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SUB-CHAPTER 4.3 – NUCLEAR DESIGN

0. SAFETY REQUIREMENTS

0.1. SAFETY FUNCTIONS

The safety functional requirements met by the neutronic core design are:

- Control of core reactivity to enable the chain reaction to be stopped under all circumstances and to return the reactor to a safe state
- Removal of heat produced in the fuel via the coolant fluid
- Containment of radioactive substances (actinides and fission products) inside the first barrier.

The nuclear design must ensure these safety functions are achieved for all design basis operating conditions, Plant Condition Category PCC-1 to PCC-4 (see Chapter 14), and contribute to achieving the safety functions in conditions corresponding to the Risk Reduction Categories RRC-A and RRC-B (see Sub-chapters 16.1 and 16.2).

0.2. FUNCTIONAL CRITERIA

0.2.1. Controlling core reactivity

Core reactivity must be controlled under all normal operating conditions from start-up to shutdown with the help of two methods that are functionally diverse.

One consists of the Rod Cluster Control Assembly (RCCA), the other of variations in the concentration of soluble boron in the coolant.

When the core is critical and whatever the power level, the neutronic feedbacks must be such that the reactor is inherently stable in the event of a power excursion.

0.2.2. Removal of heat produced in the fuel

At any point in the core, the heat produced must be limited so that:

- Its removal can be guaranteed under normal and incident operating conditions by maintaining efficient heat transfer between the fuel rod and the coolant
- It does not cause degradation in the geometry of the core in the event of an accident
- It remains within limits compatible with the mechanical design of the fuel assembly.

0.2.3. Containment of radioactive products

To guarantee containment, the thermal-mechanical conditions imposed on the fuel cladding must be such that its integrity is ensured under normal and incident operating conditions.

0.3. DESIGN REQUIREMENTS

The safety functions related to nuclear design require the application of a quality assurance program whose aim is to document and monitor activities related to the design.

0.4. TESTING

0.4.1. Pre-operational tests

Compliance of the core with design studies must be verified by physical tests at the beginning of each cycle.

0.4.2. In-service monitoring

Compliance of the core with design studies must be verified during the entire cycle by monitoring critical boron concentration and obtaining regular core flux maps.

0.4.3. Periodic tests

Not applicable.

1. DESIGN BASES

This section describes the design bases and functional requirements used in the nuclear design of the fuel and reactivity control system.

The reactivity variations are presented in pcm: reactivity in pcm = 10^5 $\Delta\rho$ where $\Delta\rho$ is calculated from two state point values of K_{eff}:

$$\Delta\rho = \frac{1}{K1} - \frac{1}{K2}$$

where: K1 = K_{eff} of initial state

K2 = K_{eff} of final state

As stated in section 0.1, safety functions must be fulfilled in all conditions of plant operation.

The four major plant operation categories consist of the normal operational states anticipated for normal plant operation and are enlarged by systematically looking for abnormal events having the potential to disturb safety functions.

These events are divided into four PCCs, according to their estimated frequency of occurrence (see Sub-chapter 14.0).

The safety criteria are the criteria that must be met in the safety analysis. They are defined in terms of radiological limits.

In addition to these safety criteria, it is convenient to introduce in practice some decoupling criteria. These decoupling criteria are defined in terms of behaviour of the barriers. They provide a guaranty that the safety criteria (i.e. the radiological limits; see Sub-chapter 14.0) will be met. In fact, they provide a decoupling between the thermal-hydraulic calculations and the radiological calculations, so that these two types of calculations can be easily performed separately.

The core design power distribution limits, related to safety criteria for plant condition category 1 occurrences, are met through conservative design and maintained by the action of the control system. The requirements for plant condition category 2 occurrences are met by providing an adequate protection system which monitors reactor parameters.

1.1. FUEL BURNUP

Basis

The nuclear design basis is to provide sufficient reactivity in the fuel to attain the expected region discharge burnup.

Discussion

Fuel burnup is a measure of fuel depletion, which represents the integrated energy output of the fuel and is a convenient means for quantifying fuel exposure criteria.

The cycle length or design discharge burnup is achieved by providing sufficient initial excess reactivity in each fuel region and by following a fuel replacement program (such as those described in section 2) that meets all safety-related criteria in each cycle of operation.

The initial reactivity excess in the fuel, although not a design basis, must be sufficient to maintain core criticality at full power operating conditions throughout the cycle life with equilibrium xenon, samarium, and other fission products present. The end of design cycle life is defined to occur when the chemical poison concentration is essentially zero with control rods present to the degree necessary for operational requirements. In terms of soluble boron concentration, this represents approximately 10 ppm with no control rod insertion.

A limitation on initial excess reactivity is not required other than that defined in terms of other design bases, such as core reactivity coefficients and shutdown margin, as discussed below.

1.2. REACTIVITY COEFFICIENTS

Basis

The fuel temperature coefficient is negative and the moderator temperature coefficient of reactivity is, in principle, kept negative from hot zero power to nominal conditions with all the control rods out of the core. The coolant void coefficient is required to be negative for all conditions. Nevertheless, some fuel management regimes could lead to high boron concentrations at the beginning of life of the core and consequently to a positive moderator temperature coefficient.

Discussion

When compensation for a rapid increase in reactivity is considered, there are two major effects. These are the resonance absorption effects (Doppler) associated with changing fuel temperature and the reactivity effects (variations in spectrum and boron absorption) resulting from changing moderator density. These basic physics characteristics are usually represented as reactivity coefficients. The use of slightly enriched uranium ensures that the Doppler coefficient of reactivity is negative. This coefficient provides the most rapid reactivity compensation. The core is also designed to have in principle a negative moderator temperature coefficient of reactivity from hot zero power to nominal power so that average coolant temperature or void content provides another slower negative feedback effect.

The negative moderator temperature coefficient can be achieved through use of fixed burnable absorber and/or control rods to limit the soluble boron concentration. Burnable absorber content (quantity and distribution) is not stated as a design basis, other than by its contribution to achieving a negative moderator temperature coefficient.

The fuel management regimes presented in this document have been designed on the basis of a negative moderator temperature coefficient at hot zero power with all rod cluster control assemblies out.

1.3. CONTROL OF POWER DISTRIBUTION

Basis

The nuclear design basis is that, with at least a 95% confidence level:

a) Fuel linear power density at the hot spot is not greater than the limit given in Sub-chapter 4.3 - Table 1 under normal operating conditions.

b) Under abnormal conditions, including the maximum overpower condition, the fuel peak power will not cause melting,

c) The fuel will not operate with a power distribution that violates the departure from nucleate boiling (DNB) design basis under plant condition category 1 and 2 events, including the maximum overpower condition,

d) Fuel management will be such as to produce rod powers and burnups consistent with the assumptions used in the fuel rod mechanical integrity analysis.

Discussion

The analysis of extreme power shapes which affect fuel design limits is performed with proven methods (see Appendix 4). The conditions under which limiting power shapes are assumed to occur are chosen conservatively with regard to any permissible operating state.

Even though there is good agreement between the peak power calculations and the measurements, a nuclear uncertainty (see section 3.1) is applied to the calculated peak local power. Such an uncertainty is provided both for the analysis of normal operating states and for anticipated transients.

1.4. MAXIMUM CONTROLLED REACTIVITY INSERTION RATE

Basis

The maximum reactivity insertion rate due to withdrawal of rod cluster control assemblies at power or by boron dilution is limited. For normal operation at power the maximum rate of change of reactivity due to accidental withdrawal of control banks is set (see Sub-chapter 4.3 - Table 1) such that the peak heat generation rate and the Departure from Nucleate Boiling Ratio (DNBR) do not exceed the limits at overpower conditions.

Discussion

Reactivity addition associated with accidental withdrawal of a control bank (or banks) is limited by the maximum rod speed (or travel rate) and by the worth of the bank(s) (see section 6). The maximum control rod speed (see Sub-chapter 4.3 - Table 1) ensures that the maximum rate of change of reactivity due to accidental withdrawal of control banks is lower than the design limit. During normal operation at power, the maximum rate of change of reactivity is less than the design value of the maximum controlled rate of change of reactivity.

The reactivity change rates are conservatively calculated assuming pessimistic axial power and xenon distributions. The peak xenon burnout rate is significantly lower than the maximum reactivity addition rate for normal operation (see Sub-chapter 4.3 - Table 1).

1.5. SHUTDOWN MARGINS

Basis

An adequate shutdown margin and a sub-critical core are required in the at-power and shutdown conditions, respectively.

Discussion

Two independent reactivity control systems are provided, namely control rods and soluble boron in the coolant.

Movement of the control rods compensates for the reactivity effects of the fuel and moderator temperature changes accompanying power level changes over the range from full-load to no-load. In addition, the control rods provide the minimum shutdown margin under plant condition category events and are capable of making the core sub-critical rapidly enough to prevent fuel damage from exceeding acceptable limits, assuming that the highest worth control rod is stuck out following trip.

Changes in the soluble boron concentration in the reactor coolant compensate for all xenon depletion and density reactivity changes and enable the reactor to go to and maintain cold shutdown. Thus, shutdown is provided by both a mechanical and a chemical poison control system.

1.6. SUB-CRITICALITY

Basis

When fuel assemblies are in the pressure vessel and the vessel head is opened or being removed, the core must be maintained sufficiently sub-critical to guarantee the safety of the reactor in case of an accidental transient occurring in this state. The accidental transients considered are boron dilution and removal of all rod cluster control assemblies.

Discussion

The boron concentration required to meet the refuelling shutdown criteria is specified in Sub-chapter 4.3 - Table 4.

1.7. STABILITY

Basis

The plant is inherently stable to power oscillations at the fundamental mode. Spatial power oscillations within the core with a constant core power output, should they occur, can be reliably and readily detected and suppressed.

Discussion

Oscillations of the total power output of the core, from whatever cause, are readily detected by the control and protection systems. The core is protected by these systems with successive countermeasures (to reduce power), culminating in a reactor trip which would occur if the power increases unacceptably, to preserve the design margins to fuel design limits. The stability of the turbine/steam generator/core systems and the reactor control system is such that total core power oscillations are not normally possible. The redundancy in the trip system ensures an extremely low probability of exceeding design power limits.

The core is designed so that radial and azimuthal oscillations due to spatial xenon effects are self-damping and no operator action or control action is required to suppress them. The stability to radial oscillations is such that this excitation is highly improbable. Convergent azimuthal oscillations can be excited by, among other things, prohibited movement of individual control rods. Such oscillations are readily observable, and are alarmed using the in-core flux measuring instrumentation. In all presently proposed cores, these oscillations in the horizontal plane are self-damping by virtue of reactivity feedback effects designed into the core.

However, axial xenon spatial power oscillations may occur during operation. The control banks are provided for control and monitoring of axial power distributions. The protection system ensures that fuel design limits are not exceeded.

2. OVERALL DESCRIPTION OF THE CORE

First of all, it must be noted that the fuel management regimes presented in the following have been defined as conservative alternatives to support the design. Cycle lengths of 12, 18 and 22 months, INOUT fuel management types, and uranium or MOX fuel are considered in order to provide high flexibility. This leads to a conservative set of data for the plant design and safety studies. This also covers other particular fuel management regimes, following verification that their corresponding parameters remain compatible with the design set of data.

The reactor core consists of a specified number of fuel rods which are held in bundles by spacer grids and top and bottom fittings. The fuel rods consist of uranium or MOX (uranium plus plutonium) pellets stacked in M5 cladding tubes plugged and seal welded to encapsulate the fuel. The bundles, known as fuel assemblies, are arranged in a pattern within the core which approximates a right circular cylinder.

Each fuel assembly contains a 17x17 rod array composed of 265 fuel rods and 24 guide thimbles.

The fuel rods within a given fuel assembly without gadolinium have the same uranium enrichment in both the radial and axial planes. In fuel assemblies with gadolinium, the gadolinium rods do not have the same U-235 enrichment as the normal rods. At the present stage it is assumed that the fuel rods within a given MOX assembly have the same uranium and plutonium enrichment in the axial direction. For the example considered the MOX assemblies are divided into three different radial enrichment zones (see Sub-chapter 4.3 - Figure 1) in order to minimise the interface power peaks induced by the proximity of a uranium fuel assembly. The MOX fuel consists of a depleted uranium matrix and several plutonium isotopes (see Sub-chapter 4.3 - Table 2).

Uranium fuel assemblies of different enrichments are used in the initial core loading to establish a favourable radial power distribution. Sub-chapter 4.3 - Figure 2 shows the uranium fuel loading pattern to be used in the first core. Two regions consisting of the two lower enrichments are interspersed so as to form a checkerboard pattern in the central portion of the core. The third region is arranged around the periphery of the core and contains the highest enrichment. The enrichments for the first core are shown in Sub-chapter 4.3 - Table 2. *

The reloading pattern, initial and final positions of fuel assemblies, and the number of fresh fuel assemblies and their placement depend on the energy requirement for the next cycle and the burnup and power histories of the previous cycles. The enrichment of the fresh fuel assemblies for the next cycle depends on the type of fuel management regime and is shown in Sub-chapter 4.3 - Table 2. High enrichments are used which allow increased burnup. Enrichments up to 5% wt are used for uranium fuel. Single fuel assemblies of this type remain sub-critical (K_{eff} less than 0.95) when placed in pure water up to this enrichment limit. For MOX fuel, the maximum anticipated fissile Pu enrichment in one fuel rod is limited due to fabrication constraints to 7.44%wt. To maintain a negative void coefficient, the fissile Pu enrichment is limited to 7.0%wt on the average in every MOX assembly. The reloading patterns of the equilibrium cycles of each type of fuel management regime are provided in Sub-chapter 4.3 - Figure 3 to Sub-chapter 4.3 - Figure 5.

The core average enrichment is determined by the amount of fissionable material required to provide the desired cycle length and energy requirements. The physics of the burnout process are such that operation of the reactor depletes the amount of fuel available due to the absorption of neutrons by the fissile atoms and their subsequent fission. The rate of depletion of fissile atoms is directly proportional to the power level at which the reactor is operated.

