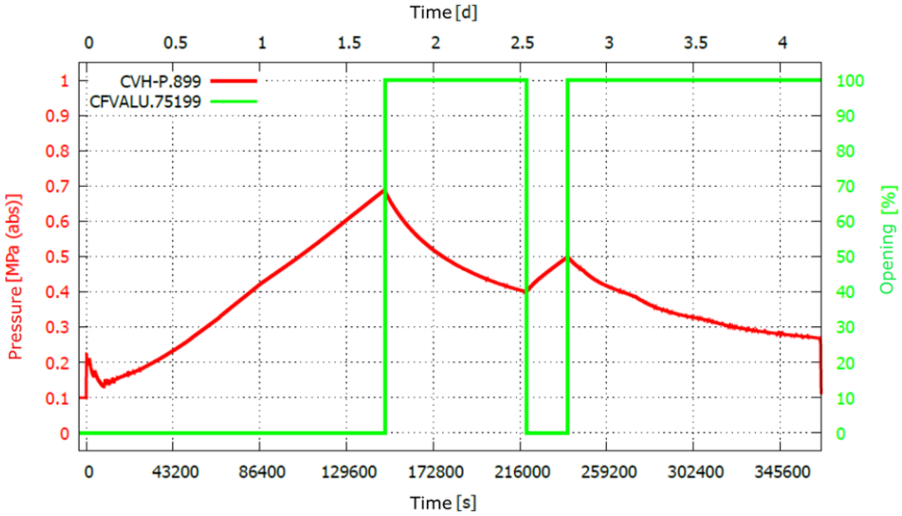


Figure 3 Pressure evolution (red curve) in a CTMT during an SA with corium top flooding. Application of filtered CTMT venting to prevent over-pressurization in the CTMT (fraction of the FCVS – green curve). MELCOR simulation of LOCA-initiated SA scenario for the VVER-1000 unit [4]

PARs

To prevent hydrogen explosions inside the CTMT, which is not filled with an inert atmosphere (e.g., nitrogen), igniters or PARs (Figure 4) installed at appropriate compartments inside a CTMT help decrease dangerous H_2 concentrations. The indicators of an air mixture's susceptibility to hydrogen deflagration or even detonation are shown in a Shapiro diagram (Figure 5); however, with decreasing O_2 concentration (with ongoing H_2 recombination or burning), its accuracy ceases to be sufficient since the atmosphere inside is no longer air composed of O_2:N_2 at a ratio of 1:4. Therefore, σ and λ criteria for evaluating the risk of flame acceleration and detonation, respectively, are used to assess the risk of occurrence of dangerous H_2 burn conditions. A typical evolution of hydrogen mass inside the CTMT is shown in Figure 6. Whereas up to several hundreds of kilograms of H_2 may be generated during the core degradation phase (more for an SBO scenario due to coolant availability, and less for a LOCA scenario due to steam starvation), tonnes of H_2 may be generated during a long-term MCCI process. Moreover, in-vessel H_2 generation is accompanied by thermal output which may be up to one order greater than instantaneous decay power. During the late ex-vessel phases of an accident, the risk of hydrogen deflagration reduces since oxygen is consumed by ongoing recombination/deflagration. However, hydrogen blasts might be an issue when the CTMT is opened several days into the accident (or even when the FCVS is applied).

Figure 4 Active, battery-powered igniter (left). PAR unit with visible instrumentation and exhaust grate (middle), and catalytic Pt-coated foils inside the PAR body (right) [6]

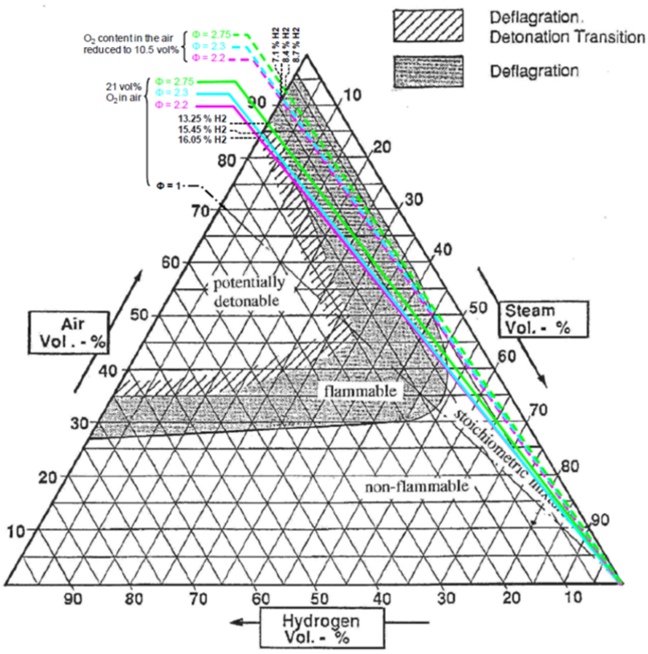


Figure 5 Shapiro (ternary) diagram with flammability and detonation areas indicated. The magenta, cyan and green lines represent regions wih unpaired PAR effectivity. [7]

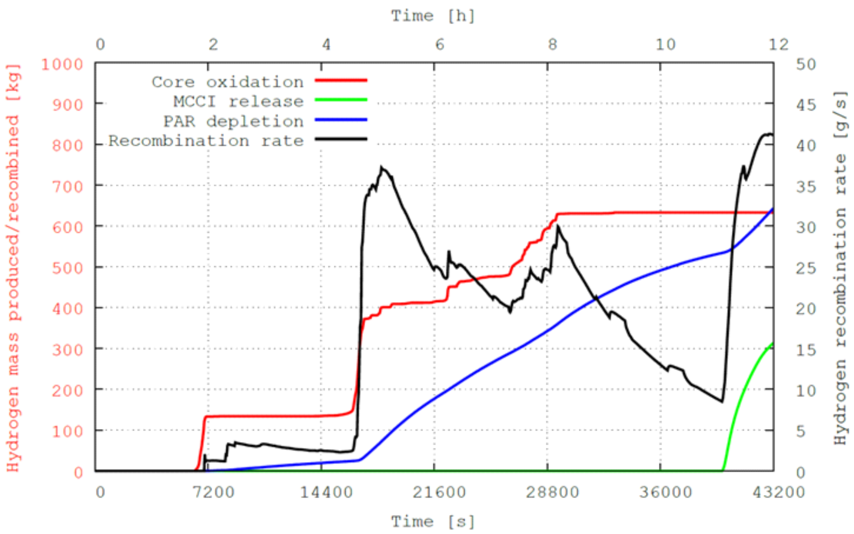


Figure 6 Left axis: H_2 production due to core material oxidation (red) and MCCI (green), H_2 depletion due to PAR unit operation (blue). Right axis: instantaneous H_2 depletion rate (black). MELCOR simulation of an SBO-initiated SA scenario for the VVER-1000 unit [8].

IVMR

One of the common strategies for mitigating an SA, especially for low thermal-power output reactors, is the IVMR strategy (Figure 7). Although it is questionable whether such a strategy is applicable to reactors of electrical power output of around 1000 MWel and greater, it has been well proven with a high degree of confidence that such a strategy would be successful for reactors of 600 MWel and less. Indeed, not only Gen-III concepts (e.g., Westinghouse AP-600) adopt the IVMR/ERVC strategy, but Gen-II reactors are also back-fitted with provisions for IVMR (VVER-440, e.g., Loviisa NPP in Finland, Paks NPP in Hungary, and Dukovany NPP in the Czech Republic). The ultimate goal in application of the IVMR strategy is to achieve a substantial margin of critical heat flux (CHF) at the outer RPV surface, i.e., not attaining a boiling crisis (Figure 8). Thermo-chemical separation of corium (i.e., molten core materials) into two or three layers (Figure 9)[2] in the lower plenum (LP) affects heat flux density and the possibility of occurrence of the so-called "focusing effect". This dangerous effect develops if a light metallic layer sits on top of the corium pool, focusing the heat flux into a narrow part of the RPV inner surface due to its enhanced thermal conductivity and thinness. Computational tools are commonly used to assess the evolution over time of heat fluxes from corium layers into the inner RPV surface. ASTEC (an integral LP code for SA analyses; concurrent to MELCOR) simulation results for the VVER-1000 reactor and an LB LOCA scenario with the application of IVMR are shown in Figure 10, Figure 11 and Figure 12. It is worth noting that the usual remaining thickness of the RPV lower head (LH) for a quasi-steady advanced phase of an SA with the application of IVMR may be as low as ~2 cm.

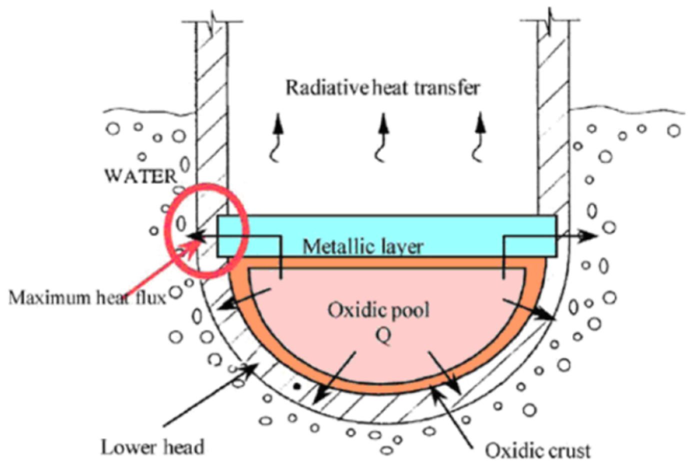


Figure 7 Concept of IVMR through ERVC [9].

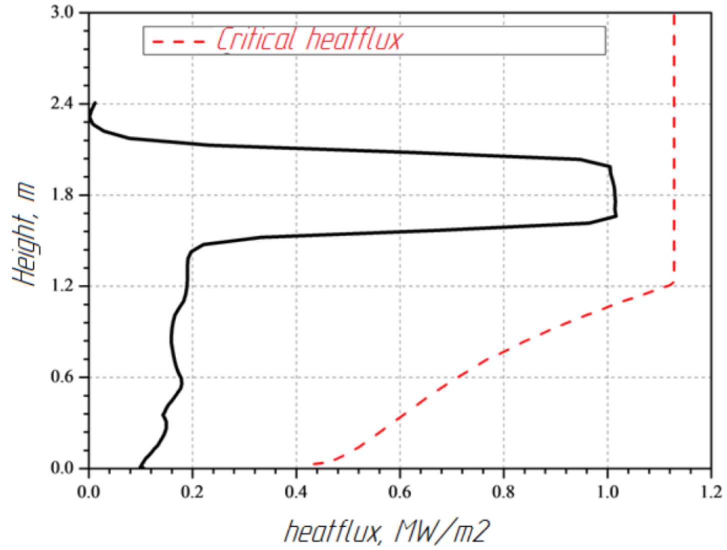


Figure 8 Heat flux density profile (black) and the theoretcal CHF (red-dashed). [10]

2 Two corium layers usually exist for "slow" accident scenarios, e.g., SBO, when enough oxidant is available for Zr oxidation. A large amount of ZrO_2 is therefore produced, while less Zr is available for UO_2 reduction (U being the main component of a heavy metallic layer). However, three corium layers develop for a rather fast accident scenario, e.g., LB LOCA, when steam starvation is responsible for less zirconium oxidation and the metallic Zr then reduces the UO_2, thus creating a heavy metallic layer, uranium being the most common constituent.

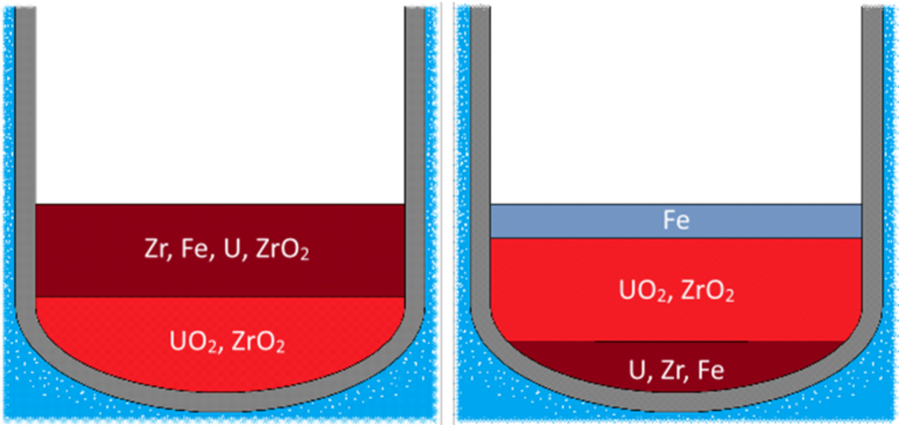


Figure 9 Two-layer (oxides and metals; left) and three-layer (heavy metals, oxides, light metals; right) molten pool configuration [10]

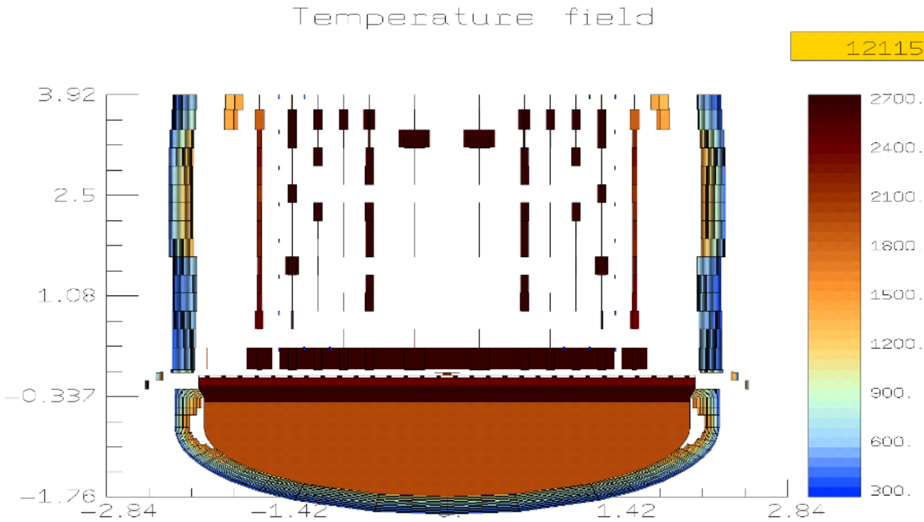


Figure 10 Three-layer configuration of corium (with solid debris floating at the top) together with the temperature field and the shape of the ablated wall at 12 115 s (the instant of maximum heat flux obtained, see below) [11]

