ABB

February 25, 1994
LD-94-016

Docket No. 52-002

Attn: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: System 80+™ Information for Issue Closure

Dear Sirs:

The attachments to this letter provide material to close follow-on questions to DSER responses. Attachment 1 provides several minor revisions to Chapter 4 that resulted from a recent consistency review for that chapter. Based on our judgment and a discussion with Mr. L. Kopp of the NRC, it is our belief that none of these revisions affects the conclusions in the Advanced FSER. For example, changes to the fuel temperature correlation on page 4.3-13 were made to be consistent with the correlation that was actually used. Changes to Table 4.4-1 were faxed to Mr. S. Sun and transmitted separately in letter LD-94-015.

Attachment 2 transmits changes to Table 1.9-1 initiated by a request from Mr. T. Boyce. The title for that table may be revised based on further discussions with NRC.

Attachment 3 presents revisions to Certified Design Material related to non-1E loads, Operations Support Center, Containment Spray and Service Water Systems, and piping design. These revisions should be given to Mr. T. Boyce.

Attachment 4 provides revisions to Section 14.2 of CESSAR-DC to be consistent with the ITAAC change for containment spray in Attachment 3.

Attachment 5 transmits revisions to the references listed in Section 1.6

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of CESSAR-DC. Additional changes may be made after a consistency check with the Advanced FSER.

If you have any questions, please call me or Mr. Stan Ritterbusch at (203) 285-5206.

Very truly yours,

COMBUSTION ENGINEERING, INC.

S. E. Ritterbusch for

C. B. Brinkman
Acting Director
Nuclear Systems Licensing

CBB/ser
cc: J. Trotter (EPRI)
T. Wambach (NRC)
P. Lang (DOE)

ATTACHMENT 1

LIST OF FIGURES (Cont'd)

CHAPTER 4

4.2 FUEL SYSTEM DESIGN

4.2.1 DESIGN BASES

4.2.1.1 Fuel Assembly

The fuel assemblies are required to meet design criteria for each design condition listed below to assure that the functional requirements are met. Except where specifically noted, the design bases presented in this section are consistent with those used for previous designs.

A. Nonoperation and Normal Operation (Condition I)

Condition I situations are those which are planned or expected to occur in the course of handling, initial shipping, storage, reactor servicing and power operation (including maneuvering of the plant). Condition I situations must be accommodated without fuel assembly failure and without any effect which would lead to a restriction on subsequent operation of the fuel assembly. The guidelines stated below are used to determine loads during Condition I situations:

1. Handling and Fresh Fuel Shipping

   Loads correspond to the maximum possible axial and lateral loads and accelerations imposed on the fuel assembly by shipping and handling equipment during these periods, assuming that there is no abnormal contact between the fuel assembly and any surface, nor any equipment malfunction.

2. Storage

   Loads on both new and irradiated fuel assemblies reflect storage conditions of temperature, chemistry, means of support and duration of storage.

3. Reactor Servicing

   Loads on the fuel assembly reflect those encountered during refueling, inspection, and reconstitution. Irradiation effects on material properties are considered when analyzing the effects of handling loads which occur during refueling. Additional information regarding shipping and handling loads is contained in Section 4.2.3.1.5.

D. Nomenclature

The symbols used in defining the allowable stress levels are as follows:

P_m = Calculated general primary membrane stress[a]

P_b = Calculated primary bending stress

S_m = Design stress intensity value as defined by Section III, ASME Boiler and Pressure Vessel Code[b]

S_u = Minimum unirradiated ultimate tensile strength

F_s = Shape factor corresponding to the particular cross section being analyzed[c]

S'_m = Design stress intensity value for faulted conditions

The definition of S'_m as the lesser value of 2.4 S_m and 0.7 S_u is contained in the ASME Boiler and Pressure Vessel Code, Section III.

(a) P_m and P_b are defined by Section III, ASME Boiler and Pressure Vessel Code.

