Introduction to Reactor Physics

Joint ICTP-IAEA Essential Knowledge Workshop on Deterministic Safety Assessment and Engineering Aspects Important to Safety

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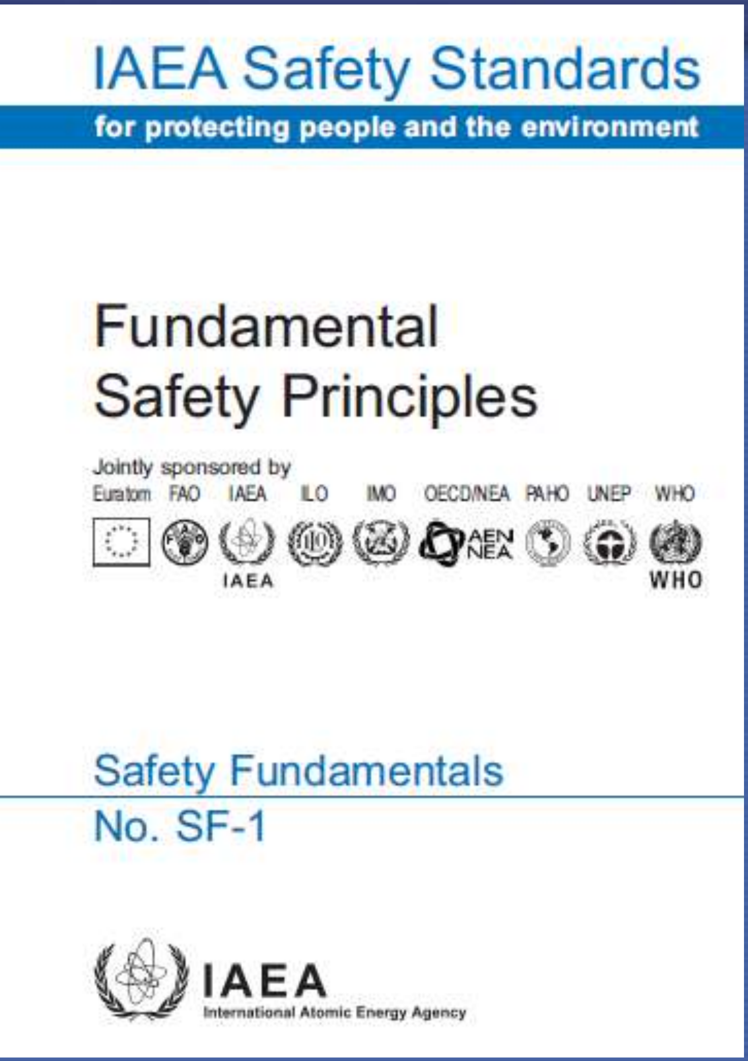
Safety Fundamentals SF-1



The fundamental safety objective is to protect people and the environment from harmful effects of ionizing radiation.

<u>Measures to be taken:</u>

a) To control the radiation exposure of people and the release of radioactive material to the environment;
b) To restrict the likelihood of events that might lead to a loss of control over a nuclear reactor core, nuclear chain reaction, radioactive source or any other source of radiation;
c) To mitigate the consequences of such events if they were to occur.

<u>Principle 8: Prevention of accidents</u>

All practical efforts must be made to prevent and mitigate nuclear or radiation accidents.

- To prevent the loss of, or the loss of control over, a radioactive source or other source of radiation.

Neutronic Safety Consideration in the Reactor Core

APoS


IAEA Safety Standards
for protecting people and the environment

Safety of Nuclear Power Plants: Design

Specific Safety Requirements
No. SSR-2/1

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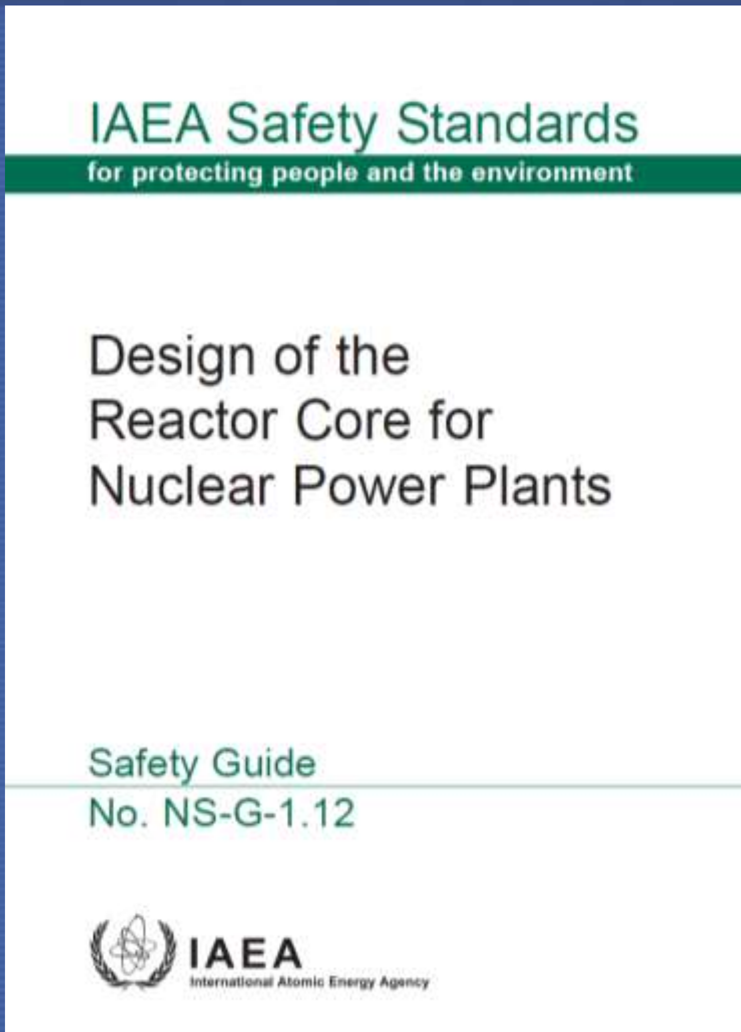
- **Requirement 43: Performance of fuel elements and assemblies**: Fuel elements and assemblies for the nuclear power plant shall be designed to maintain their structural integrity, and to withstand satisfactorily the anticipated radiation levels and other conditions in the reactor core, in combination with all the processes of deterioration that could occur in operational states.

- **Requirement 45: Control of the reactor core:** Distributions of neutron flux that can arise in any state of the reactor core in the nuclear power plant, including states arising after shutdown and during or after refuelling, and states arising from anticipated operational occurrences and from accident conditions not involving degradation of the reactor core, shall be inherently stable. The demands made on the control system for maintaining the shapes, levels and stability of the neutron flux within specified design limits in all operational states shall be minimized..

Neutronic Safety Consideration in the Reactor Core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Design of the Reactor Core for Nuclear Power Plants

Safety Guide
No. NS-G-1.12

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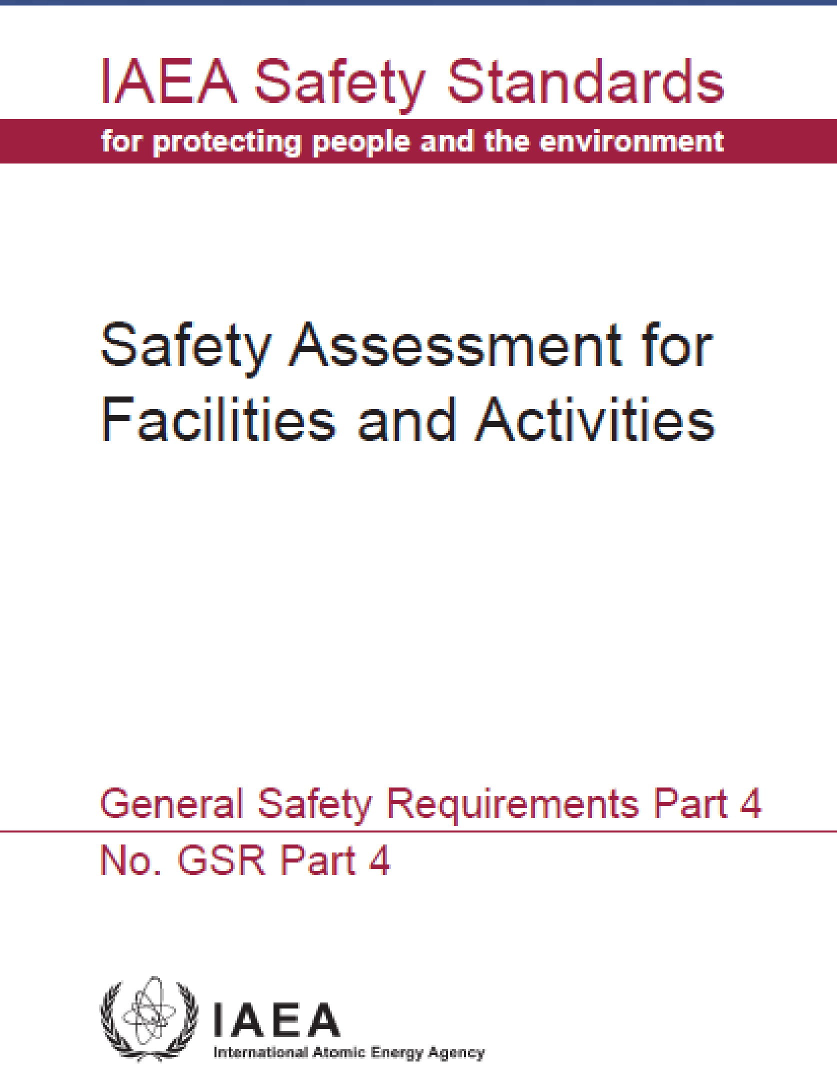
- The reactor power should be controlled by a combination of the inherent neutronic characteristics of the reactor core, its thermal-hydraulic characteristics and the capability of the control and shutdown systems to actuate for all operational states and in design basis accident conditions.
- …" the maximum insertion rate for positive reactivity in operational states and in design basis accidents should be limited…"
- Calculation of the core power distribution should be performed in the design for representative operational states to provide information for use in determining: (a) operational limits and conditions; (b) action set points for safety protection systems; (c) operating procedures ….

