Lecture No 08

Title:

TH Design of LWR Fuel – Safety Limits

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Outline of the Lecture

- Thermal-hydraulic design requirements
- Hot channel factors
- Safety margins
- CHF limit: a more realistic approach

Thermal-hydraulic Design Requirements (1)

- The thermal-hydraulic design requirements depend on how frequent given conditions persist
- Most frequent occurrences must yield little or no radiological risk to the public
- Extreme situations having the potential for greatest consequences to the public shall be those least likely to occur

Thermal-hydraulic Design Requirements (2)

- Thus full spectrum of plant operational and fault conditions is divided into four categories, and each category has its specific requirements

Condition	Criteria
1) Normal operation and operational transients, normally expected	Fuel damage is not expected, but very small number of rod failures can occur
2) Faults of moderate frequency of once per year	Possible rod failures are small and within the limit of plant cleanup system
3) Infrequent faults that may happen once in the lifetime of the plant	Reactor can be brought to a safe state, after possibly some outage time
4) Limiting faults, not expected, but analyzed and designed for	Reactor can be brought to a safe state; core is kept subcritical; $T_{Cmax} < 1478$ K

Thermal-hydraulic Design Requirements (3)

- To satisfy the criteria, the following design basis have been established:
  - Critical Heat Flux (both DNB and dryout) design base: at least 99.9% of fuel pins will not experience CHF during Condition 1 and 2
  - Fuel temperature design base: the maximum fuel temperature shall be less than the melting temperature of UO_2 during conditions 1 and 2 and selected conditions 3 and 4 (for example Loss-of-Coolant Accident – LOCA condition)
  - Core flow base: a specified lower limit flow must pass through core
  - Hydrodynamic stability design base: operation in conditions 1 and 2 should not lead to flow oscillations

Hot Channel Factors (1)

- One of the main goals of the thermal-hydraulic core analysis is to ensure that the thermal limitations on the core behavior are not exceeded

- To exclude melting of fuel, the linear power density must be limited

$$q' < q'_{\max}$$ At any location in the core

- Another limitation is dictated by the requirement that the surface heat always remains below the critical limit -CHF

$q'' < q''_{CHF}$ At any location in PWR core

$x < x_{cr}$ At any location in BWR core

Hot Channel Factors (2)

- A thorough thermal-hydraulic analysis of the core requires a detailed, three-dimensional calculation of the core power distribution, including
  - the effects of fuel burnup,
  - fission product buildup,
  - control rod distributions,
  - moderator density variations over core life
- This information is next used to determine the coolant flow and temperature distribution throughout the core
- Even though such types of calculations are performed nowadays, they are quite expensive and time consuming
- Especially for fast transient applications they are prohibitively expensive

Hot Channel Factors (3)

- To make the thermal-hydraulic core analysis more practical, a common approach is to investigate how closely the ***hot channel*** in the core approaches the operating limitations
- Then if one can ensure that the thermal conditions of this channel remain below the core limitations, the remaining channels will presumably fall within design limitations
- One usually defines the **hot channel** in the core as that coolant channel in which the core **heat flux and enthalpy rise are maximum**
- Associated with this channel are various **hot channel** or ***hot spot* factors** relating the performance of this channel to the average behavior of the core

Hot Channel Factors (4)

- The fuel assembly having the maximum power output is defined as the ***hot assembly***

- The hot spot in the core is the point of maximum heat flux or linear power density, while the hot channel is defined as the coolant (sub-) channel in which the hot spot occurs or along which the maximum coolant enthalpy increase occurs

- The ***nuclear hot channel*** is defined to take into account the variation of the neutron flux and fuel distribution within the core

Hot Channel Factors (5)

- The ***radial nuclear hot channel factor*** is defined as

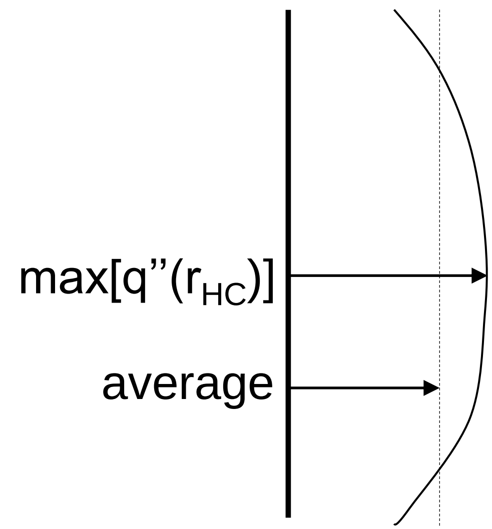
$$F_R^N = \frac{\text{average heat flux of the hot channel}}{\text{average heat flux of all channels in core}} = \frac{\dfrac{1}{H}\displaystyle\int_{-H/2}^{H/2} q''(\mathbf{r}_{HC})dz}{\dfrac{1}{N_C}\displaystyle\sum_{i=1}^{N_C}\dfrac{1}{H}\int_{-H/2}^{H/2} q''(\mathbf{r}_i)dz}$$

Here N_C is the total number of channels in core, H is channel height and $q''(\mathbf{r})$ is the heat flux distribution in the core; ***r*** – radial position of the channel in the core

Hot Channel Factors (6)

- In a similar manner, the ***axial nuclear hot channel factor*** is defined as

$$F_Z^N = \frac{\text{maximum heat flux of the hot channel}}{\text{average heat flux of the hot channel}} = \frac{\max\limits_{z}\left[q''\left(\mathbf{r}_{HC}\right)\right]}{\frac{1}{H}\int\limits_{-H/2}^{H/2} q''\left(\mathbf{r}_{HC}\right)dz}$$