(b) With the exception of zirconium base alloys, the design stress intensity values, S_m, of materials not tabulated by the Code are determined in the same manner as the Code. The design stress intensity of zirconium base alloys shall not exceed two-thirds of the unirradiated minimum yield strength at temperature. Basing the design stress intensity on the unirradiated yield strength is conservative because the yield strength of zircaloy increases with irradiation. The use of the two-thirds factor ensures 50% margin to component yielding in response to primary stresses. This 50% margintogether with its application to the minimum unirradiated properties and the general conservatism applied in the establishment of design conditions is sufficient to ensure an adequate design.

add space

(c) The shape factor, F_s, is defined as the ratio of the "plastic" moment (all fibers just at the yield stress) to the initial yield amount (extreme fiber at the yield stress and all other fibers stressed in proportion to their distance from the neutral axis). The capability of cross sections loaded in bending to sustain moments considerably in excess of that required to yield the outermost fibers is discussed in Timoshenko (see Reference 1).

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These new data are shown in Figure 4.2-1, along with O'Donnell's curve and Weber's data. This curve was then adjusted because of differences in anisotropy, stress states and strain rates, and the design limit was set at 1%.

The conservatism of the clad strain calculations is provided by the selection of adverse initial conditions and material behavior assumptions, and by the assumed operating history. The acceptability of the 1% unrecoverable circumferential strain limit is demonstrated by data from irradiated Zircaloy-clad fuel rods which show no cladding failures (due to strain) at or below this level, as illustrated in Figure 4.2-1.

C. The clad will be initially pressurized with helium to an amount sufficient to prevent gross clad deformation under the combined effects of external pressure and long-term creep. For conservatism, the clad design will not rely on the support of the holddown spring in the plenum region.

D. Cumulative strain cycling usage, defined as the sum of the ratios of the number of cycles in a given effective strain range ($\Delta\epsilon$) to the permitted number (N) at that range, as taken from Figure 4.2-2, will not exceed 0.8.

The cyclic strain limit design curve shown on Figure 4.2-2 is based upon the Method of Universal Slopes developed by S. S. Manson (Reference 12) and has been adjusted to provide a strain cycle margin for the effects of uncertainty and irradiation. The resulting curve has been compared with known data on the cyclic loading of Zircaloy and has been shown to be conservative. Specifically, it encompasses all the data of O'Donnell and Langer (Reference 13).

As discussed in Section 4.2.1.2.5, the fatigue calculation method includes the effect of clad creep to reduce the pellet-to-clad diametral gap during that portion of operation when the pellet and clad are not in contact. The same model is used for predicting clad fatigue as is used for predicting clad strain. Therefore, the effects of creep and fatigue loadings are considered together in determining end-of-life clad strain. Moreover, the current fatigue damage calculation method includes a factor of 2 which is applied to the calculated strain before determining the allowable number of cycles associated with that strain. This, in combination with the allowable fatigue usage factor 0.8, ensures a considerable degree of conservatism (see Figure 4.2-2).

where:

C_p = specific heat, BTU/ft^3-°F; and,

T = temperature, °F.

4.2.1.2.4.5 Mechanical Properties

A. Young's Modulus of Elasticity

The static modulus of elasticity of unirradiated fuel of 97% TD and deformed under a strain rate of 0.097 hr^{-1} is given by Reference 24:

$$E = 14.22\ (1.6715 \times 10^6 - 924.4T),$$

where:

E = modulus of elasticity in psi; and,
T = temperature in °C in the range of 1000 to 1700°C.

B. Poisson's Ratio

The Poisson's Ratio of polycrystalline UO_2 has a value of 0.32 at 25°C based on Reference ~~26~~ 79. The same reference notes a 10% decrease in value over the range of 25 to 1800°C. Assuming the decrease is linear, the temperature dependence of the Poisson's Ratio is given by:

$$\nu = 0.32 - 1.8 \times 10^{-5}\ (T-25),$$

where:

ν = Poisson's Ratio
T = temperature in °C in the range of 25 to 1800°C.