GSR Part 4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Safety Assessment for Facilities and Activities

General Safety Requirements Part 4
No. GSR Part 4

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4.19. The possible radiation risks associated with the facility or activity include the level and likelihood of radiation exposure of workers and the public, and of the possible release of radioactive material to the environment, that are associated with anticipated operational occurrences or with accidents that lead to a loss of control over a nuclear reactor core, nuclear chain reaction, radioactive source or any other source of radiation.

Analysis of LWR Core Design

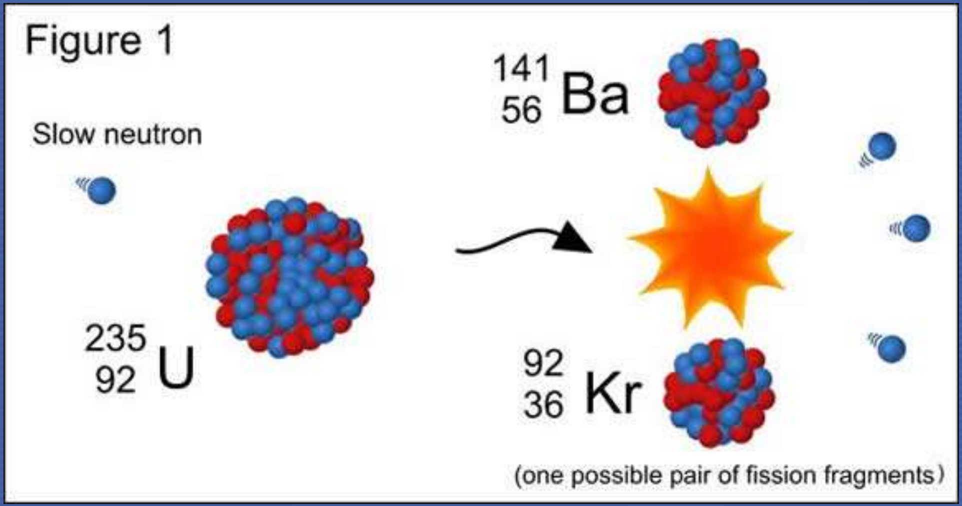
- Fuel Procurement Analysis:
  - Enrichment specification
  - Burnable absorber design
  - Economics analysis
- Reload Core Design:
  - Selection of “optimum” fuel loading pattern
  - Selection of coolant flow and control rod strategy (BWR)
  - Computations of margins to design safety limits
- Safety Analysis:
  - Calculations of nominal and off-nominal power shapes
  - Calculations of rod worth, shutdown margins, reactivity coefficients
  - DNBR analysis
  - Power input in transient/accident analysis

Needs for Analytical Solution

- In-core fuel management and core design
  - Calculation of fuel depletion, in realistic core conditions, in multiple fuel cycles, and taking into account certain design limitations.
  - Cross-section generation
  - Calculation of cross-section dependencies on burn-up
  - Thermal-hydraulic conditions, and control absorber(s) presence.

- Core neutronic dynamics, in normal and accident conditions
  - Global and local power generation

Fission Energy Production Principle



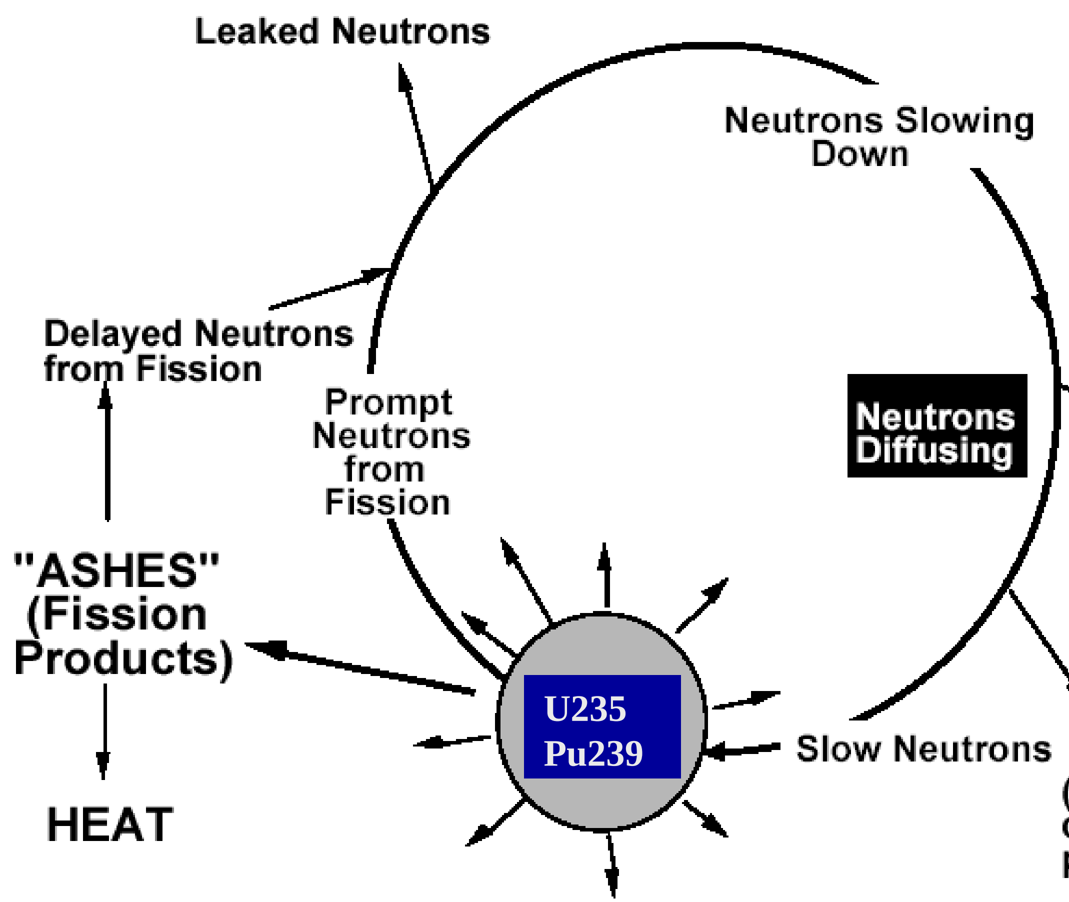
U-235 + Neutron (n) → Fission Products (FP) + Xn

ΔM = Mass (U-235) + Mass (n) - Mass (FP) - Mass (Xn) $\neq 0$

Energy Released = $\Delta MC^2 \cong$ **200 E6 ev/Fission**
(32 E-12 J)

Energy Released From Combustion Process $\cong$ **2 ev / Reaction**
C + O2 → CO2

Introduction:
Neutron Cycle in Thermal Reactor



- When Number of Slow Neutrons is constant, the system is critical.
- Delayed Neutrons appear after ~ 10 seconds.
- Fast Neutrons slow down in ~ 1 millisecond.

Fission

- Basic types of neutron-nucleus interactions:
  - Scattering (elastic, inelastic)
  - Absorption (fission, capture)
- Few nuclides can fission – U^{235}, U^{233}, Pu^{239}, Pu^{241}
- Energy per fission ~ 200 MeV (85% energy of fission products, 15% kinetic energy of other particles)
- The fission products are nuclides of roughly half the mass of uranium, “neutron rich”, decay typically by β- or γ-disintegration with various half-lives
- Energy from fission products disentigration exists long after chain reaction is stopped – decay heat

Fission

- The probability of a neutron inducing fission in 235U is much greater for very slow neutrons than for fast neutrons
- Moderators – materials that slow down neutrons to thermal energies (more efficient are atoms close to neutron mass that are not neutron “eaters”)
- Several processes compete for neutrons:
  - Absorptions that end in fission
  - Non-productive absorptions
  - Leakage out of reactor
- Self-sustainability of chain reaction depends on relative rates of production and elimination of neutrons

Reactivity

- Effective reactor multiplication constant:
  k_{eff} = Rate of neutron production/Rate of neutron loss

- Reactivity :
  $\rho = \rho = 1 - 1/k_{eff}$
  (Neutron production-loss)/Neutron production

- $k_{eff} < 1$, $\rho < 0$ - subcritical reactor
  $k_{eff} = 1$, $\rho = 0$ - critical reactor
  $k_{eff} > 1$, $\rho > 0$ - supercritical reactor

- Control of reactivity crucial for safe operation

Concept of Cross Section (Probability of Neutron Interaction)

- Microscopic cross-section σ
  - probability for the interaction of neutrons with only one kind of nuclei
  - effective area presented to the neutron by 1 nuclei
  - depends on the type of nucleus and on the neutron energy
  - expressed in units for area cm^2, barn = 10^{-24} cm^2
- Macroscopic cross-section Σ
  - probability for the interaction in unit volume of the material