In addition, the fission process results in the formation of fission products, some of which readily absorb neutrons. These effects, depletion and the build-up of fission products, are partially offset by the build-up of plutonium which occurs due to the non-fission absorption of neutrons in U-238. Therefore, at the beginning of any cycle, a reactivity reserve equal to the depletion of the fissionable fuel and the build-up of fission product poisons over the specified cycle life must be "built" into the reactor. This reactivity excess is controlled by removable neutron absorbing material in the form of boron dissolved in the primary coolant and by burnable poisons.

Burnable poisons co-mixed with the fuel material itself are used to avoid an excessive soluble boron concentration and hence to avoid a positive moderator temperature coefficient at beginning of life. During operation the poison content in these rods is depleted, thus adding positive reactivity to offset some of the negative reactivity from fuel depletion and fission product build-up. The depletion rate of the burnable poison is not critical since chemical poison is always available and flexible enough to cover any possible deviations in the expected burnable poison depletion rate.

In addition to reactivity control, the burnable poison is strategically located to provide a favourable radial power distribution. Sub-chapter 4.3 - Figure 6 shows examples of burnable poison distributions within a fuel assembly for the several burnable patterns used in a 17 x 17 array as they were used in the present power distribution calculations.

Sub-chapter 4.3 - Tables 1 to 4 contain a summary of the reactor core design parameters including reactivity coefficients, delayed neutron fraction, and neutron lifetimes.

3. POWER DISTRIBUTIONS

The safety demonstration relies to a certain extent on calculated power distributions. With respect to the qualification of the tools used for this purpose (see Appendix 4), the accuracy of power distribution calculations has been confirmed by flux mapping on existing plants [Ref-1].

3.1. DEFINITIONS

Power distributions are quantified in terms of hot channel factors. These factors are a measure of the peak pellet power within the reactor core and the total energy produced in a coolant channel, and are expressed in terms of parameters related to the nuclear or thermal design.

The factors used in the discussion of power distributions in this section are defined as follows:

- Fxy(z), radial peaking factor at elevation z, defined as the ratio of the peak power density to the average power density at elevation z,
- P(z), average axial power distribution, defined as the ratio of the average linear power density at elevation z to the average linear power density,
- Q(z), maximum linear power at elevation z, is defined as the maximum local fuel rod linear power density at elevation z divided by the average linear power density:
    - $Q(z) = F_{XY}(z) \times P(z)$

- FQ, heat flux hot channel factor, defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density:
  - FQ = Max Q(z), without uncertainties and penalties.

The following uncertainties and penalties are applied to the design calculated values of F_Q:

- F_Q^E engineering heat flux hot channel factor is the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of the fuel rod and eccentricity of the gap between pellets and cladding.
- F_U^N = factor for conservatism, taking into account the uncertainties of power distribution calculations,
- F_Q^B = rod bow penalty,
- Fxe = xenon penalty for azimuthal and radial oscillations,
- F_I = total uncertainty factor taking into account the above factors

The design peaking factor, including uncertainties and penalties is thus:

$$F_Q^D = F_Q * F_I$$

$F_{\Delta H}^N$, nuclear enthalpy rise hot channel factor is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power (see Sub-chapter 4.4).

Manufacturing tolerances, hot channel power distribution, and surrounding channel power distributions are treated explicitly in the calculation of the DNB ratio described in Sub-chapter 4.4, related to thermal and hydraulic design.

3.2. RADIAL POWER DISTRIBUTION

The core radial power shape at any given height is a function of:

- The loading pattern of fuel assemblies,
- The location of poisoned rods,
- The insertion of rod cluster control assemblies,
- The core burnup,
- The power level and the moderator density,
- The concentration and the distribution of xenon and samarium.

The effect of non-uniform flow distribution is negligible.

Sub-chapter 4.3 - Figures 7 to 14 show the radial power distributions per assembly for one quarter of the core for different burnup steps of cycle 1 and equilibrium cycle for the different types of fuel management regime [Ref-1]. These distributions are obtained by integrating the nuclear power over the core height.

The power of the hot channel in the core results from the superposition of the macroscopic power distribution in the core and the pin-by-pin distribution in the assembly. For the purpose of illustration, assembly pin-by-pin power distributions at the beginning of life and end of life for cycle 1 are given for the same assembly in Sub-chapter 4.3 - Figures 15 and 16 respectively, and at the beginning of life and end of life for the 18 month equilibrium cycle in Sub-chapter 4.3 - Figures 17 and 18 respectively.

Since the position of the hot channel varies with time, a single reference radial design power distribution is selected for DNB calculations. This reference power distribution is chosen conservatively to concentrate power in one area of the core, minimising the benefits of flow redistribution. Assembly powers are normalised to core average power.

Since the detailed power distribution surrounding the hot channel varies with time, a conservatively flat assembly power distribution is assumed in the DNB analysis (see Sub-chapter 4.4), with the maximum rod integrated power artificially raised to the design value of $F^N_{\Delta H}$. Care is taken in the nuclear design of all fuel cycles and all operating conditions to ensure that a flatter assembly power distribution does not occur with limiting values of $F^N_{\Delta H}$.

3.3. AXIAL OR 3D POWER DISTRIBUTION

The axial power profile depends mainly on:

- The insertion of control rods
- The power level
- The axial xenon distribution
- The Doppler and moderator density feedback effects
- The fuel burnup
- The axial design of the fuel assembly (for optimisation of axial power shape).

For the purpose of illustration, some axial power shapes for UO_2 cycle 1 fuel management at different burnup are shown in Sub-chapter 4.3 - Figures 19 and 20.

Signals are available from in-core or ex-core flux instrumentation. These signals are used for core monitoring during normal operation to determine the average axial power distribution of the core, which is characterised by the Axial Offset (AO) or the ΔI defined below:

$$\text{Axial Offset} = AO = \frac{P_T - P_B}{P_T + P_B}$$

where P_T and P_B are the power fraction in the top and bottom halves of the core.

$$\Delta I = \frac{P_T - P_B}{(P_T + P_B)_{\text{nominal}}} = AO \times \text{Pr}$$

where Pr is the ratio of actual to nominal power level.

3.4. LIMITING POWER DISTRIBUTION

The fuel management regime and the control rod location are chosen to limit fluctuations in the radial power distribution during normal operation.

The control rod worth and insertion are chosen to limit the fluctuation of the axial power distribution.

In order to limit the axial power oscillation due to xenon, the axial power distribution is controlled by maintaining the axial offset within a target operating band. This minimises xenon transient effects on the axial power shape since the xenon distribution is kept in phase with the power distribution.

The worst or limiting power distribution which can occur during normal operation (PCC-1 events) is considered as the starting point for analysis of PCC-2 to PCC-4 events, as described in Sub-chapters 14.3 to 14.5. These limiting power distributions are generated in a conservative way; nevertheless they fulfil the surveillance limitations on the maximum linear power density Q(z) and on the DNBR. The Instrumentation and Control System is designed to ensure operation within these limits.

3.5. EXPERIMENTAL VERIFICATION

3.5.1. Power distribution analysis

The calculation uncertainty F_U^N to be applied for F_Q is 5% [Ref-1].

The calculation uncertainty to be applied for $F_{\Delta H}$ is 4% [Ref-1].

3.5.2. Core conformity

A series of physics tests is performed on the first core. The main purpose of the tests is to provide a check on the computation methods used in the predictions of the conditions of the test.

Measurements of core reactivity, control rod worth and power distribution enable experimental verification of the calculations and confirmation of the correctness of core build.

3.5.3. Online surveillance and protection systems

The information relevant to the core monitoring instrumentation is summarised in Sub-chapter 4.4 on thermal-hydraulic design. The in-core and ex-core instrumentation provide the monitoring of power distributions required for the on-line surveillance and protection systems. These systems are described in Sub-chapter 4.4.

4. REACTIVITY COEFFICIENTS

The kinetic characteristics of the reactor core determine its response to changing plant conditions or to operator adjustments made during normal operation, as well as during abnormal (including accidental) transients. The reactivity coefficients reflect the changes in the neutron multiplication due to varying plant conditions, such as power, moderator temperature, or fuel temperatures. Since reactivity coefficients change during the life of the core, ranges of coefficients are employed in transient analysis to determine the response of the plant throughout life.

Quantitative information on calculated reactivity coefficients, including fuel/Doppler coefficient, moderator coefficients (density, temperature), and power coefficient is given in the following sections.

4.1. FUEL TEMPERATURE (DOPPLER) COEFFICIENT

The fuel temperature (Doppler) coefficient is defined as the change in reactivity per degree Celsius change in effective fuel temperature, and is primarily a measure of the Doppler broadening of U-238 and Pu-240. An increase in fuel temperature increases the effective resonance absorption cross-sections of the fuel and produces a corresponding reduction in reactivity.

The Doppler temperature coefficient is shown in Sub-chapter 4.3 - Figures 21 to 24 as a function of the effective fuel temperature for cycle 1 and the equilibrium cycles of the different fuel management regimes.

When the power becomes non-negligible, the effective fuel temperature is no longer equal to the moderator temperature but varies as a function of core power. This effect is taken into account in the Doppler power coefficient. The integral of the Doppler power coefficient as a function of relative power is the Doppler contribution to the power defect defined later (assuming that the moderator temperature varies according to the part load diagram) (see section 4.3).

4.2. MODERATOR COEFFICIENT

The moderator coefficient is a measure of the change in reactivity due to a change in specific coolant parameters such as density or temperature. The coefficients so obtained are moderator density and temperature coefficients.

1. Moderator density and temperature coefficient

The moderator temperature (density) coefficient is defined as the change in reactivity per degree Celsius change in the moderator temperature. Generally, the effects of the changes in moderator density as well as the temperature are considered together. A decrease in moderator density means less moderation which results in a negative moderator coefficient. An increase in coolant temperature keeping the density constant (obtained by a pressure increase), leads to a hardened neutron spectrum and results in an increase in the resonance absorption in U-238, Pu-240 and other isotopes. The hardened spectrum also causes a decrease in the fission to capture ratio in U-235 and Pu-239. Both of these effects make the moderator coefficient more negative. Since water density changes more rapidly with temperature as temperature increases, the moderator temperature (density) coefficient becomes more negative with increasing temperature.

The soluble boron used in the reactor as a means of reactivity control also has an effect on moderator density coefficient, since both the soluble boron poison density and the water density decrease when the coolant temperature rises. A decrease in the soluble poison concentration, resulting from the moderator density reduction, introduces a positive component in the moderator coefficient.

Thus, if the concentration of soluble poison is large enough, the net value of the coefficient may be positive. With the burnable poison present, however, the initial boron concentration in hot conditions is sufficiently low that the moderator temperature coefficient meets the criterion presented in section 1.2.

With burnup, the moderator coefficient becomes more negative, primarily as a result of boric acid dilution but also to a significant extent from the effects of the build-up of plutonium and fission products.

The moderator temperature (density) coefficients are presented for cycle 1 and the equilibrium cycles of the different fuel management regimes in Sub-chapter 4.3 - Table 3 for the following core configurations:

- All rods out for hot zero power and nominal power with critical boron concentration at nominal power at beginning of life and end of life,
- All rods out for hot zero power with critical boron concentration at hot zero power at beginning of life.

2. Moderator void coefficient

The moderator void coefficient relates the change in neutron multiplication to the presence of voids in the moderator. In a PWR this coefficient is not very significant because of the low void content in the coolant. The core void content is less than half a percent and is due to local or random boiling. Typically the void coefficient value is close to -250 pcm/percent void at end of life and at operating temperatures. The negative void coefficient at operating temperature becomes more negative with fuel burnup.

4.3. POWER COEFFICIENT

The combined effect of moderator temperature and fuel temperature changes as the core power level changes (the moderator temperature varies according to the part load relationship shown in Sub-chapter 4.4). This is called the total power coefficient and is expressed as the reactivity change per percent power change.

The total power coefficient becomes more negative with burnup, reflecting the combined effects of moderator and fuel temperature coefficients with burnup. The power defect (integral reactivity effect due to a power variation) at beginning of life and end of life with the critical boron concentration at nominal power is presented in Sub-chapter 4.3 - Table 3 for cycle 1 and equilibrium cycles of the different fuel management regimes.

4.4. REACTIVITY COEFFICIENTS USED IN TRANSIENT ANALYSIS

Sub-chapter 4.3 - Table 3 gives the limiting values for the reactivity coefficients. The limiting values are used as design limits in the transient analysis. The exact values of the coefficient used in the analysis depend on whether the transient of interest is examined at the beginning of life or end of life, whether the most negative or the most positive (least negative) coefficients are appropriate, and whether spatial non uniformity must be considered in the analysis. Conservative values of coefficients, considering various aspects of analysis are used in the transient analysis.

The reactivity coefficients presented in sections 4.1 through 4.3 are best estimate values calculated for cycle 1 and the equilibrium cycle of the different fuel management regimes. The limiting values shown in Sub-chapter 4.3 - Table 3 are chosen to encompass the best estimate reactivity coefficients, including the uncertainties given in section 10.3 over appropriate operating conditions calculated for these cycles and the expected values for the subsequent cycles. The most positive as well as the most negative values are selected to form the design basis range used in the transient analysis. The need for a re-analysis of any accident in a subsequent cycle depends on whether or not the coefficients for that cycle fall within the identified range used in the analysis with due allowance for the computation uncertainties given in section 10.3.

5. CORE CONTROL

5.1. CONTROL REQUIREMENTS AND PRINCIPLES

5.1.1. Burnup

Excess reactivity is installed at the beginning of each cycle to provide sufficient reactivity to compensate for fuel depletion and the build-up of fission products during the cycle. The reactivity is controlled by the addition of soluble boron to the coolant and by burnable poison.

5.1.2. Xenon and samarium poisoning

Changes in xenon and samarium concentrations in the core occur at a sufficiently slow rate, even following rapid power level changes, that the resulting reactivity change is controlled by changing the soluble boron concentration.

5.1.3. Start-up at xenon peak

Compensation for the xenon peak build-up is accomplished by changing the concentration of soluble boron in the coolant. Start-up from the xenon peak condition is accomplished with a combination of rod withdrawal and boron dilution. The boron dilution may be made at any time, including during the shutdown period, provided the required sub-criticality for shutdown is maintained.