At temperatures above 1800°C, a constant value of 0.29 is used for Poisson's Ratio.

4.2.1.2.5 Fuel Rod Pressurization

Fuel rods are initially pressurized with helium for two reasons:

A. To preclude clad collapse during the design life of the fuel. The internal pressurization reduces stresses from differential pressure, thus extending the time required to produce clad collapse beyond the required service life of the fuel; and, normal, upset and emergency loading combinations identified in Sections 4.2.1.1 and 4.2.1.2 are highlighted as follows:

A. During normal operating and upset conditions, the maximum primary tensile stress in the Zircaloy clad shall not exceed two-thirds of the minimum unirradiated yield strength of the material at the applicable temperature. The corresponding limit under emergency conditions is the material yield strength and the limit for faulted conditions is the smaller of 1.6 times the yield strength and 0.7 times the ultimate strength.

B. Net unrecoverable circumferential clad strain shall not exceed 1% as predicted by computations considering clad creep and poison pellet swelling effects.

C. The clad will be initially pressurized with helium to an amount sufficient to prevent gross clad deformation under the combined effects of external pressure and long-term creep. For conservatism, the clad design will not rely on the support of the holddown spring in the plenum region. 2

4.2.1.3.2 Burnable Poison Rod Cladding Properties

Cladding tubes for burnable poison rods are purchased under the specification for fuel rod cladding tubes. Therefore, the mechanical, metallurgical, chemical, and dimensional properties of the cladding are as discussed in Section 4.2.1.2.2.

4.2.1.3.3 Al_2O_3-B_4C Burnable Poison Pellet Properties

The Al_2O_3-B_4C burnable poison pellets used in C-E designed reactors consist of a relatively small volume fraction of fine B_4C particles dispersed in a continuous Al_2O_3 matrix. The boron loading is varied by adjusting the B_4C concentration in the range from 0.7 to 4.0 wt% (1.0 to 6.0 vol%). The bulk density of the Al_2O_3-B_4C pellets is specified to be greater than 93% of the calculated theoretical density. Typical pellets have a bulk density of about 95% of theoretical. Many properties of the two-phase Al_2O_3-B_4C mixture, such as thermal expansion, thermal conductivity, and specific heat are very similar to the properties of the Al_2O_3 major constituent. In contrast, properties such as swelling, helium release, melting point, and corrosion are dependent on the presence of B_4C. The operating centerline temperature of burnable poison pellets is less than 1150°F, with a maximum surface temperature of 1090°F.

For definition of P_m, P_b, S_m, S'_m, S_u and F_s, see Section 4.2.1.1.1. For the Inconel 625 CEA cladding, the value of S_m is two-thirds of the minimum specified yield strength at temperature.

For Inconel 625, the specified minimum yield strength is 65,000 lb/in.^{2} at 650°F.

F_s = Mp/My where Mp is the bending moment required to produce a fully plastic section and My is the bending moment which first produces yielding at the extreme fibers of the cross-section. The capability of cross-sections loaded in bending to sustain moments considerably in excess of that required to yield the outermost fiber is discussed in Reference 1. For the CEA cladding dimensions, F_s = 1.33.

The values of uniform and total elongation of Inconel Alloy 625 cladding are estimated to be as follows:

	Fluence (E >1 MeV), nvt	
	1×10^{22}	3×10^{22}
Uniform elongation, %	3	1
Total elongation, %	6	3

4.2.1.4.4 Irradiation Behavior of Absorber Materials

A. Boron Carbide

1. Swelling

The linear swelling of B_4C increases with burnup according to the relationship:

$$\%\Delta L = (0.1)\ B^{10} \text{ Burnup, at } \%$$

This relationship was obtained from experimental irradiations on high density (≥90% TD) wafers (Reference 43) and pellets with densities ranging 44 between 71 and 98% TD (References 42 and [illegible]). Dimensional changes were measured as a function of burnup, after irradiating at temperatures expected in the design.