- The ***total nuclear hot channel factor*** or ***nuclear heat flux factor*** is then

$$F_q^N = \frac{\text{maximum heat flux in the core}}{\text{average heat flux in the core}} = F_R^N F_Z^N$$

Hot Channel Factors (7)

- **Example 1:** The power distribution in a cylindrical core is described by the following expression

For simplicity, we assume: $\tilde{R} \cong R\,,\ \tilde{H} = H$

$$q'''(z) = w_f \Sigma_f \phi_0 J_0\left(\frac{2.405 r_f}{R}\right)\cos\left(\frac{\pi z}{H}\right)$$

- The radial factor is then

$$F_R^N = \frac{J_0(0)\int_{-H/2}^{H/2}\cos\left(\frac{\pi z}{H}\right)dz}{\frac{1}{\pi R^2}\int_0^R J_0\left(\frac{2.405 r}{R}\right)2\pi r dr\int_{-H/2}^{H/2}\cos\left(\frac{\pi z}{H}\right)dz} \approx 2.32$$

- And the axial factor is

$$F_Z^N = \frac{J_0(0)\cos(0)}{J_0(0)\frac{1}{H}\int_{-H/2}^{H/2}\cos\left(\frac{\pi z}{H}\right)dz} \approx 1.57$$

- This implies the over-all factor

$$F_q^N \approx 2.32 \cdot 1.57 \approx 3.642$$

Hot Channel Factors (8)

- $F_q^N \approx 3.642$ is a very conservative (e.g. high) estimate of the factor. A zone-loaded PWR has $F_q^N \approx 2.6$

- The nuclear heat flux hot channel factor is defined assuming nominal fuel pellet and rod parameters

- No influence of manufacturing tolerances is taken into account

Hot Channel Factors (9)

- In reality, however, there will be local variation in fuel pellet density, enrichment and diameter, surface area of fuel rod and eccentricity of the fuel-clad gap due to manufacturing tolerances and operating conditions

- The more general ***heat flux hot channel factor*** or ***total power peaking factor*** F_q is defined as the maximum heat flux in the hot channel divided by the average heat flux in the core (allowing for above-mentioned variability)

Hot Channel Factors (10)

- F_q and F_q^N are related by defining an ***engineering heat flux hot-channel factor***

$$F_q^E = \frac{F_q}{F_q^N}$$

- Typically this factor is small reflecting the fact that manufacturing tolerances are small. In PWR it is ~1.03, thus:

$$F_q = F_q^E F_q^N \quad \text{where} \quad F_q^E \cong 1.03$$

Hot Channel Factors (11)

- One can also define an ***enthalpy-rise hot channel factor***

$$F_{\Delta i} = \frac{\text{maximum coolant enthalpy rise}}{\text{average coolant enthalpy rise}}$$

- This factor is a function of heat source and coolant flow distributions
- Such factor is used with significantly non-uniform coolant flow distribution, with dryout limitation ($x < x_{cr}$)

Safety Margins (1)

- Safety limit
  - Practically all systems have limits of safe operation; for example an elevator can carry a limited load
  - Safety limit of a simple system (e.g. elevator) can be calculated with a reasonable precision
  - Safety limit of a complex system is usually found in an experimental way, for instance by applying such load at which the system fails

- **Safety margin** is a difference between safety-limit value of a parameter and its actual value
  - For example, if a system fails under 1000 N load and the actual load is 800 N, the safety margin is equal to 200 N.

Safety Margins (2)