5.1.4. Load follow control and xenon control

During load follow manoeuvres, power changes are accomplished using control rod movement and dilution or boration as required. Control rod movement is limited by the control rod insertion limits. The power distribution is maintained within acceptable limits through the location of the rod bank. Reactivity changes due to the changing xenon concentration can be controlled by rod movement and/or changes in the soluble boron concentration.

5.1.5. Sub-criticality and shutdown margin

To ensure the shutdown margin under conditions where a cooldown to ambient temperature is required, concentrated soluble boron is added to the coolant which adds to the poisoning of the core by the rod cluster control assemblies. For all core conditions including refuelling, the boron concentration is well below the solubility limit.

In order to maintain a zero load state at hot or cold conditions, it is necessary to guarantee the sub-criticality of the core in the operational shutdown states. However, fault transients can occur from these states. It is thus necessary to maintain sufficient sub-criticality in order to ensure that the consequences of such faults are acceptable.

The following faults are taken into account in deriving the required shutdown margin:

- Rod ejection, steam line break and dilution transients, from zero power operational shutdown state with a closed vessel and a temperature from hot to ambient

- Inadvertent withdrawal of all rods when lifting the vessel head and boron dilution faults from the cold zero power operational shutdown state with the vessel open

For reactor trip, it is necessary to demonstrate the ability to achieve core shutdown to hot zero power assuming an unchanged xenon level. For automatic partial cooldown, the Extra Boration System (RBS [EBS]) is not automatically actuated; the control rods are therefore required to ensure sub-criticality at the end of this cooldown transient, corresponding to actuation of safety injection. This is achieved by comparing the difference between the rod cluster control assembly reactivity available, with an allowance for the worst stuck rod, with that required for control and protection purposes. The largest reactivity control requirement is at the end of life when the moderator temperature coefficient reaches its most negative value, as reflected in the larger power defect.

The control rods are required to provide sufficient reactivity to account for the power defect from full to hot zero power or to the end of the partial cooldown transient, assuming unchanged xenon level, and to provide the required shutdown margin. The reactivity addition resulting from power reduction consists of contributions from Doppler, the change in average moderator temperature, flux redistribution, and reduction in void content as discussed later.

5.2. MEANS OF CONTROL

Core reactivity is controlled by means of a chemical poison dissolved in the coolant, rod cluster control assemblies, and burnable poison rods as described below.

5.2.1. Chemical poison

Boron in solution as boric acid is used to control relatively slow reactivity changes associated with:

a) The moderator temperature defect in going from cold shutdown at ambient temperature to the hot operating temperature at zero power

a) The xenon and samarium poisoning transient, such as that following power changes or changes in rod cluster control position

b) The excess reactivity required to compensate for the effects of fissile inventory depletion and build-up of long-life fission products

c) The burnable poison depletion.

The boron concentrations for various core conditions are presented in Sub-chapter 4.3 - Table 4 for natural boron [Ref-1]. The critical boron concentration at nominal power as a function of burnup is presented in Sub-chapter 4.3 - Figures 25 to 28. Due to the high values of natural boron concentrations required, B-10 enriched boron will be used, with the objective of having boron concentrations lower than 1400 ppm at beginning of life, nominal power without xenon (see Sub-chapter 5.5 and Chapter 14).

5.2.2. Burnable poison

Burnable poison provides partial control of the excess reactivity present during the fuel cycle. In doing so, this poison allows the moderator temperature coefficient to meet the criterion presented in section 1.2. It performs this function by reducing the requirement for soluble poison in the moderator at the beginning of the first fuel cycle as described previously. For purposes of illustration, typical burnable poison patterns in the core are shown in Sub-chapter 4.3 - Figures 2 to 5 [Ref-1], while the arrangements within an assembly are displayed in Sub-chapter 4.3 - Figure 6. The poison in the rods is depleted with burnup but at a sufficiently slow rate that the resulting critical concentration of soluble boron is such that the criterion on the moderator temperature coefficient is always met.

5.2.3. Rod cluster control assemblies

The number of rod cluster control assemblies is shown in Sub-chapter 4.3 - Table 2. At the present stage of the design the core neutronics, calculations assume that they consist of AIC-B_4C but this may change. The rod cluster control assemblies are used for shutdown and control purposes to make the reactivity changes required for:

a) Reactor trip, including the required shutdown margin at hot zero power and at the end of a partial cooldown transient, assuming a stuck rod

b) An increase in power above hot zero power (to accommodate the reactivity changes due to the fuel and moderator power coefficients)

c) Variations in boron concentration, coolant temperature, or xenon concentration (with rods not exceeding the allowable rod insertion limits)

d) Load changes.

The allowed control bank reactivity insertion is limited at full power to maintain the required shutdown capability. As the power level is reduced, the control rod reactivity requirements are also reduced and more rod insertion is allowed. In addition, the rod cluster control assembly withdrawal pattern determined from these analyses is used in determining the maximum worth of an inserted rod cluster control assembly ejection accident.

The arrangement of the rod cluster control assemblies is shown in Sub-chapter 4.3 - Figure 29.

Core control

The main objective of core control is to facilitate operations by ensuring simultaneous control of temperature and Axial Offset.

Xenon oscillations which could be the consequences of perturbations of the axial power distribution could lead to loss of operating margins.

Core control limits xenon oscillations (sometimes due to slight Axial Offset perturbations) by continuously controlling Axial Offset.

Axial Offset is controlled within a band (called the Axial Offset dead band) by logic which prioritises control rod movements. Since the control of primary average temperature has priority, this control is ensured even if the control rods movements do not maintain the Axial Offset within its dead band in the very short term.

Soluble boron compensates only for slow changes in reactivity.

6. CONTROL ROD PATTERNS AND REACTIVITY WORTH

The terms "group" and "bank" are used synonymously throughout this report to describe a particular grouping of rod cluster control assemblies. The rod cluster assembly pattern is shown in Sub-chapter 4.3 - Figure 29.

The axial position of the rod cluster control assemblies may be controlled manually or automatically. The rod cluster control assemblies are all dropped into the core following a reactor trip signal.

Calculations of control rod reactivity worth versus time following reactor trip involves both control rod velocity and differential reactivity worth. For nuclear design purposes, to be conservative, the reactivity worth versus rod position is calculated with the rod of highest worth assumed stuck out of the core, and the flux skewed to the bottom of the core. The result of these calculations is shown in Sub-chapter 4.3 - Figure 30 [Ref-1].

The shutdown margin is defined as the amount by which the core would be sub-critical at hot shutdown or at the end of the partial cooldown transient if all rod cluster control assemblies are tripped, but assuming that the highest worth assembly is stuck and no changes in xenon or boron take place. The description of the method for shutdown margin calculation is presented below.

In order to calculate the shutdown margin, a conservative balance is assessed between the reactivity addition resulting from power variation and the reactivity decrease resulting from rod cluster control assemblies drop.

1) Doppler

The Doppler Effect arises from the broadening of U-238 and Pu-240 resonance peaks with an increase in effective pellet temperature. This effect is most noticeable over the range of zero power to full power, due to the pellet temperature increase with power.

2) Variable average moderator temperature

When the core is shutdown to zero power conditions, the average moderator temperature changes from the equilibrium full load value determined by the steam generator and turbine characteristics (steam pressure, heat transfer, tube fouling, etc) to the zero load value. The design temperature is conservatively increased by 2.2°C to account for the control dead band and measurement errors.

The moderator coefficient becomes more negative with fuel depletion because the boron concentration is reduced. This effect is the major contributor to the increased shutdown requirement at end of life.

3) Redistribution

During full power operation, the coolant density decreases with core height, and this results in less fuel depletion near the top of the core. Under steady-state conditions, the relative power distribution will be slightly asymmetric towards the bottom of the core. On the other hand, at zero power conditions, the coolant density is uniform up the core, and there is no flattening due to the Doppler Effect. The result will be a flux distribution which at zero power can be skewed toward the top of the core. The reactivity insertion due to this skewed distribution is calculated with an additional allowance for the effects of xenon distribution.

4) Void content

There is a small void content in the core due to nucleate boiling at full power. The void collapse coincident with power reduction makes a small reactivity contribution.

5) Negative reactivity resulting from the rod cluster control assemblies drop

This takes into account:

- The worth of all rod cluster control assemblies, but assumes that the highest worth assembly is stuck,

- The rod insertion allowance: At full power, the control bank is operated within a prescribed band of travel to compensate for small periodic changes in boron concentration, changes in temperature and Axial Offset, and very small changes in boron concentration. When the control bank reaches either limit of this band, a change in boron concentration is required to compensate for additional reactivity changes. A conservative calculation of the inserted rod worth is made which exceeds the normally inserted reactivity.

- The rod depletion effect due to the insertion of some rods for core control.

The shutdown margins at end of life for cycle 1 and for the equilibrium cycle of the different fuel management regimes are presented in Sub-chapter 4.3 - Table 5 and Table 6 [Ref-2].

The initial sub-criticality must ensure that the DNBR limit is not exceeded following Steam Line Break (see Sub-chapter 14.5); the final value is determined by transient analyses in hot zero power conditions. The shutdown criterion is 0 pcm at the end of partial cooldown.

7. CRITICALITY OF FUEL ASSEMBLIES DURING STORAGE IN FUEL POOL OR IN DRY CONDITIONS

Criticality of fuel assemblies outside the reactor is precluded by adequate design of fuel transfer, shipping, and storage facilities, and by administrative control procedures. The two principal methods of preventing criticality are limiting the fuel assembly array size and limiting the assembly interaction. The latter is accomplished by fixing the minimum separation between assemblies and/or by inserting neutron poisons between assemblies.

The following conditions are assumed in meeting the design basis:

a) Fuel assemblies with the maximum authorised enrichment will be considered, with an irradiation that gives the fuel its maximum reactivity. The possible presence of burnable poisons in the fuel may be considered, provided that the depletion of these poisons is taken into account according to the fuel burnup.

b) For flooded conditions, the moderator is pure water at the temperature within the design limits that yields the largest reactivity

c) The array is either infinite in lateral extent or is surrounded by a conservatively chosen reflector, whichever is appropriate for the design

d) Mechanical uncertainties are treated by either using "worst case" conditions or by performing sensitivity studies and obtaining appropriate uncertainties

e) Credit is taken for the neutron absorption in structural materials and in solid materials added specifically for neutron absorption

f) Where borated water is present, credit for the dissolved boron is not taken except under postulated accident conditions where the double contingency principle is applied. This principle states that it shall require at least two unlikely, independent, and concurrent events to produce a criticality accident.

The preliminary criticality design criteria are:

- For dry storage, the multiplication factor k_{eff} must not exceed 0.98 with fuel of the highest anticipated enrichment (5% U^{235}, see Sub-chapter 4.3 - Table 2) together with an assumed optimum moderation.

- For storage in the fuel pool, the multiplication factor $k_{eff,}$ including all uncertainties, must be less than 0.95 during normal operation and less than 0.98 in accident situations.

8. RESIDUAL HEAT CURVES

The residual heat in a sub-critical core arises from:

- Residual fissions due to delayed neutrons (called Term A),
- Decay of U-238 neutron capture products (called Term B),
- Fission product decay energy (called Term C).

8.1. TERM A

Residual fission energy versus time after shutdown depends on the characteristics of the delayed neutrons and the effective multiplication factor.

The variation of the multiplication factor as a function of time after a reactor trip signal depends on the characteristics of the reactor trip and the thermal-hydraulic parameters of the core.

Term A is derived for most transients using a neutron kinetics model.

In the particular cases of Loss of Coolant Accident (LOCA), Feed Water Line Break (FWLB), and Steam Generator Tube Rupture (SGTR) the Term A is provided as an input generated by using a decoupled, conservative reactor trip simulation.

8.2. TERM B + C

The ORIGEN-S code is used for the calculation of the Term B + C. ORIGEN-S calculates the nuclide inventory as a function of fuel irradiation, as well as the decay heat power of each chemical element in the core after reactor shutdown. Further description of ORIGEN-S is given in Appendix 4.

The decay heat power as a function of time is shown for the most adverse UO_2 and MOX fuel management regime equilibrium core in Sub-chapter 4.3 - Figure 31, from 1 second to 1 month after reactor shutdown [Ref-1]. These results are used in the safety analysis.

9. VESSEL IRRADIATION

A brief review of the methods and analyses used to determine neutron and gamma ray flux attenuation between the core and the pressure vessel is given below. The materials that serve to attenuate neutrons originating in the core, and gamma rays from both the core and structural components, consist of the heavy reflector, the core barrel, and the downcomer water gap, all of which are within the region between the core and the pressure vessel.

In general, few group neutron diffusion theory and nodal analysis codes are used to determine fission power density distributions within the active core, and the accuracy of these analyses is verified by in-core measurements on operating reactors. Region and rod-wise power sharing information from the core calculations is then used as source data in two-dimensional transport calculations which compute the flux distributions throughout the reactor.

The neutron flux distribution and spectrum in the various structural components varies significantly from the core to the pressure vessel. Representative values of the neutron flux distribution and spectrum are presented in Sub-chapter 4.3 - Table 7. The values listed are based on equilibrium cycle reactor core parameters and power distributions for a UO_2 - INOUT – 18 months fuel cycle at end of life, and are suitable for long-term irradiation projections and for correlation with radiation damage estimates.

The irradiation surveillance program utilises actual test samples to verify the accuracy of the calculated fluxes at the vessel.

10. METHODS AND TOOLS

10.1. GENERATION OF THE LIBRARIES OF MACROSCOPIC DATA

The APOLLO 2 code generates the two-group macroscopic cross-sections and Assembly Discontinuity Factors to be used in the SMART diffusion code. The codes are described in Appendix 4.

For the fuel, APOLLO 2 solves the Boltzman transport equation for 2-D geometries. The cross-sections and discontinuity factors are tabulated as a function of burnup, boron concentration, xenon level, water density, fuel temperature, rod cluster insertion, and one specific variable for the fuel spectrum history.