2. Thermal Conductivity

The thermal conductivity of unirradiated 73% dense B_4C decreases linearly with temperatures from 300 to 1600°F, according to the relationship:

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The second program~~, which is nearing completion~~ at ANO-2 ~~has~~ irradiated two fuel assemblies containing both standard and advanced design fuel rods to extended burnups. Both assemblies were extensively pre-characterized. One assembly was irradiated for three reactor cycles and reached an assembly-averaged burnup of 33 GWd/MTU. A second assembly was exposed to 5 cycles and reached an assembly-averaged burnup of 52 GWd/MTU (Reference 55). Both assemblies were examined after each reactor cycle. Visual examinations, oxide thickness measurements, and other dimensional measurements results in the conclusion that the performance of the fuel has been satisfactory. Destructive hot cell examinations are scheduled to complete the characterization of fuel behavior.

A surveillance program to follow the fuel performance of the System 80 design was carried out in Palo Verde-1. The program included poolside examinations after each of the first three operational cycles. The examinations included visual inspections for overall performance, dimensional measurements to characterize growth behavior, and cladding oxide measurements to track corrosion behavior of the fuel rod cladding. Results of this program indicate that the fuel behaves as expected with no indication that would alter the planned fuel management scheme for the System 80 fuel. These results are also applicable to System 80+ fuel.

4.2.2 DESCRIPTION AND DESIGN DRAWINGS

This subsection summarizes the mechanical design characteristics of the fuel system and discusses the design parameters which are of significance to the performance of the reactor. A summary of mechanical design parameters is presented in Table 4.2-1. These data are intended to be representative of the design; limiting values of these and other parameters will be discussed in the appropriate sections.

4.2.2.1 Fuel Assembly

The fuel assembly (Figure 4.2-6) consists of 236 fuel and poison rods, 5 guide tubes, 11 fuel rod spacer grids, upper and lower end fittings, and a holddown device. The outer guide tubes, spacer grids, and end fittings form the structural frame of the assembly.

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The results presented above were obtained through flow testing an oversized model of a standard 14 x 14 fuel assembly. Because of the great similarity in design between the Standard System 80 16 x 16 assembly and the earlier 14 x 14 array, these test results also constitute an adequate demonstration of the effects that flow blockage would have on the 16 x 16 assembly. This conclusion is also supported by the fact that the 16 x 16 assembly has been demonstrated to have a greater resistance to axial flow than would occur with the 14 x 14 array. The effect of the higher flow resistance, to produce more rapid flow recovery, i.e., more nearly uniform flow, is analogous to the common use of flow resistance devices (screens or perforated plates) to smooth non-uniform velocity profiles in ducts or process equipment.

4.2.3.2.15 Fuel Temperatures

Steady state fuel temperatures are determined by the FATES computer program. The calculational procedure considers the effects of linear heat rate, fuel relocation, fuel swelling, densification, thermal expansion, fission gas release, and clad deformation. The model for predicting fuel thermal performance is discussed in detail in References 15-17.

Two sets of burnup and axially dependent linear heat rate distributions are considered in the calculation. One is the hot rod, time averaged, distribution expected to persist during long-term operation, and the other is the envelope of the maximum linear heat rate at each axial location. The long-term distributions are integrated over selected time periods to determine burnup, which are in turn used for the various burnup dependent behavioral models in the FATES computer program. The envelope accounts for possible variations in the peak linear heat rate at any elevation which may occur for short periods of time and is used exclusively for fission gas release calculations.

The power history used assumes continuous 100% reactor power from beginning-of-life. Using this history, the highest fuel temperatures occur at that time. It has been shown that fuel temperatures for a given power level at any burnup are insensitive to the previous history used to arrive at the given power level.

Fuel thermal performance parameters are calculated for the hot rod. These parameters for any other rod in the core can be obtained by using the axial location in the hot rod, whose local power and burnup corresponds to the local power and burnup in the rod being examined. This procedure will yield conservatively high stored energy in the fuel rod under consideration.

