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Review of Reactor Core Analysis

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I. Overview

1. Introduction

2. Scope

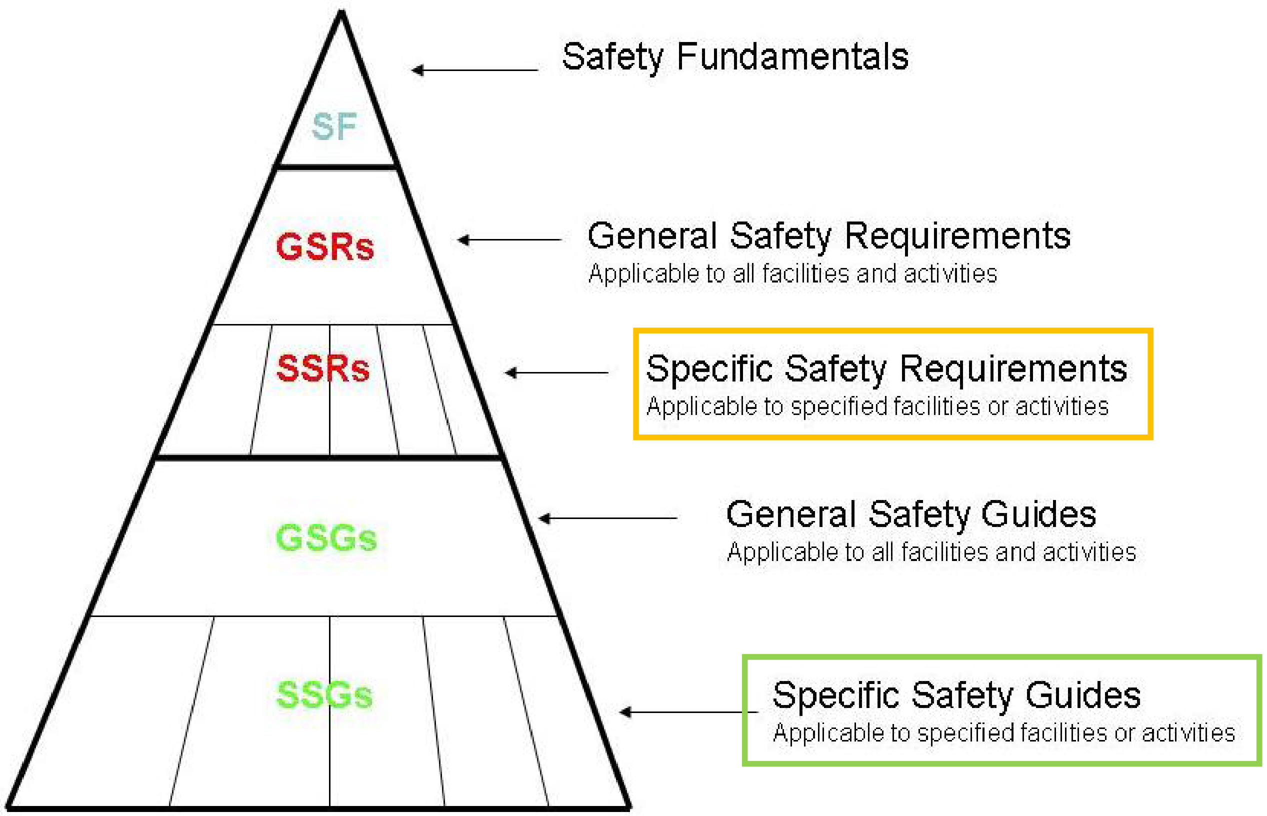
1. Introduction

□ Research Reactor? (1)

- IAEA
  - Safety of Research Reactors, Specific Safety Requirements NS-R-4
    - A research reactor is a nuclear reactor used mainly for the generation and utilization of radiation for research and other purposes, such as the production of radioisotopes.
    - This definition excludes nuclear reactors used for the production of electricity, naval propulsion, desalination or district heating.
    - The term covers the reactor core, experimental devices and all other facilities relevant to either the reactor or its associated experimental devices located on the reactor site.

1. Introduction

– NS-R-4 in IAEA Safety Standards Categories



1. Introduction

□ Research Reactor? (2)

- Korea
  - The Nuclear Laws of Korea does not define what is a research reactor.
  - If a reactor is described as a “research reactor” in the reactor type of the application documents for construction and operation permit, it is a research reactor.
    - The application documents would be reviewed by the regulatory authority, and whether the type of a research reactor is proper would be determined.

1. Introduction

□ Research Reactor? (3)

- US
  - 10CFR50
    - Nuclear reactor means an apparatus, other than an atomic weapon, designed or used to sustain nuclear fission in a self-supporting chain reaction.

<table>
<tr><th colspan="2">Type</th><th>Explanation</th></tr>
<tr><td colspan="2">Power Reactor</td><td>Commercial purpose</td></tr>
<tr><td rowspan="2">Non-power Reactor</td><td>Research Reactor</td><td>A nuclear reactor for operation at a thermal power level of 10 megawatts or less, and which is not a testing facility.</td></tr>
<tr><td>Test Reactor</td><td>Under the category of special projects</td></tr>
<tr><td colspan="2">Prototype Plant</td><td>A nuclear reactor that is used to test design features. The prototype plant is similar to a first-of-a-kind or standard plant design in all features and size, but may include additional safety features to protect the public and the plant staff from the possible consequences of accidents during the testing period.</td></tr>
</table>

1. Introduction

□ Research Reactor? (4)

- Japan
  - New Regulation Standard and Its Interpretation (by NRA)

Type	Explanation
Critical Experiment Facility	Easily changing the core structure
Water-cooled Research Reactor	Water as a primary coolant
Gas-cooled Reactor	Gas as a primary coolant
Sodium-cooled Reactor	Sodium as a primary coolant, Fast neutron used
Floating Nuclear Plant	Nuclear propulsion for ship

1. Introduction

□ Example of Research Reactors

	HANARO	JRTR	OPAL	JRR-3M	JRTR
Country	Korea	Jordan	Australia	Japan	Korea
Thermal Power (MWth)	30	5	20	20	15
Type (FA:Fuel Assembly)	Rod (32 FA)	Plate (18 FA)	Plate (16 FA)	Plate (31 FA)	Plate (22 FA)
Components	U_3Si	U_3Si_2	U_3Si_2	U_3Si_2	UMo
Uranium density (g-U/cc)	4.8	4.8	4.8	4.8	8.0

2. Scope

□ Review Category

- Chapter 5. Reactor of SAR (Safety Analysis Report)
  - Fuel design

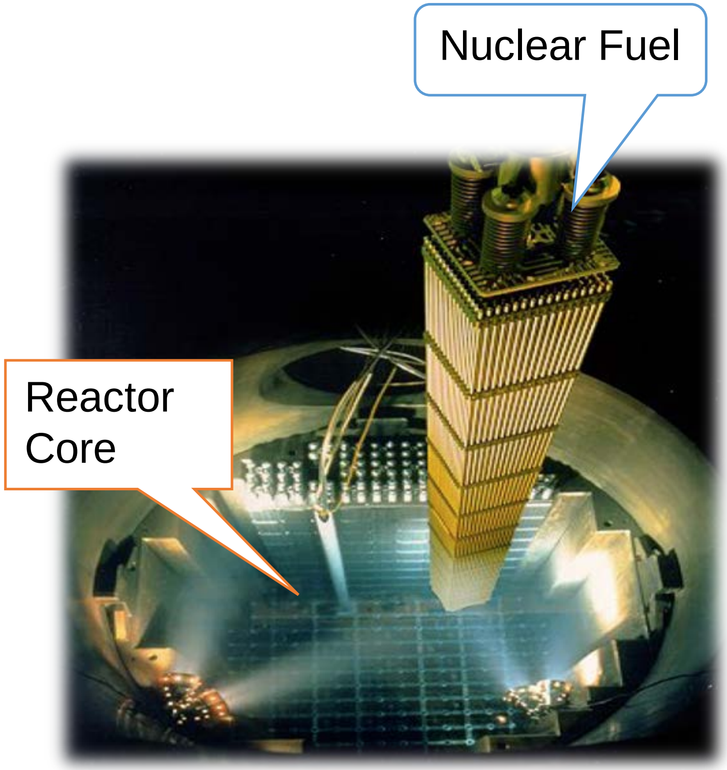
    → Nuclear fuel analysis

  - Nuclear design

    → Reactor core analysis

  - Thermal-hydraulic design

    → Reactor core thermal-hydraulic analysis



II. Nuclear Fuel Analysis

1. Nuclear Fuel?

2. Requirements

3. Design Limit

4. Review of Fuel Analysis

1. Nuclear Fuel?

□ Example of Fuel

JRTR(**Plate**)

HANARO (**Rod**)

CANDU

PWR

1. Nuclear Fuel?

□ Nuclear Fuel Components

- For research reactor fuels
  - Fuel assembly (e.g. Al alloy)
    - Fuel plate
      - Fuel meat (e.g. U_3Si_2 + Al, metal)
      - Cladding (e.g. Al alloy)

- For power reactor fuels
  - Fuel assembly (e.g. Fe alloy)
    - Fuel rod
      - Pellet (e.g. UO_2, ceramic)
      - Cladding (e.g. Zr alloy)

1. Nuclear Fuel?

□ History of Fuel Development

- Characteristics

1st Gen.
- ~`88
- Over 90% (HEU)
- 1.5 g-U/cm^3
- U_3O_8 or UAl_x

2nd Gen.
- `90s ~ `00s
- Less 20% (LEU)
- 4.8 g-U/cm^3
- U_3Si_2 or U_3Si

3rd Gen.
- ~ now and future
- Less 20% (LEU)
- Over 8 g-U/cm^3
- U-Mo

KINS

2. Requirements

□ Regulatory Requirements (1)

- Regulations on Technical Standards for Nuclear Reactor Facilities, Etc.
  - **Article 13** (External Events Design Bases)

    (2) **Design bases** as regards structures, systems, and components important to safety shall consider each of the following:

    1. The most severe natural phenomena and man-induced external events **considering the historical records** for the relevant site and surrounding areas;

    2. Combination of the effects of normal operations or accident conditions **with the effects of natural phenomena and/or man-induced external events**, considering the probability of concurrent occurrences thereof;

    3. The importance of safety functions to be performed; and

    4. Appropriate provisions to defend against the third party access to reactor facilities in the design of the buildings and site layout.