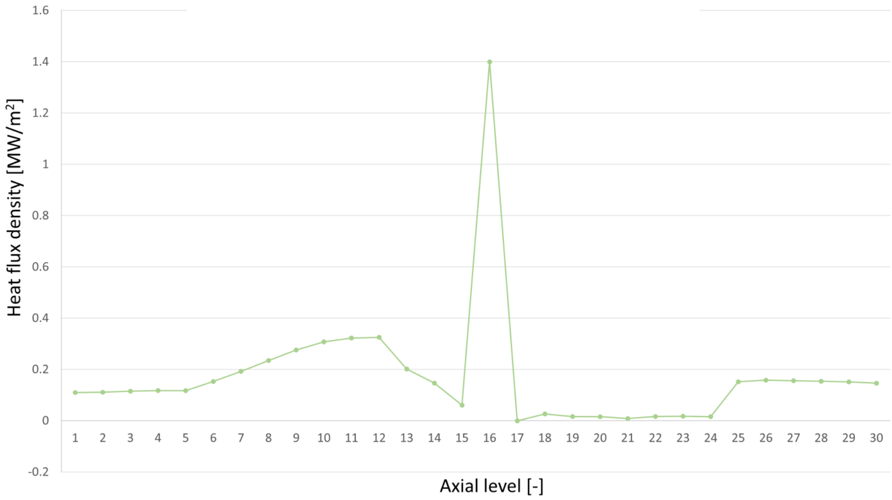


Figure 11 Heat flux density profile at the instant of its maximum attainment. The x-axis represents axial levels of LH nodalization (from bottom to top) [11]

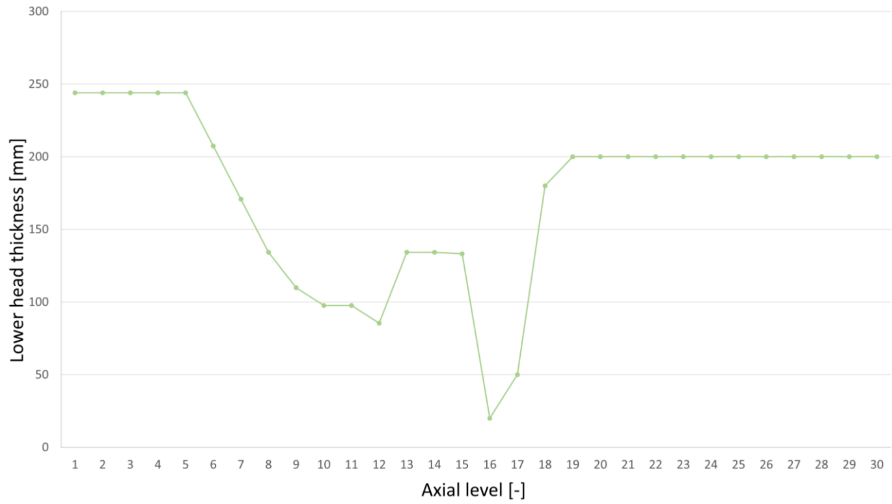


Figure 12 Long-term (after one day into the accident) LH thickness profile [11]

Example of WOG SAMGs

WOG SAMGs (or, more generally, accident management procedures and guidelines) are probably the best suited for educative/illustrative purposes of the SAMG philosophy. It has been adopted at Czech (VVER) and Slovenian (PWR) NPPs.

The following three sets of procedures/guidelines dealing with accident progression are involved:

1. Abnormal operating procedures (AOPs)

2. Emergency operating procedures (EOPs)

3. Severe accident management guideline (SAMGs)

AOPs and EOPs are applied by operators in the main control room (MCR) in the case of operational states or accident conditions involving DBAs and BDBAs without core damage. The principal means of mitigating these conditions are engineered safety features (ESF) which serve to localize, control, mitigate and terminate accidents and keep the offsite environmental exposure levels within limits. The MCR operators are supported by a technical support centre (TSC) and trained personnel who evaluate the operational and safety status of the plant. In case the EOPs are no more effective in preventing severe core damage, the transition from EOPs to SAMGs, based on precise criteria, is performed. Whereas the goal of EOPs (through either optimal recovery procedures for a known diagnosis, or through the function restoration guidelines for an unknown diagnosis) is to prevent core damage, the goal of SAMGs is to halt core degradation and lead the CTMT into a controlled stable state, terminating FP release into the environment. The main difference between EOPs and SAMGs is the nature of the actions: EOPs are rule-based, i.e., specific actions are taken for given plant conditions; SAMGs are knowledge-based and require evaluation and decision-making processes since more than one strategy may be applied to treat a situation and any action undertaken may have a positive or negative impact. Making the correct choices is the main role of the crew in the TSC. Figure 13 shows the domains where EOPs and SAMGs take place.

When the transition from EOPs to SAMGs is made, the EOPs are terminated—both may not be applied at the same time. The EOP => SAMG transition is always made from one of the following procedures:

- FR-C.1 for no reactor core cooling
- ECA-0.0 for loss of electricity supply
- FR-S.1 for anticipated transient without SCRAM (ATWS)

An indication of severe fuel damage is the core exit temperature reading, which must exceed a given value, usually 650 °C. The transition from EOPs to SAMGs first applies the severe accident control room guideline (SACRG)-1 until the TSC is operational, which means that the "evaluators" are still the staff in the MCR. Later, when TSC becomes operational, its staff becomes the evaluators and the personnel in the MCR become the "implementers". The transition of those roles is the subject of SACRG-2 (Figure 14).

Once the TSC is operational and SAMGs are applied, the evaluators begin to monitor a diagnostic flow chart (DFC). This is a diagram in which selected and crucial plant parameters are monitored. If one of those parameters exceeds a defined setpoint, an action—a severe accident guideline (SAG)—is triggered. The eight basic SAGs, together with the associated DFC parameters to be monitored, are listed in Figure 15. The evaluators must also follow a severe challenge status tree (SCST): a graphical aid which identifies dangerous challenges to CTMT integrity through monitoring, similarly to DFC, selected crucial parameters. If the setpoints of those parameters are attained, a recovery action— severe challenge guideline (SCG, Figure 16)—must be promptly undertaken. The evaluators must cease DFC monitoring and trigger appropriate SCGs instead. Indeed, a substantial difference between the SAGs and SCGs exists: as already mentioned, effectuating SAM strategies (i.e., triggering concrete SAGs) is a process which is not rigid since more strategies may lead to a desired result (terminate FP release, maintain CTMT integrity and restore core cooling); an optimum solution should be selected according to the evaluation of possible advantages and disadvantages. However, the actions of SCGs must be undertaken immediately, even without any evaluation of potential adverse outcomes (if not done promptly, CTMT integrity might be jeopardized).

The severe accident exit guidelines (SAEG) serve to properly terminate the execution of SAMGs. The SAEG-1 "TSC long-term monitoring" guides the evaluators through the process of ensuring that SAGs or SCGs may be carried out in the long term, that systems functioning before the execution of SAMGs continue their functions and that provisions, if possible, are devised to restore primary recovery functions. SAEG-2 "SAMG termination" is initiated after the plant is declared to be in a stable and controlled state. It identifies plant conditions which may prohibit recovery actions, the special needs of long-term monitoring and the conditions of radioactivity inside the CTMT.

To facilitate the decision-making process to select which SAM strategies should be triggered, the SAGs/SCGs are complemented with computational aids (CA). These are usually in the form of charts which illustrate and simplify more complex relationships of different parameters with the aim of allowing quick assessment of the situation and thus proper action to be taken.

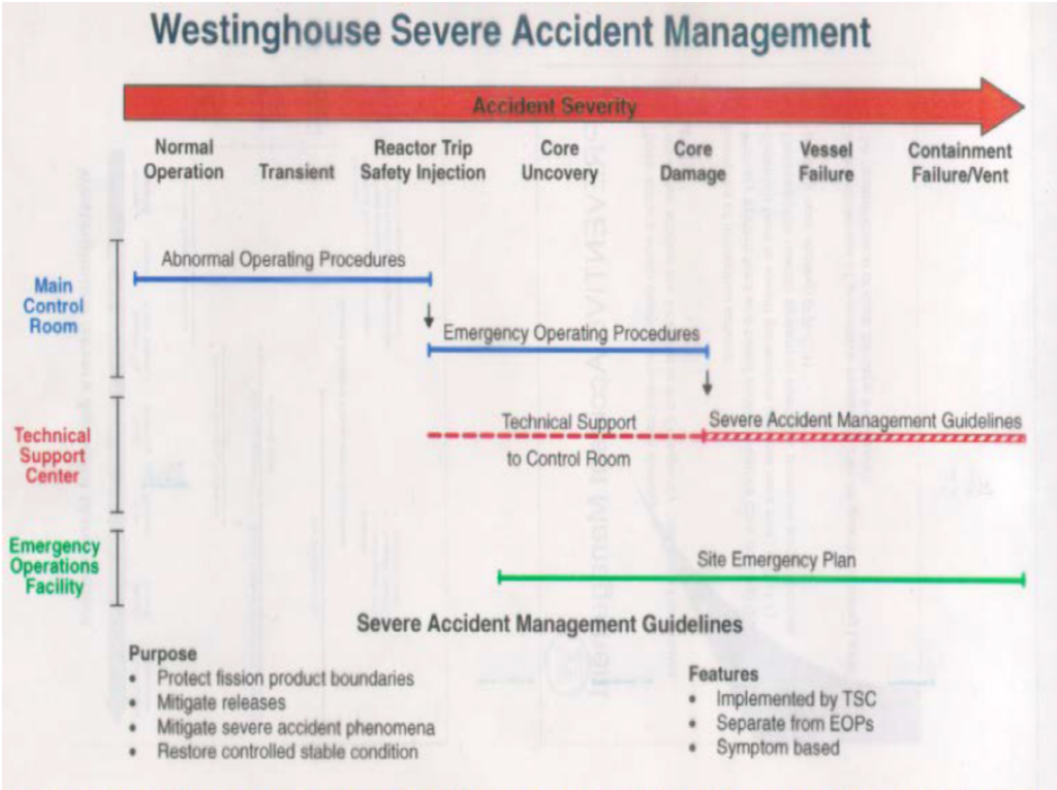


Figure 13 Accident severity correlation to the application of EOPs/SAMGs and the roles of staff [12]

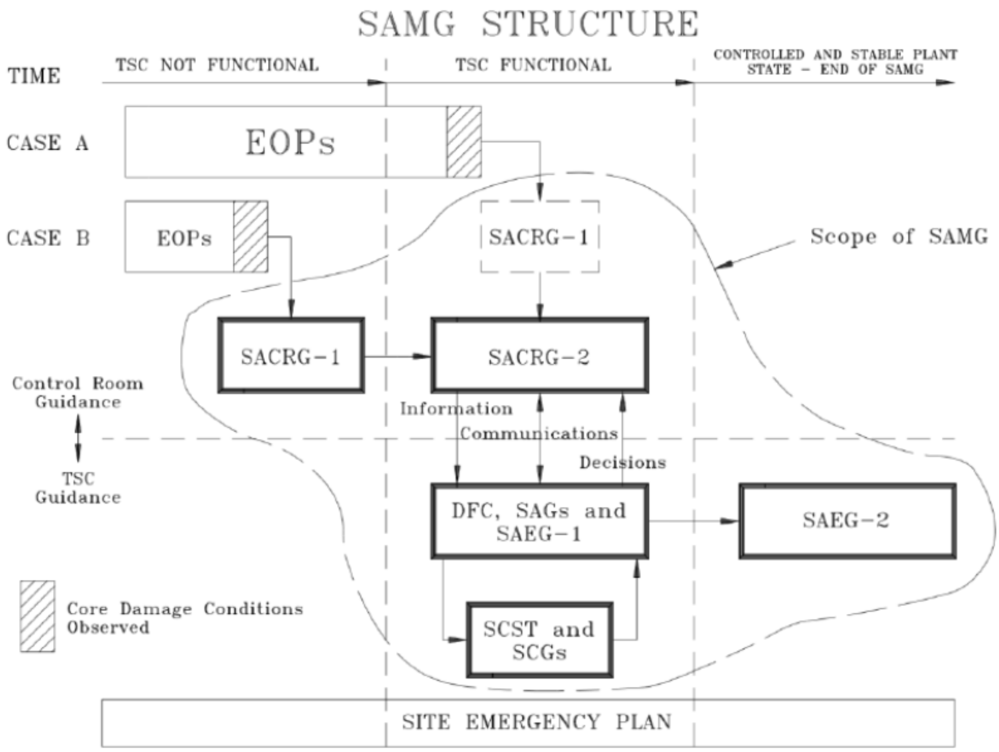


Figure 14 Schematic presentation of the transition from EOPs to SAMGs [13]

TSC guidelines and associated DFC parameter		
Guideline	Description	DFC parameter
SAG-1	Inject into steam generators	Water level in all SGs
SAG-2	Depressurize RCS	RCS pressure
SAG-3	Inject into RCS	Core temperature
SAG-4	Inject into containment	Containment water level
SAG-5	Reduce fission product releases	Site releases and SFP temperature
SAG-6	Control containment conditions	Containment pressure
SAG-7	Reduce containment hydrogen	Containment hydrogen concentration
SAG-8	Flood containment	Containment water level

Figure 15 List of SAGs and corresponding brief descriptions and monitored parameters [13]

TSC SCGs and associated SCST parameters		
Guideline	Description	SCST parameter
SCG-1	Mitigate fission product releases	Site releases and SFP level
SCG-2	Depressurize containment	Containment pressure
SCG-3	Control hydrogen flammability	Containment hydrogen below severe challenge
SCG-4	Control containment vacuum	Containment pressure

Figure 16 List of SCGs and corresponding brief descriptions and monitored parameters [13]

Examples of SAG applications

The following four subsections present the numerical results obtained from the integral LP code for SA evaluation at light water reactors (LWR) MELCOR. The plant examined in the simulations is the Czech NPP Temelin, equipped with 2 VVER-1000/320 reactors. The Temelin NPP adopted the WOG SAMGs, which are kept updated. The accident scenarios presented in the following sections are initiated with various events. Only basic results which best illustrate the effectiveness of a SAM measure effectuated are presented.