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to AMS, ASTM or C-E specifications. In addition, various CEA hardware tests have been conducted or are in progress.

During manufacturing, the following inspections and tests are performed:

A. The loading of each control element is carefully controlled to obtain the proper amounts and types of filler materials for each type of CEA application (e.g., full-strength B_4C or Ag-In-Cd; part-strength Inconel 625).

B. All end cap welds are liquid penetrant examined and helium leak tested. A sampling plan is used to section and examine end cap welds.

C. Each type of control element has unique external features which distinguish it from other types.

D. Each CEA is serialized to distinguish it from the others. See Figures 4.2-3 through 4.2-5 and Figure 4.2-14.

E. Fully assembled CEAs are checked for proper alignment of the neutron absorber elements using a special fixture. The alignment check ensures that the frictional force that could result from adverse tolerances is below the force which could significantly increase scram time.

In addition to the basic measures discussed above, the manufacturing process includes numerous other quality control steps for ensuring that the individual CEA components satisfy design requirements for material quality, detail dimensions, and process control.

After installation in the reactor, but prior to criticality, each CEA is traversed through its full stroke and tripped. A similar procedure will also be conducted at refueling intervals.

The required 90% insertion scram time for CEAs is 4.0 seconds under worst case conditions. Verification of adequacy was initially determined by testing in the C-E TF-2 flow test facility as reported in Appendix 4B. This test facility contained prototypical (System 80) reactor components consisting of fuel assemblies, CEA shroud, control element drive mechanism, and a simulation of surrounding core internal support components. The test conditions simulated the range of temperatures and flow rates predicted for System 80 normal plant operation. The required scram time has been subsequently verified to be conservative by testing at the Palo Verde (System 80) operating units.

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TABLE 4.2-1 (Cont'd)

(Sheet 2 of 7)

MECHANICAL DESIGN PARAMETERS

Fuel Assemblies (Cont'd)

Spacer Grid	
Type	Leaf spring
Material	Zircaloy-4
Number per assembly	10
Weight each, lb	2.0
Bottom Spacer Grid	
Type	Leaf spring
Material	Inconel 625
Number per assembly	1
Weight each, lb ~~(with skirt)~~	2 [redacted]
Weight of fuel assembly (nominal), lb	1461
Outside dimensions	
Fuel rod to fuel rod, inches	7.972 x 7.972
Fuel Rod	
Fuel rod material (sintered pellet)	UO_2
Pellet diameter (nominal), inches	0.3255
Pellet length, inches	0.390
Pellet density (nominal), g/cm^3	10.47
Pellet theoretical density, g/cm^3	10.96
Pellet density (nominal) (% theoretical)	95.5
Stack height density (nominal), g/cm^3	10.315
Clad material	Zircaloy-4
Clad ID, inches	0.332
Clad OD, (nominal), inches	0.382
Clad thickness, (nominal), inches	0.025
Diametral gap, (cold, nominal), inches	0.0065
Active length, inches	150
Plenum length, inches	7.938

The core average linear heat rate is also linear with power. The average effective fuel temperature dependence on the core average linear heat rate is calculated from the following semi-empirical relation:

$$T_f = T_{MOD} + \left(\sum_{i=0}^{2} B_i * M^i\right) * P + \left(\sum_{j=0}^{3} C_j * M^j\right) * P^2 \qquad (1)$$

T_{MOD} is the average moderator temperature (°F), M is the exposure in MWd/MTU, P is the linear heat generation rate in the fuel in kW/ft, and T_f is the average effective fuel temperature (°F). The coefficients B_i and C_i are determined from least squares fitting of the fuel temperature generated by FATES (References 4,5). For the fuel pins in System 80+, the following values apply:

B_o = +137.248 $\cancel{146.526}$

B_1 = +0.828149 $\cancel{0.8841}$ $* 10^{-3}$

B_2 = -0.192215 $\cancel{-0.2052}$ $* 10^{-6}$

C_o = -1.86062 $\cancel{-2.6955}$

C_1 = -0.468091 $\cancel{-0.5421}$ $* 10^{-3}$

C_2 = +0.460960 $\cancel{1.5643}$ $* 10^{-7}$

C_3 = -0.976452 $\cancel{-0.1671}$ $* 10^{-12}$

The basis for this relation is discussed in Reference 3.