2. Requirements

□ Regulatory Requirements (2)

- **Article 15** (Environmental Effects Design Bases, etc.)
  (2) The following components shall be installed in such a way that **prevents any damage caused by vibrations resulting from the circulation**, boiling, and etc. of primary or secondary coolants: fuel assembly, moderators, reflectors, and associated supports; thermal shields; and vessels, pipes, pumps, and valves that are part of primary coolant system.

- **Article 17** (Reactor Design)
  (1) The reactor core and associated coolant system, control system, and protection system shall be designed with appropriate margins to assure that **specified acceptable fuel design limits are not exceeded during normal operation conditions and anticipated operational occurrences.**

2. Requirements

□ Regulatory Requirements (3)

– **Article 35** (Reactor Core, etc.)
  Reactor core, and components adjacent to it within the reactor pressure vessel shall be designed to **withstand the loadings due to pressure, temperature, and radiation expected to occur in normal operation conditions, anticipated operational occurrences, and design basis accidents**, in appropriate combinations with the effects of earthquake, within the design basis to the extent necessary to **ensure the safe shutdown of the reactor and cooling of the core**.

2. Requirements

□ Safety Review Guidelines (1)

- Acceptance criteria of NUREG-1537*
  - The design bases for the fuel should be clearly presented, and the design considerations and functional description should ensure that fuel conforms with the bases. Maintaining fuel integrity should be the most important design objective.
  - The chemical, physical, and metallurgical characteristics of the fuel constituents should be chosen for compatibility with each other and the anticipated environment.
  - Fuel enrichment should be consistent with the requirements of 10 CFR50.64.

* NUREG-1537 Part2, Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Standard Review Plan and Acceptance Criteria (US.NRC, 1996)

2. Requirements

□ Safety Review Guidelines (2)

- The fuel design should take into account characteristics that could limit fuel integrity, such as heat capacity and conductivity, melting, softening, and blistering temperatures; corrosion and erosion caused by coolant; physical stresses from mechanical 'or hydraulic forces (internal pressures and Bernoulli forces); fuel burnup; radiation damage to the fuel and the fuel cladding or containment; and retention of fission products.

- The fuel design should include the nuclear features of the reactor core, such as structural materials with small neutron absorption cross-sections and minimum impurities, neutron reflectors, and burnable poisons, if used.

2. Requirements

□ Safety Review Guidelines (3)

- The discussion of the fuel should include a summary of the fuel development and qualification program.
- The applicant should propose technical specifications as discussed in Chapter 14 of the format and content guide to ensure that the fuel meets the safety-related design requirements. The applicant should justify the proposed technical specifications in this section of the SAR.

2. Requirements

□ Safety Review Guidelines (4)

- Acceptance criteria of IAEA Safety Requirements NS-R-4*
  - 6.79. Appropriate neutronic, thermal-hydraulic, mechanical, material, chemical and irradiation related considerations associated with the reactor as a whole shall be taken into account in the design of fuel elements and assemblies, the reflectors and other core components.
  - 6.80. Analyses shall be performed to show that the intended irradiation conditions and limits (such as fission density, total fissions at the end of lifetime and neutron fluence) are acceptable and will not lead to undue deformation or swelling of the fuel elements. The anticipated upper limit of possible deformation shall be evaluated. These analyses shall be supported by data from experiments and from experience with irradiation. Consideration should be given in the design of the fuel elements to the requirements relating to the long term management of irradiated elements.

* NS-R-4, Safety of Research Reactors (IAEA, 2005)

2. Requirements

□ Safety Review Guidelines (5)

- 6.82. The reactor core (i.e. the fuel elements, reflectors, cooling channel geometry, irradiation devices and structural parts) shall be designed to maintain the relevant parameters within specified limits in all operational states. There shall be provisions in the design to monitor the integrity of the fuel. In the event of the detection of fuel failure, an investigation shall be conducted to identify the failed fuel element. Authorized limits shall not be exceeded (see also paras 7.96–7.102) and if necessary the reactor shall be shut down and the failed fuel element shall be unloaded from the core.
- 6.83. The reactor core shall be designed so that fuel damage in DBAs would be kept within acceptable limits.

2. Requirements

□ Safety Review Guidelines (6)

- 6.84. The reactor core, including fuel elements, reactivity control mechanisms and experimental devices, shall be designed and constructed so that the permissible design limits that are specified for all operational states are not exceeded. A suitable margin, including margins for uncertainties and engineering tolerances, shall be incorporated in setting these limits.

2. Requirements

□ Safety Review Guidelines (7)

- Acceptance criteria of IAEA Safety Requirements SSG-20*
  - 2.17. In the development of the acceptance criteria, consideration should be given to the criteria listed below:

    (b) Nuclear fuel performance criteria:

    · Maximum cladding temperature below blistering temperature;

    · Maximum heat flux not exceeding the critical heat flux during a transient;

    · Maximum heat flux not exceeding the onset of significant voiding during a transient;

    · Flow conditions not exceeding the onset of flow instability;

    · Frequency limits for significant damage to fuel cladding.

* SSG-20, Safety Assessment for Research Reactors and Preparation of the Safety Analysis Report (IAEA, 2012)

2. Requirements

□ Safety Review Guidelines (8)

- Content of a Safety Analysis Report by IAEA SSG-20
  - A.2.4. The specific design requirements applied should be stated in this section.

    (10) Fuel design limits and materials design criteria, including:

    (a) Fuel design bases for mechanical, chemical and thermal design;

    (b) Safety margins for fuel design parameters;

    (c) Methods of achieving a conservative safety margin for prototypical fuels;

    (d) Verification of fuel integrity;

    (e) Design bases for mechanical, thermal and chemical design of reactor materials important to safety.

2. Requirements

□ Safety Review Guidelines (9)

– A.5.4. Basic information on fuel design and fuel properties should comprise:

(a) Fuel material, enrichment, composition and metallurgical state (oxide,alloy, etc.);

(b) Material (type, composition, etc.) of all other fuel parts, such as cladding, spacers and fittings, and burnable neutron absorbers;

(c) Fuel geometry, dimensions, tolerances, etc. (together with drawings);

(d) The material properties required for the analyses mentioned in paras A.5.5–A.5.8;

(e) The maximum temperatures to which the fuel elements can be subjected without deformation (due to blister formation or mechanical weakening);

(f) Fuel element instrumentation, if any.

2. Requirements

□ Safety Review Guidelines (10)

- A.5.5. An analysis should be provided that shows that the fuel elements can withstand the thermal conditions to which they are subjected throughout their normal operational life cycle. This life cycle should comprise not only nuclear applications in the reactor core but also the periods of storage, handling and transport.

- A.5.6. An analysis should be provided that shows that the fuel elements can withstand the mechanical forces to which they are subjected (hydraulic forces, differential thermal expansion effects, etc.) without breach of mechanical integrity or undue deformation. The anticipated effects should be quantified.

2. Requirements

□ Safety Review Guidelines (11)

– A.5.7. An analysis should be provided that shows that the fuel element cladding can withstand the chemical environment to which is subjected during use and storage, with account taken of the effects of temperature and irradiation.

– A.5.8. An analysis should be provided that shows that the intended irradiation conditions and limits (fission, density, total fissions at the end of lifetime, etc.) are acceptable and will not lead to undue deformation or swelling of components that may contain fissile material. The anticipated upper limit of the eventual deformation (e.g. expressed as minimum cooling channel width) should be provided for the thermal safety analysis.

– A.5.9. These analyses and this information should be supported by a report on experimental measurements and irradiation experience, and should include the entire fuel cycle (storage, transport, etc.).

2. Requirements

□ Safety Review Guidelines (12)

- Item to be considered in the description by IAEA SSG-20
  - III–4. The fuel used, including the uranium enrichment and the type of fuel, needs to be specified. The description of the fuel element, supported by drawings, and the main characteristics of the fuel elements are to be presented, such as:
    (a) Thickness of cladding;
    (b) Length of active zone;
    (c) Width of coolant channel;
    (d) Number of fuel plates and/or pins;
    (e) Cladding material;
    (f) Uranium loading.
  - If fuel elements are used that contain channels for the movement of neutron absorbing blades or neutron absorbing rods, they are to be described in the same section. A summary of the experience with the fuel is a part of the section regarding the fuel elements.

3. Design Limit

□ Design Parameters for Fuel Integrity

- Fuel plate
  - Swelling, Blistering, Corrosion
- Fuel assembly
  - Stress, Vibration
- Core Coolability
  - Swelling, Blistering, Critical heat flux

3. Design Limit

□ Design Limit for KJRR (1)

Parameter	Design requirement	Design limits
Swelling	Swelling shall not cause a significant narrowing of the channel gap.	
Blistering	Bonding shall be ensured between fuel meat and cladding.	
Oxidation	Oxidation growth shall be limited to prevent spallation.	
Critical heat flux	Heat flux shall be limited to prevenent critical heat flux.	