    $\Sigma = N\sigma$ (N - atomic density N atoms/$cm^{-3)}$
  - Expressed per neutron path length
  - For different types of nuclei in the material, sum of partial products $N\sigma$ gives Σ

Simplistic Treatment of Power Changes

- $P = P_0 \exp(\rho t/T)$
  T - average time interval between successive neutron generations
- Without delayed neutrons mean generation time leads to prompt-neutron lifetime (fraction of microsecond)
- Delayed neutrons, although only ~0.6 %, reduce rate of power change considerably
- Correct treatment requires solving coupled set of equations for the time-dependent flux distribution and the concentrations of the individual delayed-neutron precursor atoms

Moderator

- Fission neutrons are fast, small probability for new interaction if the number of fission atoms is low
- Need to slow down (moderation) the neutrons with as few as possible collisions without loss of neutrons (absorption)

Moderator	No. of collisions
H	18
D	25
C	114

Moderator	Moderating ratio
H	62
D	165
C	5000

Moderation ratio = ratio of the slowing-down power of the material/ neutron absorption cross section

Fuel Burnup

- Cumulative quantity of fission energy produced per mass of nuclear fuel during its residence time in the core, MWd/tU
- Practically linar with time spent in neutron flux
- Important economic quantity, high burnup signifies low fuel consumption

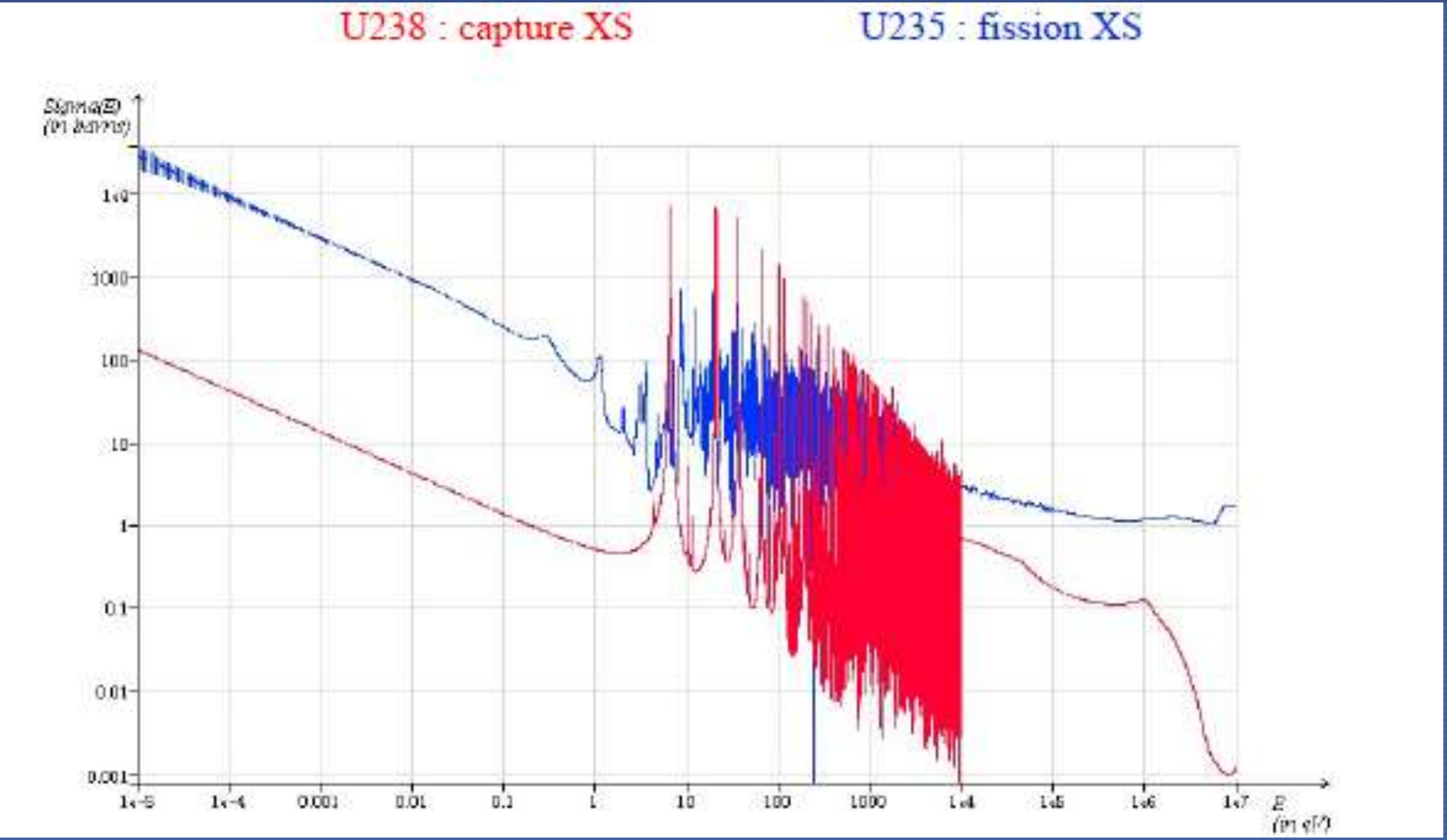
Boltzmann Equation

- Reactor is described in terms of its geometry, composition and cross-sections
- Purpose of a neutron physics calculation is to compute the reaction rates, therefore the neutron density or flux
- Neutron population is very large and treated as a whole by comparing its behavior to a fluid
- Equation formulated by Ludwig Boltzmann working on statistical mechanics in 1879
- Study and numerical processing of the Boltzmann equation for neutrons is one of the main challenges faced by neutron physicists

Boltzmann Equation (cont.)

- Change of the number of neutrons in the arbitrary “volume” considered during time is caused by:
  - Streaming
    the gains and losses of neutrons due to streaming, that is the neutrons that enter the volume from the outside or that leave the volume, during the time t.
  - Collisions
    the gains and losses of neutrons due to collisions
  - Sources
    the gains of neutrons due to the production within the volume

Cross Section vs. Energy



Computational challenge: ANGLE

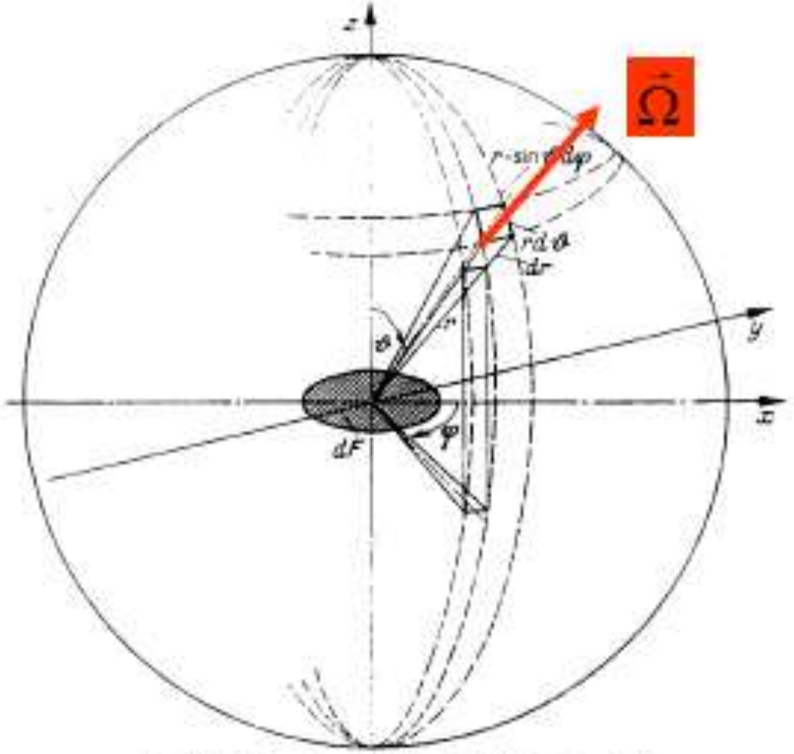
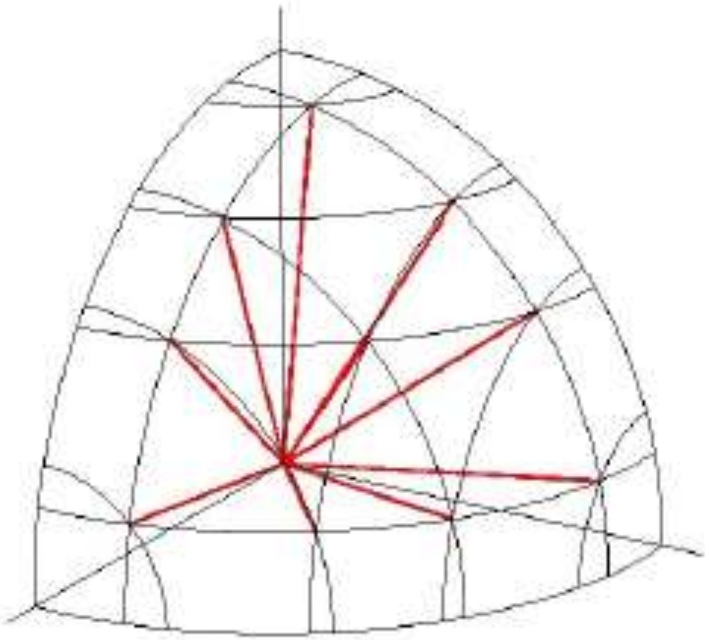