The total power coefficient at a given core power can be determined by evaluation, for the conditions associated with the given power level, of the following expression:

$$\frac{d\rho}{dp} = \frac{\delta\rho}{\delta T_f}\,\frac{\delta T_f}{\delta p} + \frac{\delta\rho}{\delta T_m}\,\frac{\delta T_m}{\delta p} \qquad (2)$$

The first term of the equation (2) provides the fuel temperature contribution to the power coefficient, which is shown as a function of power in Figure 4.3-45.

The first factor of the first term is the fuel temperature coefficient of reactivity discussed in Section 4.3.2.3.1 and shown in Figure 4.3-41. The second factor of the first term is obtained by calculating the derivative of Equation (1).

$$\frac{\delta T_i}{\delta p} = \left(\sum_{i=0}^{2} B_i * M^i\right) + 2\left(\sum_{j=0}^{3} C_j * M^j\right) * P \qquad (3)$$

77. Inchikawa, Uchida, Yanagisawa, Nakajima, Nakamura and Kawaski, JAERI; Hayevik, Knudsen and Kolstad, Halden, "Studies of LWR Fuel Performance Under Power Ramping and Power Cycling Utilizing In-Pile Measurement and Fuel Modeling", Proceedings of the ANS Topical Meeting, Williamsburg, Virginia, April 1988.

78. "Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWD/kgU for Combustion Engineering 16x16 PWR Fuel", CEN-386-P-A, August 1992.

79. Marlowe, Mo. O., "High Temperature Isothermal Elastic Modulii of UO_2", Journal of Nuclear Materials, Vol. 33 (1969), Pages 292 - 244.

TABLE 4.3-1

(Sheet 1 of 2)

NUCLEAR DESIGN CHARACTERISTICS

Item	Value
General Characteristics	
Fuel management	3-batch,mixed central zone
Core Average Burnup (MWD/MTU), 10 ppm soluble boron	16,000
Core Average U-235 Enrichment (wt%)	2.6
Core Average H_2O/UO_2 volume ratio, first cycle, hot (core cell)	2.06
Number of control element assemblies	
Full strength	68
Part strength	25
Burnable Poison Rods	
Number	11,680
Material	Er_2O_3
Worth % $\Delta\rho$, at BOC	
Hot, 587°F	5.3
Cold, 68°F	4.0
Dissolved Boron	
Dissolved boron content for criticality, ppm, (CEAs withdrawn, BOC)	
Cold, 68°F	1431
Hot, zero power, clean, 565°F	~~1400~~ 1414
Hot, full power, clean, 587°F	~~1284~~ 1270
Hot, full power, equilibrium Xe	1006

TABLE 4.3-1

(Sheet 2 of 2)

NUCLEAR DESIGN CHARACTERISTICS

Item	Value
Dissolved boron content (ppm) for:	
Refueling	2150
5% subcritical, cold, first cycle (all CEAs out) 0 MWd/MTU	1837
5% subcritical, hot, first cycle (all CEAs out) 0 MWd/MTU	1920
Boron worth, ppm/% $\Delta\rho$ (BOC/EOC)	
Hot, 587°F	96/91
Cold, 68°F	78/66
Neutron Parameters	
Neutron lifetime (cycle average), microseconds	~~27.6~~ 28.4
Delayed neutron fraction (cycle average)	0.0061
Plutonium Buildup (first cycle)	
$\frac{\text{g Fissile Pu (final)}}{\text{kg U (original)}}$	4.68
$\frac{\text{g Total Pu (final)}}{\text{kg U (original)}}$	6.02