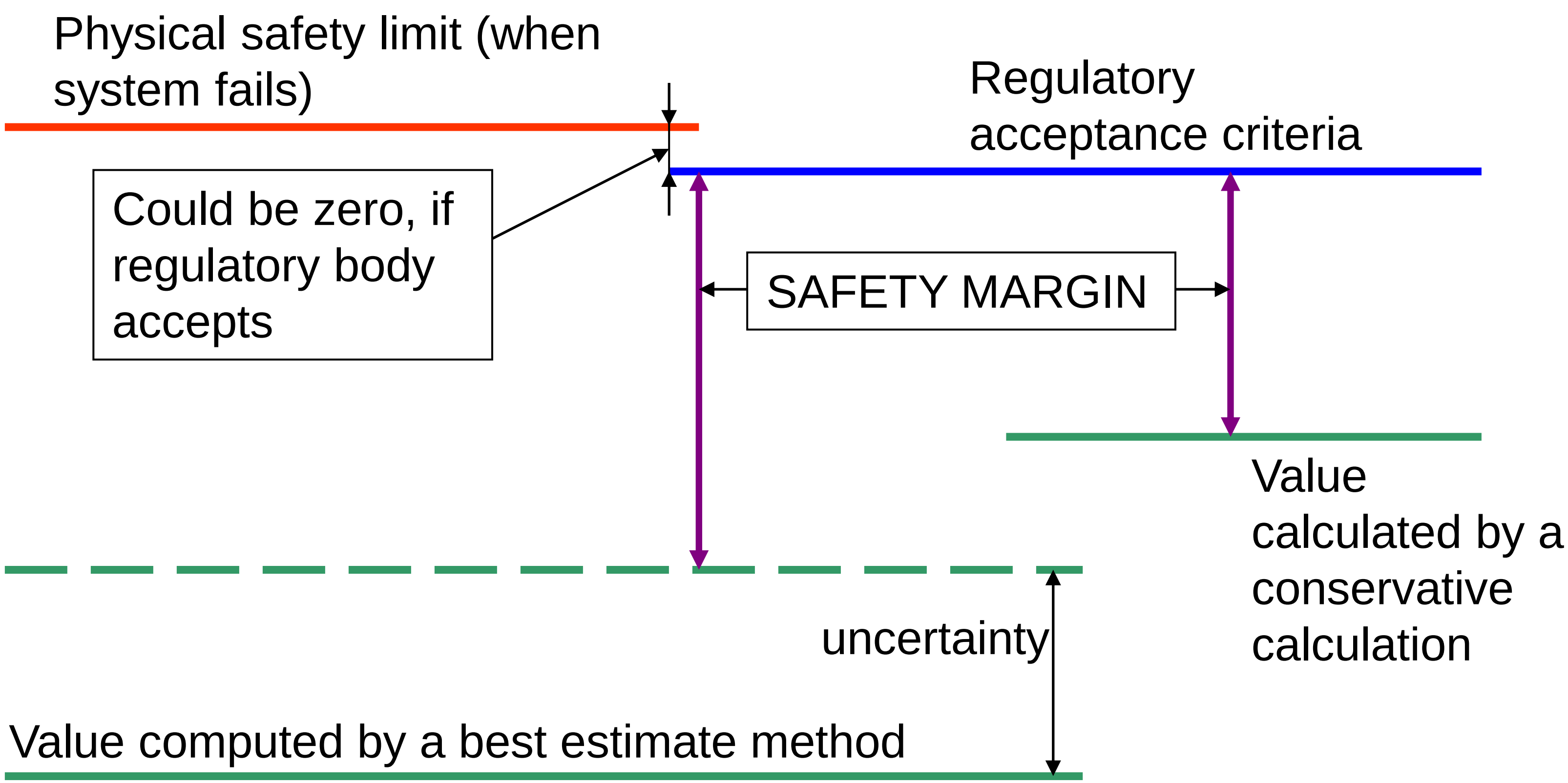
- Safety limits are specified in a technical specification of each nuclear power plant (NPP)

- Safety limits should not be exceeded under the normal operation of NPP

- In each country the safety limits in NPP are defined by the regulatory body (SSM in Sweden) as acceptance criteria

- Thus safety margin can be defined as a difference between the acceptance criteria value of a given parameter and its actual value

Safety Margins (3)



Safety Margins (4)

- The following parameters have specified acceptance value:
  - Reactor coolant system pressure
  - Minimum shutdown margins (negative reactivity after reactor shutdown)
  - Linear heat generation in fuel
  - Fuel pellet temperatures
  - Clad temperatures
  - Departure from nucleate boiling ratio (DNBR) or critical power ratio (CPR)
  - Fuel enthalpy
  - Clad strain and extend of oxidation
  - Percentage of fuel failure
  - Hydrogen generation
  - Containment pressure and temperature
  - Radiation dose to plant personnel and to the public

Safety Margins (5)

- Safety limits are set as international standards and accepted by national regulatory bodies

- For Loss-of-Coolant design basis accident, the criteria are as follows
  - Peak clad temperature (1478 K, or 1204 °C)
  - Maximum clad oxidation (17% of clad thickness)
  - Maximum hydrogen generation (not to exceed deflagration or detonation limits for containment integrity)
  - Coolable geometry of core

Safety Margins (6)

- The above mentioned criteria are based on the deterministic safety margins

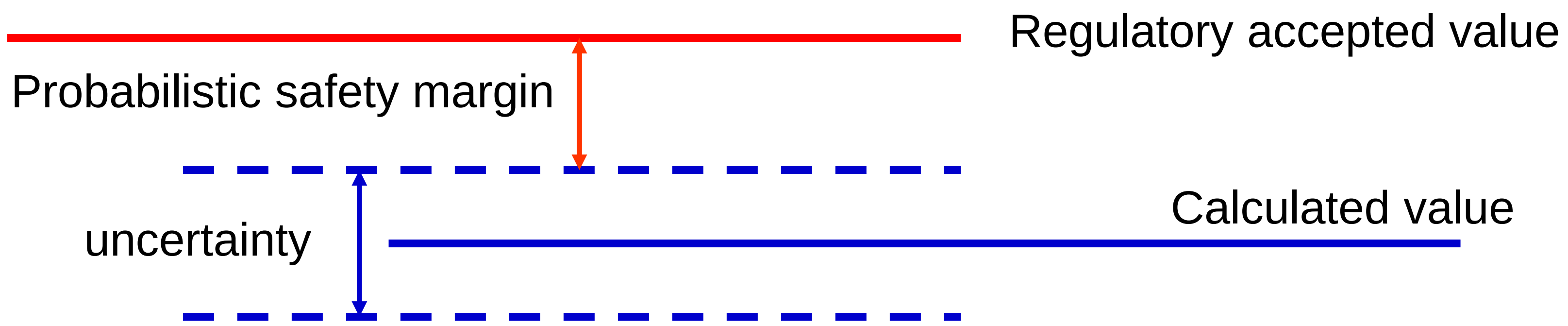
- Current international trend requires that the margins are determined using the probabilistic safety analysis (PSA) as well, to support the deterministic analysis

- PSA is used to support to make **risk informed decisions** (that is decisions based on the evaluation of risks)

Safety Margins (7)

- Probabilistic safety margins are defined in a similar manner as the deterministic safety margins

- Probabilistic safety margin is defined as a difference between the acceptable target defined by regulatory body and the calculated value of the risk parameter