3. Design Limit

□ Design Limit for KJRR (2)

Parameter	Design requirement	Design limits
Stress during normal and AOO	Stress shall be low enough to maintain the structural integrity of fuel	
Stress during SSE	The coolable geometry should be maintained.	
Vibration/ Dynamic	Coolant velocity should not cause hydraulic instability of fuel plates, and avoid resonance vibration.	

P_m : Primary Membrane Stress, P_b : Primary Bending Stress, Q : Secondary Stress
S_m : Design Stress Intensity, S_u : Ultimate Tensile Strength

4. Review of Fuel Analysis

□ Review Practice (1)

- The design criteria for the fuel must be based on sufficient test data to maintain the integrity of the fuel and to ensure the core cooling.

- A methodology that has a proven model and adequate conservativeness should be used for fuel design analysis.

- Nuclear fuel assemblies should be proved to meet the design criteria given in the scope of licensing burnup through verification of performance in the in-pile tests.

4. Review of Fuel Analysis

□ Review Practice (2)

- Points
  - Validity of Design Criteria
  - Appropriateness of Test Data
  - Suitability of Design Evaluation Methodology
  - Verification through In-reactor tests

4. Review of Fuel Analysis

□ Conclusion of Review for KJRR

- The design limits are compatible with
  - Regulations on Technical Standards for Nuclear Reactor Facilities, Etc (Article 13, 15, 17, 35)
  - KINS/GE-N10, Safety Review Guide for Research and Educational Reactors, Chapter 5.2 (Nuclear Fuel Design)

- Acceptable fuel design limits are satisfied.

III. Reactor Core Analysis

1. Characteristics of KJRR Core

2. Requirements

3. Review of Core Analysis

4. Review Results

1. Characteristics of KJRR Core

□ Requirement of Core Design

Parameters	Design Requirements	Core Design
Reactor power		
Max. thermal neutron flux		
Operation day per year		
Reactor life		
Average discharge burnup		
Cycle length		
Power defect		
Max. power peaking factor		
Fission Mo production		
NTD		
Fuel consumption per year		

1. Characteristics of KJRR Core

□ Characteristics of Nuclear Design

- Reactivity Balance
  - Eq. core of - days
  - Reactivity swing: - mk
  - Reactivity loss a day: - mk

m-k?

Fig. Reactivity Change due to Depletion

1. Characteristics of KJRR Core

□ Characteristics of Nuclear Design

- Reactivity? (1)
  - Expressing the departure of a nuclear reactor from criticality

$$\rho = \frac{K_{eff} - 1}{K_{eff}} \quad ,$$

$$where, \quad K_{eff} = \frac{number\ of\ neutrons\ in\ the\ (n+1)th\ generation}{number\ of\ neutrons\ in\ the\ (\ n\ )th\ generation}$$

$$= \frac{number\ of\ neutron\ porduction}{number\ of\ neutron\ absorption\ and\ leakage}$$

K_{eff} *is called "effective multiplication factor"*

1. Characteristics of KJRR Core

□ Characteristics of Nuclear Design

- Reactivity? (2)
  - If k_{eff} is greater than 1, reactivity is positive and reactor power increases, and vice versa
  - Changes in fuel temperature, coolant(moderator) temperature also affect the chain reaction and the rate of reactivity change is given by

$$\dot{\rho} = \dot{\rho}_{cs} + \frac{\partial \rho}{\partial T_f}\dot{T}_f + \frac{\partial \rho}{\partial T_m}\dot{T}_m + \frac{\partial \rho}{\partial \alpha_m}\dot{\alpha}_m$$

where, $\dot{\rho}_{cs}$ *is the rate of reactivity change by control sysytem*

$\frac{\partial \rho}{\partial T_f}$ *is called "doppler fuel temperature coefficient which is always negative*

$\frac{\partial \rho}{\partial T_m}$, $\frac{\partial \rho}{\partial \alpha_m}$ *are moderator temperature coefficient (MTC) and void coefficient*

2. Requirements

□ Regulatory Requirements (1)

- Regulations on Technical Standards for Nuclear Reactor Facilities, Etc.
  - Article 17 (Reactor Design)
  - **Article 18** (Inherent Protection of Reactor)
    - The reactor core and associated coolant systems shall be designed so that, in all power operating range, the net effect of prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.
  - **Article 19** (Suppression of Reactor Power and Power Distribution Oscillations)
    - The reactor core and associated coolant system, control system, and protection system shall be designed to assure that power and power distribution oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be readily detected and suppressed.

2. Requirements

□ Regulatory Requirements (2)

– **Article 26** (Protection System)

- (1) Protection system that meet each of the following requirements shall be installed at reactor facilities:

  4. The protection system shall be separated from the control systems to ensure that **the protection system satisfies all the reliability, diversity, and independence** requirements in the following states:

     a. Failure of a single component or channel of control systems;

     b. Failure of a common component or channel of control and protection systems; and

     c. Removal from service of a single channel.

  5. The protection system shall be designed to **assure that the specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems** such as accidental withdrawal of control rods

2. Requirements

□ Regulatory Requirements (3)

– **Article 28** (Reactivity Control System)

- (1) Reactivity control systems shall be installed to meet each of the following requirements:

  1. Reactivity control systems shall be capable of reliably controlling anticipated reactivity changes under normal operations and anticipated operational occurrences, and capable of maintaining operating states **without exceeding specified acceptable fuel design limits**.

  2**. Two independent reactivity control systems of different design principles shall be provided** and one of the systems shall use control rods.

  3. One of the systems as provided in the foregoing Subparagraph 2 shall be capable of **rendering the reactor subcritical from normal operation and maintaining the core subcritical under cold condition**.

2. Requirements

□ Safety Review Guidelines (1)

- Acceptance criteria of IAEA Safety Requirements NS-R-4*
  - 6.81. All foreseeable reactor core configurations from the initial core through to the equilibrium core for various appropriate operating schedules shall be considered in the core design.
  - 6.85. The reactor core shall be designed so that the reactor can be shut down, cooled and held subcritical with an adequate margin for all operational states and for DBAs. The state of the reactor shall be assessed for selected BDBAs.
  - 6.86. Wherever possible, the design of the reactor core should make use of inherent safety characteristics to minimize the consequences of accident conditions (those that are produced by transients and instabilities).

* NS-R-4, Safety of Research Reactors (IAEA, 2005)

2. Requirements

□ Safety Review Guidelines (2)

- 6.87. Sufficient negative reactivity shall be available in the reactivity control devices(s) in order that the reactor can be brought into a subcritical condition and maintained subcritical in all operational states and in DBA conditions, with account taken of the experimental arrangements with the highest positive reactivity contribution. In the design of reactivity control devices, account shall be taken of wear-out and the effects of irradiation, such as burnup, changes in physical properties and the production of gas.

* NS-R-4, Safety of Research Reactors (IAEA, 2005)

2. Requirements

□ Safety Review Guidelines (3)

- Content of a Safety Analysis Report by IAEA SSG-20*
  - A.2.4. The specific design requirements applied should be stated in this section.

    (8) Reactivity control and core design criteria, including:

    (a) Redundant reactivity control;

    (b) Reactivity limits;

    (c) Prevention of inadvertent criticality;

    (d) Shutdown margins;

    (e) Power peaking factors;

    (f) Maintenance of fuel design margins (e.g. burn-up level balancing with experimental requirements, residence time and water chemistry);

    (g) Design provisions to prevent, or to reduce the potential for, fuel loading errors.

* SSG-20, Safety Assessment for Research Reactors and Preparation of the Safety Analysis Report (IAEA, 2012)

2. Requirements

□ Safety Review Guidelines (4)

- A.5.13. An analysis should be provided that shows that the nuclear conditions in the reactor core are acceptable throughout its anticipated core cycle. The analysis should include the steady state and the dynamic nuclear and thermal characteristics of the reactor.

- A.5.14. Basic information on the nuclear design should include:
  (a) Core configuration and composition, such as the type and anticipated loading pattern of fuel elements, control elements and other components that affect the nuclear properties of the core. Since core configurations for research reactors may change with the changing experimental applications and requirements, the analysis may use a standard core configuration that has conservative properties with respect to all other configurations. An explanation of the intended fuel replacement strategy should complement this information. The information should be supported by drawings.

3. Review of Core Analysis

□ Review of Design Adequacy

- Core Power Distribution of Fuel Loading Plan
- Core Reactivity, Rod Worth, Reactivity Coefficients
- Power Stability and Reactivity Control System

□ Main Point of Review of Nuclear Design

- In normal operation state and AOO, fuel design limits must be satisfied
- Reactor pressure boundary and core cooling capability must be maintained during the anticipated reactivity accidents

3. Review of Core Analysis

□ Detailed Review Points

- Design Requirement
- Core Power Distribution
- Reactivity Coefficients
- Reactivity Control Requirements
- Control Assembly Arrangement and Reactivity Worth
- Core Stability
- Neutron Fluence for Pressure Vessel
- Criticality of Fuel Assembly
- Core Analysis Methodologies

3. Review of Core Analysis

□ Core Power Distribution

- Core Power Distribution
  - Core power distribution is directly related to fuel design limits.
  - In normal operation and AOO, maximum linear power density must be limited to the point where the fuel melting is not happened.
  - DNBR must be limited so as fuel cladding is not failed.
  - Radial and axial peak powers are used for accident analysis and DNBR analysis
- Maximum Peak Power
  - Considered calculation and measurement uncertainty 10%
  - Design Requirement Satisfied

3. Review of Core Analysis

□ Reactivity Coefficients

- Combination of FTC, MTC, MDC, void coefficient and power coefficient
- FTC : always negative because of Doppler feedback effect
  - Inherent Safety : Rapid Power Increase is suppressed in power transient accidents
- Power Coefficient : Negative in power operation range
  - Reactivity control is stable in power oscillation
- FTC, MTC and MDC is main input parameters for accident analysis
  - Related to reactivity feedback effects during AOO
  - Reactivity coefficients are all negative in equilibrium cycles (calculation uncertainness considered)

3. Review of Core Analysis

□ Reactivity Worth (1)

- Reactor core design must accept the reactivity change
  - Due to the fuel and moderator temperature change, fuel depletion and increase of fission products
  - In beginning of cycle, core has a big extra reactivity worth and the reactivity worth is controlled by control rods and burnable poison rods.
- Control Rod Assemblies
  - 4 CARs operated by CRDM(Control Rod Driving Mechanism)
  - 2 SSRs operated by SSDM(Shutdown Rod Driving Mechanism)

3. Review of Core Analysis

□ Reactivity Worth (2)

- Individual control rods are designed adequately.
  - The worth satisfies the limits with 10% uncertainties by design code and manufacturing tolerance.
- Control rod worth related to the accident analysis
  - Shutdown reactivity worth, Rod withdrawal reactivity
  - In the accident analysis, conservative values are used.
  - Control rod worth must be designed as it can satisfy the condition used in the accident analysis.