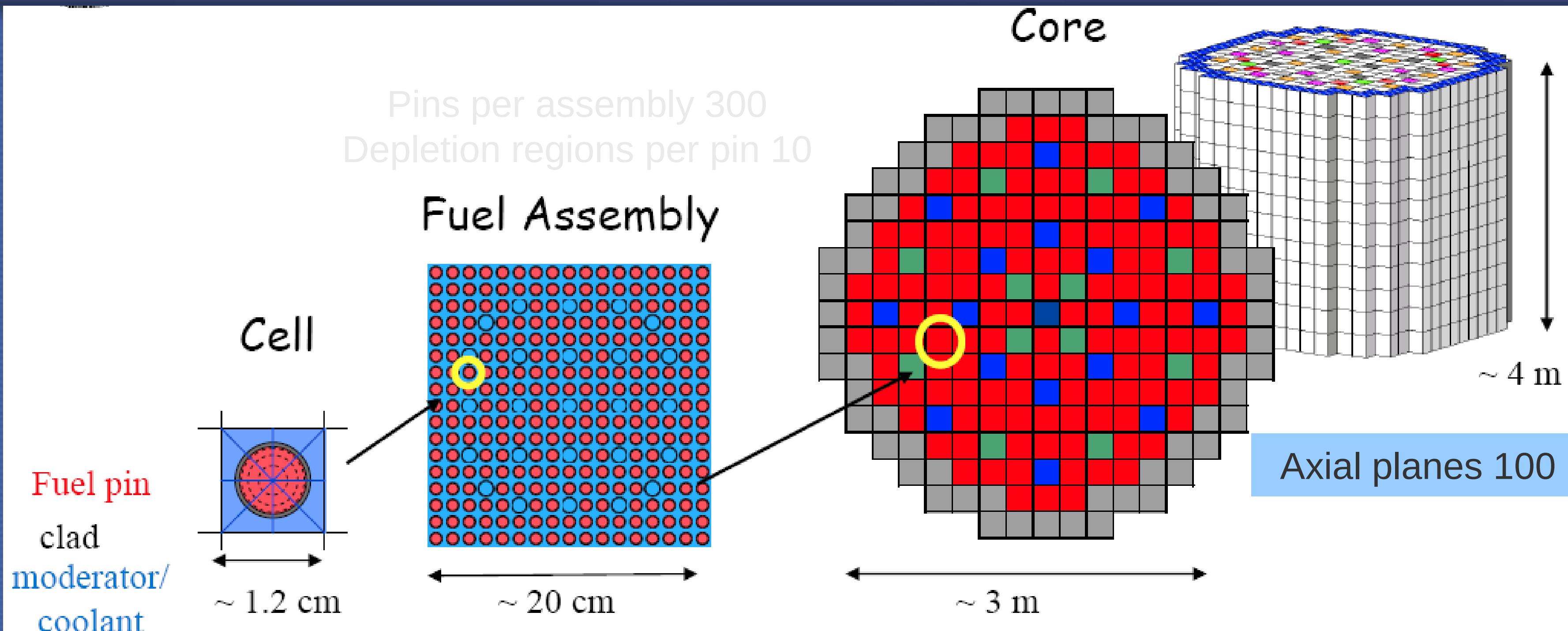
Fig. 6.1.1. Elementary derivation of FICK's law (see text)



about 80-200 angles

Scale of the LWR Core Problem

APoS



Axial planes 100

Number of fuel Assemblies 200

Angular directions 100
Neutron energy groups 100

To follow resonance absorption in heavy metals, one would need about 10,000 energy groups, with any prior knowledge

History of Reactor Physics

Simple Core Models

Four/Six Factor Formulas:

$$k_{eff} = (\eta_{th} \bullet f \bullet \varepsilon \bullet p) \bullet L_{th} \bullet L_{fast}$$

where,

$$\eta_{th} = \nu \frac{\Sigma_f^{fuel}}{\Sigma_a^{fuel}},$$ **Fuel thermal "eta"**

$$f = \frac{\Sigma_a^{fuel}}{\Sigma_a^{fuel} + \Sigma_a^{clad} + \Sigma_a^{\text{moderator}} + \ldots}$$ **Thermal utilization factor**

$\varepsilon =$ **Fast fission factor**

$p =$ **Resonance escape probability**

$L_{th} =$ **Thermal non-leakage probability (geometry)**

$L_{fast} =$ **Fast non-leakage probability (geometry)**

Diffusion Equation

- An approximation of the transport equation
- Applicable as netrons undergo a lot of collisins and interactions
- Few groups (in typical LWR solution) of neutrons computed
- Based on Ficks law

$$\bar{J} = -\frac{1}{3\Sigma_{tr}}\bar{\nabla}\phi = -D\bar{\nabla}\phi$$

Diffusion coefficient

- Not a very good approximation for large flux gradient
- Neglects boundary layers, so solving of additional equation is required

Multigroup Diffusion Equation

Two-step process.

1. Computing the multigroup cross-section from the microscopic cross-sections:
   - They are problem-dependent. They depend on the neutron spectrum, the temperature, the flux gradient, etc.
   - In order to obtain these cross-sections, we need to solve “local” (in space and/or energy) problems.
2. Solving the system of multigroup equations
   - Discretize the space
     (Finite elements, finite differences, finite volumes)
   - Discretize the angle
   - Solve the resulting linear system

In-Core Fuel Management

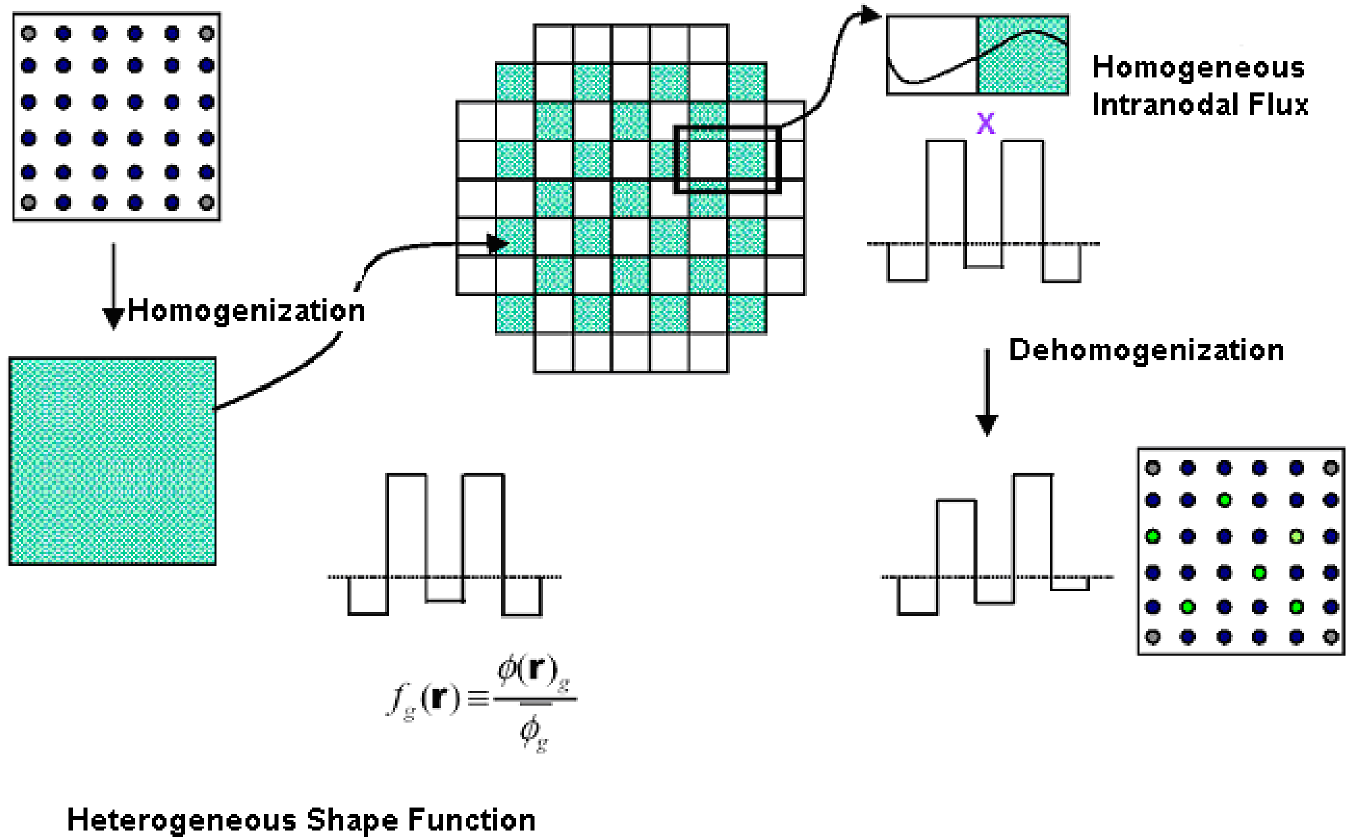
- Calculation of fuel depletion in realistic core conditions in multiple fuel cycles and taking into account certain design limitations
- Calculation should cover whole core or part of the core depending on the simetry
- All fuel assemblies should be included in depletion calculation and some kind of book keeping should be implemented (fuel assembly burnup and history effects)
- Depletion at the cell or fuel assembly level and depletion at core level are two different things