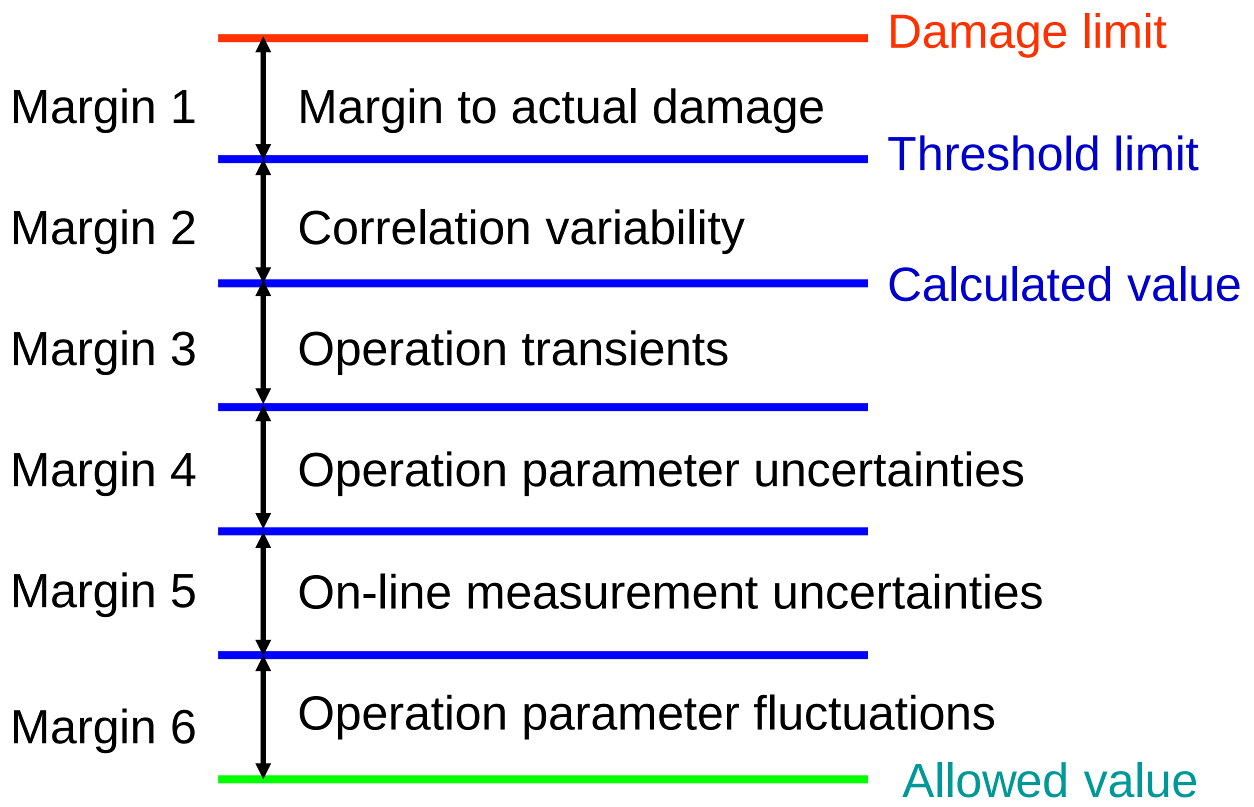
Safety Margins (8)

- Examples of regulatory probabilistic safety targets
  - Shutdown system unavailability: $< 10^{-6}$ per demand
  - Engineering safety systems unavailability: $< 10^{-3}$ per demand
  - Core damage frequency: $< 10^{-5}$ per reactor and year
  - Probability of large radioactivity release: $< 10^{-6}$ per reactor and year
  - Individual risk of fatality: $< 10^{-6}$ per reactor and year

- These criteria may vary for various countries

Safety Margins (9)

- Components of safety margins



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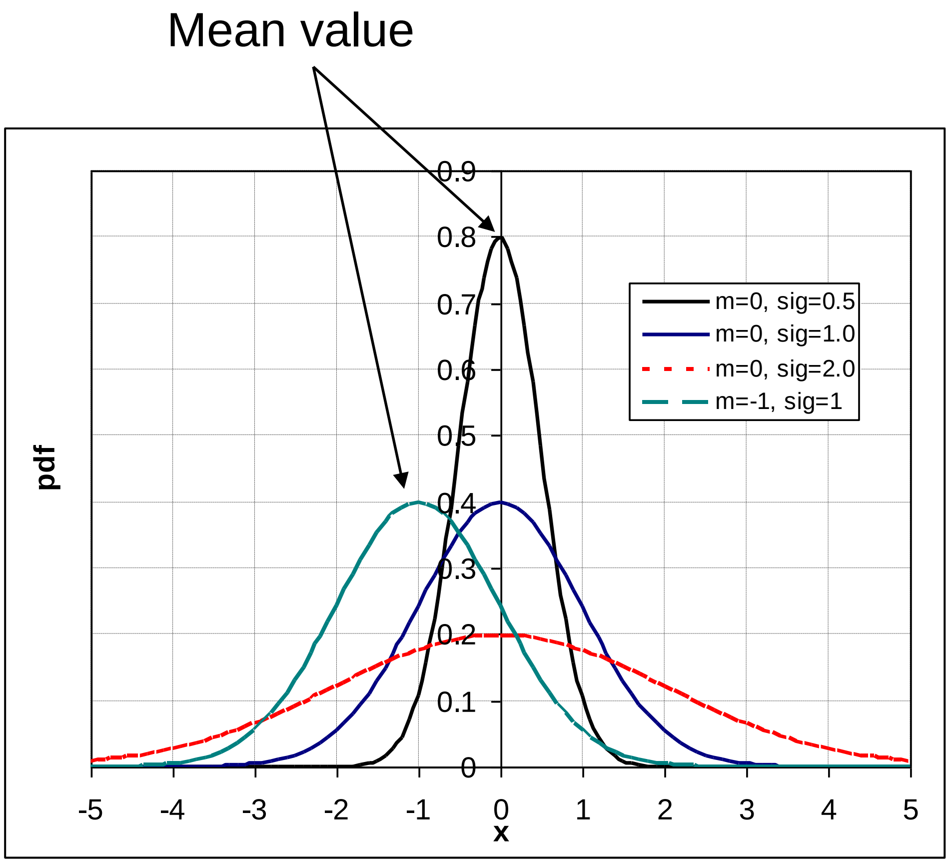
Probabilistic Assessment of CHF (1)