For the heavy reflector, the cross-sections and discontinuity factors are based on 1D APOLLO 2 calculations, followed by a nodal equivalence between heterogeneous and homogeneous calculations; these cross-sections and discontinuity factors are tabulated as a function of boron concentration and water density.

10.2. CORE CALCULATIONS

The core calculations are performed with the SMART code.

There are four radial nodes per assembly, and 20 axial nodes (including 18 in the heated length). The heavy reflector material is represented by an additional row of fuel assemblies that is assumed to surround the active core.

10.3. UNCERTAINTIES

The following uncertainties are applied for the safety analysis:

- ± 20% for the fuel temperature coefficient [Ref-1]
- ± 3.6 pcm/°C for the moderator temperature coefficient [Ref-1]
- ± 100 ppm for the critical boron concentration [Ref-2]
- ± 10% for the rod cluster control assembly worth. [Ref-1]

SUB-CHAPTER 4.3 - TABLE 1

Nuclear Design Parameters

Region average discharge burnup for UO_2 (GWd/tU)	From > 55 (for 18 months) to < 65 (for 22 months) [Ref-1]
Fuel average linear power density at cold conditions (W/cm) (without uncertainties)	163.4 [Ref-2]
Total heat flux hot channel factor F_Q (limit for LOCA analyses)	2.82 (= 470 W/cm) [Ref-3]
Total enthalpy rise hot channel factor $F_{\Delta H}^N$ (limit for LOCA analyses)	1.80 [Ref-3]
Total enthalpy rise hot channel factor $F_{\Delta H}^N$ (target value for defining the fuel management regime)	1.61 [Ref-2]
Reactivity change rate:	
Maximum control rod speed (cm/min)	75 [1]
Maximum controlled reactivity change rate (pcm/s)	40 [2]
Maximum rate for accidental withdrawal of control bank at power (pcm/s)	90 [2]
Maximum xenon burnout (pcm/min)	10 [2]

$\Delta\rho = \frac{1}{K_1} - \frac{1}{K_2}$ where $K_1 = K_{eff}$ of initial state

K2 = K_{eff} of final state

[1] 1 step = 1 cm
[2] Reactivity in pcm = $10^5\ \Delta\rho$, where $\Delta\rho$ is calculated from the two state point values of K_{eff}

SUB-CHAPTER 4.3 - TABLE 2 (1/4)

Reactor Core Description

- Active core (Dimensions are at cold conditions (20°C)):	
▪ Equivalent diameter (mm)	3767
▪ Average active height of the core fuel (mm)	4200
▪ Height/diameter ratio	1,115
▪ Total surface area (cm^2)	111440
- Radial heavy reflector (preliminary):	
▪ Thickness (mm)	Between 77 and 297 (average 194)
▪ Composition (% volume)	About 95.6% steel – 4.4% water
- Fuel assemblies (preliminary) (Dimensions are at cold conditions (20°C)):	
▪ Number	241
▪ Rod array	17x17
▪ Number of rods per assembly	265
▪ Lattice pitch (mm)	12.6
▪ Assembly overall dimensions (mm)	214x214
▪ Weight of fuel for each assembly (kg)	598 UO_2, 527.5 U
▪ Number of grids per assembly	10
▪ Composition of grids	Zircaloy & Inconel
▪ Number of guide thimbles per assembly	24
▪ Composition of the guide thimbles	Zircaloy
▪ Diameter of guide thimbles, upper part (mm)	11.45 inside 12.45 outside
- Fuel rods (preliminary) (Dimensions are at cold conditions (20°C)):	
▪ Number	63865
▪ Outside diameter (mm)	9.50
▪ Diametrical gap (mm)	0.17
▪ Thickness of the cladding (mm)	0.57
▪ Cladding material	M5 type

SUB-CHAPTER 4.3 - TABLE 2 (2/4)

Reactor Core Description

- Fuel pellet (preliminary) (Dimensions are at cold conditions (20°C)):	
▪ Material	UO_2 or MOX
▪ Density of the UO_2 (% of theoretical density)	95
▪ Density of the UO_2 + PuO_2 (% of theoretical density)	94.5
▪ Diameter (mm)	8.19
▪ Theoretical density of the UO_2 (g/cm³)	10.96
▪ Theoretical density of the PuO_2 (g/cm³)	11.46
▪ Enrichment of fuel for the UO2 assemblies (% by weight)	
Zone 1 of cycle 1	2.1%
Zone 2 of cycle 1	3.2%
Zone 3 of cycle 1	4.2%
New assemblies for the UO_2 – IN/OUT – 18 months	5.0%
New assemblies for the UO_2 – IN/OUT – 22 months	5.0%
New assemblies for the MOX – IN/OUT – 18 months	5.0%
▪ Enrichment of fuel for the MOX assemblies (% by weight)	
Maximum enrichment of fissile Pu for zone 1	7.44% [3]
Average enrichment of fissile Pu for zone 2	6.44% [3]
Minimum enrichment of fissile Pu for zone 3	3.44% [3]
Mean enrichment for fissile Pu	7.0 % [3]
Enrichment assumption for the UO_2 in the MOX fuel (% U-235 in volume)	0.2 % [3]
▪ Pu vector for the MOX fuel assemblies from a UO_2 fuel burned to 60 GWd/t (% by weight):	
Pu-238	4.0
Pu-239	50.0
Pu-240	23.0
Pu-241	12.0
Pu-242	9.5
A-241	1.5

[3] enrichment of the fissile Pu is defined as being $e = \dfrac{Pu^{239} + Pu^{241}}{U + Pu + Am}$

SUB-CHAPTER 4.3 - TABLE 2 (3/4)

Reactor Core Description

▪ Burnable poison within the fuel:	
Material	Gd_2O_3
Enrichment of gadolinium (% by weight)	8
Enrichment in UO_2 vector (% U-235 by weight)	
- in assemblies enriched to 2.1%	1.2
- in assemblies enriched to 3.2%	1.9
- in assemblies enriched to 4.2%	2.2
- in assemblies enriched to 5.0 %	3.0

SUB-CHAPTER 4.3 - TABLE 2 (4/4)

Reactor Core Description
(Dimensions are at cold conditions (20°C))

▪ Absorber - See also Sub-chapter 4.1 - Figure 1:	
(1) AIC part : AIC composition (%wt) Ag/In/Cd AIC density (g/cm^3) AIC upper part absorber outer diameter (mm) AIC upper part length (mm) AIC lower part absorber outer diameter (mm) AIC lower part length (mm)	 80/15/5 10.17 8.66 2400 8.53 500
(2) B4C part : B4C composition B4C density (g/cm^3) B4C part absorber outer diameter (mm) B4C part length (mm)	 19.9% atomic wt. of B-10 1.79 8.47 1340
(3) Cladding : Cladding outer diameter (mm) Cladding inner diameter (mm) Cladding thickness (mm) Cladding material	 9.68 8.74 0.47 Stainless steel
(4) Lower end plug material	Stainless steel
(5) Distance between the bottom of the active height and the bottom of the absorber column Cluster fully inserted (mm) Cluster fully removed (mm)	 90 4200
(6) Number of Rod Cluster Control Assemblies	89
(7) Number of absorber rods per cluster	24

SUB-CHAPTER 4.3 - TABLE 3 (1/3)

Nuclear Design Parameters (Reactivity Coefficient Results)

	Beginning of life	End of life
-Reactivity coefficients, best estimate values:		
Moderator temperature coefficient, hot zero power, all rods out, critical boron concentration at nominal power (pcm/°C):		
Cycle 1	-13.0	-48.1
UO_2 - INOUT - 18 months	-9.6	-58.8
UO_2 - INOUT - 22 months	-8.8	-58.1
MOX - INOUT - 18 months	-14.9	-57.0
Moderator temperature coefficient, nominal power, all rods out, critical boron concentration at nominal power (pcm/°C):		
Cycle 1	-20.1	-60.6
UO_2 - INOUT - 18 months	-24.1	-82.6
UO_2 - INOUT - 22 months	-20.6	-79.4
MOX - INOUT - 18 months	-28.3	-79.6

SUB-CHAPTER 4.3 - TABLE 3 (2/3)

Nuclear Design Parameters (Reactivity Coefficient Results)

	Beginning of life	End of life
Moderator temperature coefficient hot zero power, all rods out, critical boron concentration at hot zero power (pcm/°C):		
Cycle 1	-10.35	
UO_2 - INOUT - 18 months	-5.28	
UO_2 - INOUT - 22 months	-4.8	
MOX - INOUT - 18 months	-10.4	
Power defect, all rods out, critical boron concentration at nominal power (pcm):		
Cycle 1	1103	2245
UO_2 - INOUT - 18 months	1396	2858
UO_2 - INOUT - 22 months	1270	2840
MOX - INOUT - 18 months	1471	2706

SUB-CHAPTER 4.3 - TABLE 3 (3/3)

Nuclear Design Parameters (Reactivity Coefficient Results)

Reactivity coefficients, design limits:		
▪ Doppler-only power coefficients (pcm/% power):		
. upper limit (0% to 100% relative power)	-28.3 to -10.5	[4]
. lower limit (0% to 100% relative power)	-6.4 to – 5.1	[4]
▪ Doppler temperature coefficient (pcm/°C)	-4.03 to – 1.98	[4]
▪ Moderator temperature coefficient (pcm/°C)	≤ 0	[4]
▪ Rodded moderator density coefficient (pcm/g.cm-³)	$\leq 0.515\ 10^5$	[4]
▪ Boron coefficient, nominal power, beginning of life (pcm/ppm):		
Cycle 1	-9.7	
UO_2 - INOUT - 18 months	-6.1	
UO_2 - INOUT - 22 months	-6.0	
MOX - INOUT - 18 months	-5.1	
▪ Boron coefficient, nominal power, end of life (pcm/ppm):		
Cycle 1	-9.3	
UO_2 - INOUT - 18 months	-7.1	
UO_2 - INOUT - 22 months	-7.1	
MOX - INOUT - 18 months	-5.8	
▪ Delayed neutron fraction and lifetime:		
β_{eff} upper limit	0.0073	[4]
β_{eff} lower limit	0.0045	[4]
ℓ^* upper limit (µs)	23.3	
ℓ^* lower limit (µs)	10.8	

[4] Including uncertainties

SUB-CHAPTER 4.3 - TABLE 4 (1/2)

Nuclear Design Parameters (Boron Concentration Results) [Ref-1]

- Natural boron concentrations (ppm), without uncertainties:	
▪ Zero power, k_{eff} = 1, cold, rod cluster control assemblies out, beginning of life:	
UO_2 fuel management regimes	2195
MOX fuel management regimes	2439
▪ Design basis refuelling boron concentration, minimal required values:	
UO_2 fuel management regimes	2195
MOX fuel management regimes	2439
▪ Nominal power, no xenon, k_{eff} = 1, rod cluster control assemblies out, beginning of life:	
Cycle 1	1026
UO_2 - INOUT - 18 months	2042
UO_2 - INOUT - 22 months	1971
MOX - INOUT - 18 months	2095
▪ Nominal power, equilibrium xenon, k_{eff} = 1, rod cluster control assemblies out, beginning of life:	
Cycle 1	697
UO_2 - INOUT - 18 months	1610
UO_2 - INOUT - 22 months	1535
MOX - INOUT - 18 months	1649

SUB-CHAPTER 4.3 - TABLE 4 (2/2)

Nuclear Design Parameters (Boron Concentration Results) [Ref-1]

- Minimum required boron concentrations (ppm), without uncertainties:	
▪ Zero power, no xenon, k_{eff} < provision, hot, rod cluster control assemblies in, beginning of life:	
Cycle 1	392
UO_2 - INOUT - 18 months	1224
UO_2 - INOUT - 22 months	1142
MOX - INOUT - 18 months	1339
▪ Zero power, no xenon, keff < provision, cold, rod cluster control assemblies in, beginning of life:	
Cycle 1	925
UO_2 - INOUT - 18 months	1528
UO_2 - INOUT - 22 months	1425
MOX - INOUT - 18 months	1599
▪ Reduction with fuel burnup	See Sub-chapter 4.3 - Figure 25 to Sub-chapter 4.3 - Figure 28

SUB-CHAPTER 4.3 - TABLE 5

Shutdown Margin at Hot Zero Power Conditions [5] at End of Life

	Cycle 1	UO_2 INOUT 18 months	UO_2 INOUT 22 months	MOX INOUT 18 months
1. Estimated rod cluster control assembly worth (pcm)				
a - all full length assemblies inserted	12258	11374	10930	10077
b - all but one (highest worth) assemblies inserted	9224	9784	9558	8812
c - estimated rod cluster control assembly credit with 10% adjustment to accommodate uncertainties	8302	8805	8602	7931
2. Reactivity addition resulting from power variation (pcm)				
Doppler, moderator temperature and redistribution	2243	2856	2838	2706
3. Uncertainty and allowances (pcm)				
- rod insertion	400	400	400	400
- rod depletion	100	100	100	100
- void effect	50	50	50	50
- accuracy of control system	200	200	200	200
- xenon perturbation	400	400	400	400
- Doppler uncertainty	100	100	100	100
- moderator uncertainty	41	41	41	41
4. Shutdown margin available (pcm) (= 1c - 2 - 3)	4768	4658	4473	3934

[5] For hot zero power temperature, 303.3°C is used

SUB-CHAPTER 4.3 - TABLE 6

Shutdown Margin at Partial Cooldown Conditions [6] at End of Life

	Cycle 1	UO_2 INOUT 18 months	UO_2 INOUT 22 months	MOX INOUT 18 months
1. Estimated rod cluster control assembly worth (pcm)				
a - all full length assemblies inserted	11317	10556	10121	9344
b - all but one (highest worth) assemblies inserted	8025	8813	8679	8041
c - estimated rod cluster control assembly credit with 10% adjustment to accommodate uncertainties	7223	7932	7811	7237
2. Reactivity addition resulting from power variation (pcm)				
Doppler, moderator temperature and redistribution	3884	4881	4834	4691
3. Uncertainty and allowances (pcm)				
- rod insertion	400	400	400	400
- rod depletion	100	100	100	100
- void effect	50	50	50	50
- accuracy of control system	200	200	200	200
- xenon perturbation	400	400	400	400
- Doppler uncertainty	100	100	100	100
- moderator uncertainty	197	197	197	197
4. Shutdown margin available (pcm) (= 1c - 2 - 3)	1892	1604	1530	1098