Current methodology: Divide & Conquer

- Instead of using brute force for the 3-D multigroup transport problem, the problem is split into different levels or scales (divide & conquer)
  - Assembly level
  - Core level
  - Additional step called homogenization

Assembly calculation > homogenization of results > core calculation > dehomogenization

Homogenization / Dehomogenization



Assembly Calculation

- Neutron in an assembly does not have knowledge of the boundaries really, most of them will die (capture or fission) before leaking
- The assembly is very long in axial dimension
  - Need to treat very precisely the assembly using 2-D fuel assembly computations in an infinite lattice
- Precisely means
  - Transport equation
  - Fine spatial mesh
  - Fine angular mesh
  - Fine energy mesh

Fuel Assembly Homogenization

- With the knowledge of the microscopic neutron distribution, we generate 2-group cross-sections to be used in the core level calculation.
- This step is called homogenization
  - Few energy groups
  - Coarser spatial representation

FA homogenization

pin-by-pin homogenization

Core Calculation

- We then use the homogenized cross-sections and solve the 3D problem with multigroupdiffusion theory
  - Fewer spatial meshes (40 planes x 200 assyx 4 ~ 32,000)
  - Few energy groups ~2
- ~ 60,000 unknowns
- It is fast [thousands of these computations are run for a reload]

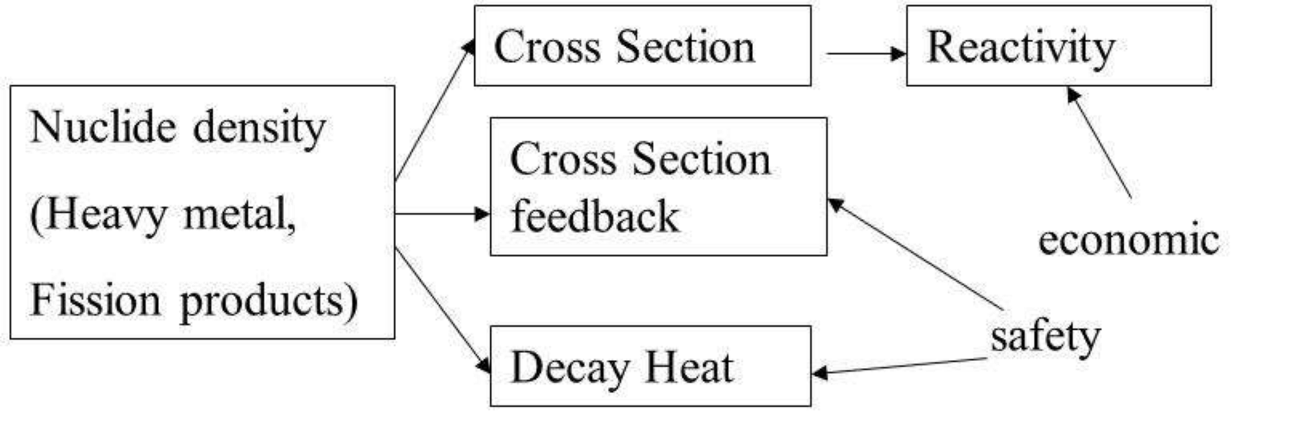
Cross Section (XS) Library

- Important part of any code used for core calculation
- Multi-parameter XS data homogenized at fuel assembly level
- Transport code (collision probability, discrete ordinates) used in cross section calculation (2D)
- Macroscopic base depletion calculation + branch points (variation of TH data)
- Burnup dependent XS library or material compositions based on some burnup distribution
- Multi linear or higher order interpolation of XS data
- Correction for history effects (burnup weighting for history variables)
- Version with and without inserted control rods needed

Depletion

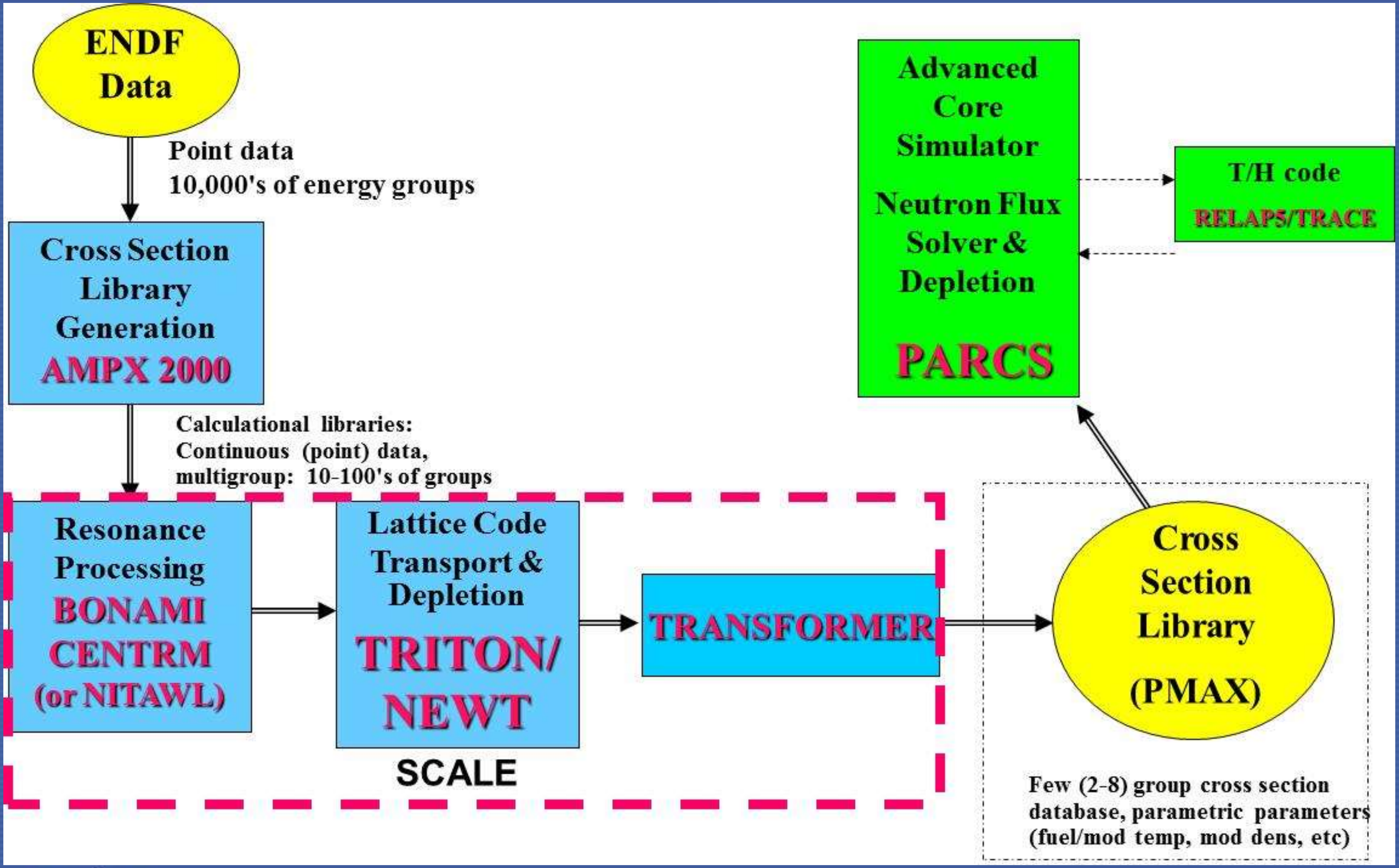
- Nuclide density change in nuclear reactor core when operated at power

- Related changes



Depletion code system must solve coupled nuclide/neutron and temperature/fluid field equations

Example of Full Calculation Cycle

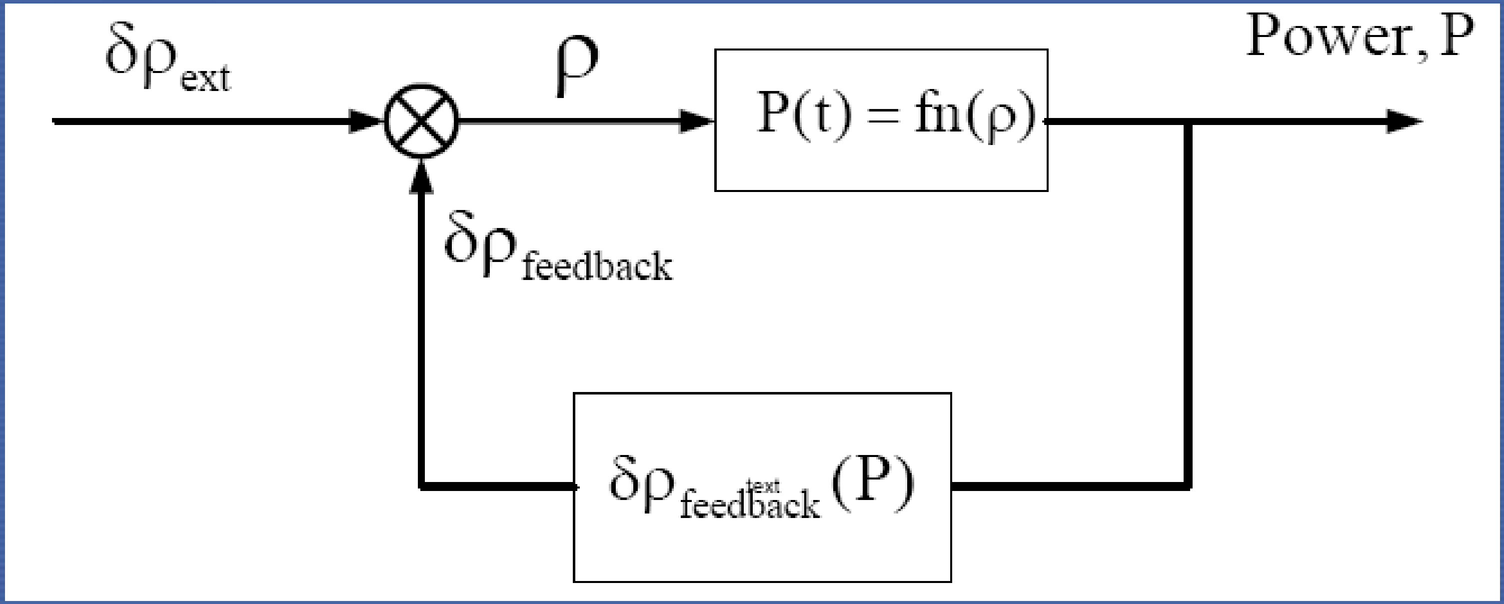


Dynamic Core Calculation (Kinetics)