3. Review of Core Analysis

□ Verification of Design Code (1)

– Nuclear Design Code : McCARD

- McCARD : Monte-Carlo Code for Advanced Reactor Design
- H. J. Shim, et al., "McCARD: Monte Carlo Code for Advanced Reactor Design and Analysis," Nuclear Engineering and Technology, 44[2], 161-176, 2012.
- Accuracy and Uncertainty of McCARD is verified through the independent calculation and review of the uncertainty analysis report during the Construction License
- In addition, verification report for separate licensing of the McCARD will be suggested in course of the Operational License

3. Review of Core Analysis

□ Verification of Design Code (2)

- Nuclear Data : ENDF/B-VII.0
  - ENDF : Evaluated Nuclear Data File
  - ACE Format through NJOY Code
  - ENDF is widely used and fully verified in the nuclear design of reactor core.
- McCARD calculation
  - Without Feedback effects, Considering the uniform temperature distribution (uniform by regions)
  - The accuracy and adequacy is verified through additional calculations with various temperature distribution in the operational range.

3. Review of Core Analysis

□ Conclusion of Review for KJRR

- Core power distribution of KJRR is designed conservatively with the uncertainties as it satisfies the fuel design limits.
- Reactivity control system can provide the reactivity worths required in all operational conditions and shutdown the core with enough margin.
- Fuel design limits can be satisfied because reactivity control system limit the reactivity worth induced in the single failure.
- Nuclear design code system is used in the calculation of various reactor physics parameters and verified fundamentally.
- Nuclear design of KJRR satisfies the Korean Regulatory Requirements.
  - Regulations on Technical Standards for Nuclear Reactor Facilities, Etc (Article 17, 18, 19, 26, 28)
  - KINS/GE-N10, Safety Review Guide for Research and Educational Reactors, Chapter 5.5 (Nuclear Design)

IV. Reactor Core Thermal-hydraulic Analysis

1. Introduction

2. Requirements

3. Design Limit

4. Design Method

1. Introduction

□ KJRR Core Design

2. Requirements

□ Core T/H Design Acceptance Criteria Safety

- There should be at least a 95-percent probability at the 95-percent confidence level that the hot rod in the core does not experience a DNB(Departure of Nucleate Boiling) or transition condition during normal operation or AOOs.
- Uncertainties in the values of process parameters (e.g., reactor power, coolant flow rate, core bypass flow, inlet temperature and pressure, nuclear and engineering hot channel factors), core design parameters, and calculational methods used in the assessment of thermal margin should be treated with at least 95-percent probability at the 95-percent confidence level.
- Sub-channel hydraulic analysis codes should be used to calculate local coolant conditions within fuel assemblies for use in PWR DNB correlations.

2. Requirements

□ Core T/H Design Acceptance Criteria Safety

- The acceptability of such codes must be demonstrated by measurements made in large lattice experiments or power reactor cores.
- The design should address core oscillations and thermal-hydraulic instabilities.
- Methods for calculating single-phase and two-phase fluid flow in the reactor vessel and other components should include classical fluid mechanics relationships and appropriate empirical correlations.
- The proposed technical specifications should ensure that the plant can be safely operated at steady-state conditions under all expected combinations of system parameters.

2. Requirements of LWR

□ 10CFR50 Appendix A, GDC10

- The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that SAFDL(Specified Acceptable Fuel Design Limits) are not exceeded during any condition of normal operation, including the effect of AOO(Anticipated Operational Occurrences)

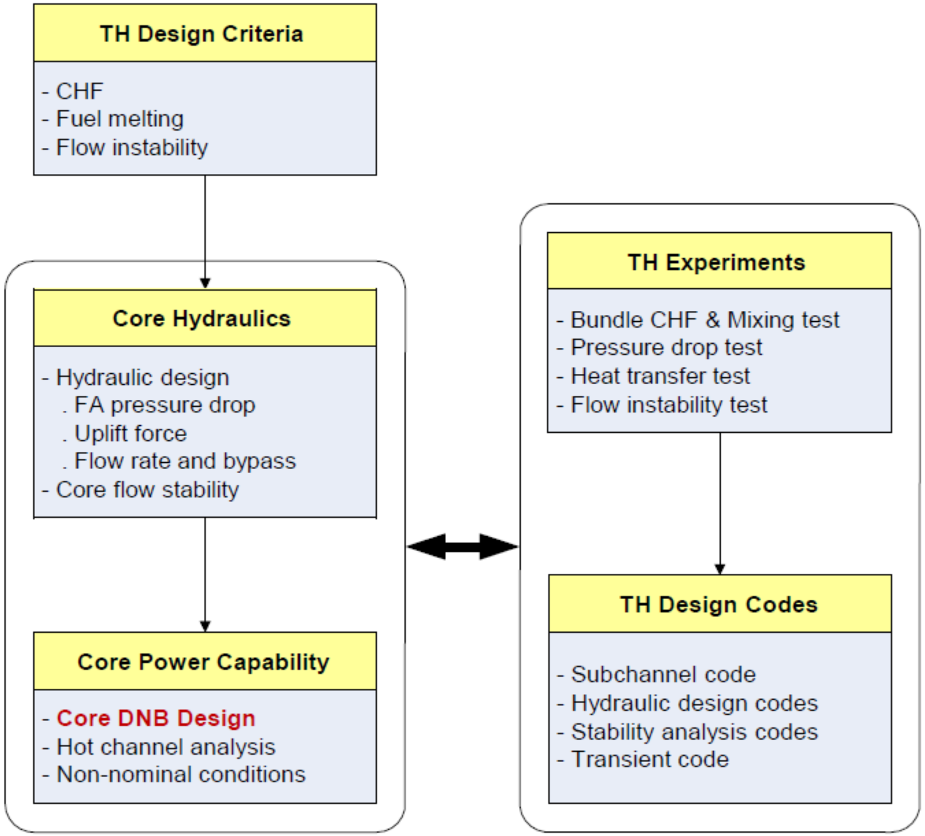
□ SAFDL (Specified Acceptable Fuel Design Limits)

- Departure from Nucleate Boiling Ratio(DNBR)
  - To prevent overheating of fuel cladding
- Linear Heat Generation Rate(LHGR)
  - To prevent fuel centerline melting

□ Acceptance Criteria to meet GDC 10 (SRP 4.4)

3. Core T/H Design Limits

□ Overview of Core T/H Design



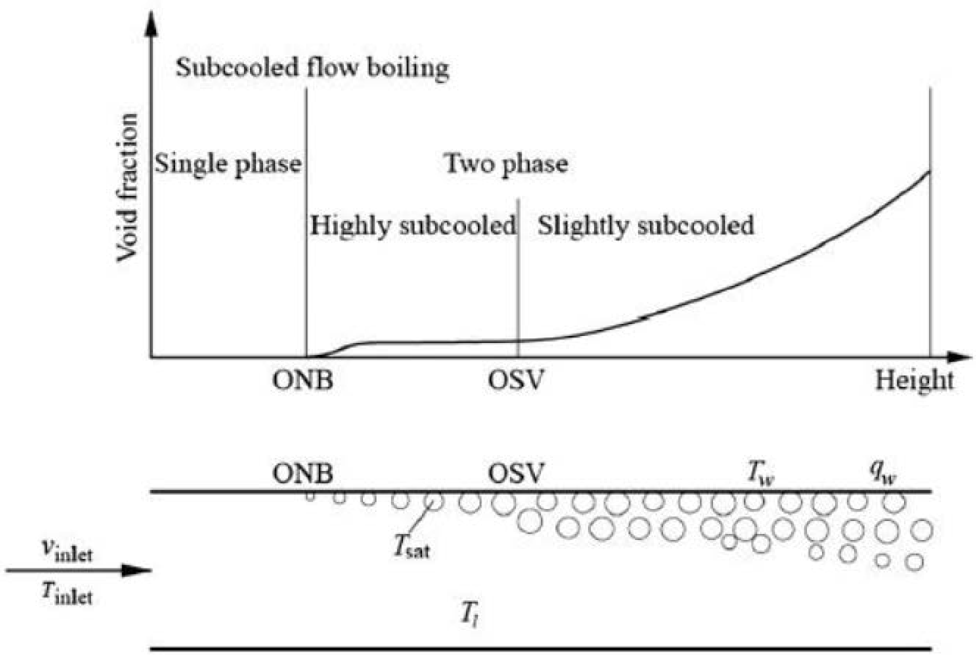
3. Core T/H Design Limits

□ Onset of Nucleate Boiling Margin

- In order to avoid a reactivity effect on the core by a small amount of bubbles, the Onset of Nucleate Boiling (ONB) temperature serves as a design limit for the normal and training operation modes.
- The ONB margin is defined as a difference between the fuel surface temperature at the starting point of a nucleate boiling and the fuel surface temperature at a local cooling condition.
- If the nucleate boiling is prevented, a large thermal margin can be also ensured for Critical Heat Flux (CHF).

