3.2 DESIGN BASIS

3.2.1 PERFORMANCE OBJECTIVES

The full-power thermal rating of the core is 2737 MWt. The physics, thermal and hydraulic information presented in this section is based on this power level.

3.2.2 DESIGN OBJECTIVES

The reactor core, together with its control systems and the RPS, is designed to function over its lifetime without exceeding fuel damage limits of excessive fuel temperature, cladding strain, and cladding stress (Section 3.2.3) during normal operating conditions and Design Basis Events (DBEs).

In the power operating range, the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in power. At the beginning of cycle (BOC) a slightly positive Moderator Temperature Coefficient (MTC) may occur. If power oscillations occur, their magnitude will be such that the fuel damage limits are not exceeded.

Reactivity control is provided by two independent systems: (1) the CEDS, and (2) the CVCS. The CEDS controls short-term reactivity changes and is used for rapid shutdown. The CVCS is used to compensate for long-term reactivity changes and can make the reactor subcritical without the benefit of the CEDS. The design of the core and the RPS prevents fuel damage limits from being exceeded for any single malfunction in either of the reactivity control systems.

The maximum reactivity addition rate from the withdrawal of the CEAs is limited by the core excess reactivity, CEA worth, and CEDS design. These limitations prevent sudden large reactivity increases. The design restraints are such that reactivity increases will not result in exceeding the fuel damage limits, rupture of the reactor coolant pressure boundary, or disruption of the core or other internals sufficient to impair the effectiveness of emergency cooling.

3.2.3 DESIGN LIMITS

3.2.3.1 Nuclear Design Limits

The design of the core is based upon the following nuclear limitations:

a. The limitation on fuel burnup is determined by material design, mechanical design and nuclear considerations. The mechanical integrity of the fuel remains satisfactory beyond the planned discharge burnup.

b. In the power operating range, the effect of the prompt inherent nuclear feedback characteristic [Fuel Temperature Coefficient (FTC)] compensates for rapid increases in power.

c. CEAs are moved in groups to satisfy the requirements of shutdown, power level changes and operational maneuvering. The control systems are designed to produce power distributions that are within the acceptable limits of overall Nuclear Heat Flux Factor (F_q^N) and DNBR. The RPS and administrative controls ensure that these limits are not exceeded.

d. Axial xenon oscillations, when they occur, will be manually controlled by regulating CEAs using information provided by the neutron flux detectors. The xenon oscillation period, about one day, allows ample time for operator action before the RPS trip setpoint is exceeded.

3.2.3.2 Reactivity Control Design Limits

The control system and operating procedures provide for adequate control of the core reactivity and power distributions such that the following limits are met:

a. Sufficient CEAs are withdrawn to provide an adequate shutdown reactivity margin;
b. The shutdown margin is maintained with the highest worth CEA assumed stuck in its fully withdrawn position;
c. The CVCS is capable of adding boric acid to the reactor coolant at a rate sufficient to maintain the shutdown margin during a RCS cooldown at the design rate following a reactor trip.

3.2.3.3 Thermal and Hydraulic Design Limits

The principal basis of the thermal and hydraulic design is to avoid thermally-induced fuel damage during normal operation, and Design Basis Event (DBE). It is recognized that there is a small probability of limited fuel damage in certain unlikely situations as discussed in Chapter 14.

The following corollary thermal and hydraulic design bases are established, but violation of either is not necessarily equivalent to fuel damage:

a. There is a high confidence level that Departure from Nucleate Boiling (DNB) is avoided during normal operation and DBEs. This is achieved by setting a design lower limit on the Minimum Departure from Nucleate Boiling Ratio (MDNBR) calculated according to the Asea Brown Boveri, Inc. (ABB)-NV correlation for each cycle. Starting with Unit 1 Cycle 17, the ABB-TV correlation was used. Starting with Unit 2 Cycle 19 and Unit 1 Cycle 21, the high thermal performance (HTP) correlation was used to determine DNBR for AREVA/Framatome fuel.
b. The melting point of the UO_2 fuel is not reached during normal operation nor during DBEs.

The RPS provides for automatic reactor trip before these design limits are exceeded.

Reactor internal flow passages and fuel coolant channels are designed to prevent hydraulic instabilities. Flow maldistributions are limited by design to be compatible with the specified thermal design criteria.

3.2.3.4 Mechanical Design Limits

The reactor internals are designed to perform their functions safely during steady state conditions and DBEs. The internals can safely withstand the forces due to deadweight, handling, system pressure, flow-induced pressure drop, flow impingement, temperature differential, shock, and vibration. The structural components satisfy stress values given in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III.