TABLE 4.3-2

EFFECTIVE MULTIPLICATION FACTORS AND REACTIVITY DATA[a]

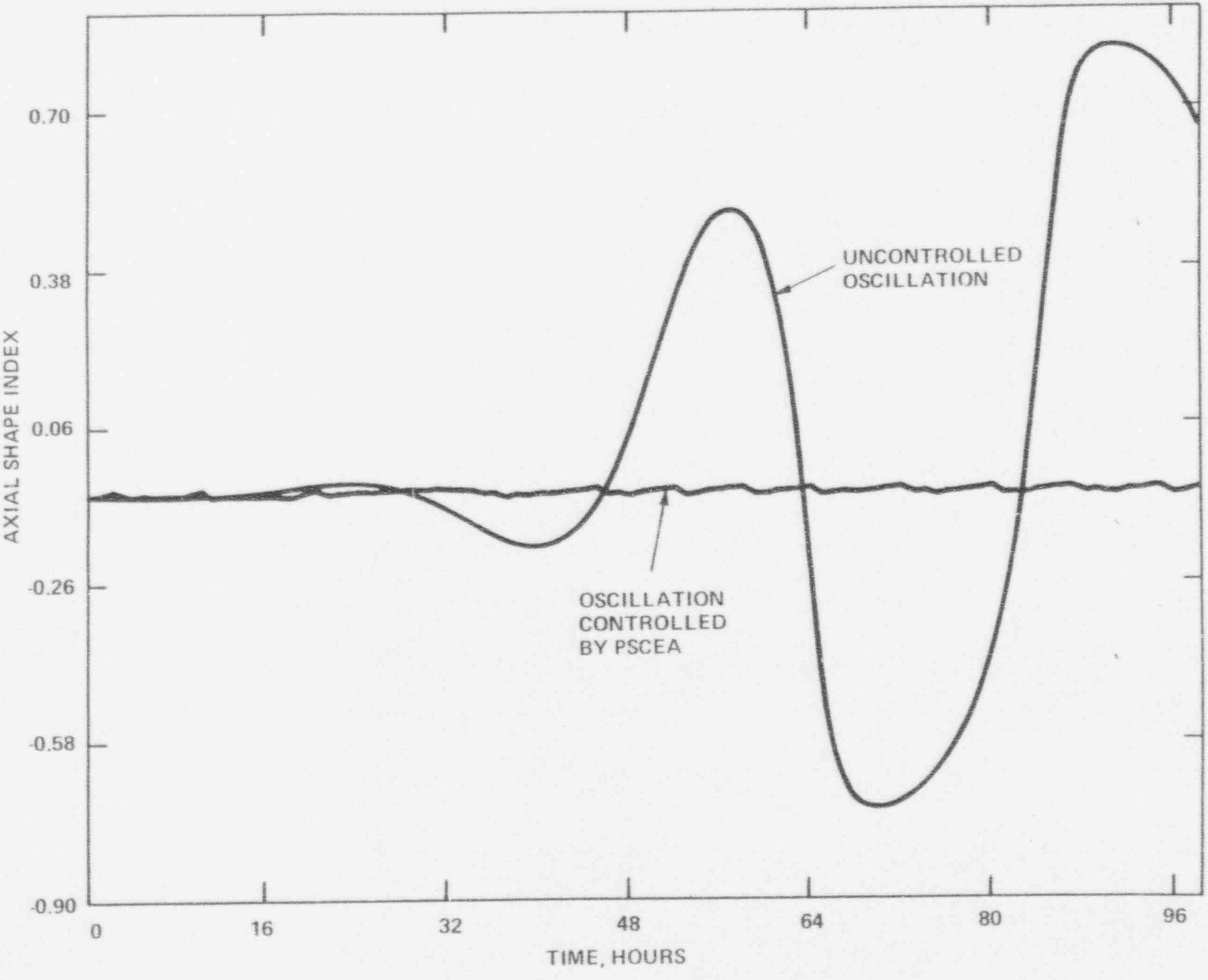
Condition	K_{eff}	ρ
Cold, 68°F (OPPM), BOC1	1.233	0.189
Cold (68°F) at minimum refueling boron concentration (2150 ppm), BOC1	0.916	-0.092
Hot, 557°F, zero power, clean (0 ppm), BOC1	1.173	0.148
Hot, full power, no Xe or Sm, 587°F (0 ppm), BOC1	1.148	0.129
Hot, full power, equilibrium Xe (0 ppm)	1.111	0.100
Hot, full power, equilibrium Xe and Sm (0 ppm)	1.107	0.096
Reactivity decrease, hot		
Zero to full power, BOC (911 ppm)		0.014
Fuel temperature		0.012
Moderator temperature		0.002
Reactivity decrease, hot		
Zero to full power, EOC (0 ppm)		0.020
Fuel temperature		0.011 ~~0.009~~
Moderator temperature		0.009 ~~0.011~~