3. Core T/H Design Limits

□ Onset of Nucleate Boiling Margin



ONB : Onset of Nucleate Boiling OSV : Onset of Significant Void

3. Core T/H Design Limits

□ Determination of ONB condition

- Surface heat transfer coefficient for non-boiling forced convection

$$h_{db} = \frac{0.023k}{De}(N_{Re})^{0.8}(N_{Pr})^{0.4} \qquad \text{(Dittus-Boelter correlation)}$$

- Temperature drop across surface film

$$\Delta T_{film} = q'' / h_{db}$$

- For nucleate boiling regime, surface temperature of cladding

$$T_{wall} = T_{sat} + 60(q'' \times 10^{-6})^{0.25}\,[\exp(-P/900)] \quad \text{(Jens-Lottes correlation)}$$

- Nucleate boiling is assumed to exist if T_{wall} is less than the sum of $T_{coolant} + \Delta T_{film}$

3. Core T/H Design Limits

□ Critical Heat Flux Ratio

- At the Critical Heat Flux (CHF) point, a large amount of vapor forming as a thin film covers the cladding surface.
- Decreasing the heat transfer coefficient by the thin vapor film leads to very high cladding temperature.
- The safety margin should be ensured that the Minimum Critical Heat Flux Ratio (MCHFR) is satisfied under the normal operation mode
- The MCHFR is defined as the minimum ratio of the CHF to the fuel plate heat flux

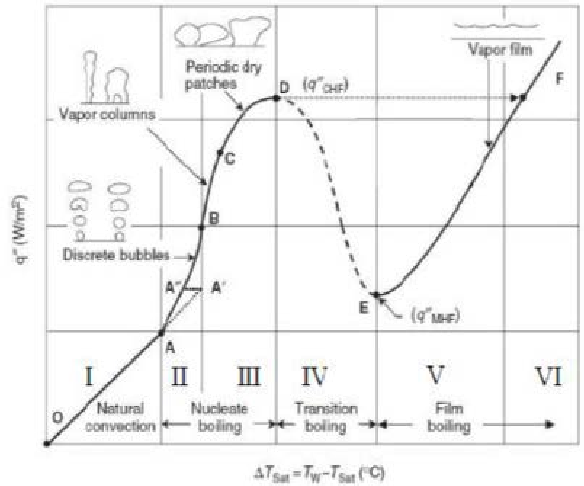
□ Fuel Temperature

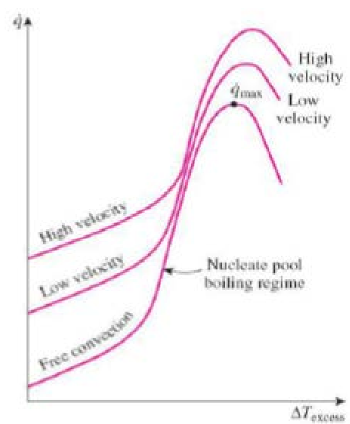
- The fuel temperature should be maintained low enough not to induce a rapid swelling during the normal operation

3. Core T/H Design Limits-CHF Phenomenon

□ Boiling Curve

- At a certain value of wall superheat (point A), vapor bubbles appear on the heater surface (onset of nucleate boiling)
- In region II, discrete bubbles are released from randomly located active sites on the heater surface
- Transition from isolated bubbles to fully developed nucleate boiling (B-C) occurs when bubbles begin to merge
- At point D, the steam forms an insulating layer over the surface and raises surface temperature.
- This is “boiling crisis” or “burnout” and the point called Critical Heat Flux (CHF)
- The phenomenon can be classified as Departure from Nucleate Boiling (DNB) in the subcooled or low quality region, and Dryout in the high quality region
- Transition boiling region (D-E) is characterized by the existence of an unstable vapor blanket over the heater surface and film boiling region (E-F) represents a stable film boiling



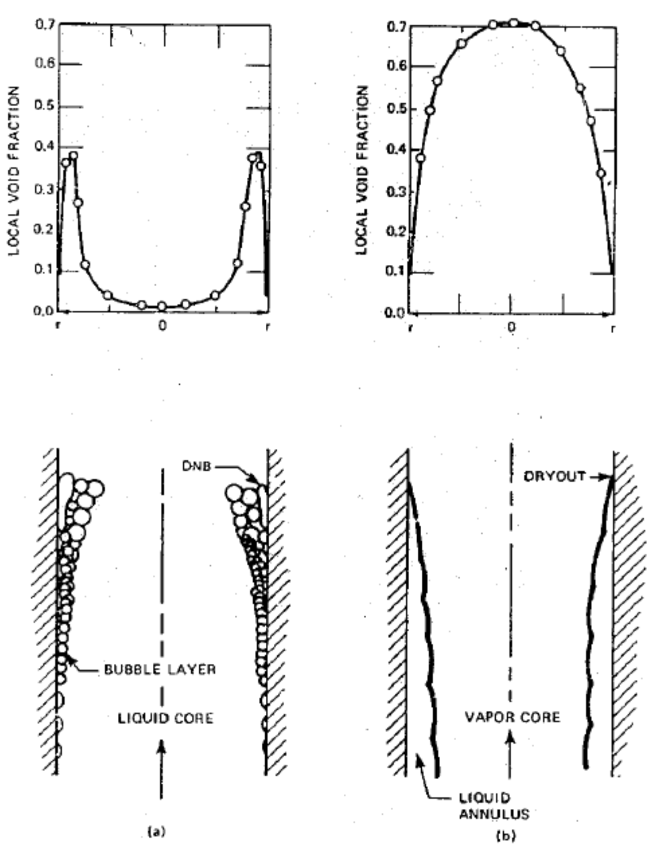


Effect of forced convection on flow boiling

3. Core T/H Design Limits-CHF Phenomenon

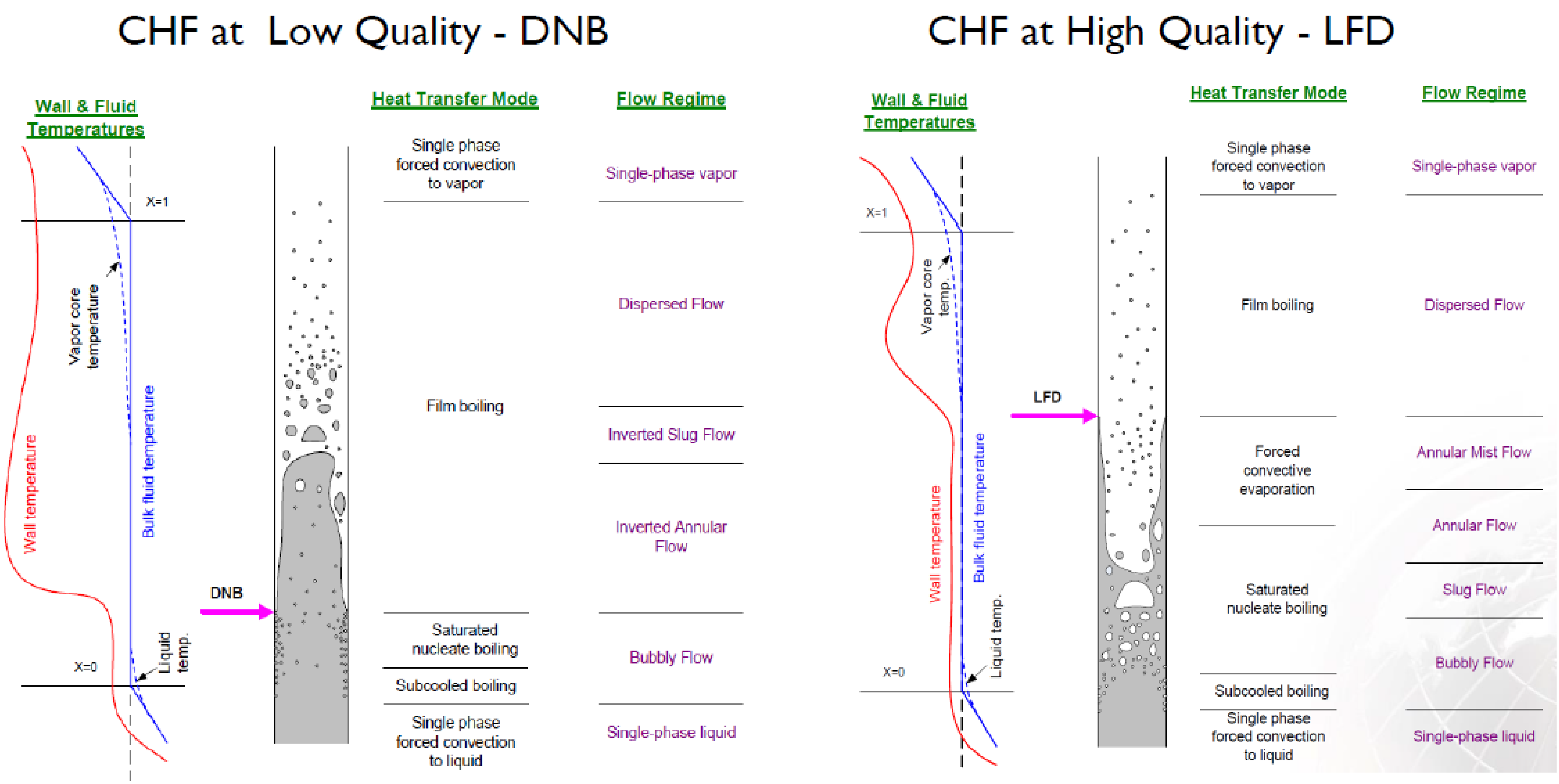
□ Mechanisms in Rod Bundles

- DNB Mechanism
  - DNB occurs when many bubble nucleation sites are simultaneously activated
  - Bubble can be initiated only when the fluid temperature exceeds a certain value greater than the saturation temperature
- Dryout Mechanism
  - Annular flow regime
  - (Liquid evaporated + droplet entrained in steam) > Droplets deposited on liquid film or wall
  - Entrainment of droplets results from the liquid surface waves and bursting of bubbles through the liquid film