SAG-1 "Inject into the steam generators (SG)"

The positive effect of SG flooding is shown in Figure 17 for an SBO scenario. Without emergency feedwater (EFW), the residual heat produced in the core is not evacuated via SGs, hence pressure in the reactor coolant system (RCS) rises rapidly, and a relatively short time after depressurization of the primary circuit (PC), the RPV fails. However, if the EFW is operational, heat is evacuated through the SGs, which is accompanied by an initial pressure decrease. After the EFW stops injecting, the scenario is similar and consists of the following events: pressure rise, pilot-operated relief valve (PORV) cycling, PC depressurization and RPV failure. The longer the period of EFW operation, the longer the delay before the RPV melts through.

SAG-2 "Depressurize the RCS"

The positive effect of PC depressurization is demonstrated in the SGTR scenario (the break at the PC/SC interface equivalent to a diameter of 40 mm). The SA sequence is defined as high-head injection into the RCS, which is operational for a limited time, and the relief valve on the steam line, connected to the failed SG, being stuck in an open position. By depressurizing the PC via PORV after execution of SAMGs, most of the FPs are released into the CTMT (because of the backflow pattern induced in the water inventory from the failed SG into RCS) instead of bypassing the CTMT. This helps to reduce the source term into the environment (as illustrated in Figure 18).

SAG-3 "Inject into the RCS"

The proof of the effectiveness of coolant injection into the partially degraded core is the course of a TMI-2 accident, for which meltdown progression (Figure 19) is halted with the aid of high pressure injection of water (at a rate of 3.78 kg/s) into the RPV at a late phase (15.8 hrs) of progress of the accident. A similar numerical study was carried out for a hypothetical LB LOCA accident at the Temelin NPP, though with a higher flow rate of the coolant (~13 kg/s). The state of the core a few minutes after the onset of coolant injection is depicted in Figure 20, which is also the final structural state attained after several hours of continuous core cooling.

SAG-4 "Inject into containment"

One of the goals of CTMT flooding is to cool down the corium which attacks the CTMT basemat if an ex-vessel scenario takes place. Applying water to the spread corium surface assists through the phenomena of melt eruption and water ingression in evacuating a substantial part of the decay and oxidation heat from the melt. If not halting the progression of MCCI, top-flooding of the melt substantially slows down concrete ablation, as shown in Figure 21 (the factor for axial melt penetration depth into the CTMT basemat being ~3).

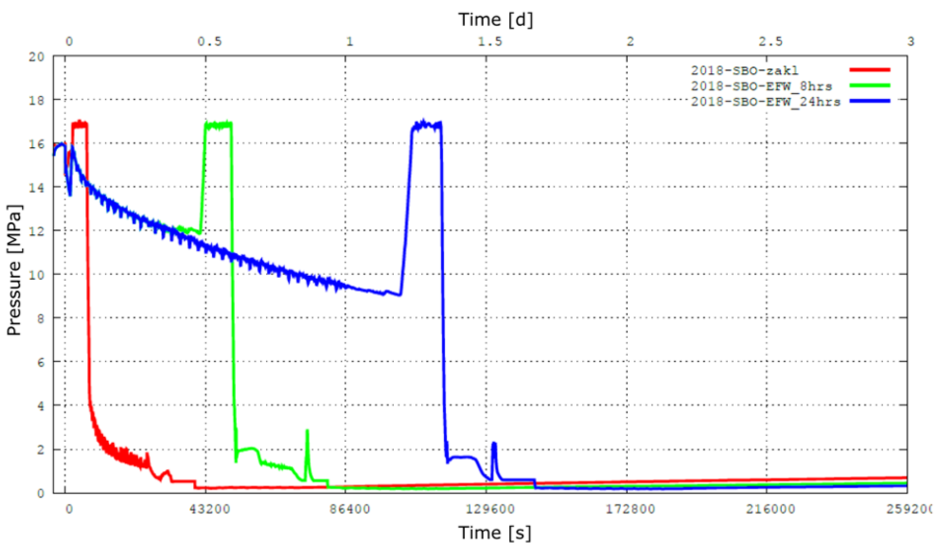


Figure 17 Primary pressure for an SBO scenario without SGs flooding (red), with emergency flooding operational for 8 hours (green) and 24 hours (blue) [14]

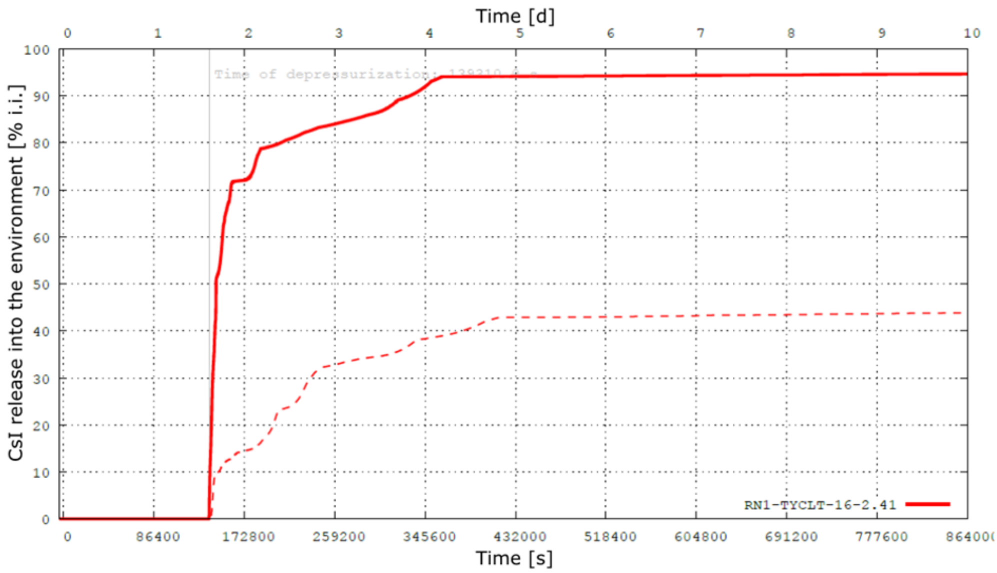


Figure 18 Release of caesium iodide (related to its initial inventory) into the environment via CTMT bypass: thick solid line – without PC depressurization, thin dashed line – with PC depressurization [15]

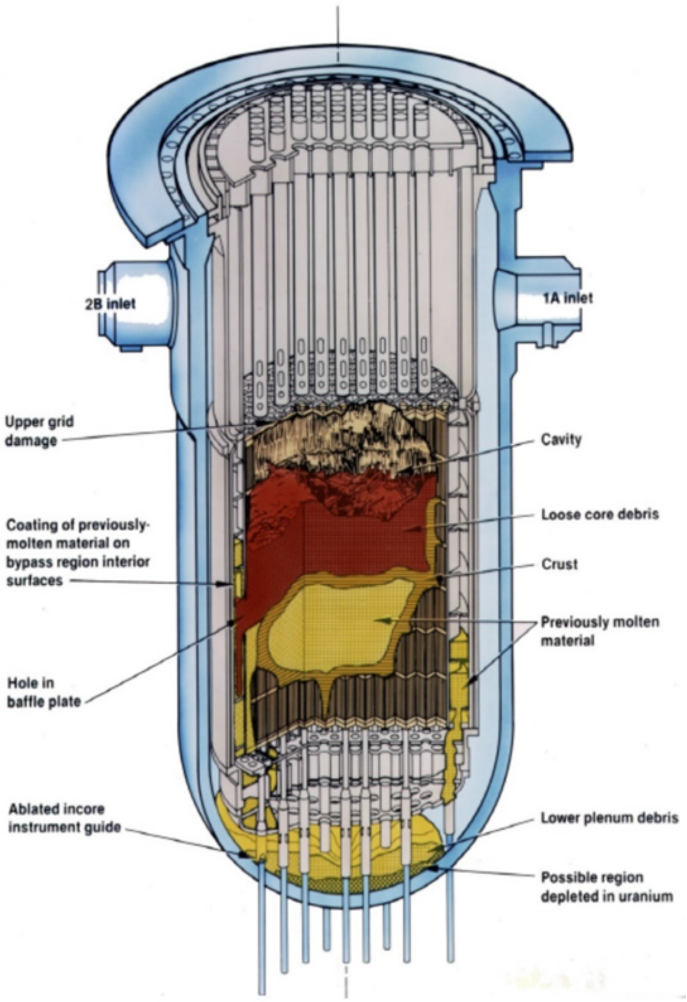


Figure 19 Stabilized state of the partially relocated and melted core in a TMI-2 accident. [16] operational for 8 hours (green) and 24 hours (blue) [14]

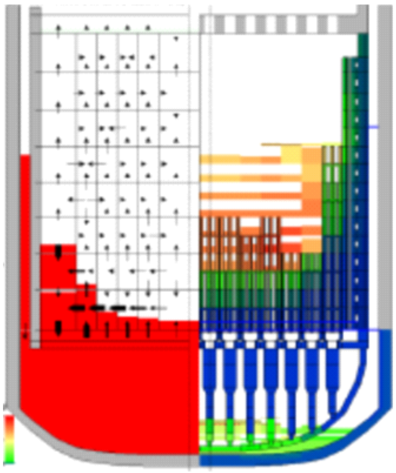
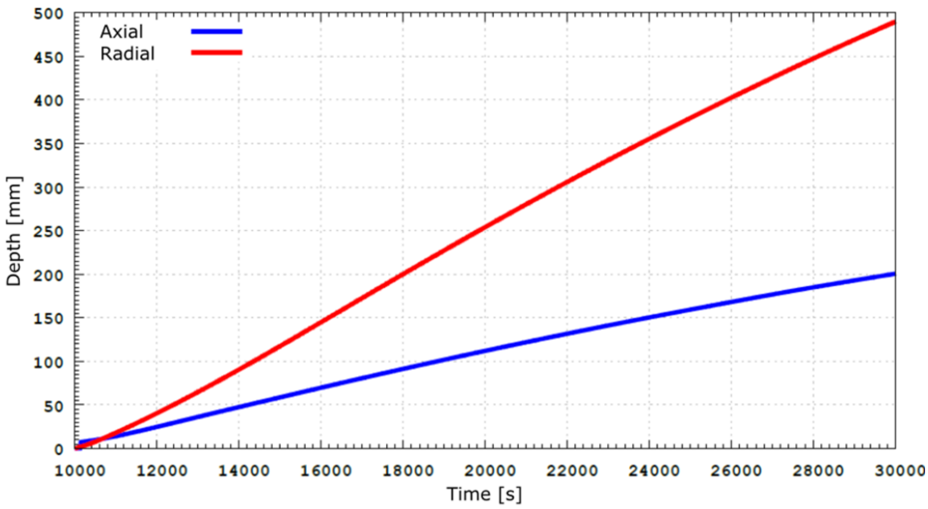


Figure 20 State of the partially degraded core saved by massive water injection into the RPV. Left: water level and flow pattern. Right: structures and temperatures. Time instant at 2500 s, i.e., 500 s after the onset of coolant injection [17]



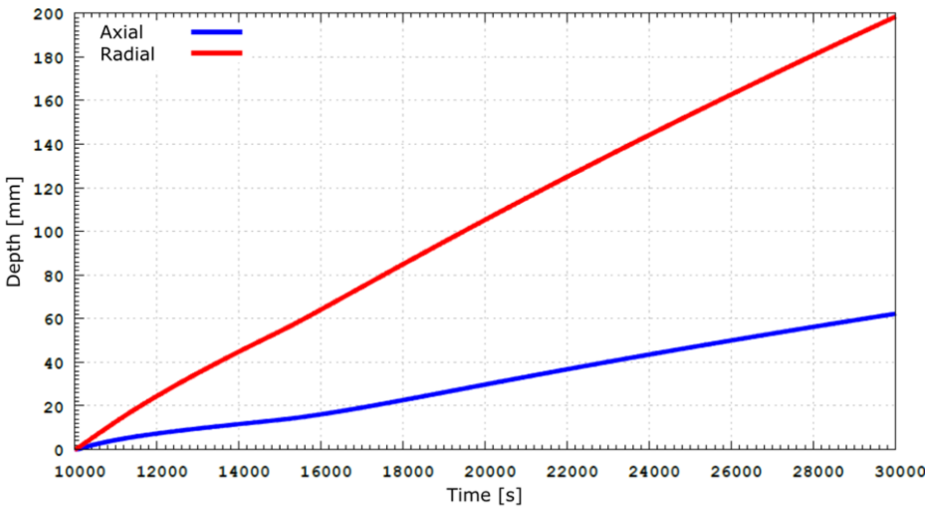


Figure 21 Melt progression in the radial/lateral (red curve) and axial (blue curve) direction in the CTMT concrete basemat. Upper figure: dry corium surface, lower figure: flooded corium surface [18]