SUB-CHAPTER 4.3 - TABLE 7

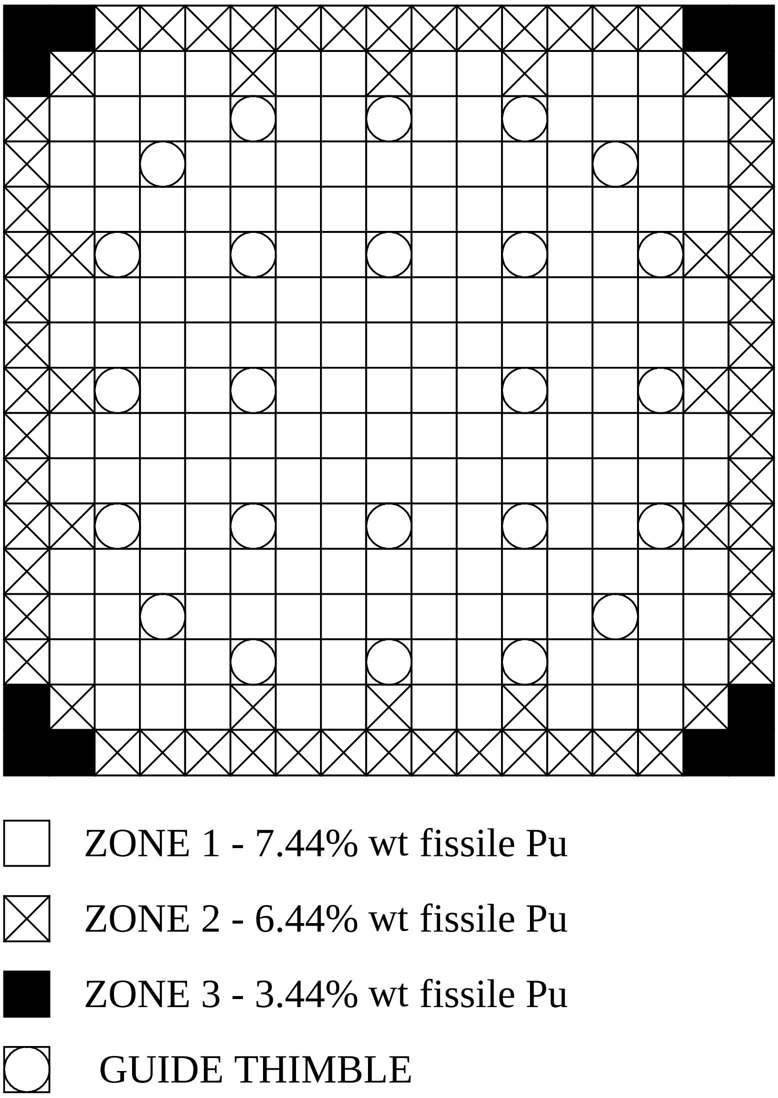
Typical Neutron Flux Levels (N/cm^2.s) at Full Power

ITEM	E > 1.0 MeV	E > 0.1 MeV	E > 0.625 eV	E < 0.625 eV
Central fuel assembly at mid-height			2.1E14	2.9E13
Central fuel assembly at the top			1.9E14	2.5E13
Central fuel assembly at the bottom			1.7E14	2.6E13
Outer fuel assembly closest to the pressure vessel at mid-height			1.4E14	1.8E13
Pressure vessel inner wall, azimuthal peak, core height average value	5.3E9	1.2E10		

[6] For end of partial cooldown temperature, 260°C is used as a decoupling value.

SUB-CHAPTER 4.3 - FIGURE 1

Radial Description of a MOX Assembly



SUB-CHAPTER 4.3 - FIGURE 2

First Core Loading Pattern

	J	K	L	M	N	P	R	S	T
09	2.10%	2.1% 8Gd	3.2% 16Gd	2.1% 8Gd	3.2% 20Gd	2.1% 8Gd	3.2% 20Gd	2.1% 8Gd	4.2% 16Gd
08	2.1% 8Gd	3.2% 16Gd	2.1% 8Gd	3.2% 20Gd	2.10%	3.2% 20Gd	2.1% 8Gd	4.2% 16Gd	4.2% 16Gd
07	3.2% 16Gd	2.1% 8Gd	3.2% 16Gd	2.1% 8Gd	3.2% 16Gd	2.1% 8Gd	3.2% 20Gd	2.1% 8Gd	4.2% 16Gd
06	2.1% 8Gd	3.2% 20Gd	2.1% 8Gd	3.2% 20Gd	2.10%	3.2% 20Gd	2.1% 8Gd	4.2% 16Gd	4.2% 16Gd
05	3.2% 20Gd	2.10%	3.2% 16Gd	2.10%	3.2% 20Gd	2.1% 8Gd	2.1% 8Gd	4.2% 16Gd	
04	2.1% 8Gd	3.2% 20Gd	2.1% 8Gd	3.2% 20Gd	2.1% 8Gd	3.2% 16Gd	4.2% 16Gd	4.2% 16Gd	
03	3.2% 20Gd	2.1% 8Gd	3.2% 20Gd	2.1% 8Gd	2.1% 8Gd	4.2% 16Gd	4.2% 16Gd		
02	2.1% 8Gd	4.2% 16Gd	2.1% 8Gd	4.2% 16Gd	4.2% 16Gd	4.2% 16Gd			
01	4.2% 16Gd	4.2% 16Gd	4.2% 16Gd	4.2% 16Gd					

The first cycle loading pattern is composed of:
- 17 FA : UO2 2.10% 0Gd
- 80 FA : UO2 2.10% 8Gd
- 24 FA : UO2 3.20% 16Gd
- 48 FA : UO2 3.20% 20Gd
- 72 FA : UO2 4.20% 16Gd

SUB-CHAPTER 4.3 - FIGURE 3

Reloading Pattern for UO_2 – INOUT – 18 Month Equilibrium Cycle

	J	K	L	M	N	P	R	S	T
09	N5	P14	M9	**5.0% 16gd**	P9	**5.0% 16gd**	K11	**5.0% 12gd**	H11
08	P4	J5	S6	R6	P7	N7	R4	**5.0% 16gd**	J1
07	J6	M2	J7	R5	**5.0% 16gd**	L16	N6	**5.0% 12gd**	M8
06	**5.0% 16gd**	M3	N3	J2	S8	P8	**5.0% 12gd**	**5.0% 8gd**	T7
05	J4	L4	**5.0% 16gd**	K2	M6	R8	**5.0% 8gd**	M7	
04	**5.0% 16gd**	L5	B7	K4	K3	**5.0% 12gd**	**5.0% 8gd**	S5	
03	L8	P3	M5	**5.0% 12gd**	**5.0% 8gd**	**5.0% 8gd**	J8		
02	**5.0% 12gd**	**5.0% 16gd**	**5.0% 12gd**	**5.0% 8gd**	L6	N2			
01	L10	R3	K6	L1					

The CY7 (INOUT UO2 18 months) reload in composed of:
- 24 FA : UO2 5.0% 8Gd
- 24 FA : UO2 5.0% 12Gd
- 24 FA : UO2 5.0% 16Gd

SUB-CHAPTER 4.3 - FIGURE 4

Reloading Pattern for UO_2 – INOUT – 22 Month Equilibrium Cycle

	J	K	L	M	N	P	R	S	T
09	K4	S9	M9	**5.0% 20Gd**	K9	**5.0% 20Gd**	P9	**5.0% 20Gd**	L9
08	J2	L7	**5.0% 20Gd**	L8	**5.0% 24Gd**	S8	**5.0% 20Gd**	**5.0% 16Gd**	M8
07	J6	**5.0% 20Gd**	P4	**5.0% 24Gd**	R8	S6	S7	**5.0% 20Gd**	N8
06	**5.0% 20Gd**	K7	**5.0% 24Gd**	N5	R6	R5	**5.0% 20Gd**	**5.0% 20Gd**	R7
05	J8	**5.0% 24Gd**	K3	M3	**5.0% 24Gd**	R4	**5.0% 16Gd**	M7	
04	**5.0% 20Gd**	K2	M2	N3	P3	**5.0% 20Gd**	**5.0% 20Gd**	P5	
03	J4	**5.0% 20Gd**	L2	**5.0% 20Gd**	**5.0% 16Gd**	**5.0% 20Gd**	M6		
02	**5.0% 20Gd**	**5.0% 16Gd**	**5.0% 20Gd**	**5.0% 20Gd**	L6	N4			
01	J7	K6	K5	L3					

The fuel management is composed of:
- 16 UO2 fuel assemplies 5.0% 16Gd
- 64 UO2 fuel assemblies 5.0% 20Gd
- 20 UO2 fuel assemblies 5.0% 24Gd

SUB-CHAPTER 4.3 - FIGURE 5

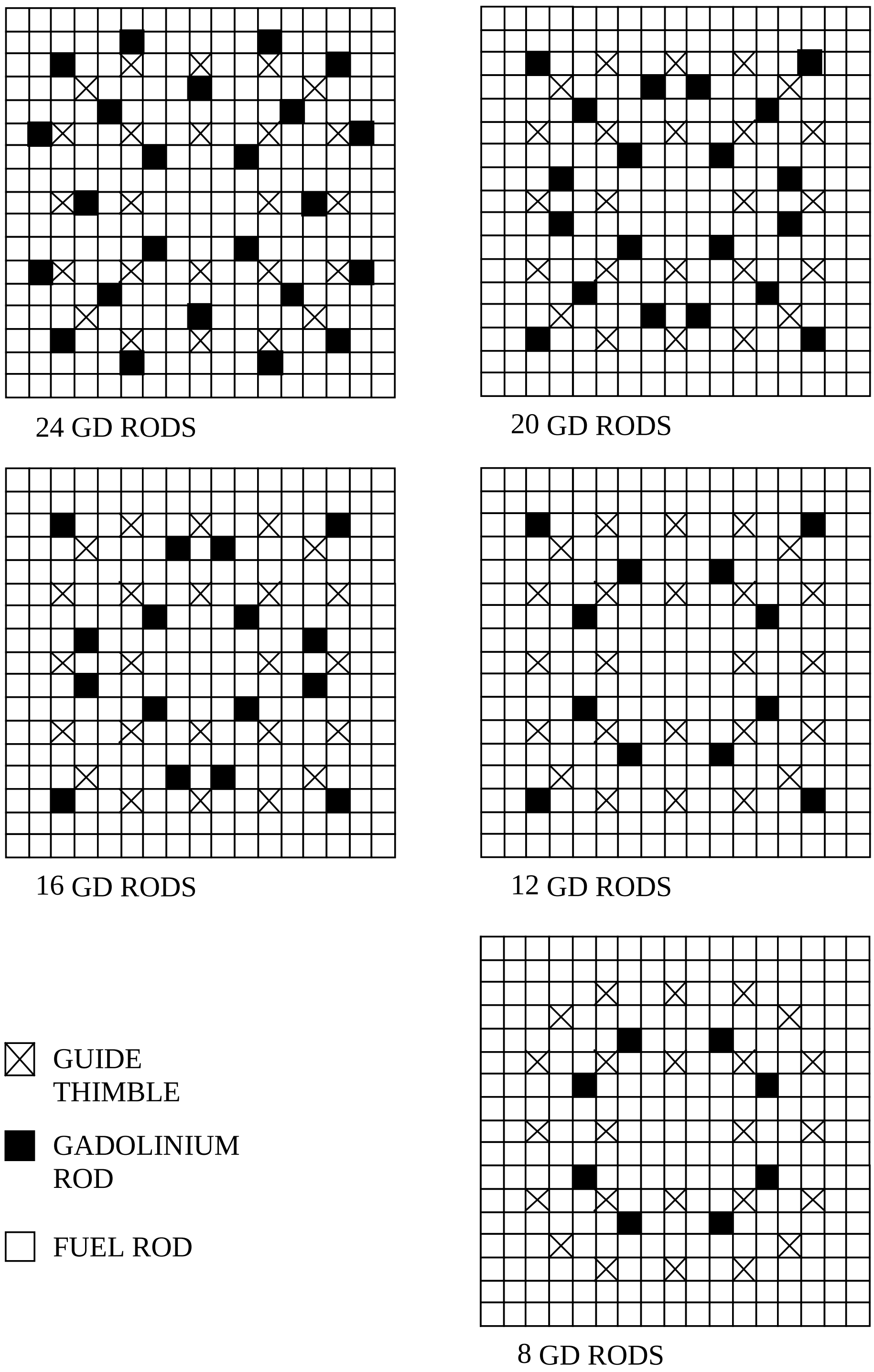
Reloading Pattern for 30% MOX – INOUT – 18 Month Equilibrium Cycle

	J	K	L	M	N	P	R	S	T
09	N2	P14	M9	**5.0% 20Gd**	P9	**5.0% 20Gd**	K11	**MOX 7.0%**	L10
08	P4	J5	S6	R5	P7	R6	N7	**5.0% 20Gd**	T7
07	J6	M2	J7	R4	**5.0% 20Gd**	S7	N6	**MOX 7.0%**	M8
06	**5.0% 20Gd**	N3	P3	N5	S8	P8	**5.0% 16Gd**	**5.0% 8Gd**	R7
05	J4	L4	**5.0% 20Gd**	K2	J2	R8	**MOX 7.0%**	M7	
04	**5.0% 20Gd**	M3	L2	K4	K3	**5.0% 8Gd**	**5.0% 16Gd**	P6	
03	L8	L5	M5	**5.0% 16Gd**	**MOX 7.0%**	**5.0% 16Gd**	J8		
02	**MOX 7.0%**	**5.0% 20Gd**	**MOX 7.0%**	**5.0% 8Gd**	L6	M4			
01	K7	L1	K6	L3					

The fuel management is composed of:
- 68 MOX fuel assemplies (20 new ones)
- 173 UO2 fuel assemblies (52 new ones)