3. Core T/H Design Limits-CHF Phenomenon

□ CHF and Flow Regime



3. Core T/H Design Limits-CHF Phenomenon

□ CHF Factors

- CHF decreases if
  - Flow decreases
    - Reduction in flow rate results in an increase in coolant temperature
  - Flux (Power) increase
    - High local power densities produce higher heat flux, and higher coolant and cladding temperature
  - Temperature increases
    - Closer to saturation conditions
  - Pressure decreases
    - Operating at lower pressures allows DNB to occur at lower temperatures

3. Core T/H Design Limits-CHF Phenomenon

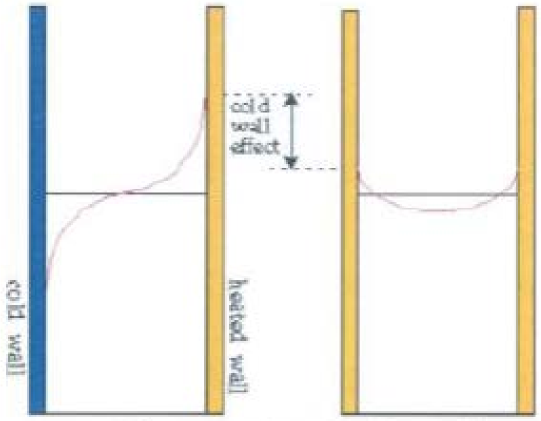
□ Effects of parameters on CHF in rod bundle

- Effects of interaction between neighboring channels
  - Mass, momentum and/or energy exchange due to :
    - Diversion cross flow
    - Turbulent inter-channel mixing
    - Forced cross flow due to spacers such as wire-wrap or helical fin
    - Forced mixing due to mixing vanes, etc.
  - Radial power distribution
    - The beneficial effect of turbulent interchange on limiting subchannel enthalpy rise is proportional to the lateral power gradient
    - A “flat” hot assembly power distribution means more heat input and consequently more diversion crossflow out of the hot assembly, thus produces conservative min. DNBR

3. Core T/H Design Limits-CHF Phenomenon

□ Cold wall effects

- In a channel containing an unheated wall (e.g., subchannel adjacent to guide tube), liquid film builds up along the cold wall
- This fluid is not effective in cooling the heated surface and the fluid cooling the hot surface is effectively at a higher enthalpy than calculated by an energy balance
- Thus, CHF in a channel with an unheated wall is generally lower than that in a channel with all sides heated and at the same bulk exit enthalpy
- Tong extended his design correlation (W-3) to the case of a channel with an unheated wall by defining a cold wall factor (CWF)



at the same local fluid condition

3. Core T/H Design Limits-CHF Phenomenon

□ Effects of non-uniform axial heat flux

- At PWR conditions, DNB will not occur at the tube exit with a cosinusoidal or skewed cosinusoidal flux distribution
- Most correlations based on uniform heating slightly overpredict the data from nonuniformly heated channels at PWR conditions
- The flow, particularly in the boundary layer region, coming from the upstream region carries superheat and bubbles with it
- Thus, upstream heat flux distributions affect the boundary layer at the DNB position, and this “memory effect” is conveyed by Tong’s F factor
- In the subcooled region, the F factor is small and local heat flux determines the boiling crisis
- At high qualities, the average heat flux or enthalpy rise primarily determines the boiling crisis

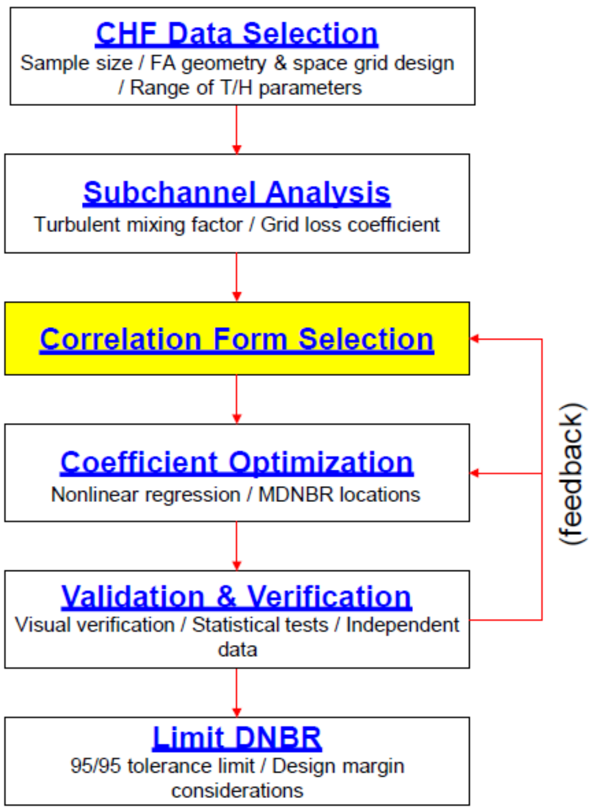
3. Core T/H Design Limits-CHF Prediction

□ Empirical correlations for rod bundles

- Mixed Flow (Bundle Average) Correlation Approach
  - CHF experiments using prototypical test bundles (BWR, CANDU).
  - Not require TH codes for the local TH conditions (GEXL, Bowring, Xc-Bl, BLA, etc.)
- Sub-channel Approach
  - CHF experiments using small scale test bundle (PWR)
  - Calculate local TH conditions using subchannel codes (WRB-1, CE-1, BAW-2, EPRI, etc.)

3. Core T/H Design Limits-CHF Prediction

□ Empirical DNB correlations



Type - I: (Linearity of CHF vs. quality)	$q''_{CHF} = A - B \cdot \chi$ $A = f(G, P, L, D_{he}, D_{hy}, g_{sp}, d_g)$ $B = f(G, P, L)$ $\chi = \dfrac{h - h_f}{h_g - h_f} = \dfrac{h - h_f}{h_{fg}}$: equilibrium quality ex.) CE-1, WRB-1, ERB-3, BAW-2, etc.
Type - II: (Type-I + Heat balance)	$q''_{CHF} = \dfrac{A + B \cdot \Delta h_{in}}{C + L \cdot Y'}$ or $q''_{CHF} = \dfrac{(A - \chi_{in})}{C + (\chi - \chi_{in})/q''_L}$ ex.) WSC-2, EPRI, etc.

3. Core T/H Design Limits-CHF Prediction

□ Application of tube CHF prediction models

- AECL CHF Lookup Table (1986, 1995)
- W-3 R grid correlation

3. Core T/H Design Limits-CHF Prediction

□ Application of tube CHF prediction models

- AECL CHF Lookup Table (1986, 1995)
  - The table provides CHF values for 8 mm tubes at discrete values of pressure, mass flux and dryout quality covering the ranges 0.1 to 20.0 MPa, 0.0 to 8.0 Mg/m^2s and -0.5 to +1.0 respectively
  - Linear interpolation is used to determine the CHF for conditions between the tabulated values
  - The empirical correction factors are introduced to extend the table to tubes of diameter values other than 8 mm and to address other effects such as heated length, non-uniform axial flux distribution, grid spacer, and bundle effects, etc.
  - The 1995 look-up table predicts the data with the root-mean-square (rms) errors of 7.82% for 22,946 data points

3. Core T/H Design Limits-CHF Prediction

□ Application of tube CHF prediction models

- AECL CHF Lookup Table (1986, 1995)