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Acronyms

AOP	abnormal operating procedures
ATWS	anticipated transient without SCRAM
B&WOG	Babcock & Wilcox owners group
BDBA	beyond design basis accident
BWR	boiling water reactor
BWROG	BWR owners group
CA	computational aid
CEOG	Combustion Engineering owners group
CHF	critical heat flux
CTMT	containment
DBA	design basis accident
DEC	design extension conditions
DF	decontamination factor
DFC	diagnostic flowchart
DFM	dry filter method
DiD	defence in depth
EFW	emergency feedwater
EOP	emergency operating procedures
EPRI	electric power research institute
ERVC	external reactor vessel cooling
ESF	engineered safety features
FCVS	filtered containment venting system
FLEX	diverse and flexible mitigation strategies
FP	fission product
IVMR	in-vessel melt retention
LOCA	loss of coolant accident
LH	lower head

LP	lower plenum/lumped parameter
LWR	light water reactor
MCCI	molten core concrete interaction
MCR	main control room
NPP	nuclear power plant
PAR	passive autocatalytic recombiner
PC	primary circuit
PORV	pilot-operated relief valve
PR(S)A	probabilistic risk (safety) assessment
PWR	pressurized water reactor
PWROG	PWR owners group
RCS	reactor coolant system
RPV	reactor pressure vessel
SA	severe accident
SACRG	severe accident control room guideline
SAEG	severe accident exit guideline
SAG	severe accident guideline
SAM	severe accident management
SAMG	severe accident management guideline
SBO	station blackout
SC	secondary circuit
SCG	severe challenge guideline
SCST	severe challenge status tree
SG	steam generator
SGTR	steam generator tube rupture
TSC	technical support centre
VVER	water-cooled water-moderated energetic reactor
WOG	Westinghouse owners group

Simulation workshop

Slovak University of Technology in Bratislava
Slovakia

In the design studies of innovative fast nuclear reactor systems, it is desirable to estimate and reduce the uncertainties of design parameters. The combination of a validated and comprehensive sensitivity processing tool with nuclear data covariances may serve as a powerful utility to identify the uncertainty resulting from inaccurate data in calculated quantities of interest. These data deviations of measured physical results from reality arise from random and systematic legitimate errors. Defective data may originate from many sources, such as discrete computer processing, overlooking of important corrections, failure or improper calibration of equipment or the application of low statistics in measurement. A reasonable estimate of the underlying cross section probability distribution can be performed by applying the central limit theorem, i.e., assuming that the cross sections are distributed according to normal distribution. Further application of the generalized least squares method combined with integral experimental data measured in critical assemblies and experimental reactor cores may improve the cross section data and help achieve better prediction accuracy in the responses and parameters of the target core. The conventional cross section adjustment (CA) method is considered a promising design method suitable for achieving this goal.

In the past six years, two new codes have been developed within the framework of cross section uncertainty and adjustment: PORK, which is a sensitivity and perturbation analysis code, and STUUP, a code designed to evaluate the response uncertainty due to deviations in cross section data. These tools in parallel serve as the backbone of the computational system in fast core design development. These codes can also be applied to other reactor designs. During the workshop, these codes will be applied to gas-cooled fast reactor technology.

This text aims to provide important information about the methods and assessment implemented in the STUUP and PORK codes, developed for uncertainty and sensitivity analyses and application of the conventional cross section adjustment method.

The conventional cross section adjustment method

Main principle

The main principle of this method is the adjustment of cross section data as much as possible within their error limits and taking into account correlations, in such a manner that a better agreement between the calculated results and the measured integral data is obtained.

An important precondition of the cross section adjustment method is that a linear relationship always exists between the variations of the integral data R and differential data T in Eq. (1).

$$\delta R = S\,\delta T \tag{1}$$

It should be kept in mind that the method is only a first order approximation, and therefore serious errors may be encountered if the assumption of linearity is not sufficiently fulfilled.

The conventional cross section adjustment method is based on Bayes' theorem. From this theorem, the probability that cross section set T takes a true value when integral experiment data R_e are given is expressed as:

$$P(T \vee R_e) = \frac{P(R_e \vee T)P(T)}{P(R_e)} \tag{2}$$

where P(T) is the probability that the cross section set takes a true value, $P(R_e)$ is the probability that experimental data takes a true value, and $P(R_e|T)$ is the probability that the integral experimental data are true, under the condition that the cross section set is given. Assuming that the cross section set has a Gaussian distribution, the following equation is satisfied (multivariate normal distribution):

$$P(T) \propto e\,x\,p\left[\frac{-[T-T_0]^T U^{-1}[T-T_0]}{2}\right] \tag{3}$$

where T_0 is the a priori cross section set and U is its covariance matrix. Superscript "T" indicates that the matrix is transposed. When the integral experimental data are distributed around the calculation values of the integral experiments $R_c(T)$ obtained by T with the variance $V_e + V_m$, where indexes "e" and "m" represent the experimental and calculated values respectively, the following equation is satisfied:

$$P(R_e|T) \propto e\,x\,p\left[\frac{-[R_e - R_c(T)]^T[V_e+V_m]^{-1}[R_e-R_c(T)]}{2}\right] \tag{4}$$

By substituting equations Eq. (3) and Eq. (4) into Eq. (2), the following equation is satisfied:

$$P(T \vee R_e) \propto \frac{e\,x\,p\left[\frac{-[R_e-R_c(T)]^T[V_e+V_m]^{-1}[R_e-R_c(T)]+[T-T_0]^T U^{-1}[T-T_0]}{2}\right]}{e\,x\,p\left[\frac{-[R_e-R_e^0]^T[V_e]^{-1}[R_e-R_e^0]}{2}\right]} \tag{5}$$

To maximize the probability $P(T|R_e)$ under the conditions of the above-mentioned assumptions, the cross section set T should minimize the following J function:

$$J = [R_e - R_c(T)]^T[V_e+V_m]^{-1}[R_e-R_c(T)]+[T-T_0]^T U^{-1}[T-T_0] \tag{6}$$

After necessary mathematical adjustments, the main equations for the adjusted cross section set – T' Eq. (7), the post covariance matrix U' Eq. (8) and the adjusted response $R_c(T_a)$ Eq. (9) can be derived as follows:

$$T' = T_0 + US^T[SUS^T + V_e + V_m]^{-1}(R_e - R_c(T_0)) \tag{7}$$

$$U' = U - US^T[(V_e + V_m) + SUS^T]^{-1} \quad (8)$$

$$R_c(T_a) = R_c(T_0) + SUS^T[SUS^T + V_e + V_m]^{-1}[R_e - R_c(T_0)] \quad (9)$$

The subsequent equations Eq. (10) and Eq. (11) demonstrate the application of the adjusted cross section set to the target core:

$$R_A{}^{(2)} = R_{c0}{}^{(2)} + S^{(2)}US^{(1)T}\left[S^{(1)}US^{(1)T} + V_e{}^{(1)} + V_m{}^{(1)}\right]^{-1}\left[R_e{}^{(1)} - R_{c0}{}^{(1)}\right] \quad (10)$$

$$V^{(2)}\left[R_A{}^{(2)}\right] = S^{(2)}U_A S^{(2)T} + V_m^{(2)} \quad (11)$$

where symbol T stands for the cross section set, S are the sensitivity profiles, U is the covariance matrix, and the term (V_e + V_m) represents the calculation and modeling errors. Symbols with A prime represent adjusted parameters.

Uncertainty and similarity evaluation

Uncertainty analysis involves the assessment of the potential impact on the evaluated result due to the use of inexact (or inaccurate) quantities or techniques in its determination. In many applications, the major source of uncertainty lays in the calculated response due to uncertainties contained in evaluated nuclear data, such as microscopic cross sections, fission spectra, neutron yield, and scattering distributions. These arise from uncertainties in the experimental nuclear data measurements or uncertainties in the evaluation process itself, which in general combines differential experimental information with nuclear physics theory to generate the evaluated data. These uncertainties are governed by probability distributions. The actual probabilities are unknown, but the evaluated data values are assumed to represent the mean of the distribution, and the evaluated variance represents a measure of the distribution width. Correlations and uncertainties in nuclear data can have a significant impact on overall uncertainty in the calculated response; thus, it is important to include covariances and variances within the uncertainty analysis.

In the case of an integral parameter R, once the sensitivity coefficient matrix S_R and the covariance matrix U are known, the uncertainty ΔR of the integral parameter can be evaluated by a sandwich equation:

$$\Delta R^2 = S_R U S_R^T \quad (12)$$

The diagonal elements of ΔR^2 represent the relative variance values for each system under consideration, and the off-diagonal elements are the relative covariances between the given experiments.

The correlation coefficients (c_k) calculated by STUUP in this system provide a common means of normalizing shared uncertainty and may thus characterize the neutron similarity of systems under comparison. A much simpler way of determining neutronic similarity (E parameter) is based on only the comparison of the magnitude and shape of sensitivity profiles under interest.

$$E = \frac{S_a S_e^T}{|S_a||S_e|} \tag{13}$$

To evaluate the individual contribution to the uncertainty associated with a single cross section σ_{lmn} (specific isotope l, reaction m, and energy group n), ΔR_{lmn} is derived as:

$$\Delta R_{l\,m\,n} = \sqrt{S_{l\,m\,n}^2 u_{l\,m\,n}^2 + \sum_{k}^{N\,c\,o\,r\,r} C\,o\,r\,r_k S_{l\,m\,n} u_{l\,m\,n} S_k u_k} \tag{14}$$

where u are the elements of covariance matrix, k is the index for all N_{corr} (i.e., the total number of other cross sections which correlate to σ_{lmn}). In this approach (Ishikawa approach in STUUP), a combination of sensitivity coefficients and the Corr term can give rise to an imaginary value of the above-mentioned expression. In this case, only the negative value of the real part is used. In this manner, it is clear that σ_{lmn} provides the negative contribution to the total uncertainty. The summation over groups or reaction or isotopes is calculated statistically, i.e., as the square root of the sum of the squares:

$$\Delta R_{l\,m} = \sqrt{\sum_n \Delta R_{l\,m\,n}^2} \tag{15}$$

The uncertainty contribution for each reaction is printed in an output file only for the target core. In all the uncertainty calculations, the total reaction (the total cross section) is excluded, because it is the sum of the other processes and its inclusion would increase the variance from its actual value.

The relative covariance in the response due to the reaction pair is defined as:

$$\Delta R_{l,m_1,m_2}^2 = S_{l,m_1} U_l S_{l,m_2}^T \tag{16}$$

The value of ΔR^2 is then computed as the sum of the variances (diagonal terms) plus twice the sum of the covariances. The standard deviation of response is then simply the square root of ΔR^2.

The total uncertainty to k_{eff} induced by the cross section data is calculated by this approach in STUUP code. This method is called the "ORNL SCALE approach" in the STUUP output file. The uncertainty contribution of each reaction pair is printed in the output file for a target core.

Sensitivity profiles

The PORK sensitivity analysis code has been under development within the framework of cross section uncertainty and adjustment at STU for more than six years. PORK calculates the sensitivity coefficients linked to TRANSXSBJ, which is a code for generating effective self-shielded cross sections, and PARTISN, which solves the discrete ordinate neutron transport equation in 1D, 2D and 3D geometries.

Within the field of reactor analysis and design calculations, the sensitivity analysis offers a nuclear engineer unique insight into the investigated system. Estimation of the change in the system response due to the change in an input parameter can easily identify important processes and evaluate the influence of variation in this parameter. Definition of the critical eigenvalue or reactivity as our response allows us to use standard perturbation theory to determine the sensitivity coefficients of the system under investigation. The main constraint in this methodology is the assumption that a considered perturbation is small enough to cause any change in the neutron flux. If this is the case, then it is not necessary to perform a new calculation for the perturbed system, which in turn decreases the requirements on computational time. The sensitivity coefficients can be used later to calculate the uncertainty of the response arising from the cross section data and for the identification of individual contributors to the integral uncertainty. In the cross section adjustments method, sensitivities play a significant role in connection with covariance data as a carrier of information concerning investigated experiments. The standard perturbation theory is a special case of the generalized perturbation theory, and therefore PORK can be also used, with minor modifications, to determine the sensitivities of reaction rates or reaction rate ratios.

Standard perturbation theory

As one of the most important core parameters, the reactor multiplication factor can be defined as the fundamental eigenvalue in the neutron balance equation for a multiplying system:

$$L\,\Phi(x) - \lambda\,P\,\Phi(x) = 0 \tag{17}$$

where x represents all independent variables, such as the space, energy and direction coordinates. L and P are net loss and production Boltzman operators, respectively, and λ is the lambda mode eigenvalue ($\lambda = 1/k_{eff}$).