- Point Kinetics
  - Calculation of core averaged point kinetics data (TH feedback reactivity tables and delayed neutron data)
  - It is important to calculate point kinetics data to be consistent with fuel reload calculations
- 3D Neutronics
  - Use of nodal codes that are based on differential equations in which neutrons are not intrinsically treated as particles, but as fields.
  - Use of discontinuity factor to overcome discontinuity problem at boundaries
  - Cross-section libraries characterize 3D space of the core

Reactivity Feedback

- In accident and transient conditions, reactor power is characterized by so called reactivity coefficients that describe impact of various preturbations on the chain reaction within the core



NS-G-1.12 Requirements on Reacivity Coefficients

3.38. On the basis of the geometry and the fuel composition of the reactor core, the nuclear evaluations for design provide steady state spatial distributions of neutron flux and of the power, core neutronic characteristics and the efficiency of the means of reactivity control for normal operation of the plant at power and at shutdown conditions.

3.40. Key reactivity parameters such as reactivity coefficients should be evaluated for each core state and for the corresponding strategy for fuel management Their dependence on the core loading and the burnup of fuel should be taken into account.

Reactivity Coefficients

- The kinetic characteristics of the reactor core determine the response of the core to changing plant conditions or to operator adjustments made during normal operation, as well as the core response during abnormal or accidental transients.

- These kinetic characteristics are quantified in reactivity coefficients.

- The reactivity coefficients reflect changes in neutron multiplication due to varying plant conditions such as power, moderator or fuel temperatures, introduction of control rods, change of soluble boron concentration, or (less significantly) a change in pressure or void fraction.

Temperature Reactivity Coefficients

- The change in reactivity per degree change in temperature is called the temperature coefficient of reactivity.
- Different materials in the reactor have different reactivity changes with temperature and the various materials are at different temperatures during reactor operation, so several different temperature coefficients are used.
- Two dominant temperature coefficients are the moderator temperature coefficient and the fuel temperature coefficient.

Moderator Temperature Coefficient

- The change in reactivity per degree change in moderator temperature.
- The magnitude and sign (+ or -) of the moderator temperature coefficient is primarily a function of the moderator-to-fuel ratio:
  - under moderated reactor has negative moderator temperature coefficient.
  - over moderated reactor has positive moderator temperature coefficient.
- A negative moderator temperature coefficient is desirable because of its self-regulating effect.

Moderator Temperature Coefficient (cont.)

- Water-moderated reactors are designed to operate in an under moderated condition.
- The soluble boron used in the reactor has an effect since its concentration is increased when the coolant temperature is lowered (if the concentration is large enough, the net value of the coefficient may be positive).
- With burnup, the moderator temperature coefficient becomes more negative primarily as a result of the reduced boron concentration but also to a lesser extent from the effects of the buildup of plutonium and fission products.

Fuel Temperature Coefficient

- The fuel temperature coefficient is the change in reactivity per degree change in fuel temperature.
- It is negative for LWR since increase of fuel temperature increases the neutron resonance absorption cross-section of U^{238} (the Doppler effect).
- Fuel temperature coefficient, has a greater effect than the moderator temperature coefficient because an increase in reactor power causes an immediate change in fuel temperature.

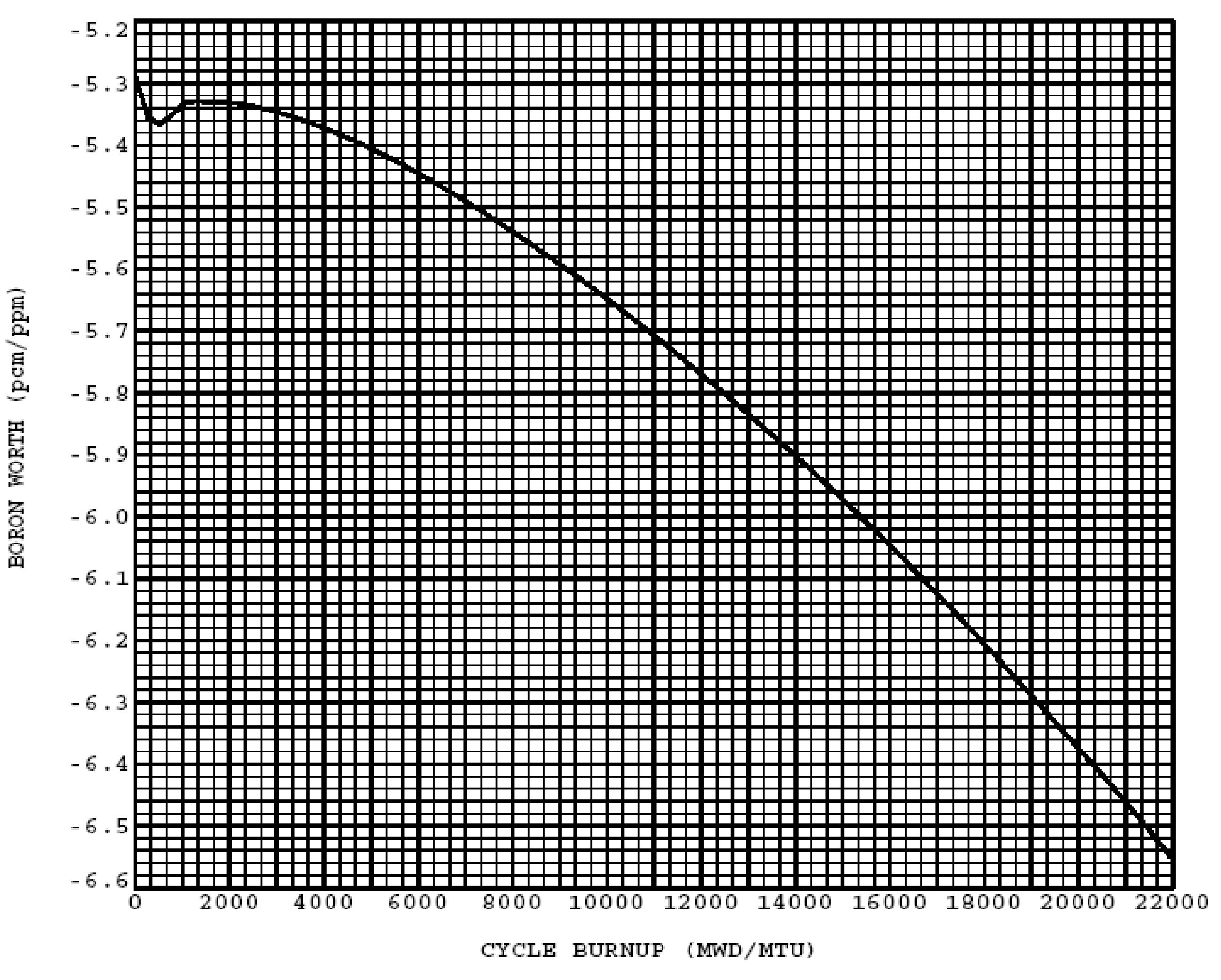
Pressure (Densitiy) Coefficient

- Pressure coefficient of reactivity is defined as the change in reactivity per unit change in pressure.
- It is the result of the effect of pressure on the density of the moderator so it is sometimes referred to as the moderator density reactivity coefficient.
- In reactors that use water as a moderator, the absolute value of the pressure reactivity coefficient is seldom a major factor because it is very small compared to the moderator temperature coefficient of reactivity.