SUB-CHAPTER 4.3 - FIGURE 6

Typical Burnable Poison Rod Arrangement within an Assembly



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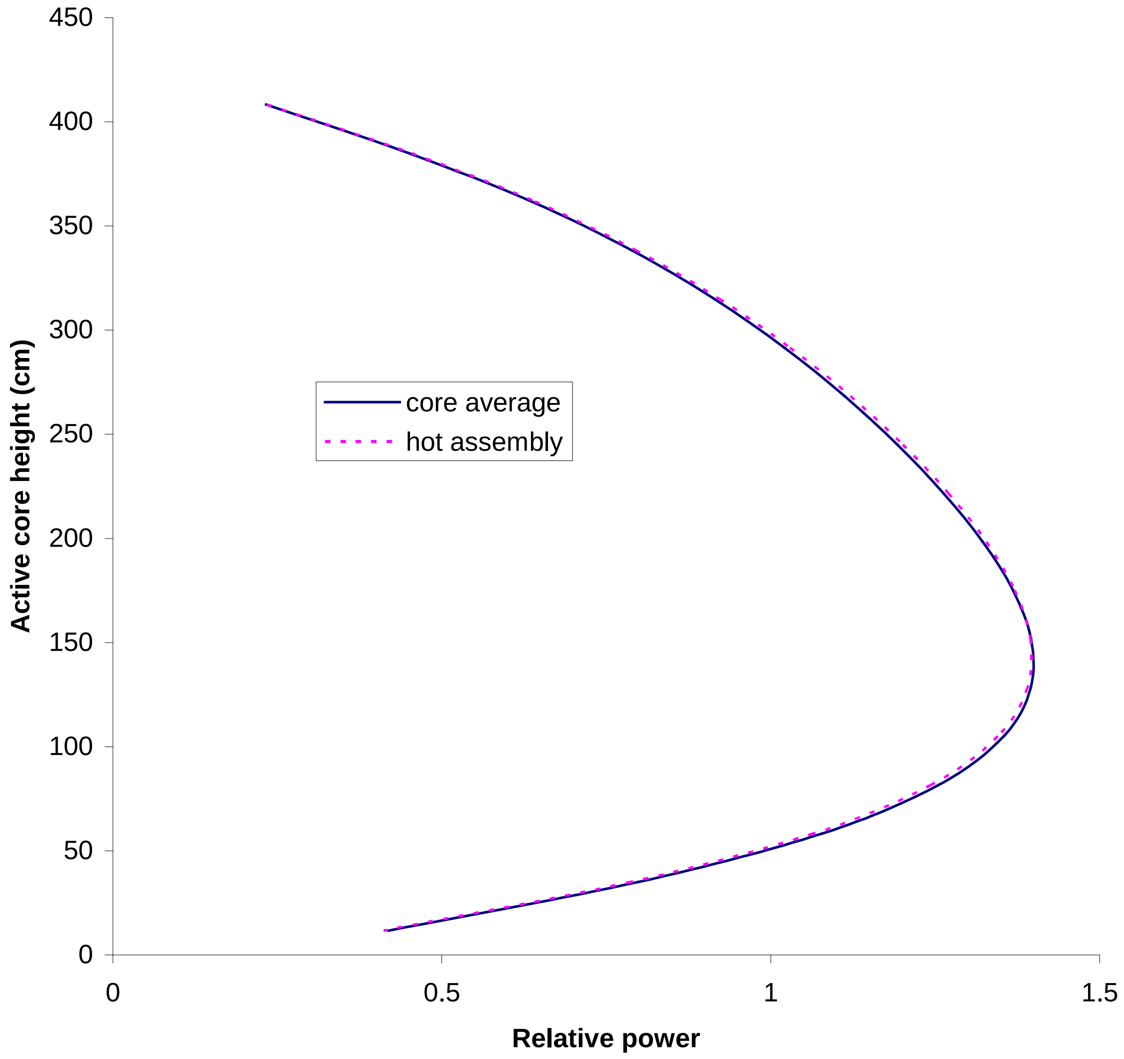
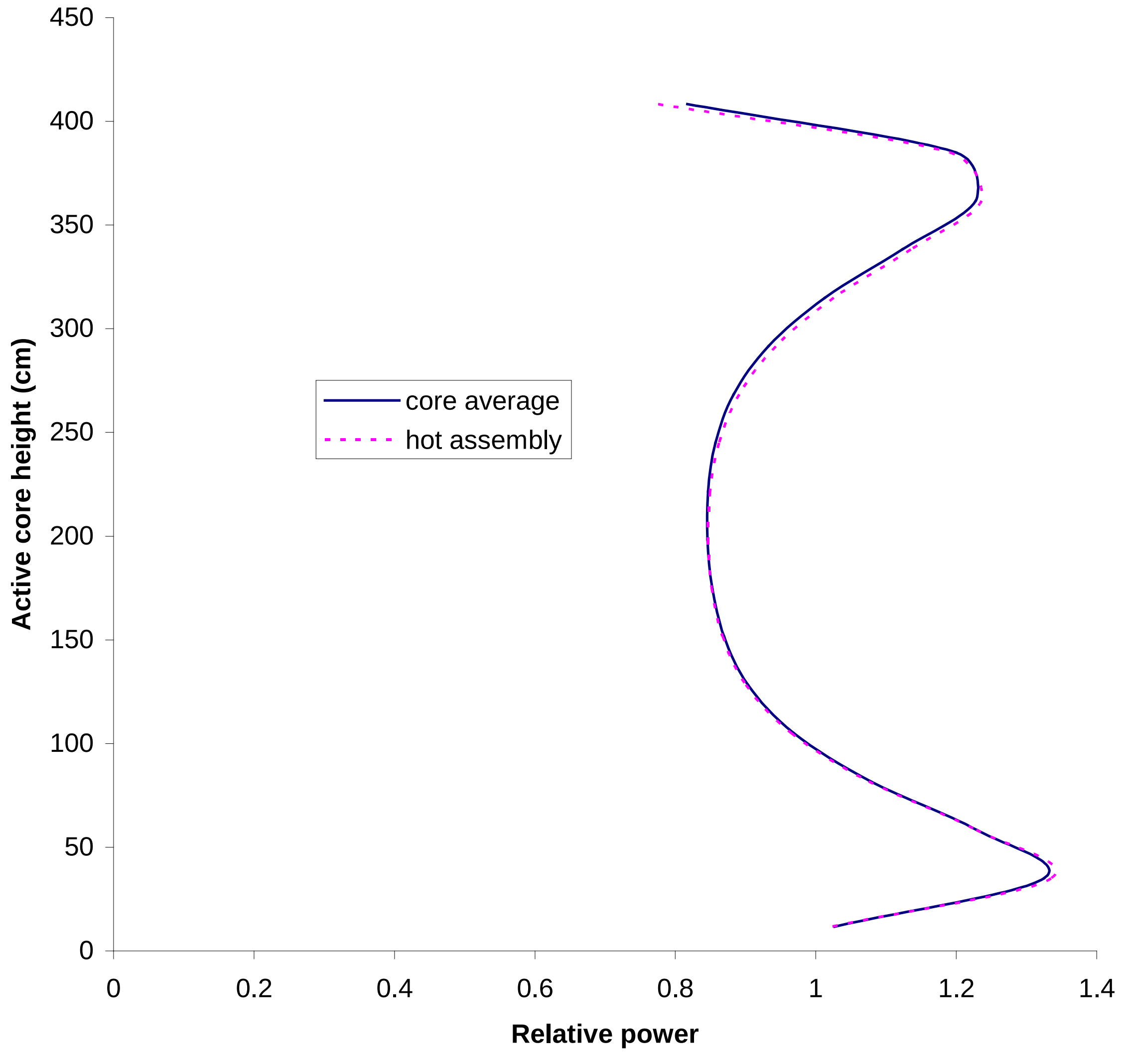
SUB-CHAPTER 4.3 - FIGURE 19

Typical Axial Power Shape, Equilibrium Xenon Occurring at Beginning of Life for UO_2 Cycle 1 Fuel Management



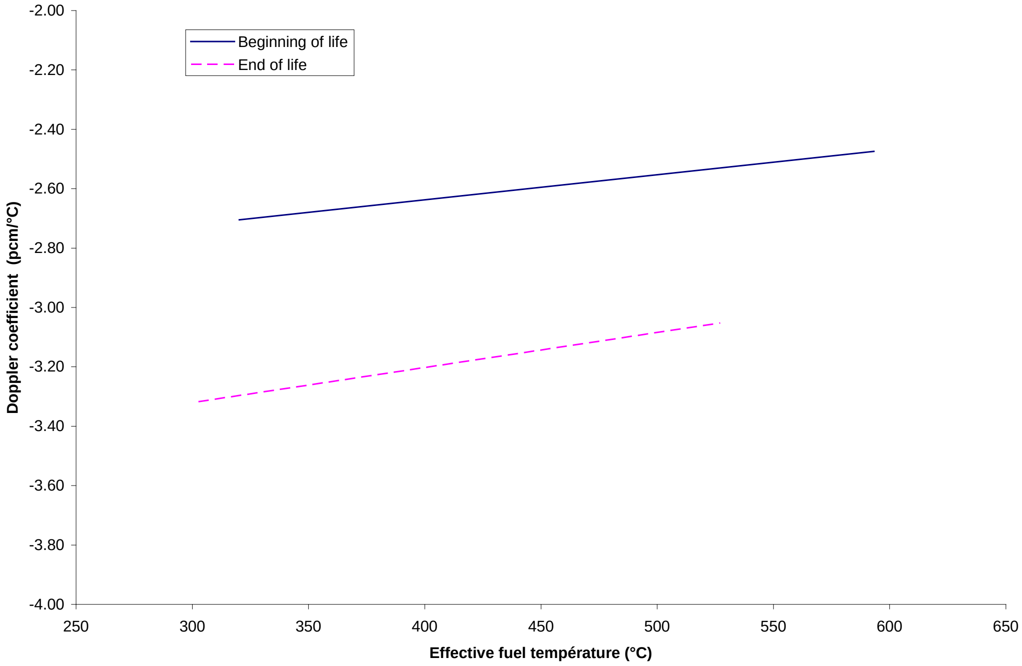
SUB-CHAPTER 4.3 - FIGURE 20

Typical Axial Power Shape, Equilibrium Xenon Occurring at End of Life for UO_2 Cycle 1 Fuel Management



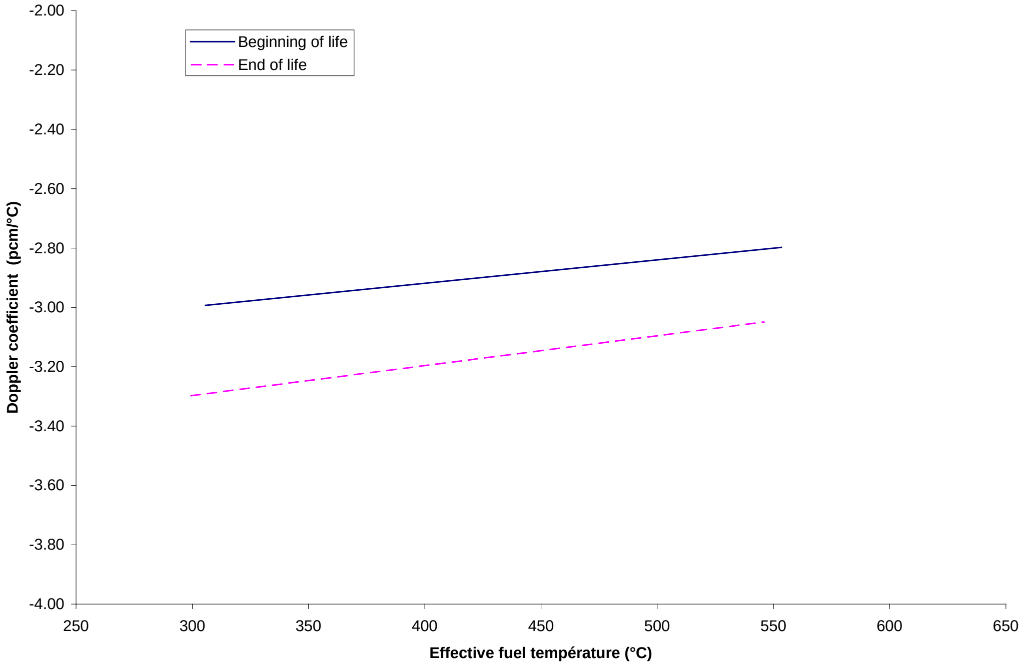
SUB-CHAPTER 4.3 - FIGURE 21

Doppler Coefficient as Function of Effective Temperature for Cycle 1



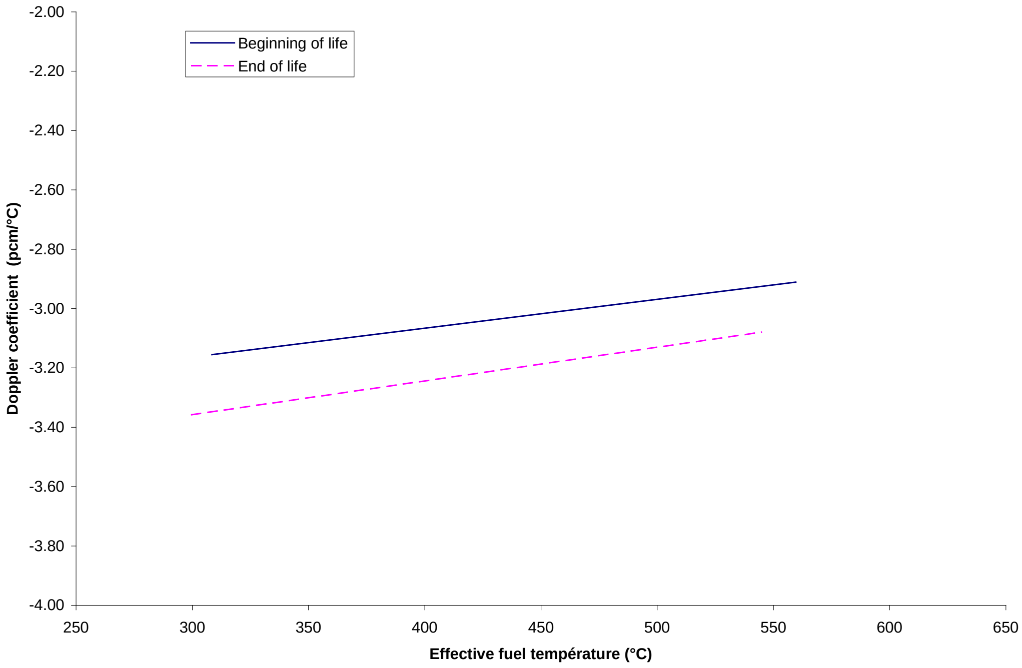
SUB-CHAPTER 4.3 - FIGURE 22

Doppler Coefficient as Function of Effective Temperature for Equilibrium Cycle UO_2 – INOUT – 18 Month