A change in an input parameter α appearing in the L and P operators will perturb the neutron balance, which will also change the eigenvalue. By introducing the perturbed parameters into Eq. (18), the perturbed system equation can be written in the following form:

$$(L + \Delta\,L)(\Phi + \Delta\,\Phi) = (\lambda + \Delta\,\lambda)(P + \Delta\,P)(\Phi + \Delta\,\Phi) \tag{18}$$

where ΔL and ΔP represent small linear perturbations in the corresponding operator, written in the following form:

$$\Delta\,L = \frac{\partial\,L}{\partial\,\alpha}\Delta\,\alpha \tag{19}$$

$$\Delta\,P = \frac{\partial\,P}{\partial\,\alpha}\Delta\,\alpha \tag{20}$$

The result of Eq. (18) multiplied by a non-zero weighting function w(x), integrated over phase-space and solved for the change in the eigenvalue can be represented as follows:

$$\Delta\,\lambda = \frac{\langle w(\Delta\,L - \lambda\,\Delta\,P)\Phi\rangle + \langle w(L - \lambda\,P)\Delta\,\Phi\rangle}{\langle wP\Phi\rangle} \tag{21}$$

where all products of perturbations are neglected, and angle brackets <> represent integration over phase-space (volume, energy and direction). Eq. (20) represents the first-order estimate for the eigenvalue perturbation. If w(x) is set equal to the λ-mode adjoint flux Φ*, which is a solution of L*Φ*(x)-λP*Φ*(x)=0, the second term in the numerator of Eq. (21) vanishes, since <Φ*(L-λP)ΔΦ>=<ΔΦ(L*-λP*)Φ*>=0. Then Eq. (20) is reduced to:

$$\frac{\Delta\,\lambda}{\lambda} \cong \frac{\left\langle \Phi^*\ (\Delta\,L - \lambda\,\Delta\,P)\Phi \right\rangle}{\lambda\left\langle \Phi^* P\Phi \right\rangle} \tag{22}$$

It is common in perturbation analysis to deal with relative changes, and we are looking for the expression for k_{eff} Eq. (23) by introducing Eq. (19) and (20) into Eq. (22):

$$\frac{\Delta\,k}{k} \cong \frac{\Delta\,\alpha}{\alpha}\frac{\left\langle \Phi^*\ \left(\frac{1}{k}\frac{\partial\,P}{\partial\,\alpha}\alpha - \frac{\partial\,L}{\partial\,\alpha}\alpha\right)\Phi \right\rangle}{\frac{1}{k}\langle \Phi^*\ P\ \Phi\rangle} \tag{23}$$

where

$$S_{k,\alpha} = \frac{\alpha}{k}\frac{\Delta\,k}{\Delta\,\alpha} \tag{24}$$

is the sensitivity coefficient of k_{eff} with respect to α.

Equivalent Generalized Perturbation Theory

The Equivalent Generalized Perturbation Theory (EGPT) is a special part of GPT, where the generalize forward and adjoint solutions are replaced by solutions of homogeneous equations. EGPT in the case of reactivity responses is then equivalent to applying the eigenvalue perturbation theory at the two states (unperturbed and perturbed states). It means that the sensitivity coefficients of the eigenvalue determined for two states of one system can be used to calculate the sensitivity coefficients of a reactivity describing this difference, such as the sodium coefficient, rod worth or Doppler Effect. Thus, the sensitivity coefficient of reactivity to parameter α is defined as follows:

$$S_{\rho,\alpha} = \frac{\Delta\,\rho}{\Delta\,\alpha}\frac{\alpha}{\rho} = \frac{1}{\rho}\left[\frac{S_{k\,2}}{k_2} - \frac{S_{k\,1}}{k_1}\right] \tag{25}$$

where ρ is the reactivity change between two pre-defined states, expressed in Eq. (26), and $S_{k1,2}$ is the sensitivity coefficient of k_{eff} to parameter α.

$$\rho = \rho_2 - \rho_1 = \frac{k_2 - k_1}{k_2 k_1} \tag{26}$$

In Eq. (26) parameters ρ_2, ρ_1, k_2, k_1 are static reactivity and k_{eff} for states 2 and 1, respectively. Static reactivity is defined as the relative change of keff from the unity. Since the EGPT method is related to change in k_{eff}, the application of this method has its limitations. Small or large changes of k_{eff} can significantly overestimate or underestimate the final values of sensitivity coefficients.

Covariance data

Usually, the ERRORJ code implemented in the NJOY system is used to transform evaluated data in the ENDF6 format into the energy-averaged cross section covariance. ERRORJ can process the covariance data of cross sections, including the resonance parameters and angular and energy distributions of secondary neutrons. In our case, the ENDF/B-VII.1 library [1] of evaluated data was used as the source for covariance data calculation. Using the COVERX or COVFIL interface files, the covariance data are transferred to the STUUP code.

Overview of the computational scheme

In order to give a thorough overview of the incorporated methods and data flow used in our approach, a simplified calculation scheme is presented in Figure 1. The ZZ-KAFAX-E70 [2] multi-group cross section library was processed at the Korea Atomic Energy Research Institute with the NJOY code [3] based on ENDF/B-VII.0 evaluated data [4] into MATXS format in 150 neutron energy group structure. Use of the ZZ-KAFAX-E70 library for GFR systems is shown in the studies [5], [6]. The ZZ-KAFAX-E70 library was used as input for the TRANSX code to process a self-shielded cross section library into ISOTXS format. Two calculation cycles were performed, according to the scheme in Figure 1. The first calculation cycle served to determine the weighting functions by the transport solver PARTISN. Based on the calculated scalar and angular flux, group collapsing was performed in TRANSX, and the 25-group cross section library was re-created in ISOTXS format. The second calculation cycle was performed, in most cases, with the DIF3D diffusion solver, and through this calculation the input data were prepared for PORK, which was used to calculate the sensitivity coefficients. It is possible to replace the DIF3D diffusion solver with the PARTISN transport solver, however the main advantage of DIF3D is the possibility to use a 3D nodal model in a triangle lattice, a feature which is not available in PARTISN. Sensitivity coefficients together with covariance data are inputted into STUUP to calculate the induced uncertainty in integral parameters from cross section data and to apply the cross section adjustment method.

Description of the target core

The target core which the above-mentioned methods is applied to is the gas cooled fast reactor, referred to as GFR 2400. Research of GFR 2400 is directed towards fulfilling the ambitious long-term goals of Generation IV concepts. This reactor is the newest on the evolutionary path of reactors using fully ceramic fuel. Structural materials in the form of silicon carbide fiber ensure high temperatures and good mechanical stability across a wide range of temperature gradients. An important

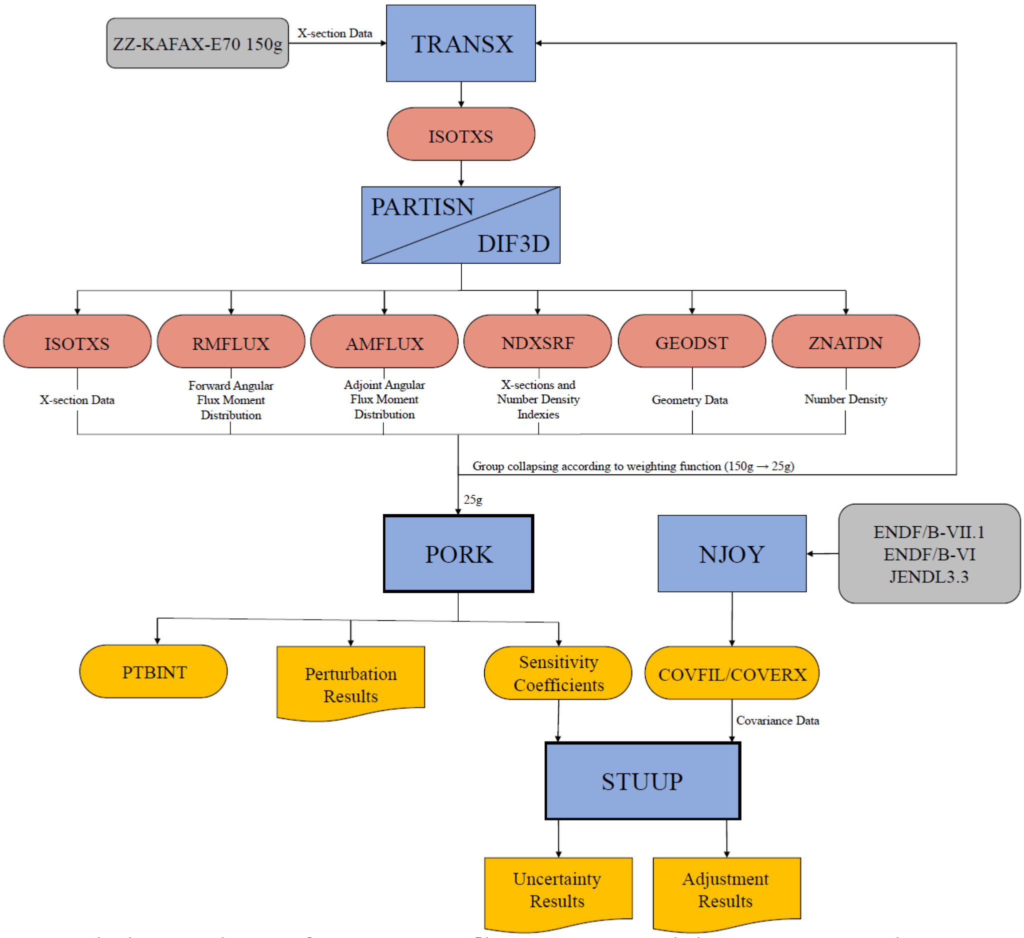
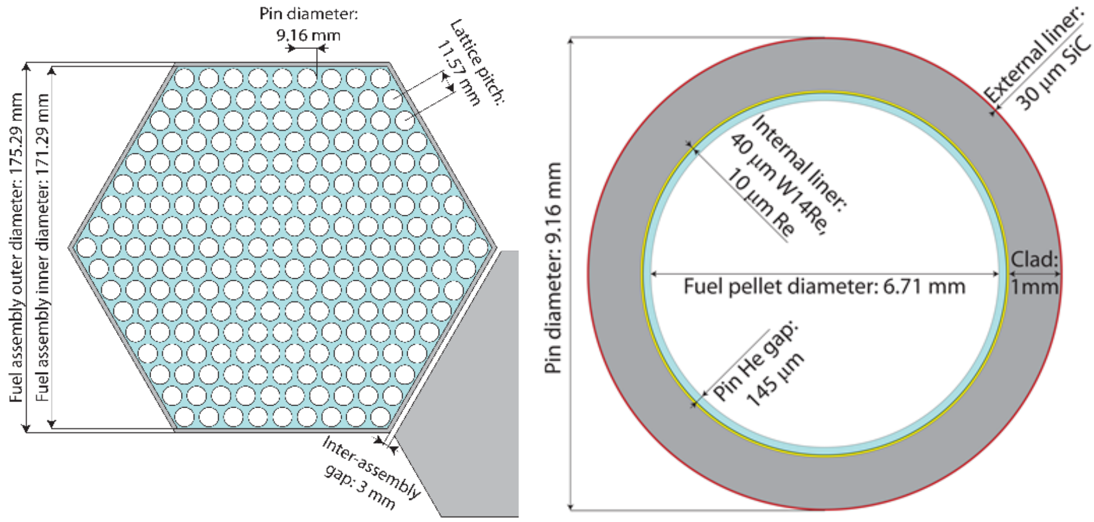


Figure 1 Calculation scheme of sensitivity profile processing and the uncertainty and cross section adjustment results.

innovation of the current design is the application of refractory metallic liners in the form of W14Re layers to enhance the fission product retention of the cladding, resulting in a significant neutronic penalty during normal operation. However, this concept is advantageous under transient conditions and involves spectrum softening. Helium is used as an efficient primary coolant. Since it introduces low moderation, the GFR's neutron spectrum is one of the hardest among the fast reactors, making it ideal for recycling all actinides, including minor actinides (MAs). Helium is inert and transparent, eliminating most problems related to coolant interaction with structural materials and enabling online visual inspection of the core. Also, the void reactivity effect is low due to the coolant's neutronic transparency. The core outlet temperature is not limited by the coolant characteristics, making it attractive for potential heat applications [7], [8], [9].

As mentioned in [10], the European experience of gas cooled reactor technology has been exceptional, with more than a thousand reactor years of gas thermal reactor operating experience and a number of in-depth design studies developed for gas-cooled fast rectors. The evolution of fuel designs includes designs of coated particle fuel with or without a binding matrix, silicon carbide blocks with dispersed microparticle fuel inside, the concept of silicon carbide plates with fuel pellets arranged in a honeycomb structure, finally arriving at the current design of a hexagonal lattice of cylindrical fuel rods consisting of a column of fuel pellets inside the composite silicon carbide cladding (Figure 2-a). The dimensions of the fuel pin are shown in Figure 2-b.



a) Design of fuel assembly.

b) Cross sectional view of the fuel pin.

Figure 2 Fuel assembly geometry [7], [8].

The cross-sectional view of the GFR 2400 core is shown in Figure 3. The core consists of 516 hexagonal fuel assemblies, with 217 fuel pins located in each assembly. The fuel material is (U-Pu)C with additional americium content, and the entire core is divided into two regions. The outer fuel core consists of 252 fuel assemblies with a volumetric enrichment of PuC 17.65 %; the inner fuel core consists of 264 assemblies with an enrichment of 14.12 %. The fuel pin's cladding material is made from a mixed structure of SiC/SiC_f. This material was selected because of its good thermo-mechanical properties. However, this material can be easily penetrated by fission gas products, and therefore an additional thin layer of the W14Re and Re is placed on the inner wall of the cladding. This material enhances fission gas product retention within the fuel pin and prevents the carbonization effect of the fuel-cladding interaction [7]. More details can be found, for example, in [8] and [11].

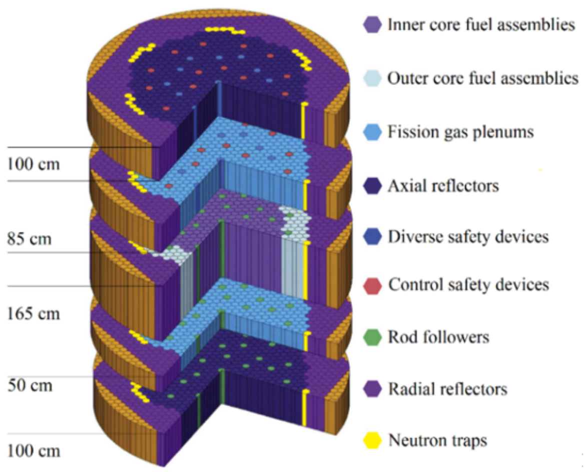


Figure 3 Cross sectional view of the GFR 2400 core [7], [8].