The following limitations on stresses or deformations are employed to ensure capability of a safe and orderly shutdown in the combined event of earthquake and major loss-of-coolant accident (LOCA). For reactor vessel internal structures, the stress criteria are given in Table 3.2-1. The intent of the limits in this table is as follows:

a. Under design loading plus design earthquake forces the critical reactor vessel internal structures are designed within the stress criteria established in the ASME B&PV Code, Section III, Article 4;
b. Under normal operating loadings plus maximum hypothetical earthquake forces, the design criteria permits a small amount of local yielding;
c. Under normal operating loadings plus reactor coolant pipe rupture loadings plus maximum hypothetical earthquake forces, permanent deformation is permitted by the design criteria.

The following typical values are selected to illustrate the conservatism of this approach for establishing stress limits. Units are 10^3 lbs/in^2.

Material	S_y[a]	S_u	S_D	S_L
SA 106B	25.4	60.0[b]	25.4	36.9
SA 533B	41.4	80.0[b]	41.4	54.3
304 SS	17.0	54.0[c]	18.35	29.3
316 SS	18.5	58.2[c]	22.2	31.7

[a] From ASME B&PV Code, Section III, at 650°F
[b] Minimum value at room temperature, which is approximately the same at 650°F for ferritic materials
[c] Estimated

S_u = Minimum tensile strength of material at temperature
S_L = $S_y + (1/3)(S_u - S_y)$
S_y = Tabulated yield at temperature from ASME B&PV Code, Section III
S_D = Design stress

To properly perform their functions, the critical reactor internal structures are designed to satisfy the additional deflection limits described below, in addition to the stress limits given in Table 3.2-1.

Under normal design loadings plus design earthquake forces or normal operating loadings plus maximum hypothetical earthquake forces, deflections are limited so that the CEAs can function and adequate core cooling is maintained. Under normal operating loadings plus maximum hypothetical earthquake forces plus pipe rupture loadings, the deflection design criteria depend on the size of the piping break. If the equivalent diameter of the pipe break is no larger than the largest line connected to the main reactor coolant lines, deflections are limited so that the core is held in place, the CEAs function normally, and adequate core cooling is maintained. Those deflections which would influence CEA movement are limited to less than two-thirds of the deflection required to prevent CEA function. For pipe breaks larger than the above, the criteria are that the fuel is held in place in a manner permitting core cooling and that adequate coolant flow passages are maintained. For these major pipe break sizes, CEA insertability is not required to achieve shutdown because the rapid voiding during the ensuing blowdown and the subsequent refill with the borated safety injection water ensures adequate shutdown margin for the reactor. For the larger break sizes, critical components are restrained from buckling by further limiting the stress levels to two-thirds of the stress level calculated to produce buckling
.

3.2.3.5 Fuel Assembly Design Limits

The fuel assemblies are designed to maintain their structural integrity under steady state conditions, DBEs, normal handling loads, shipping stresses, and refueling loads. The design takes into account differential thermal expansion of fuel rods, thermal bowing of fuel rods and CEA guide tubes, irradiation effects, and wear of all components. Mechanical tolerances and clearances have been established on the basis of the functional requirements of the components. All components including welds are highly resistant to the corrosive action of the reactor environment.

The fuel rod design accounts for external pressure, differential expansion of fuel and clad, fuel swelling, clad creep, fission and other gas releases, thermal stress, pressure and temperature cycling, and flow-induced vibrations. The structural criteria are based on the following:

a. The maximum primary stress during steady state operation, expected transients, and depressurization is limited to two-thirds of the minimum yield strength of the material at operating temperature.
b. The predicted total strain of the cladding at the End of Life (EOL) is less than 1.0%.

AREVA/Framatome has performed the mechanical design analyses starting with the Unit 2 Cycle 19 and Unit 1 Cycle 21 fuel assembly design. These evaluations used the Nuclear Regulatory Commission (NRC)-approved mechanical analysis codes and methodology to demonstrate compliance with the NRC-approved design criteria. (References 1, 2, 3, and 4)

3.2.3.6 Control Element Assembly Design Limits

The CEAs are designed to maintain their structural integrity under all steady state conditions, DBEs and handling, shipping and refueling loads. Thermal distortion, mechanical tolerances, vibration and wear of the CEA are all accounted for in the design. Clearances and corresponding fuel assembly alignment are established so that possible accumulation of mechanical tolerances and thermal distortion will not result in frictional forces that could prevent reliable operation of the system. The structural criteria are based on limiting the maximum stress intensity to those values specified in Section III of the ASME B&PV Code.

The clearance between the CEA fingers and the guide tubes is designed for actuating within the prescribed time under steady state conditions, during DBEs, under maximum hypothetical earthquake, and temperature conditions in combination with various factors which cause a reduction in diametral clearance. These factors include adverse dimensional tolerances, bowing and twisting of CEA and guide tubes and possible enlargement of the poison rod diameter due to swelling of B_4C pellets at maximum burnup conditions. The design diametral change due to swelling of B_4C is based on the pellets being rigid and the high strength clad offers no restraint to pellet diametral growth.