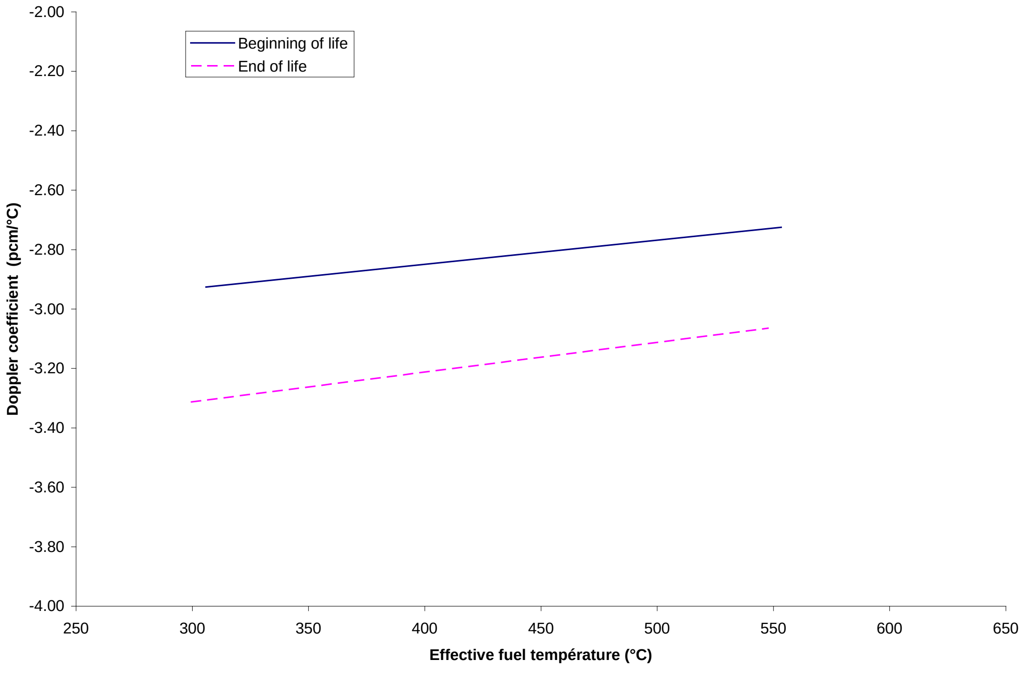
SUB-CHAPTER 4.3 - FIGURE 23

Doppler Coefficient as Function of Effective Temperature for Equilibrium Cycle UO_2 – INOUT – 22 Month



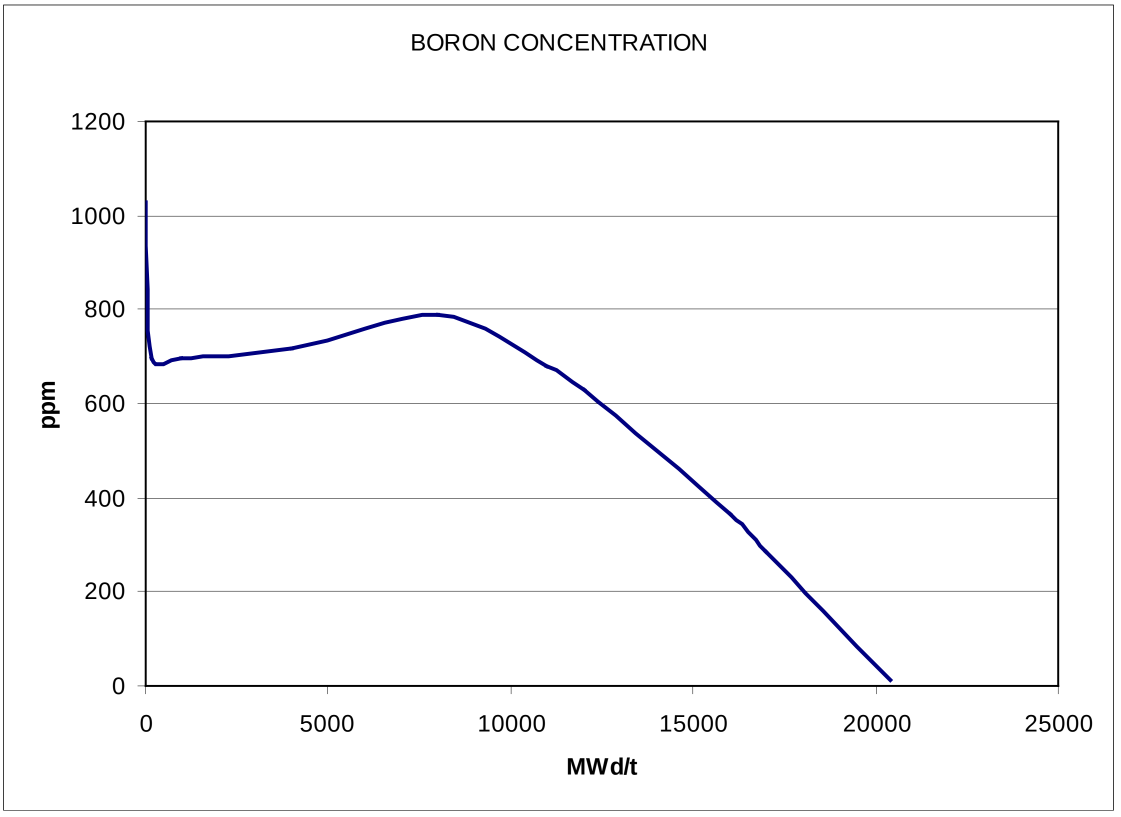
SUB-CHAPTER 4.3 - FIGURE 24

Doppler Coefficient as Function of Effective Temperature for Equilibrium Cycle 30% MOX – INOUT – 18 Month



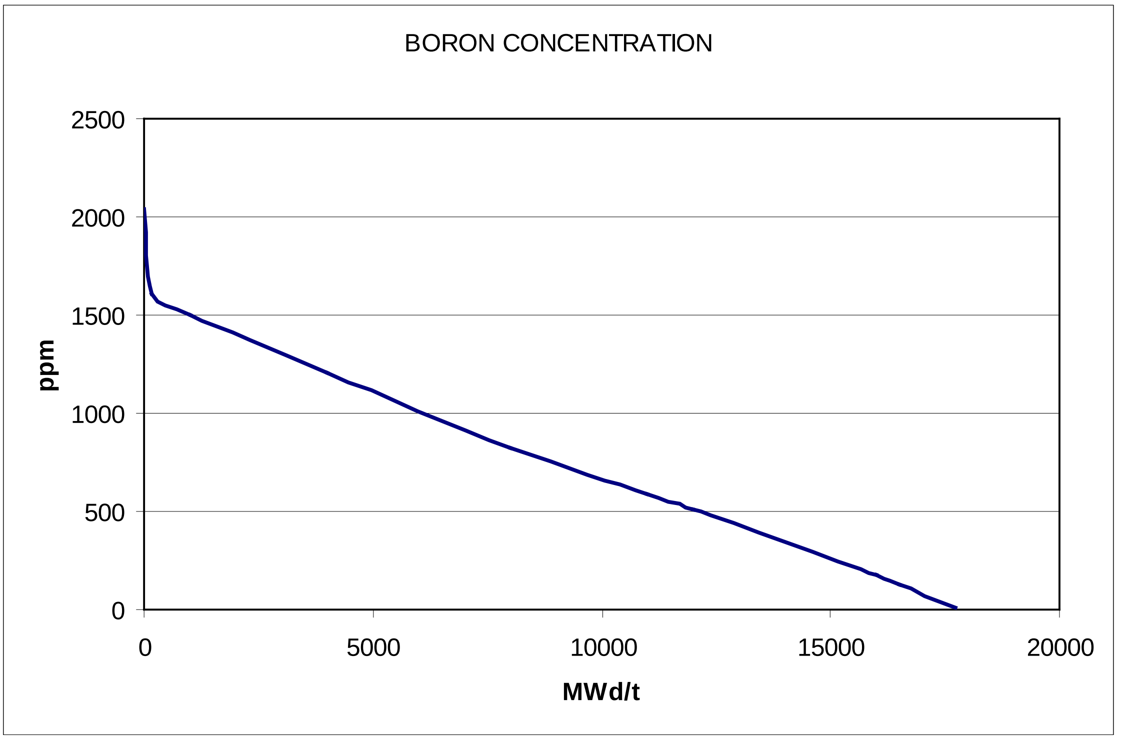
SUB-CHAPTER 4.3 - FIGURE 25

Critical Boron Concentration versus Burnup for Cycle 1



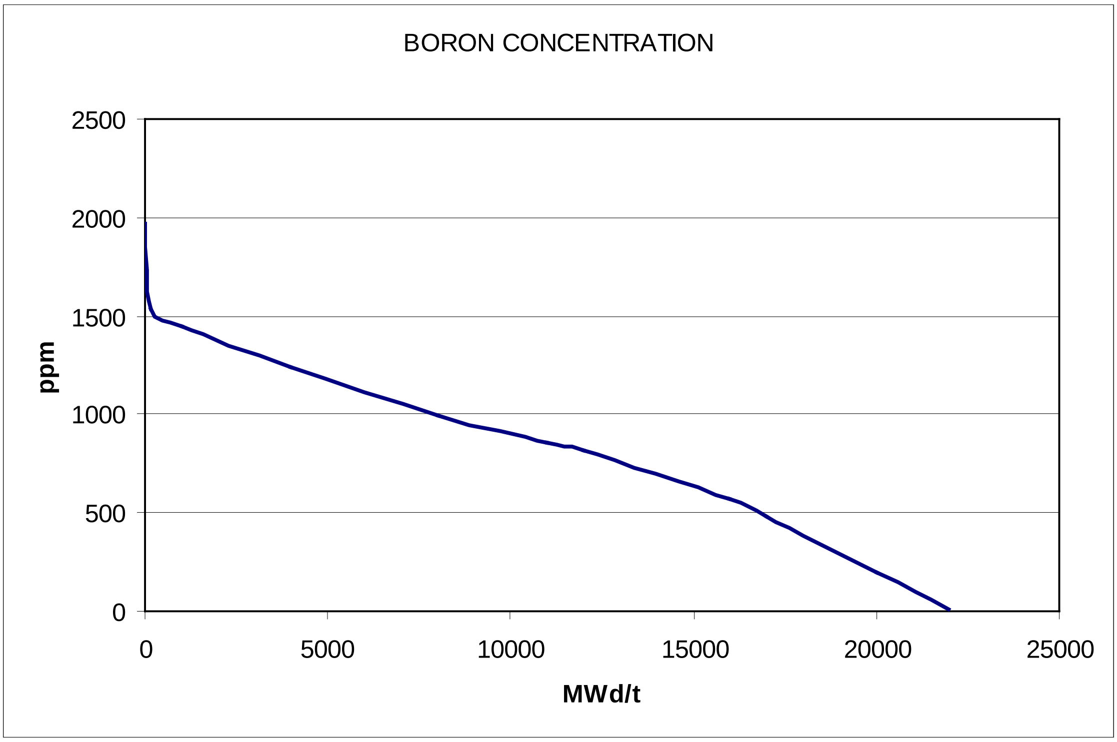
SUB-CHAPTER 4.3 - FIGURE 26

Critical Boron Concentration versus Burnup for Equilibrium Cycle UO_2 – INOUT – 18 Month



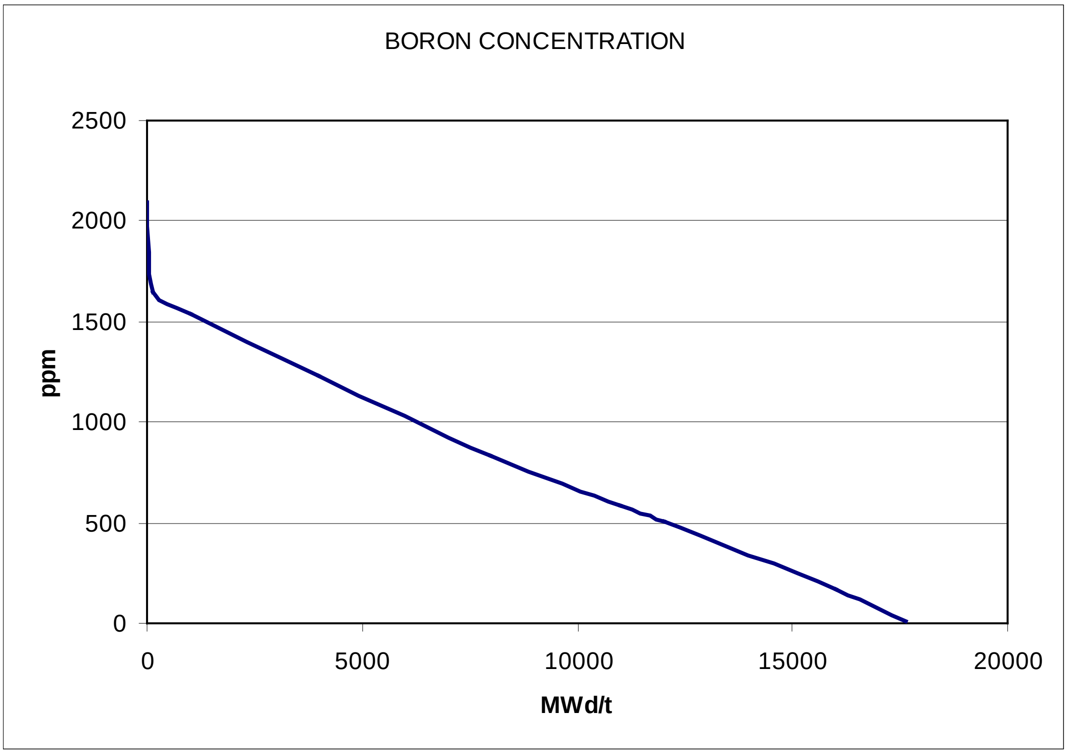
SUB-CHAPTER 4.3 - FIGURE 27

Critical Boron Concentration versus Burnup for Equilibrium Cycle UO_2 – INOUT – 22 Month