The thermal-hydraulic parameters of the GFR 2400 are shown in Table 1. The current design employs an indirect Brayton cycle, where the primary circuit consists of three loops with three blowers. The primary coolant is He, and the secondary coolant is a mixture of 20 % He and 80 % N2. The secondary circuit is connected to the tertiary via a water-steam heat exchanger. The estimated efficiency of this cycle is 45 %.

Table 1 Thermal-hydraulic properties of GFR 2400 [7], [8]

Parameter	Value	Parameter	Value
Thermal power [MW]	2 400	Primary coolant	He
Primary pressure [MPa]	7	Pressure drop in core [MPa]	0.143
Coolant mass flow rate [kg/s]	1 213	Coolant bypass flow rate [kg/s]	60
Coolant inlet temperature [°C]	400	Core outlet temperature [°C]	780

Integral experiments

The purpose of the integral experiments in the above-mentioned calculation methods is to assess the similarity level of these experiments with the target core. In reality, hundreds of the benchmarks are compared with the target core in a highly time-consuming process. For the purpose of the workshop, five representative integral experiments were selected to work with. The description of all integral experiments is based on the International Handbook of Evaluated Criticality Safety Benchmark Experiments [12].

PU-MET-FAST-006 – FLATTOP

In the mid-1960s, a critical experiment was performed at Los Alamos Scientific Laboratory using a spherical delta-phase plutonium core reflected by normal uranium. Delayed criticality (referred to as DC, meaning that criticality was achieved with the contribution of delayed neutrons, which may take up to a few minutes) was achieved. The results of this experiment are considered acceptable as a benchmark critical experiment. The Flattop assembly machine was used for several other experiments.

The experiment was performed using the Flattop critical assembly machine. This Flattop assembly has a core of delta-phase plutonium metal alloy enclosed in a thick normal uranium reflector (Reference 1). The core is composed of two hemispheres of plutonium metal. Both halves combine to form a sphere with an outside diameter of 3.586 inches, including the Ni coating (Reference 2). Table 2 shows the dimensions and composition of each of the two major plutonium alloy core parts [12].

Table 2 Pu-Alloy Core Parts

Mass (b) [g]	240Pu [wt. %]	Glory Hole Radius [cm]	Outside Radius [cm]
2962.05	4.77	0.648 ± 0.005	4.529 ± 0.005
2961.84	4.82	0.648 ± 0.005	4.529 ± 0.005

Several mass adjustment plugs (MA plugs) are available for reactivity adjustment. Mass adjustment plugs are placed in voids located on the inside surface of the reflector.

The reflector is composed of one stationary hemishell and two movable quadrants which make up the other hemishell. When assembled, the reflector is spherical in shape, with an outside diameter of approximately 48 cm. Three normal uranium control rods enter the stationary hemishell from below the assembly. The large rod has a reactivity worth about 1.6$, and the two smaller rods have a reactivity worth about 0.4$ each.

The core halves sit on a normal uranium pedestal support which rides on a track. The two movable quadrants are also situated on tracks. Assembly is accomplished by moving the core into place inside the stationary reflector hemishell and then moving the individual quadrants into place around the previously placed core. Disassembly is accomplished by rapidly removing the two normal uranium quadrants using hydraulic pressure at a rate of 17.58 cm/s for the first cm of travel and approximately 0.17 cm/s thereafter. Two independent gas accumulators are used as a backup to ensure that disassembly is accomplished in the event of a loss of power. A picture of the assembly is shown in Figure 4.

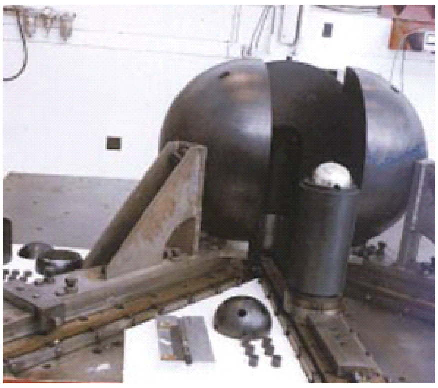


Figure 4 The Flattop Assembly [12].

The derived critical mass, corrected for the small gap between the core and reflector, is a sphere with a total mass of 6.06 ± 0.03 kg Pu alloy metal, 4.80 wt.% ^{240}Pu, with an average density of 15.53 g/cm^3 and reflected by 19.60 cm of normal uranium at a density of 19.0 g/cm^3. Since delayed criticality was obtained, the experimental was 1.0000. Using the given experimental uncertainty of ±30 g in the Pu mass, an uncertainty in of ±0.0016 can be derived using the ONEDANT code.

PU-MET-FAST-001/002 - Jezebel

In the mid-1950s, the ^{239}Pu Jezebel critical assembly was fabricated and operated at the Los Alamos Scientific Laboratory (LASL). Three Jezebel assemblies were built, one using Pu (4.5 at.% ^{240}Pu) and referred to as the ^{239}Pu Jezebel, one using Pu (20 at.% ^{240}Pu) and referred to as the ^{240}Pu or "dirty" Jezebel, and one using ^{233}U and referred to as the ^{233}U Jezebel. Only the ^{239}Pu Jezebel is described here. The ^{239}Pu Jezebel was a minimally reflected δ-phase ^{239}Pu critical assembly, nearly spherical in shape. The ^{239}Pu Jezebel was successfully operated for several years, and hundreds of experiments were performed, including a large number of reactivity worth measurements. The plutonium parts existed until at least 1981.

The Jezebel ^{239}Pu and Jezebel ^{240}Pu experiments both had the same construction, shown in Figure 5. For criticality safety purposes, the nearly spherical mass was constructed in four major pieces, each of roughly the same mass, which were assembled to provide three-part subdivision for operational safety. Because of the toxicity of plutonium, all the parts were nickel plated. The assembly was designed to be highly reproducible and to have minimum reflection, while retaining experimental flexibility.

The computational model of the Jezebel ^{239}Pu experiment represents a bare sphere of plutonium alloy with a critical mass of 17.02 ± 0.100 kg and density of 15.61 g/cm^3. The radius of the sphere is 6.3849 cm. In the case of the Jezebel ^{240}Pu assembly, the

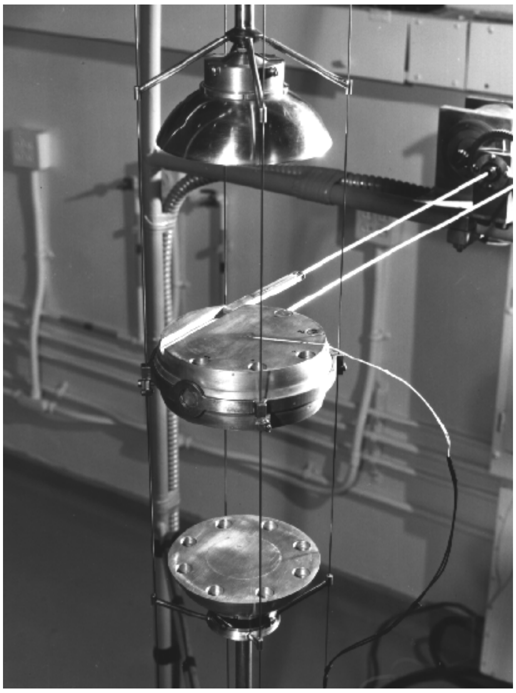


Figure 5 Jezebel in the "safe" configuration

critical ball thickness is 19.460 ± 0.156 kg and the ball radius is 6.6595 cm. The density of the material is 15.73 g/cm^3. The relatively large uncertainties of the masses of the spheres result from the effective value of the density of plutonium in the delta phase, which is more variable than in the case of uranium. The experiments were performed at room temperature in both cases. The experimentally determined value of reached the same value of 1.000 ± 0.002 in the case of the Jezebel ^{239}Pu and Jezebel ^{240}Pu experiments.

JOYO-LMFR-RESR-001 - JOYO

JOYO, Japan's first experimental fast reactor [13], was built at the O-arai Engineering Center (OEC) of the Institute of Nuclear Reactors and Nuclear Fuel Development (now JAEA) to gain the necessary experience in thermo-hydraulics, neutrons and liquid sodium-cooled fast reactor safety systems. The construction of the reactor began in 1970, and the first criticality was reached on April 24, 1977. The core fuel consisted of uranium and plutonium oxides (MOX). The radial and axial reflectors were oxides of depleted uranium. Heat removal from AZ was provided by liquid sodium. Power operation at a level of 50 MWt lasted from April 1978 to February 1979. Subsequently, the reactor was operated from July 1979 at a power level of 75 MWt until December 1980. The core configuration used in the mentioned period is called MK-1. Later, the core of the JOYO reactor underwent several configuration changes, but for the needs of an international set of physical experiments, the configuration of the MK-1 core proved to be more suitable.

Two critical sets are available in the MK-1 core configuration. The first minimum critical assembly with control cartridges above the AZ consists of 64 fuel assemblies, and the second assembly consists of 70 fuel assemblies with control assemblies at approximately half the depth of insertion. The criticality of the core and the low power operation were performed at a temperature of approximately 250 °C (523.16 K). Power operation was performed at coolant temperatures greater than 370 °C (643.16 K). A vertical section of the JOYO reactor is shown in Figure 6.

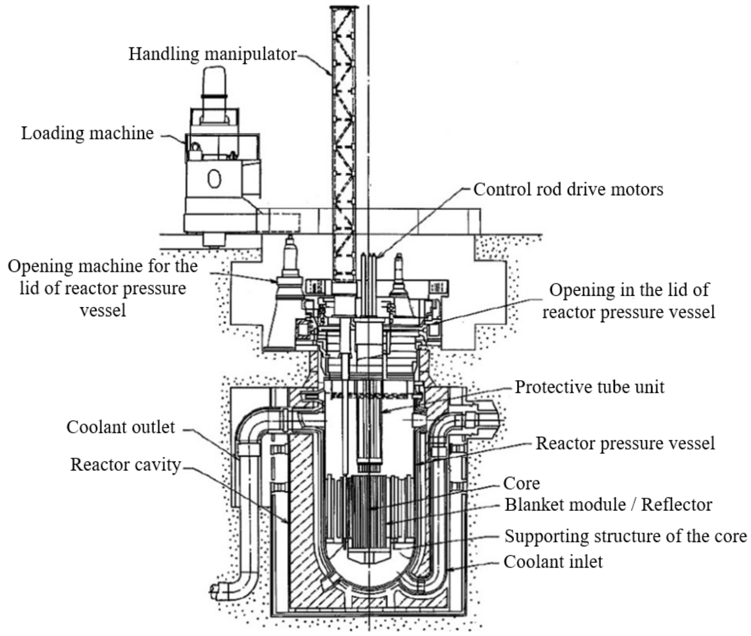


Figure 6 Vertical cross-section of the JOYO reactor [13]

For our purposes, we used a minimum critical core consisting of 64 fuel assemblies, the first criticality being reached on April 24, 1977, in the MK-1 configuration. A radial section of this configuration is shown in Figure 7. The experimentally determined value of the effective multiplication factor after considering correction to a temperature of 250 °C reaches 1.0011 ± 0.0018.

As the IHECSBE does not contain pre-prepared simulation input files for individual JOYO reactor configurations, we created our model based on the available information. All geometric dimensions and material compositions of the model were used for a temperature of 250 °C. The parts of the created model and the level of detail in the MCNP code environment are shown in Figure 8.

The created reactor model includes heterogeneous modeled areas, namely: fuel zone, axial blanket, radial blanket, control and safety assemblies (rods). The axial and radial reflectors were modeled homogeneously together with the neutron source. The fuel

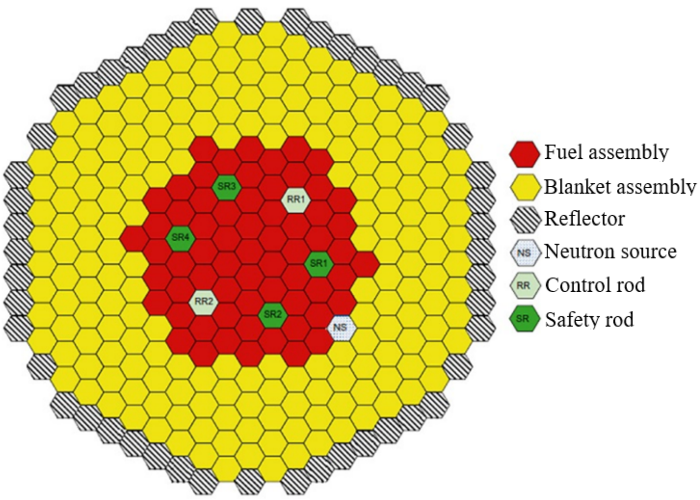
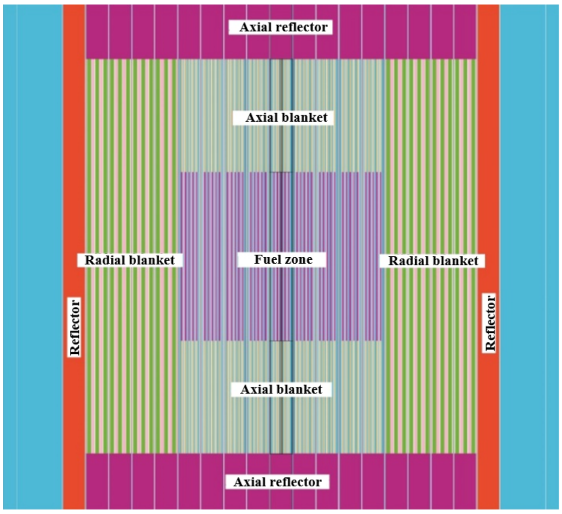
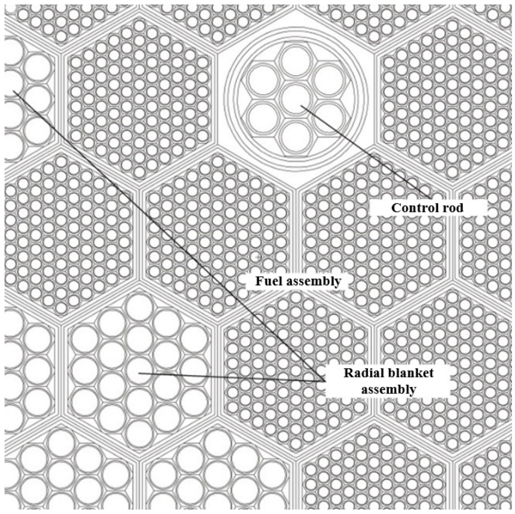