- Once measuring experimentally CHF, we will get a distribution of CHF values, with some mean value and a certain standard deviation

- If we perform the measurement many times, we will see that the distribution will approach the normal distribution as shown to the right

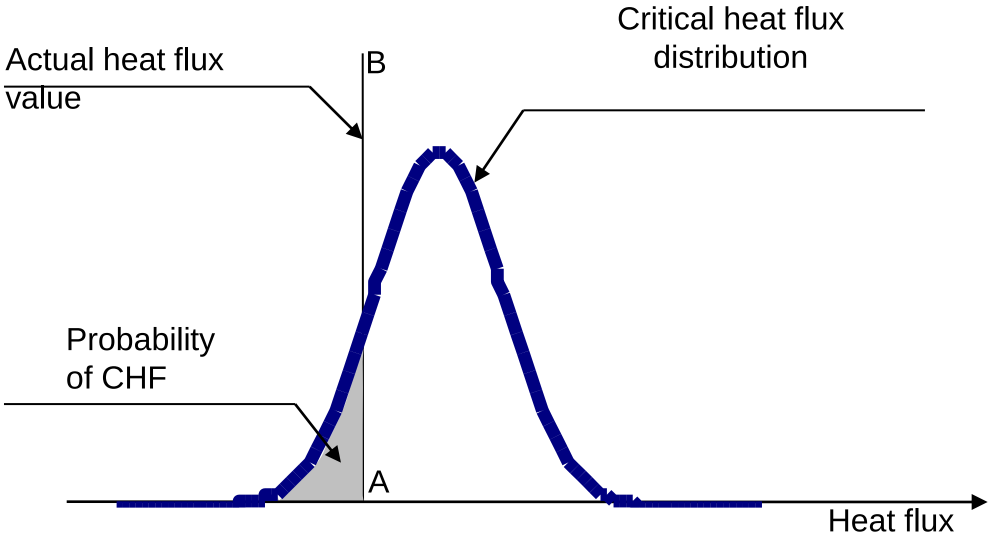


Probabilistic Assessment of CHF (2)

- Once we calculate a CHF value from a correlation, this will correspond to the mean (expected) value.

- However, the CHF may take any value in a certain range around the mean value

- Consider the situation as in the figure. The question is: what is the probability of CHF if we know the mean value of CHF, its standard deviation and the actual value of heat flux.



Probabilistic Assessment of CHF (3)

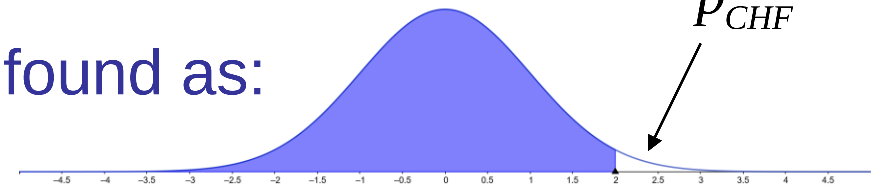
- The probability of dryout is found as:

Actual value of heat flux

Expected value of CHF

$$p_{CHF} \equiv p\left(q''_{CHF} < q''_{AB}\right) = \frac{1}{\sigma\sqrt{2\pi}} \int_0^{q''_{AB}} e^{-\frac{\left(q''_{CHF} - \overline{q''_{CHF}}\right)^2}{2\sigma^2}} dq''_{CHF}$$

- The CHF probability can be found as:

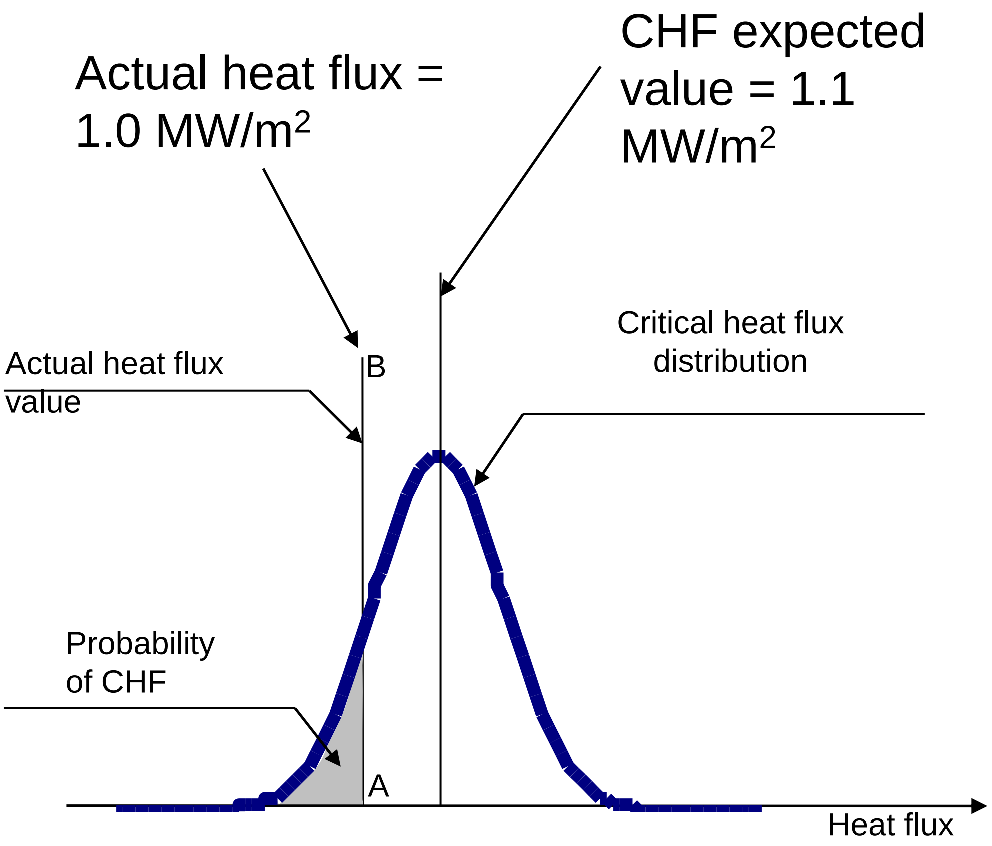


p_{CHF}

$$\underbrace{p_{CHF} = 1 - \Phi\left(\frac{\overline{q''_{CHF}} - q''_{AB}}{\sigma}\right)}_{\text{Given in Compendium}} \quad \text{where} \quad \underbrace{\Phi(z) = p(Z < z) = \frac{1}{\sqrt{2\pi}} \int_{-\infty}^{z} e^{\frac{-\xi^2}{2}} d\xi}_{\text{Cumulative standard normal distribution}}$$

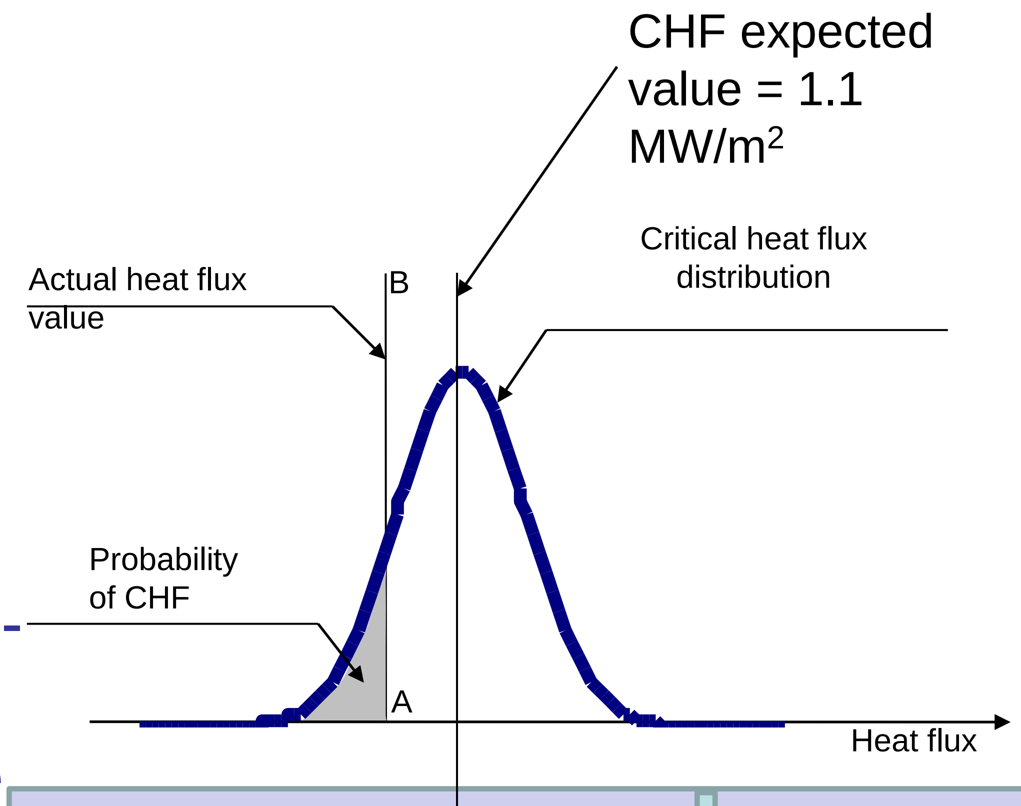
Probabilistic Assessment of CHF (4)

- **Example:** A single fuel rod operates at a constant and uniform heat flux equal to 1 MW/m^2. With the current cooling conditions, the calculated critical heat flux is 1.1 MW/m^2. The correlation developer specified that the standard deviation for the correlation is equal to 5%. Assuming the normal distribution of the CHF probability density function, calculate the probability of the CHF occurrence for the rod.