P	G	X																					
kPa	kg/m²s	-0.5	-0.4	-0.3	-0.2	-0.15	-0.1	-0.05	0.0	0.05	0.1	0.15	0.2	0.25	0.3	0.35	0.4	0.45	0.5	0.6	0.7	0.8	0.9
9000	0	4836	4571	4301	4034	3902	3783	3639	3352	2503	1977	1637	1290	1179	992	843	739	680	608	560	471	431	349
9000	50	5422	5114	4813	4547	4432	4335	4213	3960	3071	2528	2304	2110	1945	1810	1699	1610	1536	1457	1152	971	938	650
9000	100	5916	5571	5251	4988	4892	4811	4704	4462	3658	3134	2976	2841	2597	2575	2458	2373	2317	2165	1554	1306	1300	1040
9000	300	6366	5916	5527	5300	5260	5239	5236	5194	4944	4270	3692	3478	3344	3257	3065	2894	2713	2334	1672	1380	1197	899
9000	500	6641	6100	5668	5412	5332	5205	5147	5114	4925	4243	3625	3364	3160	2909	2887	2741	2361	1983	1583	1282	904	626
9000	1000	7221	6432	5947	5625	5398	4936	4705	4703	4607	4204	3520	3118	2854	2661	2545	2118	1553	1143	984	729	346	311
9000	1500	7976	7023	6356	5856	5535	4958	4693	4467	4189	3730	3227	2850	2712	2495	2034	1429	1081	892	546	350	131	119
9000	2000	9111	8171	7160	6258	5863	5299	4835	4196	3569	3097	2746	2448	2312	1931	1428	1027	745	603	345	186	97	48
9000	2500	10108	9252	8185	7162	6749	6025	5004	4119	3339	2797	2521	2241	1942	1373	1029	809	570	346	211	136	96	48
9000	3000	10887	10041	8998	7755	7210	6399	5097	4045	3240	2604	2289	2038	1724	1354	967	708	508	393	282	169	103	49
9000	3500	11622	10683	9660	8227	7395	6453	5140	3995	3155	2487	2124	1896	1643	1276	952	650	457	419	305	184	107	54
9000	4000	12336	11283	10200	8756	7669	6674	5286	3983	3066	2349	1931	1750	1604	1375	982	664	558	499	331	190	108	55
9000	4500	13022	11857	10680	9235	7924	6936	5469	4047	3026	2245	1827	1719	1631	1385	1037	771	670	615	381	205	109	56
9000	5000	13683	12418	11127	9685	8259	7305	5645	4183	3080	2249	1836	1755	1663	1416	1152	908	816	717	445	239	116	57
9000	5500	14332	12927	11580	10180	8672	7818	5764	4272	3210	2355	1963	1856	1760	1542	1290	1071	935	801	512	281	130	62
9000	6000	14961	13402	11908	10758	9759	8777	6149	4331	3309	2421	2126	2045	1937	1699	1500	1219	1072	892	578	329	157	73
9000	6500	15570	13909	12387	11336	10230	9257	6353	4416	3344	2492	2346	2317	2197	1965	1710	1431	1170	977	646	378	186	88
9000	7000	16165	14403	12779	11578	10702	9737	6355	4542	3482	2788	2527	2402	2245	2070	1785	1497	1258	1058	712	426	216	103
9000	7500	16748	14881	13106	11824	10747	9739	6394	4637	3660	3076	2877	2745	2567	2142	1800	1557	1389	1140	778	475	248	118
9000	8000	17323	15340	13551	12107	10854	9741	6556	4714	3838	3419	3342	3289	3162	2736	2042	1620	1420	1218	842	525	281	134

A portion of the CHF look-up table for 9 MPa

Factor	Form
Cross-section	$K_1 = (8/d_{hy})^{1/2}$ *for* $3 < d_{hy} < 25$ mm, $K_1 = 0.6$ *for* $d_{hy} > 25$ mm
Grid spacer	$K_3 = 1 + 1.5 \cdot K_{grid}^{0.5} \cdot \left(\frac{G}{1000}\right)^{0.2} \cdot \exp\left(-0.1 \times \frac{L_{sp}}{d_{hy}}\right)$
Heated length	$K_4 = \exp\left(e^{2\alpha} \cdot d_{hy}/L\right)$ $\alpha = \chi/[\chi + (1-\chi)\rho_g/\rho_f]$
Non-uniform APS	$F_{BLA} = \left[\int_{z_b}^{z_c} q''(z)dz\right] / \left[q''_{loc}(z_c - z_b)\right]$ *or* Tong's F-factor

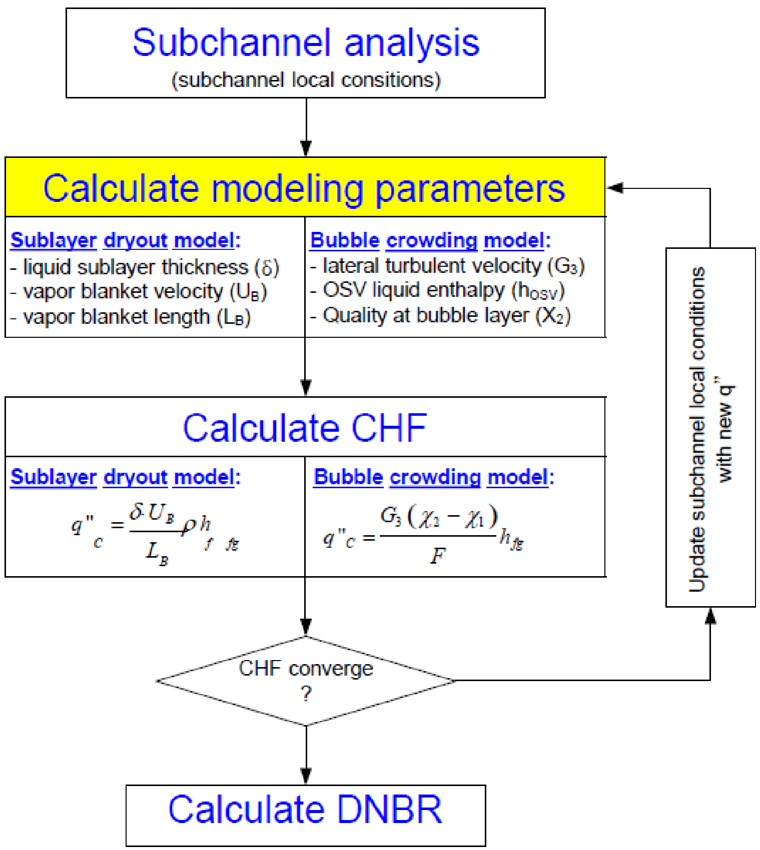
3. Core T/H Design Limits-CHF Prediction

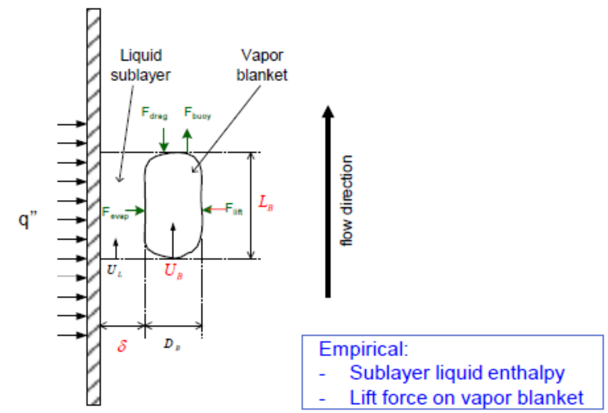
□ Phenomenological Approach

- LFD model for Annular flow (Dryout)
- Hydrodynamic instability model (DNB)
  - Far-field model: Zuber(1958)
  - Near-surface model: Lee & Mudawar(1988)
- Bubble Crowding Model (DNB)
  - Boundary layer separation: Tong(1968)
  - Vapor removal limit: Weisman et al.(1985)

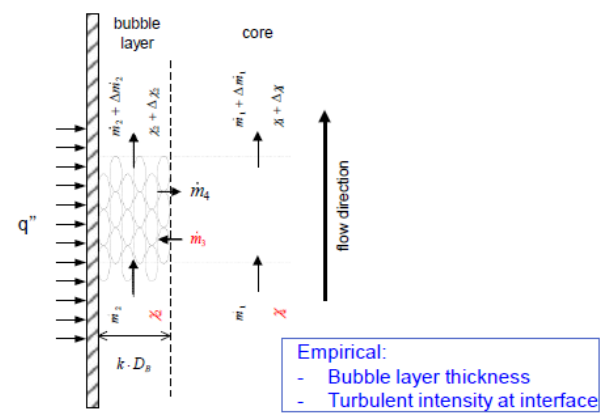
3. Core T/H Design Limits-CHF Prediction

□ Phenomenological Approach





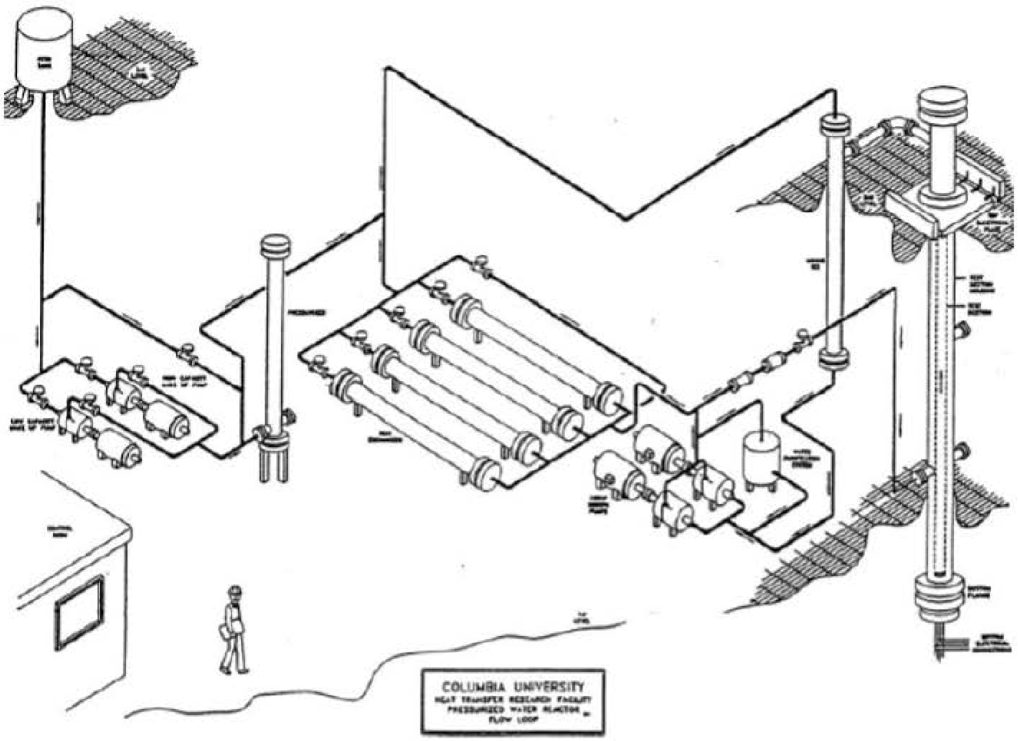
Sublayer dryout model (Lee, 1988)