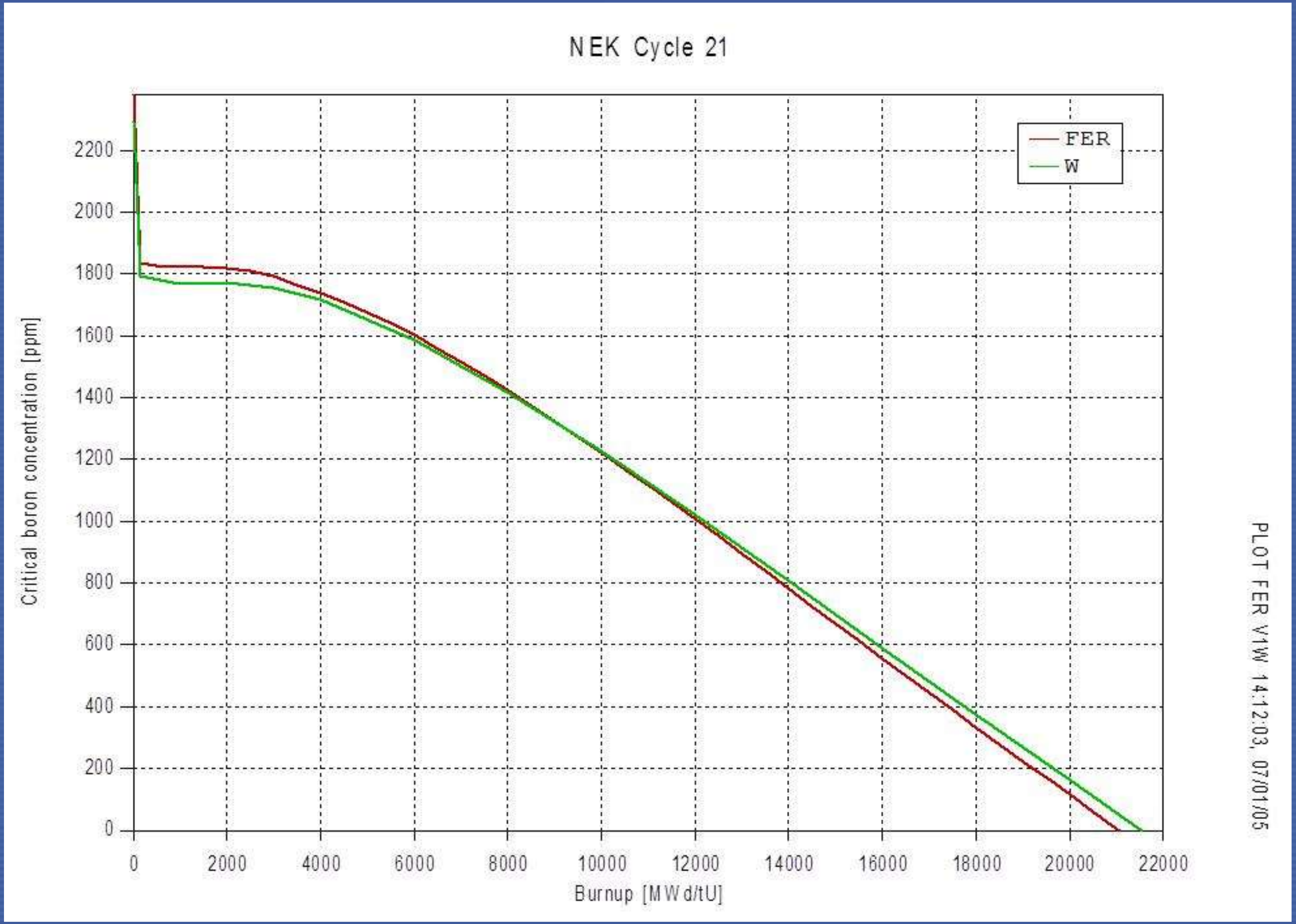
Boron Coefficients (Boron Worth)

- Defined as the change in reactivity due to a unit change in boron concentration.
- Primarily a function of the ratio of boron absorption to total absorption.
- Because the boron coefficient is a strong function of boron absorption in the thermal energy range, the magnitude of the boron coefficient also varies inversely with the fast-to-thermal flux ratio.

Boron Worth versus Burnup



Critical boron concentration

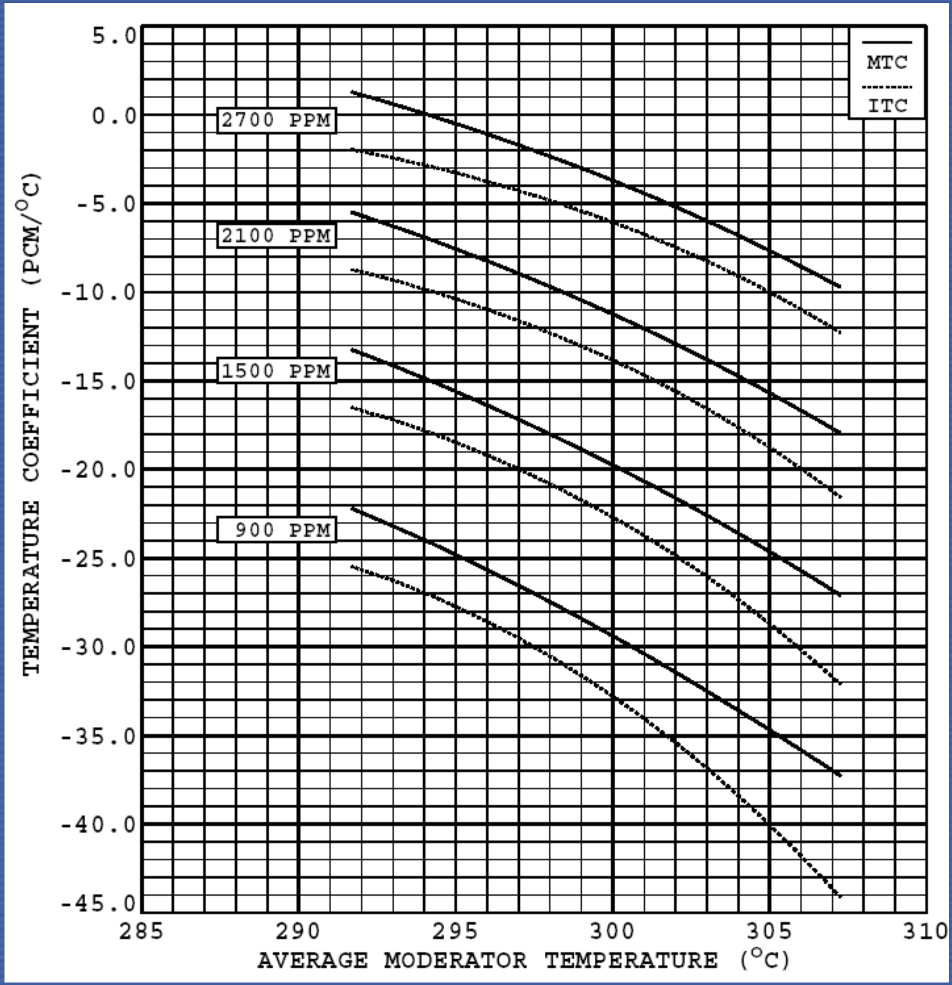


Core Coefficients

	Cycle 20	Cycle 21
Boron Concentration (ppm)		
HZP-0 MWD/MTU, No Xe, Peak Sm, ARO	2497	2581
HZP-0 MWD/MTU, No Xe, Peak Sm, Control Bank D in	2353	2451
HFP-0 MWD/MTU, No Xe, Peak Sm, ARO	2221	2290
HFP-150 MWD/MTU, Eq. Xe, ARO	1743	1794
HZP-0 MWD/MTU, ARI, $K \leq 0.984$	≥1660 *	≥1912 *
Refueling C_B, ARI, $K \leq 0.95$ (21°C)	≥2483 *	≥2713 *
Technical Specification Refueling C_B	3000	3000
Control Rod Worths (%Δρ)		
HZP-0 MWD/MTU, Control Bank D In, No Xenon	0.82	0.69
HZP-0 MWD/MTU, All Rods In, No Xenon	6.85	5.60
HFP-150 MWD/MTU, Control Bank D In	0.86	0.80
HZP-EOL (21957 MWD/MTU), All Rods In, with HFP Eq. Xe	7.51 [@]	7.54
Moderator Temperature Coefficients (pcm/°C)		
HZP-0 MWD/MTU, All Rods Out, No Xenon	-3.31	-0.58
HFP-150 MWD/MTU, All Rods Out, Eq. Xe	-25.20	-22.47
HFP-EOL (21550 MWD/MTU), All Rods Out, Eq. Xenon	-71.71 [#]	-72.89
Doppler Temperature Coefficients (pcm/°C)		
HZP-0 MWD/MTU, All Rods Out, No Xenon	-3.30	-3.24
HFP-18698 MWD/MTU, All Rods Out, Eq. Xenon, 300 ppm	-11.00 [&]	-10.74
Nuclear Enthalpy Rise Hot Channel Factor		
HZP-0 MWD/MTU, All Rods Out, No Xe	1.55	1.51
HFP-0 MWD/MTU, All Rods Out, No Xe	1.49	1.46
HFP-150 MWD/MTU, Control Bank D in, Eq. Xe	1.51 **	1.54 **