Figure 7 Fuel loading pattern of AZ JOYO MK-1





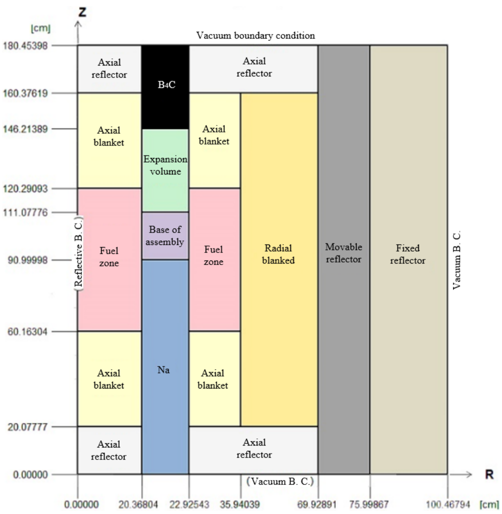
a) b)

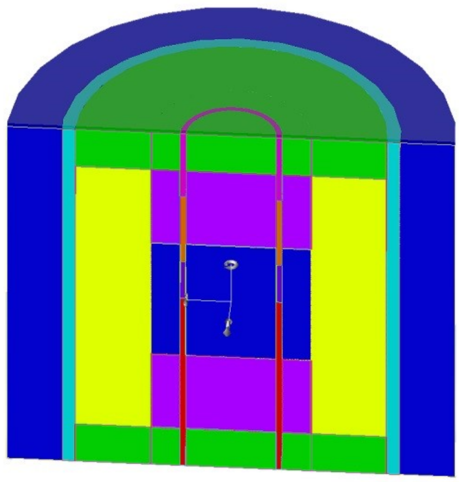
Figure 8 a) Vertical cross-cut of the detailed model; b) Detail of the fuel blanket boundary.

area consists of three material zones, namely the fuel itself, the components (SUS32 stainless steel) and the coolant. The presence of the guide rods of the fuel rods is considered by increasing the weight of their coverage. The radial blanket consists of rods filled with depleted UO_2. The material covering the rods is identical to the material in the fuel section.

For the needs of deterministic modeling, a simplified homogenized RZ variant of the geometric model was created for the JOYO MK-1 reactor, according to the recommendations of the international project WPEC Subgroup 33 [14]. A view of the specification and the subsequently created MCNP model visualized in the MCAM environment is shown in Figure 9.

The simplified model can then be used in deterministic codes DANTSYS or DIF3D to calculate the integral characteristics of AZ, represented by the value of k_{eff} and spectral indices.





a) Specification of simplified model [13]. b) 3D model in MCAM environment.

Figure 9 Simplified RZ model of JOYO MK-1.

ZPPR-LMFR-EXP-002 - ZPPR9

The Zero Power Plutonium Physical Reactor (ZPPR) is an important source of data for the development of fast breeder reactors, as it is the largest experimental fast breeder reactor ever, encompassing a wide range of geometric arrangements. The US Department of Energy (DOE) and the Japanese Atomic Energy Agency (JAEA) participated in the construction and operation of ZPPR. The fuel in the case of the ZPPR assembly consists of uranium and plutonium oxides (MOX) and the refrigerant is sodium.

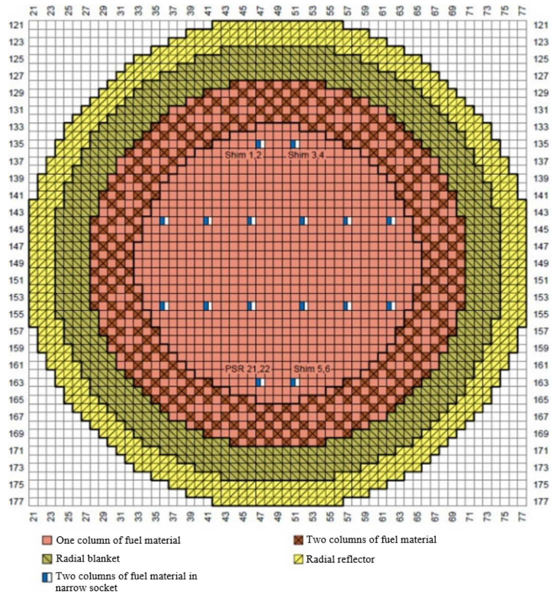
For our purposes, we use the first of a series of experiments, namely the ZPPR9 experiment, with an electrical output of 650 MWe. From a geometric point of view, the ZPPR9 is a cylindrical reactor with a volume of 4599 liters. It consists of two separate sections measuring 4.3 x 4.3 x 3 m, where one half is stationary and the other can be moved. During the critical experiment, the two halves are in close proximity, the gap between them being less than 1 mm. During the outage, the distance between the moving and stationary parts is 2.1 m. Each of the halves contains 5929 steel tubes into which drawers consisting of sheets of materials forming the reactor are horizontally inserted. The tubes are 1.5 m long with an internal dimension of 5.5 cm, while the wall thickness is 1 mm. The position of the tube is given by the respective half in which the tube is located and by the row and column of the given half. The sockets contain materials such as enriched uranium, natural or depleted uranium, sodium or steel. The performance of the assembly is controlled by two neutron detectors located in the lower corners of the stationary section. The reactor can be shut down by parking safety rods consisting of B_4C, which are, if necessary, immediately inserted into specific drawers with a narrow free space between the plates and the wall of the tube. The ZPPR9 kit contains 26 such drawers. Reactivity control is provided through the use of four additive control rods.

Both halves of the experimental assembly have the tubes arranged as a mirror image of the opposite half. The core of the reactor is formed by a fuel zone divided into two regions differing in plutonium content. The internal fuel core represents 55 % of the volume of the fuel zone. It consists of 1626 tubes containing a single column of fuel material (Pu-U-Mo), known as Single Core Fuel (SCF). In addition to 760 SCF, the outer zone also contains 576 DCF (from Double Core Fuel). The DCF socket consists of two columns of fuel material. The ZPPR9 reactor also includes a breeding blanket consisting of four rings containing U_3O_8, depleted uranium, and sodium. At the periphery, the core is surrounded by three rings of steel reflectors. A historical photograph of the ZPPR9 assembly is shown along with a diagram of the location of the various types of drawers in Figure 10.

The detailed calculation model consists of two fuel regions and does not contain any safety or control rods or positions for their insertion. During the creation of the model, corrections were made for the size of the gap between the two halves of the assembly in the critical state, as it is not possible to determine this precisely, and also for the dimensions of the pipes and drawers. The same nuclide composition of one type of drawer is also contemplated.

The temperature of the critical experiment was estimated to 26.7 ± 1.0 °C, which corresponds to the core data for 293 K.

The effect of asymmetrically distributed depleted uranium sheets in some tubes was also considered. Due to the complexity of modeling individual drawers, the individual

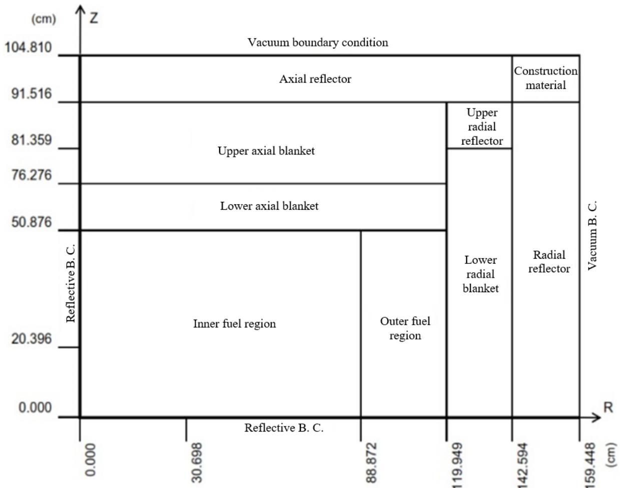


a)

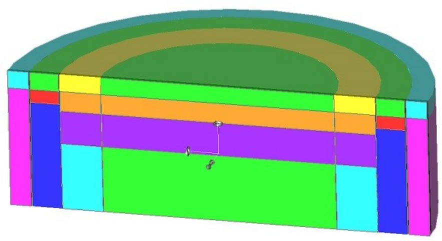
b)

Figure 10 a) Fuel loading pattern ZPPR9; b) Historical photograph of ZPPR.

types of drawers in the reference model are created as a homogeneous mixture of the respective materials. The gaps between the sheets of materials and also between the individual tubes are neglected. Homogeneous models are created for the SCF, DCF, and axial and radial blankets. The three-dimensional model of the assembly is then created from the homogeneous drawers, the radians, and the axial blanket, consisting of only one part with the average composition of the blank of the real assembly. The experimental value of the test configuration's , taking into account the necessary corrections, reached 1.00106 ± 0.00116. When part of the sodium was removed from the core test task (Mark 3), the effective multiplication coefficient reached 1.000015549 ± 0.001170000 (reactivity 29.39 cents at βeff 0.003550). For the possibility of modeling the core in deterministic codes, a homogenized RZ model of the core was again created for both investigated cases. The specification of the assignment according to WPEC Subgroup 33 [14] and the adequate MCNP model visualized in the MCAM environment are shown in Figure 11.



a) Simplified RZ model specification [14].

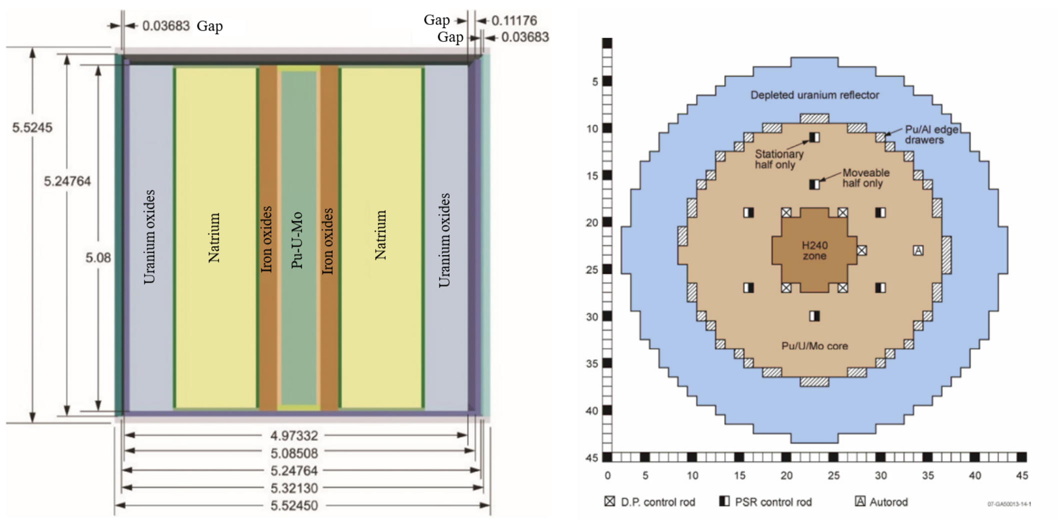


b) 3D model in MCAM environment [15].

Figure 11 Simplified RZ model of the ZPPR9.

ZPR-LMFR-EXP-001/002 - ZPR6

The ZPR6 critical assembly with configuration No. 7 (ZPR6/7) was operated at the Argonne National Laboratory USA between 1970 and 1971. Configuration 7 simulated a large sodium-cooled fast reactor with plutonium and uranium oxide fuel, with a depleted uranium radial and axial reflector simulating energy reactor blanket zones with an effective height to zone diameter (H/D) ratio of approximately one. In terms of the computational reproducibility of experimental data, the simplest possible material and geometric design of the task is selected. The core of the critical assembly consists of plates of depleted uranium, sodium, iron oxides, depleted U_3O_8 and Pu-U-M alloy stored in steel drawers. The neutron properties of the core are determined mainly by the presence of isotopes ^{238}U and ^{239}Pu. The reference configuration is named ZPR-LMFR-XP-001. The cross-section of the drawer and its composition in the central part of the zone of the reference configuration and the radial section of the core with increased ^{240}Pu content are shown in Figure 12.

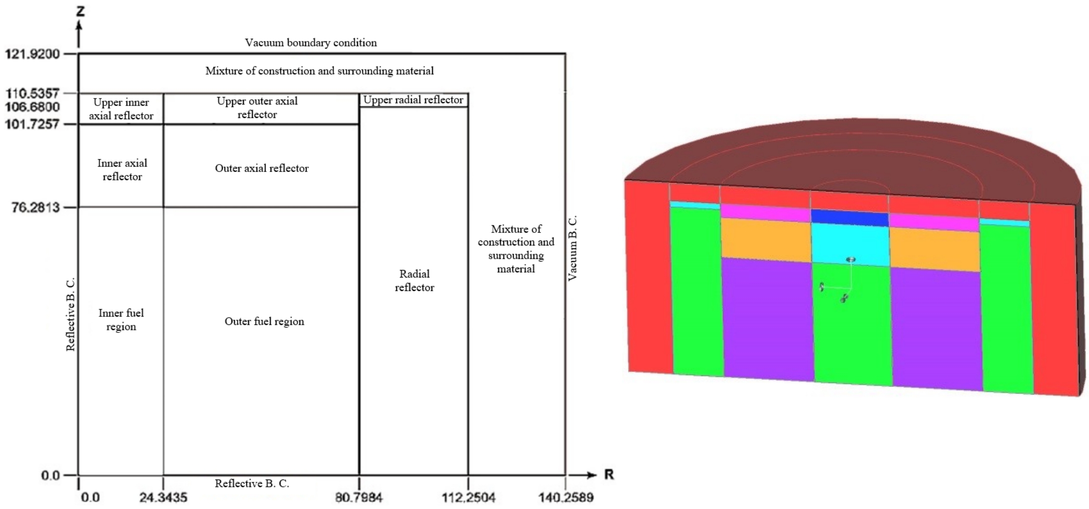


a) b)

Figure 12 a) Drawer cross-section; b) Radial cross-section of experiment with higher enrichment ^{240}Pu (H240).