Probabilistic Assessment of CHF (5)

- **Solution:** The CHF standard deviation is equal to 0.05*1.1 = 0.055 MW/m^2. The argument of the standard cumulative function is equal to (1.1-1)/0.055 = 1.81818… ~= 1.82. From the table in APPENDIX C, the probability of CHF is found equal to 0.03438.



z	0	0.01	0.02	0.03 ...
...	...	...	...	...
1.8	0.0359	0.0351	0.03438	
....				

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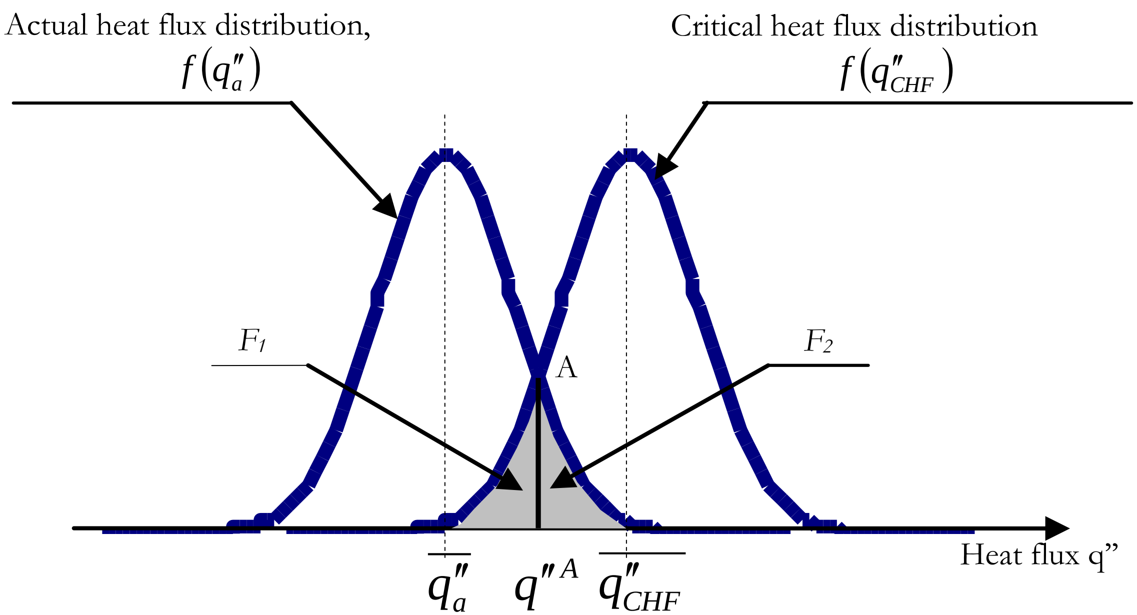
Exercise

- BWR fuel assembly has a uniform power distribution in axial and lateral directions. Calculate the maximum allowed total power in the assembly if the following condition has to be satisfied:

$$CPR >= 1.3$$

- Use the GE-CISE correlation for dryout: H = 3.76 m, G = 1200 kg/m^2.s, p = 7.17 MPa, inlet subcooling ΔT = 10 K. Assembly has 8x8 fuel rods with diameter 12.52 mm and pitch 16.3 mm and flow area A = 0.01 m^2.

CHF Limit – More Realistic Approach



Actual values of parameters in a nuclear reactor is not known exactly:

For example, the actual value of heat flux has some uncertainly

Thus, to find the probability of CHF, we need to take into account uncertainties of correlation and operation parameters.

CHF Limit – More Realistic Approach

- Since both the critical heat flux and the actual heat flux are random variables, their difference is a random variable as well:

$$\Delta q''_{CHF} \equiv q''_{CHF} - q''_a$$ is a random variable

- and it has the following distribution:

$$f\left(\Delta_{CHF}\right) = \frac{1}{\sqrt{2\pi}\cdot\sigma_\Delta}\exp\left[-\frac{\left(\Delta_{CHF} - \overline{\Delta_{CHF}}\right)^2}{2\cdot\sigma_\Delta^2}\right]$$ where

$$\overline{\Delta_{CHF}} = \overline{q''_{CHF}} - \overline{q''_a}$$

$$\sigma_\Delta^2 = \sigma_{CHF}^2 + \sigma_a^2$$

CHF Limit – More Realistic Approach

- Thus, the probabilistic formulation of CHF condition is now:

$$p\left(\Delta_{CHF} <= 0\right) = \int_{-\infty}^{0} \frac{1}{\sqrt{2\pi} \cdot \sigma_{\Delta}} \exp\left[-\frac{\left(\Delta_{CHF} - \overline{\Delta_{CHF}}\right)^2}{2 \cdot \sigma_{\Delta}^2} \right] d\left(\Delta_{CHF}\right)$$

- Clearly, to use the above expression we need to determine the standard deviations σ_{CHF}^2 and σ_a^2

Example

A nuclear reactor core contains 10000 rods operating at constant and uniform heat flux 1.0 MW/m^2. The standard deviation of the actual heat flux is known and equal to 4%. With the current cooling conditions, the calculated CHF is calculated as 1.3 MW/m^2. The standard deviation of the correlation is given as 5%. Assuming normal distributions of both the actual and the critical heat flux, calculate the number of rods that will experience CHF.