Temperature Coefficients

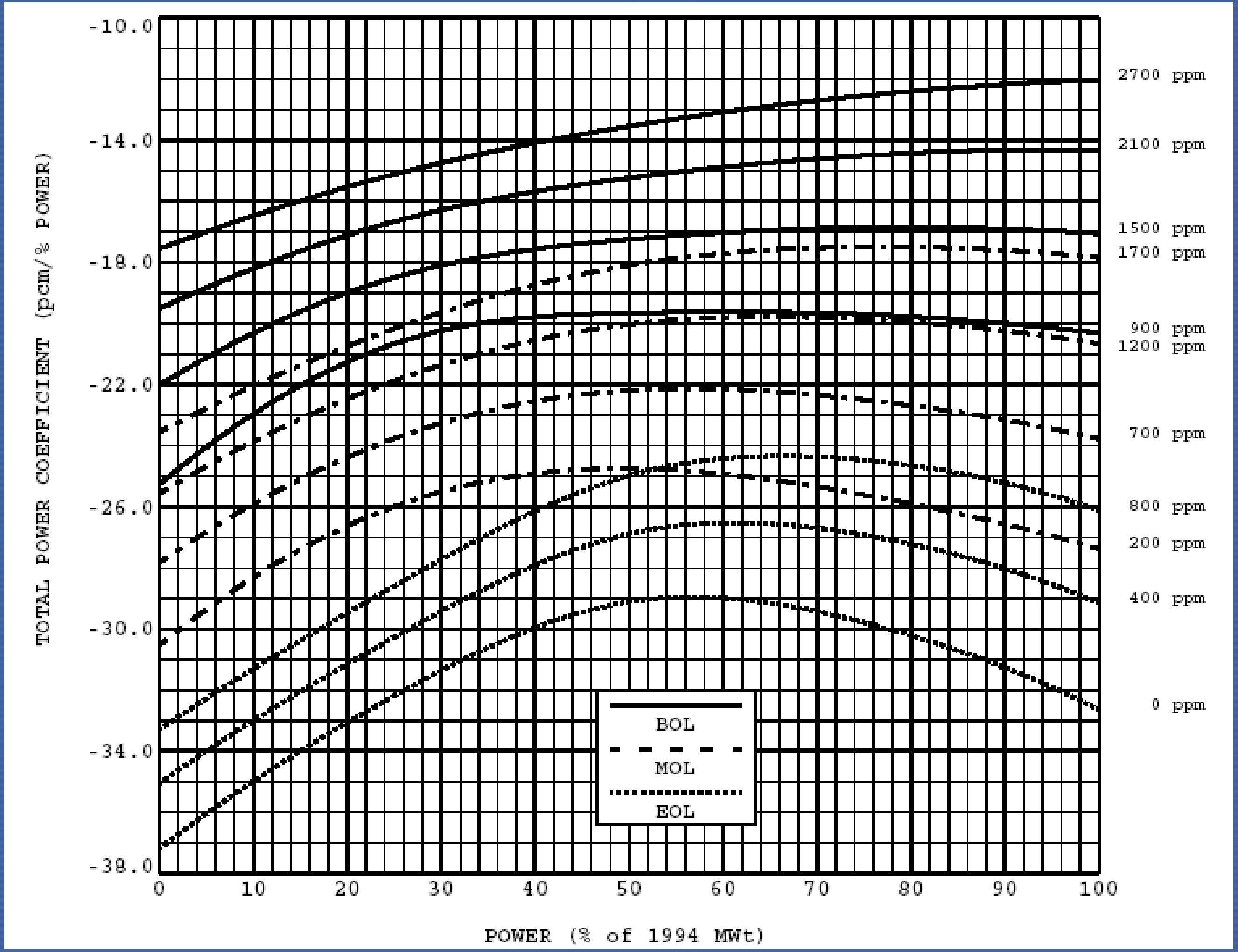
at 150 MWD/MTU



Total Power Coefficients and Defects

- The combined effect of moderator temperature changes, fuel temperature changes, and axial reactivity redistribution as the core power level changes is called the total power coefficient and is expressed in terms of reactivity change per percent power change.
- Calculated values of the total power coefficient are useful for predicting the behavior of the core during small changes of the core power.
- The total power defect is the integral of the total power coefficient over the appropriate power range.
- Calculations of the total power defect as a function of power level, cycle burnup and boron concentration permit the prediction of the behavior of the core during changes in core power level and the compensating changes to the boron concentration or the control rod positions.

Total Power Coefficient vs Power Level at BOL, MOL, and EOL



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Reactivity Redistribution Defects

- The Doppler and moderator defects do not capture all of the reactivity changes associated with a change to core power levels and core temperature distribution.
- There are reactivity differences due to changes in core axial power.
- The redistribution defect is defined as the difference between the total power defect and the sum of the Doppler and the moderator defects.
- The axial reactivity redistribution defect results from changes in axial power distribution which accompany changes in core power level.

Reactivity Redistribution Defects

- At HZP conditions the moderator temperature is uniform at all elevations.
- When generating power, the moderator temperature increases along the core height.
- At HZP, the very top-skewed flux distribution gives a high importance weighting to the more reactive top region of the core.
- At HFP, the more symmetric flux distribution gives a high importance weighting to the less reactive center region of the core. This difference serves to make the core more reactive at HZP than at HFP. The difference is the reactivity redistribution defect.

Reactor Physics in Safety Analysis Report Chapter 15

- 15.0 – General information on the analysis
  - Plant Characteristics and Initial Conditions Assumed in the Accident Analyses
  - Reactivity Coefficients Assumed in the Accident Analysis
  - Rod Cluster Control Assembly Insertion Characteristics
  - Protection and Safety Monitoring System Setpoints and Time Delays to Trip Assumed in Accident Analyses
  - Instrumentation Drift and Calorimetric Errors, Power Range Neutron Flux
  - Plant Systems and Components Available for Mitigation of Accident Effects
  - Fission Product Inventories
  - Residual Decay Heat
  - Computer Codes Used

PWR Plant FSAR interfaces

APòS

Chapter 13: Conduct of Operations

Chapter 14: QA

Organizational
Training
Emergency Planning
Review and Audit

Chapter 15: Accident Analyses

- Assumptions and inputs (table 15.0.3-2a/b)
- Computer codes and initial conditions summary (table 15.0.3-3)
- Trip setpoints and time delays (table 15.0.6-1)
- Systems and Equipment Available for Transients and Accidents (table 15.0.8-1)
- Source Term (table 15.0.9-1)
- Atmospheric Conditions (table 15.0.12-1)
- Single Failures Assumed in Accident Analyses (table 15.0.13-1)

Chapter 2: Site Characteristics
Chapter 4: Reactor (T&H Design)
Chapter 5: RCS

Chapter 6: Engineering Safety Features

- Containment
- ECCS (SI and RHR)
- MCR Habitability Systems

Chapter 3: Design of SSC

- Classification
- Misile Protection
- Seismic Design
- Environmental Design
- Flooding, Wind, etc.

Support Systems

- **Chapter 7: Instrumentation and Control (RTS and ESFAS)**
- **Chapter 8: Electrical Systems**
- **Chapter 9: Auxiliray systems**
- **Chapter 10: Steam and Power Conversion systems**
- **Chapter 11: Radioactive Waste Management**

Normal Operations

HEAT REMOVAL AND PRODUCTION

Normal Production: - Core (Fission & Decay Heat)

Normal Removal:
- Heat to the RCS
- to the Steam Generators
- to the Turbine

INHERENT NEGATIVE REACTIVITY

Additional heat absorbed by the fuel, clad, RCS, and the S/Gs

- RCS and Fuel Temperature adds negative reactivity
- Reactor goes subcritical
- Heat production decreases
- Heat Production = Heat Removal
- Criticality is not restored

Normal Operations

CONTROL SYSTEMS

HEAT MISMATCH > Inherent Reactivity Control

ROD CONTROL SYSTEM

REACTOR PROTECTION

HEAT MISMATCH > Inherent Reactivity Control
& Control Systems

REACTOR TRIP

ENGINEERED SAFETY FEATURES SYSTEMS

If normal-operation heat removal systems fail
Temperatures decrease
Fission product barriers are protected

Acceptance Criteria

Specific acceptance criteria for AOOs, examples from NUREG-0800:

- Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code.

- Fuel cladding integrity shall be maintained by ensuring that the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit for PWRs.

- An AOO should not generate a postulated accident without other faults occurring independently or result in a consequential loss of function of the RCS or reactor containment barriers.

Selection Of Initial And Boundary Conditions

Reactivity feedback depends on the direction of the change (increase or decrease) of the parameter under consideration.
The direction may change during the course of the accident, and therefore the influence of feedback coefficients may also vary during the process.

TABLE II. CONSERVATIVE SELECTION OF NEUTRONIC PARAMETERS LEADING TO OVERESTIMATION OF REACTOR POWER

Parameter change	Reactivity feedback: Fuel temperature coefficient (FTC)	Reactivity feedback: MTC[a] + void	Reactivity feedback: Boron concentration coefficient (BCC)	Fraction of delayed neutrons	Prompt neutron lifetime
Increase of coolant temperature	Strong	Weak	Weak	Max.	Max.
Decrease of coolant temperature	Weak	Strong	Weak	Min.	Min.
Reactivity increase by CRs	Weak	Weak	Weak	Min.	Min.
Reactivity decrease by CRs	Strong	Strong	Weak	Max.	Max.
Void fraction in the core during LOCA	Strong	Weak	Strong	Max.	Max.
Boron dilution	Weak	Weak	Strong	Min.	Min.

[a] MTC: moderator temperature coefficient.

Selection Of Initial And Boundary Conditions

- In Table II, ‘weak’ means minimum absolute value of a feedback coefficient and ‘strong’ means maximum absolute value of a feedback coefficient. Table II is only illustrative. The selected parameters need to be checked carefully for their influence on the results of the analysis, case by case before each application.

Questions?

- References:
- T. Bajs presentation at IAEA Safety Assessment Essential Knowledge Workshop: JNRC, Amman, Jordan, 10 - 14 November 2013
- I. Basic various presentations at IAEA Safety Assessment Essential Knowledge Workshop
- IAEA SF-1, Safety Principles
- IAEA NS-G-1.2, Design of the Core for Nuclear Power Plant Safety assessment
- IAEA SSR-2.1 Special Safety Requirements