The ZPR6-7 critical assembly covered over 139 unique core configuration variants, allowing simulation of criticality, reactivity of control assemblies, reactivity from sodium loss, and measurement of spectral indices. The high ^{240}Pu isotope enrichment configuration (ZPR-LMFR-EXP-002) simulating a power reactor core with a higher achieved fuel burnup contained an increased amount of ^{240}Pu isotope (27 %) in the Pu-Mo-U fuel plates compared to the reference configuration (11 % ^{240}Pu). The average core temperature during the experimental measurements was 19.4 °C, which corresponds to the evaluated data for 293 K. The experimental value of the effective multiplication factor, taking into account corrections, reached the value k_{eff} = 1.00051 ± 0.00087 in the case of the reference configuration (ZPR-LMFR-EXP-001). In the case of the critical set with increased ^{240}Pu isotope concentration, the effective multiplication factor, taking into account corrections, reached 1.00080 ± 0.00090.

As in previous cases, a homogeneous geometric RZ model was created for the ZPR6-7 reactor according to the specification given in the international project WPEC Subgroup 33 [14]. The specification of the reference configuration (ZPR-LMFR-EXP-001) and the MCNP model shown in the MCAM environment are presented in Figure 13.



a) Simplified RZ model specification (dimensions in cm) [14]. b) 3D model in MCAM environment [15].

Figure 13 Simplified RZ reference model of ZPR6-7

For the high ^{240}Pu isotope enrichment configuration (ZPR-LMFR-EXP-002), a Pu-Al ring extending to the outer axial reflector was added to the edge of the outer fuel zone. A homogenized simplified model was also created for this configuration.

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Approaching the Critical State

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Introduction

The critical state of a reactor is usually reached when the reactor is successfully placed into operation. In fact, every power change is a short-time deviation from the critical state and from returning to a new critical state. This process occurs without any potential nuclear safety issues but is only possible for a well-known arrangement of the core. A more difficult situation occurs in a new reactor core or when the reactor starts up after refuelling. In these cases, approaching the critical state is always associated with a factor of uncertainty. Neither the experience of the operators and reactor physicists nor the most precise physical calculations can guarantee the exact determination of the critical parameters (e.g., the amount of fuel in the core, the control rod positions in the critical state, the exact concentration of the absorber, etc.). Therefore, a critical experiment must be conducted at all reactors with almost identical methodology.

Sub-critical multiplication

Before the reactor is started it is in a subcritical state (i.e. the state where $k_{eff} < 1$). For a safe reactor start-up, an external neutron source is used. In a subcritical system with an external neutron source which emits S neutrons inside the core, the amount of neutrons in the core N is given by:

$$N = S + S\,k_{eff} + S\,k_{eff}^2 + S\,k_{eff}^3 + \cdots + S\,k_{eff}^m = \frac{S(1 - k_{eff}^m)}{1 - k_{eff}} \tag{1}$$

where m is the number of neutron generations. With the reactor in a sub-critical state, i.e. $k_{eff} < 1$, the final number of neutrons is the sum of the above geometric series with the quotient of k_{eff}. Regarding light-water reactors, the lifetime of one neutron generation is in the order of 10^{-4} to 10^{-5}s. Therefore, k_{eff} can be neglected in the numerator. The final equation therefore has the following form:

$$N = \frac{S}{1 - k_{eff}} = S\,M \tag{2}$$

where $M = 1/(1 - k_{eff})$ is the subcritical multiplication. Therefore, in a subcritical system, the neutron population depends solely on S and k_{eff} .

If a neutron population is observed by a neutron detector, the detector output is proportional to:

$$n = \frac{\varepsilon S}{1 - k_{eff}} \tag{3}$$

where ε is the detector efficiency.

Approaching criticality

A critical experiment is the experimental check of the calculated geometry and composition of the core. By comparing the experimental results and the results from calculation, the necessary corrections of the calculation methods, the applied constants, etc. can be derived. Knowledge of the real critical core is not only important during the first start-up of the reactor into operation, it also determines the safety of the core, the quantity of loaded fuel to reach operational reactivity, and a number of other parameters.

Let us assume that we have a core with an effective multiplication factor $k_{eff} < 1$. To approach the critical state in light-water moderated reactors, one can choose from the following options:

- Changing the fuel quantity
- Changing the moderator water-level
- Changing the neutron absorption (i.e., movement of the control absorption rods)
- Changing the absorber concentration in the coolant/moderator

In practice, k_{eff} depends on one or more of the above parameters. Let us assume that the parameter change is discontinuous in steps, numbered by the index i (or during the continuous parameter change, we take measurements step-by-step at certain parameter values, i.e., pulling out the rods to a certain position or reaching a certain moderator concentration). The reactor should be in the subcritical state with an external neutron source inserted into the core. Let us also assume that the reactor can be described by a single point approximation. This means that the thermal neutron fluxes in the core as well as in the reflector are mutually proportional at every moment. Thus, at any position in the reactor, the detector measures a value which is directly proportional to the reactor power.

Let us analyse the case of approaching the critical state by pulling out the control rod step-by-step. In step zero, i.e., when the rod is at its bottom position, the amount of neutrons in the core corresponds to:

$$N_0 = \frac{S}{1 - k_{eff,\ 0}} \tag{4}$$

After movement of the rod, k_{eff} is changed and the amount of neutrons in the core is changed accordingly:

$$N_1 = \frac{S}{1 - k_{e\,f\,f\,,\,1}} \tag{5}$$

We can apply a similar assumption for the next steps:

$$N_i = \frac{S}{1 - k_{e\,f\,f\,,\,i}} \tag{6}$$

If we assume that the detector efficiency remains constant, we have:

$$n_i = \frac{S}{1 - k_{e\,f\,f\,,\,i}} \tag{7}$$

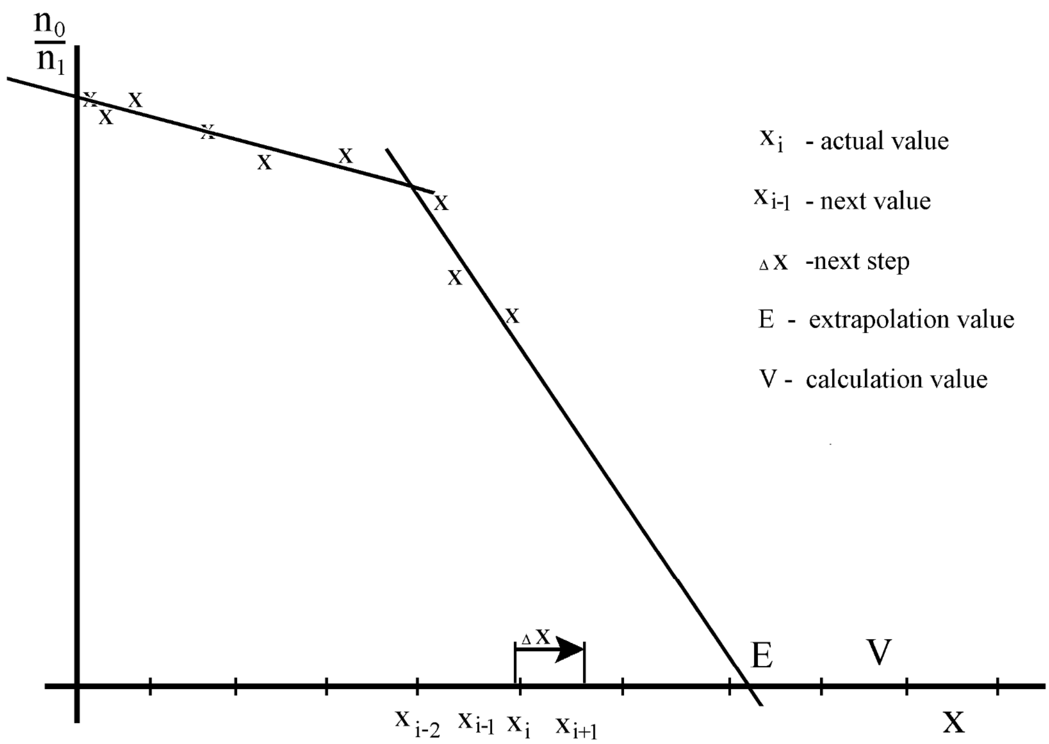

Figure 1 Critical experiment

If we are approaching the critical state, then the value of k_{eff} approaches one and the value ni (see Equation 7) increases to infinity; its inverse value $1/n_i$ approaches zero. When the inverse value is plotted with respect to k_{eff} (or generally, with respect to the variable core parameter), the curve intersects the x-axis (i.e., $1/n_i = 0$) when the critical state is reached (see Figure 1). By extrapolation of this curve, we can thus foresee the size of the variable parameter at the moment criticality is reached.

As a rule, the value of $1/n_i$ is plotted on the y-axis. When approaching criticality, $1/n_i$ approaches zero and intersects the x-axis. Any constant multiple of it also approaches

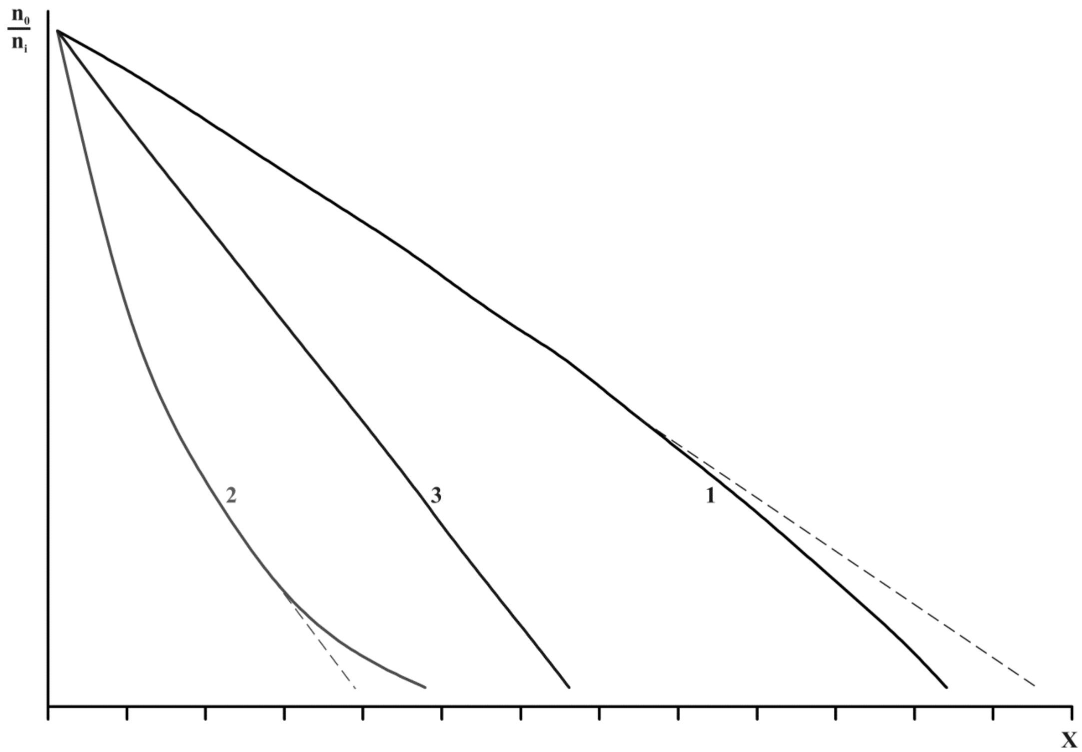

Figure 2 Approaching the critical state

zero. Therefore, it is appropriate to plot values of n_0/n_i on the graph; the initial value of n_0/n_i is 1 and there is no need to adjust the scaling.

When measurement of the ith step is finished, the positive reactivity change occurs (k_{eff} is approaches a value of 1). The ith step is measured, after the reactor power stabilizes. The value of n_0/n_i is entered and the extrapolation is plotted. The extrapolated value, i.e. where criticality can be expected with the same curve slope, is determined. This value is compared with the value determined from calculation. For safety reasons, it is required that the modified parameter increases by no more than 1/2 of the difference between the present state and the smaller value of the position determined from the previous extrapolation and the calculation (refer to Figure 1). After applying this check, the requirement on the next step becomes more accurate, the value is plotted on the graph, and the entire process is repeated. Once the value of n0/ni is approximately equal to 0.1 - 0.15, the final extrapolation is done and the operator may drive the reactor to the critical state.

Three different cases (i.e., curve shapes) may occur when the critical state is approached. In Figure 2, the ideal course is shown by Curve 3. Regarding nuclear safety, Curve 1 is the most disadvantageous since the extrapolated value is higher than the subsequent real value. Curve 2 is more advantageous than Curve 1, but the angle under which it intersects the x-axis leads to an inaccurate intersection point, and therefore the forecast of the critical state is less accurate. The shape of the curves depends on a number of factors, such the mutual position of the detector, the neutron source, and the fuel or the distance of the neutron source from the detector.

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