The core is designed to limit deflections so that the core is held in place. The CEAs function and adequate core cooling is maintained even under:

a. Normal design loadings plus design earthquake forces;
b. Normal operating loadings plus maximum hypothetical earthquake forces plus a pipe break no larger than the equivalent diameter largest line connected to the main reactor coolant lines

If the equivalent diameter of the pipe break is larger than the largest coolant line, the core is designed so that fuel is held in place to permit core cooling and adequate coolant flow is maintained.

Those deflections which would influence CEA movement are limited to less than two-thirds of the deflection required to prevent CEA function. If the equivalent diameter of the pipe break is larger than condition b above, the core is designed so that the fuel is held in place in a manner permitting core cooling and that adequate coolant flow passages are maintained. For these major pipe breaks, CEA insertion is not required to achieve shutdown because the rapid voiding during blowdown and the refilling of the vessel with borated safety injection water ensures adequate shutdown margin for the reactor.

The speed at which the CEAs are inserted or withdrawn from the core is consistent with the reactivity change requirements during reactor operation (Chapter 7). For conditions that require a rapid shutdown of the reactor, the CEDM holding coils deenergize to allow the CEAs to drop into the core. The reactivity is reduced during such a CEA drop at a rate sufficient to prevent exceeding fuel damage limits. A CEA automatic drive-down capability after a reactor trip is not required. During a trip, the RPS opens the trip circuit breakers, deenergizing the CEDM holding coils allowing the CEAs to drop by gravity into the core. To drive down a CEA stuck in the fully withdrawn position, the operator must first clear the trip condition and manually close the trip circuit breaker. Therefore, a drive-down feature would introduce the possibility of a failure which would prevent power from being removed from the CEDM holding coils. The safety analysis (Chapter 14) assumes the CEA of highest reactivity worth sticks in the fully withdrawn position.

The CEDM pressure housings are an extension of the reactor vessel, providing a part of the reactor coolant boundary, and are, therefore, designed to meet the requirements of the ASME B&PV Code, Section III, Nuclear Vessels. Pressure and thermal transients as well as steady state loadings were considered in the design analysis.

3.2.4 REFERENCES

1. ANF-88-133(P)(A), Revision 0 and Supplement 1, “Qualification of Advanced Nuclear Fuels PWR Design Methodology for Rod Burnups of 62 GWd/MTU,” December 1991

2. XN-NF-82-06(P)(A), Revision 1 and Supplements 2, 4, and 5, “Qualification of Exxon Nuclear Fuel for Extended Burnup,” October 1986

3. BAW-10133(P)(A), Revision 1, Addendum 1 and Addendum 2, “Mark-C Fuel Assembly LOCA-Seismic Analysis,” October 2000

4. EMF-92-116(P)(A), Revision 0, Supplement 1 (P)(A), Revision 0, “Generic Mechanical Design Criteria for PWR Fuel Designs,” Framatome, February 1999 and February 2015

TABLE 3.2-1

PRIMARY STRESS LIMITS FOR CRITICAL REACTOR VESSEL INTERNAL STRUCTURES

LOADING COMBINATIONS	ALLOWABLE STRESSES
Design Loading Plus Design Earthquake Forces	$P_m \leq S_m$ $P_b + P_L \leq 1.5S_m$
Normal Operating Loading Plus Maximum Hypothetical Earthquake Forces	$P_m \leq S_D$ $P_b \leq 1.5\left(1 - \left(\frac{P_m}{S_D}\right)^2\right) S_D$
Normal Operating Loadings Plus Maximum Hypothetical Earthquake Forces Plus Pipe Rupture Loadings	$P_m \leq S_L$ $P_b \leq 1.5\left(1 - \left(\frac{P_m}{S_L}\right)^2\right) S_L$

where:
LEGEND

P_m	=	Calculated Primary Membrane Stress, psi
P_b	=	Calculated Primary Bending Stress, psi
P_L	=	Calculated Primary Local Membrane Stress, psi
S_m	=	Tabulated Allowable Stress Limit at Temperature from ASME B&PV Code, Section III or ANSI B31.7, psi
S_y		Tabulated Yield Strength at Temperature, ASME B&PV Code, Section III, psi
S_D	=	Design Stress, psi
S_D	=	S_y (for ferritic steels), psi
S_D	=	$1.2S_m$ (for austenitic steels), psi
S_L	=	$S_y + 1/3\ (S_u – S_y)$, psi
S_u	=	Tensile Strength of Material at Temperature, psi