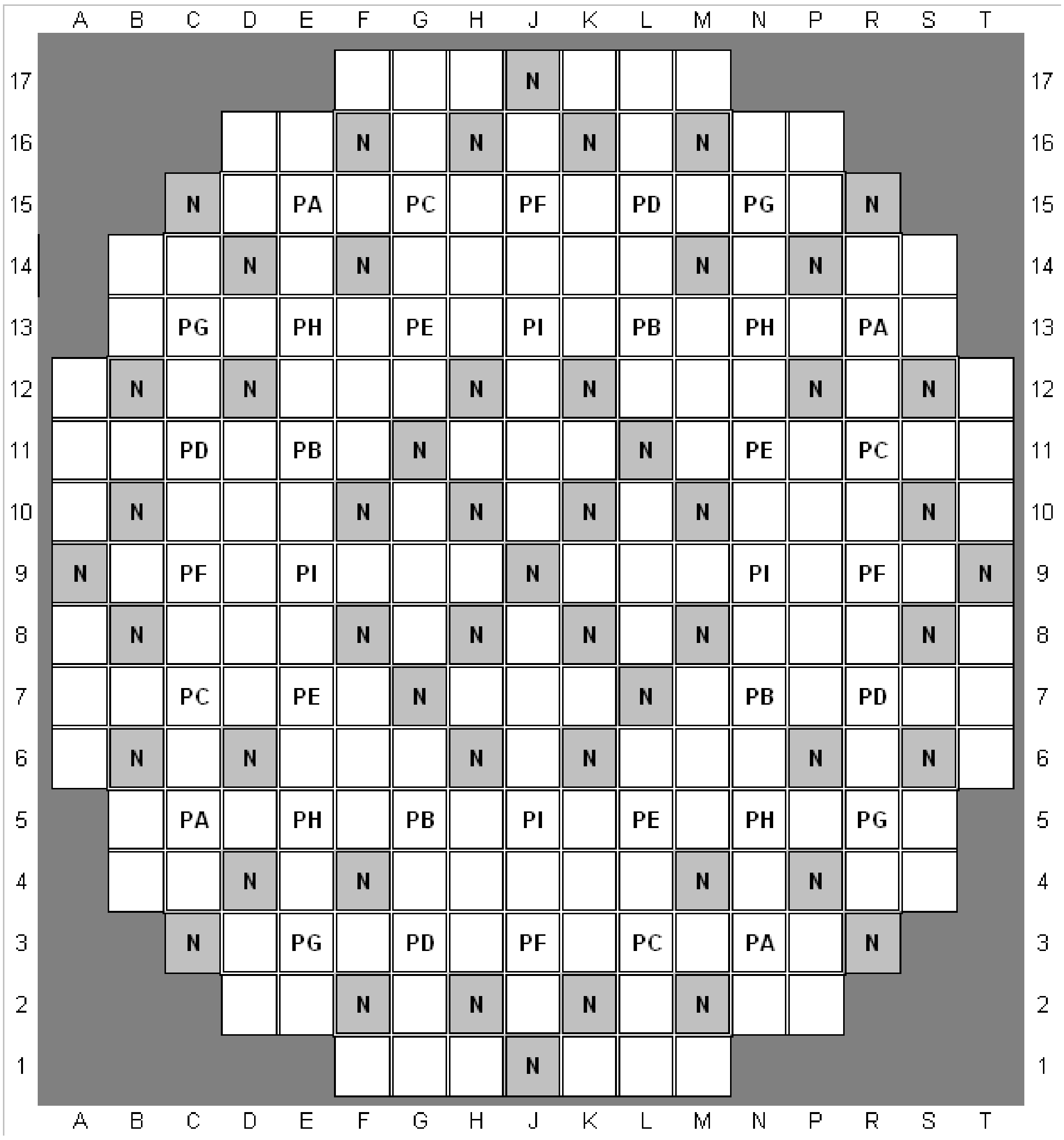
SUB-CHAPTER 4.3 - FIGURE 28

Critical Boron Concentration versus Burnup for Equilibrium Cycle 30% MOX – INOUT – 18 Month



SUB-CHAPTER 4.3 - FIGURE 29

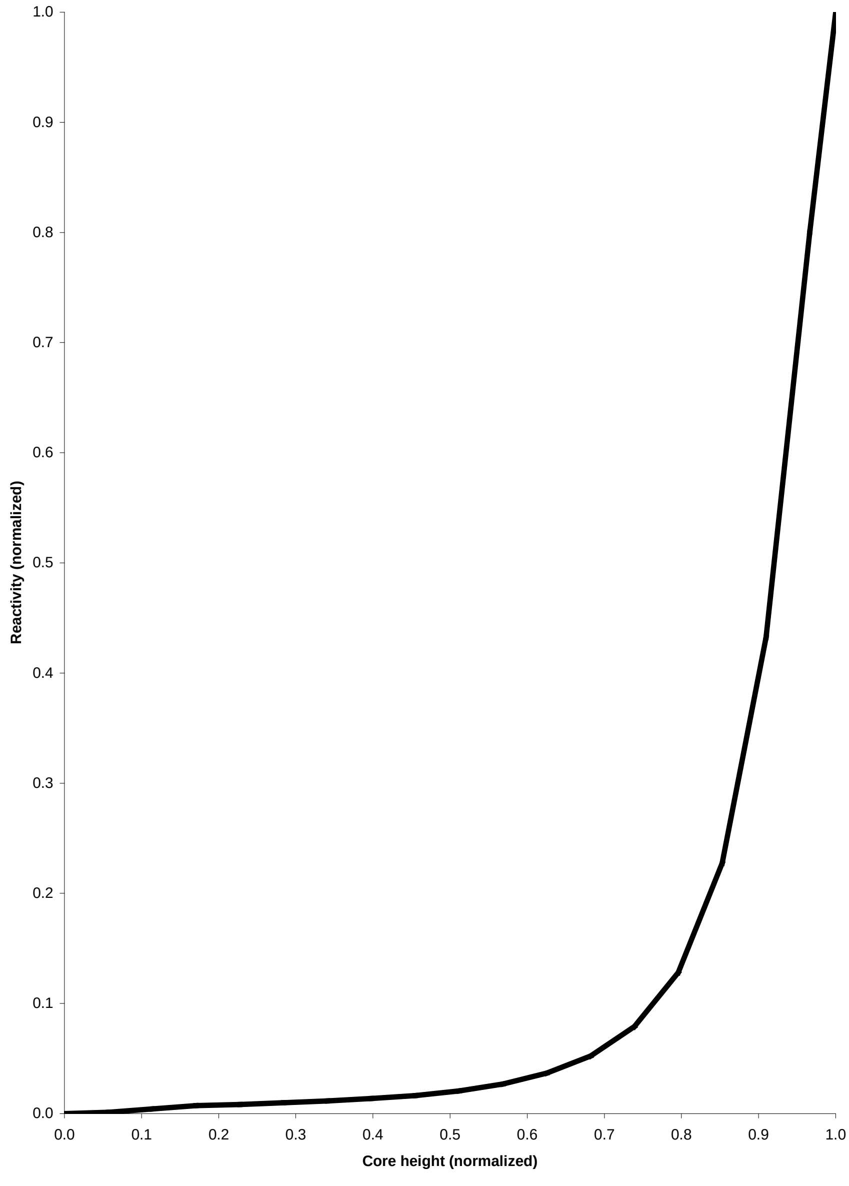
Rod Cluster Control Assembly Pattern



	P1	P2	P3	P4	P5
S1	PA	PB	PC PD	PF PE	PG PH PI
S2	PD	PH	PC PB	PA PF	PE PG PI
S3	PF	PB	PA PE	PC PD	PG PH PI
S4	PC	PH	PD PB	PA PE	PF PG PI

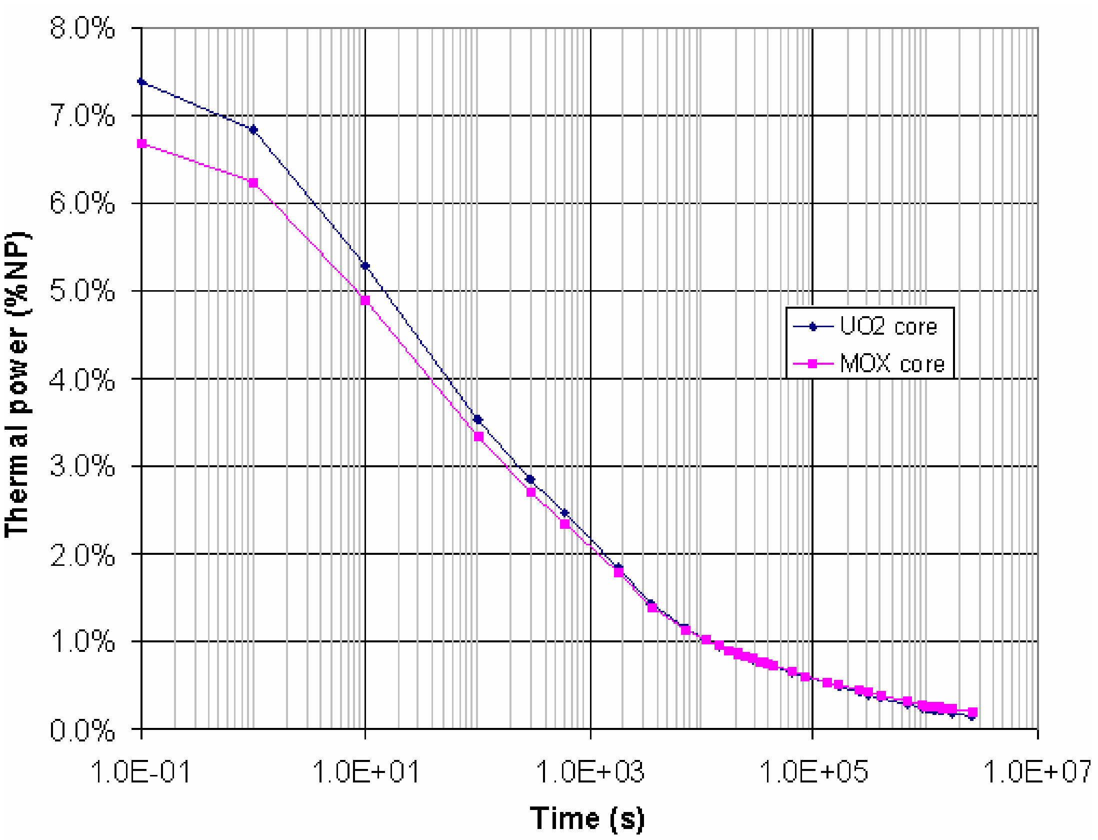
SUB-CHAPTER 4.3 - FIGURE 30

Normalised Rod Worth versus Relative Insertion, all Rods but One



SUB-CHAPTER 4.3 - FIGURE 31

Residual Power, Decay of Fission Products and Actinides



SUB-CHAPTER 4.3 – REFERENCES

External references are identified within this sub-chapter by the text [Ref-1], [Ref-2], etc at the appropriate point within the sub-chapter. These references are listed here under the heading of the section or sub-section in which they are quoted.

3. POWER DISTRIBUTIONS

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3.2. RADIAL POWER DISTRIBUTION

[Ref-1] S Laurent. Fuel Management - Neutronic Design Report (Science Calculations) (Update 4500 MWth). NFPSC DC 285 Revision A. AREVA. September 2004. (E)

3.5. EXPERIMENTAL VERIFICATION

3.5.1. Power distribution analysis

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5. CORE CONTROL

5.2. MEANS OF CONTROL

5.2.1. Chemical poison

[Ref-1] C Hove. Core Reactivity Control (Update 4500 MWth). NFPSC DC 284 Revision B. AREVA. January 2006. (E)

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[Ref-1] S Laurent. Fuel Management - Neutronic Design Report (Science Calculations) (Update 4500 MWth). NFPSC DC 285 Revision A. AREVA. September 2004. (E)

6. CONTROL ROD PATTERNS AND REACTIVITY WORTH

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[Ref-2] S Laurent. Fuel Management - Neutronic Design Report (Science Calculations) (Update 4500 MWth). NFPSC DC 285 Revision A. AREVA. September 2004. (E)

8. RESIDUAL HEAT CURVES

8.2. TERM B + C

[Ref-1] A.Delumley. Residual Decay Heat Curves For System Design And Accident Analyses (Update 4500 MWth). NFPSC DC 283 Revision C. AREVA. November 2005. (E)

10. METHODS AND TOOLS

10.3. UNCERTAINTIES

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[Ref-2] C Hove. Core Reactivity Control (Update 4500 MWth). NFPSC DC 284 Revision B. AREVA. January 2006. (E)

SUB-CHAPTER 4.3 - TABLE 1

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[Ref-2] S Laurent. Fuel Management - Neutronic Design Report (Science Calculations) (Update 4500 MWth). NFPSC DC 285 Revision A. AREVA. September 2004. (E)

[Ref-3] R Gagner. EPR Sizing 4500 MWth. EPRR DC 1685 Revision C. AREVA. February 2004. (E)

SUB-CHAPTER 4.3 - TABLE 4

[Ref-1] C Hove. Core Reactivity Control (Update 4500 MWth). NFPSC DC 284 Revision B. AREVA. January 2006. (E)