SOLUTION:

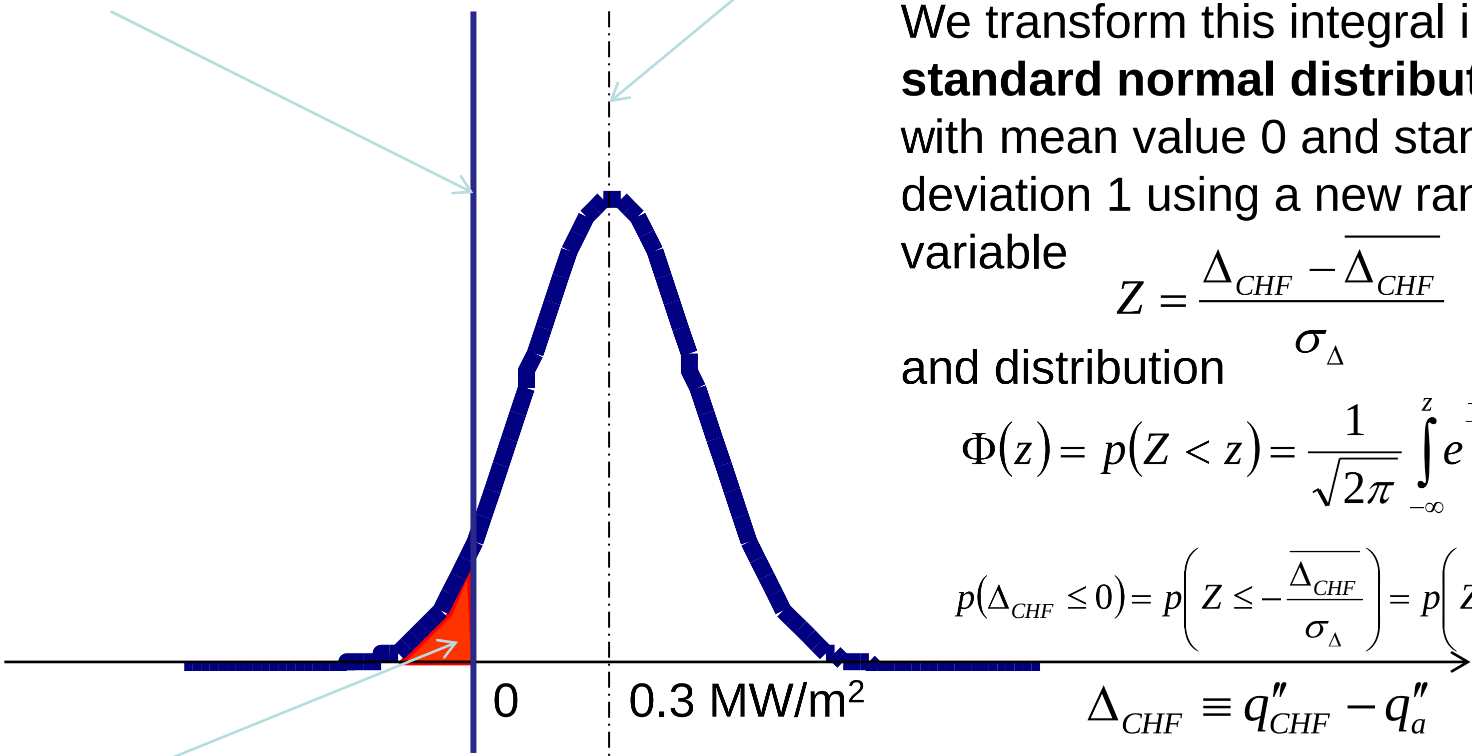
we find the standard deviations: σ_a = 0.04 x 1 MW/m^2 = 0.04 MW/m^2; σ_{CHF} = 0.05 x 1.3 MW/m^2 = 0.065 MW/m^2

thus: $\sigma_\Delta = \sqrt{(0.04)^2 + (0.065)^2} \cong 0.0763$

Example

$$\Delta_{CHF} = 0$$

$$\overline{\Delta_{CHF}} = \overline{q''_{CHF}} - \overline{q''_a}$$



$$\Delta_{CHF} \equiv q''_{CHF} - q''_a$$

We transform this integral into **standard normal distribution** with mean value 0 and standard deviation 1 using a new random variable

$$Z = \frac{\Delta_{CHF} - \overline{\Delta_{CHF}}}{\sigma_\Delta}$$

and distribution

$$\Phi(z) = p(Z < z) = \frac{1}{\sqrt{2\pi}} \int_{-\infty}^{z} e^{\frac{-\xi^2}{2}} d\xi$$

$$p(\Delta_{CHF} \le 0) = p\left(Z \le -\frac{\overline{\Delta_{CHF}}}{\sigma_\Delta}\right) = p\left(Z \ge \frac{\overline{\Delta_{CHF}}}{\sigma_\Delta}\right)$$

$$p(\Delta_{CHF} <= 0) = \int_{-\infty}^{0} \frac{1}{\sqrt{2\pi} \cdot \sigma_\Delta} \exp\left[-\frac{\left(\Delta_{CHF} - \overline{\Delta_{CHF}}\right)^2}{2 \cdot \sigma_\Delta^2}\right] d\left(\Delta_{CHF}\right)$$

Example

The argument of the cumulative standard normal distribution is:

z = [0-(1.3 – 1.0)]/0.0763 ~= -3.93

we use the following property of the standard normal dis.

$p_{CHF} = \Phi(z) = 1 - \Phi(-z)$ where $\Phi(z) = p(Z \le z) = \frac{1}{\sqrt{2\pi}} \int_{-\infty}^{z} e^{\frac{-\xi^2}{2}} d\xi$

In Appendix C of the compendium we find, that the corresponding probability is: p_{CHF} = 4.25e-5.

Thus, assuming independence of CHF for all rods, the number of rods under CHF is p_{CHF} x N_{rod} = 0.425 < 1.