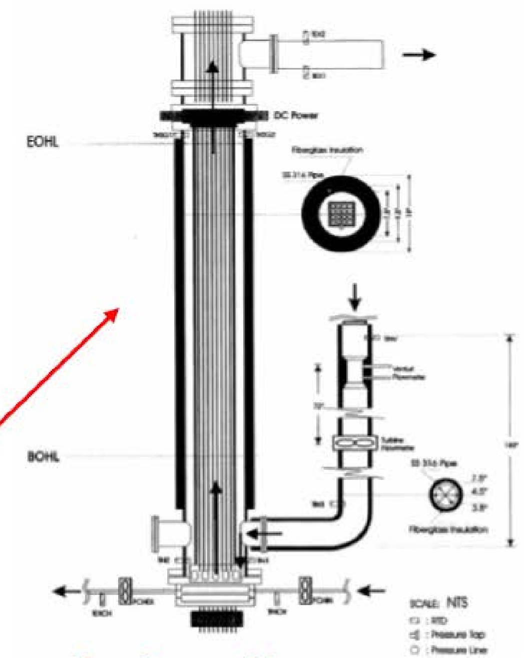


Bubble crowding model (Weisman, 1983)

3. Core T/H Design Limits-CHF Experiments

Test Loop





Test Section Housing

Grid for
Test Section

3. Core T/H Design Limits-CHF Experiments

□ Test series

- Different test runs corresponding to different rod bundle geometry
- Repeatability tests : tests at the same conditions at the different times

□ Test Matrix

- Combination of various coolant conditions (mass velocity, inlet temperature, pressure) to simulate various operating conditions corresponding to DNB

□ Test Section

- Simulation of square rod lattice of fuel assembly geometry
  - 4×4, 5×5, 6×6 array with grids (with/without mixing vane)
  - non-uniform radial power distribution
  - uniform/non-uniform axial power distribution
  - with/without guide thimbles
  - varied heated lengths
  - varied grid spacing

3. Core T/H Design Limits-CHF Experiments

□ Test Procedure

- Pre-CHF
  - Heat balance check, thermocouple (T/C) calibration, pressure drop measurement at adiabatic conditions
  - Subchannel outlet temperature measurements at single-phase conditions
- Onset of CHF
  - Setup flow condition
  - Gradually increase power or decrease flow rate
  - Identify the CHF condition based on slope of T/C trace, and decrease power
- Post test inspection
  - Check rod geometry and burn marks

3. Core T/H Design Limits-CHF Experiments

□ CHF Measurement

- Increase power infinitesimally
- Maintain other loop parameters as stable as possible
- Identify the CHF condition based on slope of T/C trace and decrease power

□ Records

- Pressure, Inlet Mass Velocity, Inlet Temperature
- Bundle Power (MW)
- CHF Locations : Heater Rod & T/C

3. Core T/H Design Limits

□ Onset of Flow Instability Ratio

- The maximum power output of from reactor fuel channels cooled by sub-cooled water may, under certain conditions, be limited by the occurrence of excursive flow (Ledinegg) instability
- An assessment of excursive flow instability entails a comparison of the slopes of the demand (S-curves) and the supply curves for the coolant channel and the criterion for excursive flow instability reduces to determining the conditions for which the S-curves have a minimum value
- The Onset of Flow Instability Ratio (OFIR) is defined as the ratio of the heat flux at onset of flow instability to the actual heat flux.

4. Core T/H Design Method

□ Uncertainty Evaluation of Design Parameters

- Thermal hydraulic design parameters uncertainty data selected for assessment are reactor operation conditions, flow distribution and cooling area, power distribution, manufacturing tolerances of the fuel assembly, correlations of thermal hydraulics, etc.
- All uncertainties are considered to determine the Engineering Hot Channel Factors (HCF) such as the bulk temperature rise of coolant (F_b), heat flux ($F_{q``}$) and film temperature rise (F_f) hot channel factors.
- Thermal hydraulic characteristics such as pressure drop, single phase heat transfer, ONB and CHF are evaluated with the appropriate thermal hydraulic correlations.

4. Core T/H Design Method

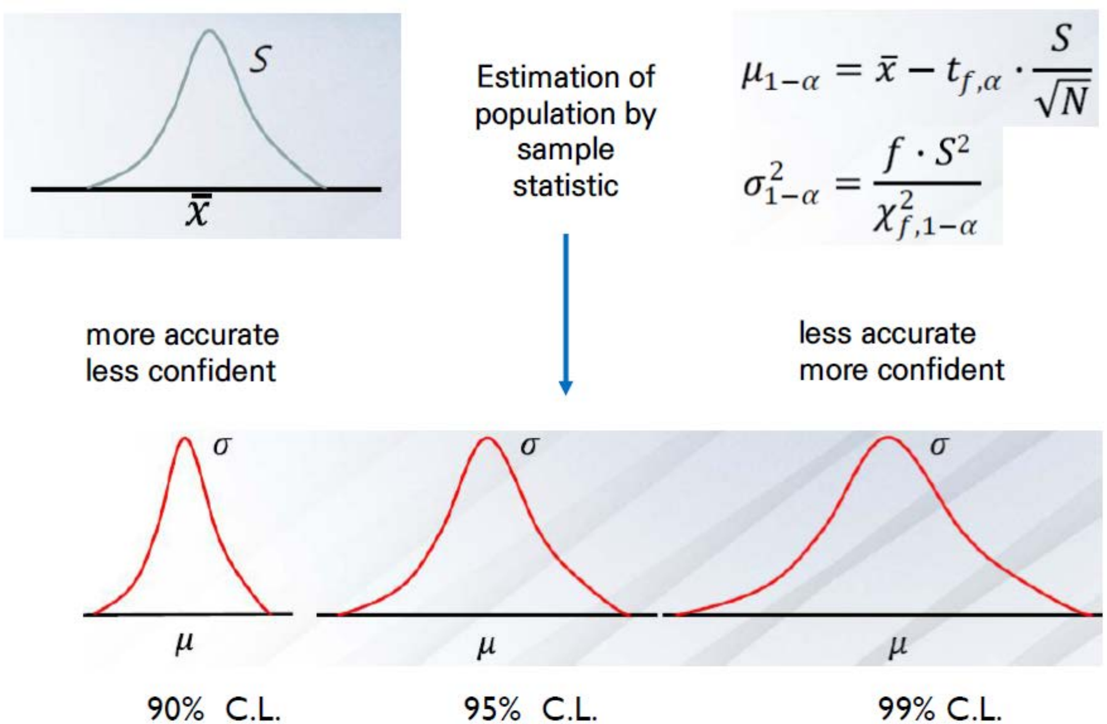
□ Uncertainty Evaluation of Design Parameters

- Treatment of Uncertainties
  - 95/95 Criterion : 95% Probability at 95% Confidence Level
  - Confidence Level
    - Probabilistic measure of assurance to estimate a population parameter from a finite number of samples
    - Confidence level expresses the probability that a population parameter estimated from a sample is within a stated range
    - Confidence level of a statistically determined relationship is the fraction of time the relationship is expected to be satisfied
    - Confidence in the estimation improves as the numbers of samples increases and confidence that a population parameter lies within a certain range increases as the range is extended

4. Core T/H Design Method

□ Uncertainty Evaluation of Design Parameters

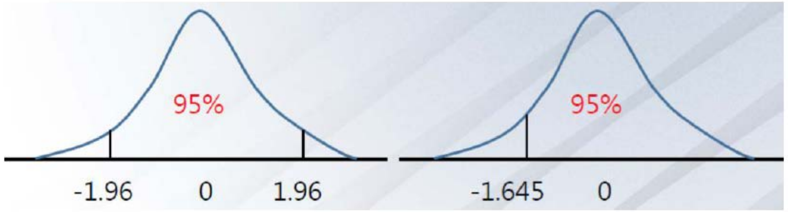
- Treatment of Uncertainties
  - Confidence Level



4. Core T/H Design Method

□ Uncertainty Evaluation of Design Parameters

- Treatment of Uncertainties
  - Confidence Interval / Limit
    - At a certain probability, the interval contains the parameter being estimated (One-sided or Two-sided)
    - End points of a confidence interval are called confidence limits (Upper or lower)
  - Tolerance Interval / Limit
    - At a certain probability, the interval contains at least a proportion P of the population (One-sided or Two-sided)
    - End points of a tolerance interval are called tolerance limits (Upper or lower)



Two-sided One-sided

4. Core T/H Design Method

□ Selection of Axial Power Distribution

- For thermal hydraulic analysis, an axial power distribution, which results in the lowest thermal margins on such as ONB and MCHFR, is required.
- Axial power distribution considered in the selection for thermal hydraulic analysis includes the initial to equilibrium cores. In each cycle, there are Beginning of Cycle (BOC) and End of Cycle (EOC).
- The axial power distributions at EOC are more flat than those at BOC since the CARs are much withdrawn at EOC.

4. Core T/H Design Method

□ T/H Design Data

- Power distribution in the core: from the nuclear design results
- Flow distribution in the core
- Pressure loss in the core
- Core inlet and outlet temperatures
- Core inlet pressure

4. Core T/H Design Method

□ Assessment of Core Thermal Margin

- Core thermal hydraulic analyses are performed for the two cooling modes (forced convection & natural convection) using the appropriate subchannel analysis code
- The thermal margins are evaluated for the ONB temperature margin, minimum CHFR, and maximum fuel temperature

- END -

Thank you!

Always we keep watching our Atomic Power

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