(a) No control element assemblies or dissolved boron except as noted, initial core.

TABLE 4.3-4

REACTIVITY COEFFICIENTS

Moderator Temperature Coefficient, $\Delta\rho/°F$	
Beginning-of-cycle (0-50 MWd/MTU)	
Cold, 68°F, Clean, 1431 ppm	-0.20×10^{-4}
Hot zero power, 557°F, no CEAs, Clean, 1400 ppm	$\boxed{-0.03}$ $\cancel{-0.26} \times 10^{-4}$
Hot full power, 587°F, no CEAs, Clean, 1284 ppm	-0.37×10^{-4}
Hot full power, 587°F, no CEAs, Equilibrium Xe, 1006 ppm	-0.65×10^{-4}
Hot zero power, 557°F, regulating CEA banks 3, 2 and 1 inserted, 50 MWd/MTU, 1006 ppm, Hot full power equilibrium Xe	-0.69×10^{-4}
End-of-Cycle (10 ppm soluble boron, 16,000 MWd/MTU)	
Cold, 68°F (approximate)	-0.04×10^{-4}
Hot zero power, 557°F, no CEAs, Hot full power equilibrium Xe	-1.70×10^{-4}
Hot full power, equilibrium Xe, no CEAs, 587°F	-2.60×10^{-4}
Hot zero power, 557°F, rodded, regulating CEA banks 3, 2 and 1 inserted, Hot full power equilibrium Xe	-2.00×10^{-4}
Moderator Density Coefficient, $\Delta\rho/gm/cm^3$	
Hot, operating, 587°F	
Beginning-of-cycle, 1284 ppm soluble boron, 0 MWd/MTU	+.031
Fuel temperature contribution to power coefficient, $\Delta\rho/(kW/ft)$, 1006 ppm, 50 MWd/MTU	
Hot zero power	$\boxed{-2.08}$ $\cancel{-2.12} \times 10^{-3}$
Full power	$\boxed{-1.77}$ $\cancel{-1.80} \times 10^{-3}$
Moderator void coefficient $\Delta\rho$/% void	
Hot, operating, 587°F	
Beginning-of-cycle, 1284 ppm soluble boron, 0 MWd/MTU	-0.22×10^{-3}
Moderator pressure coefficient, $\Delta\rho/psi$	
Hot, operating, 587°F	
Beginning-of-cycle, 1284 ppm soluble boron, 0 MWd/MTU	$+3.96 \times 10^{-6}$
Overall power coefficient, $\Delta\rho/(kW/ft)$	
Hot, operating, 587°F	
Beginning-of-cycle, 1006 ppm soluble boron, 50 MWd/MTU	-1.84×10^{-3}
End-of-cycle, 10 ppm soluble boron, 16,000 MWd/MTU	$\boxed{-4.31}$ $\cancel{-4.38} \times 10^{-3}$



SYSTEM 80+™

XENON

PSCEA CONTROLLED AND UNCONTROLLED OSCILLATION

Figure 4.3-53

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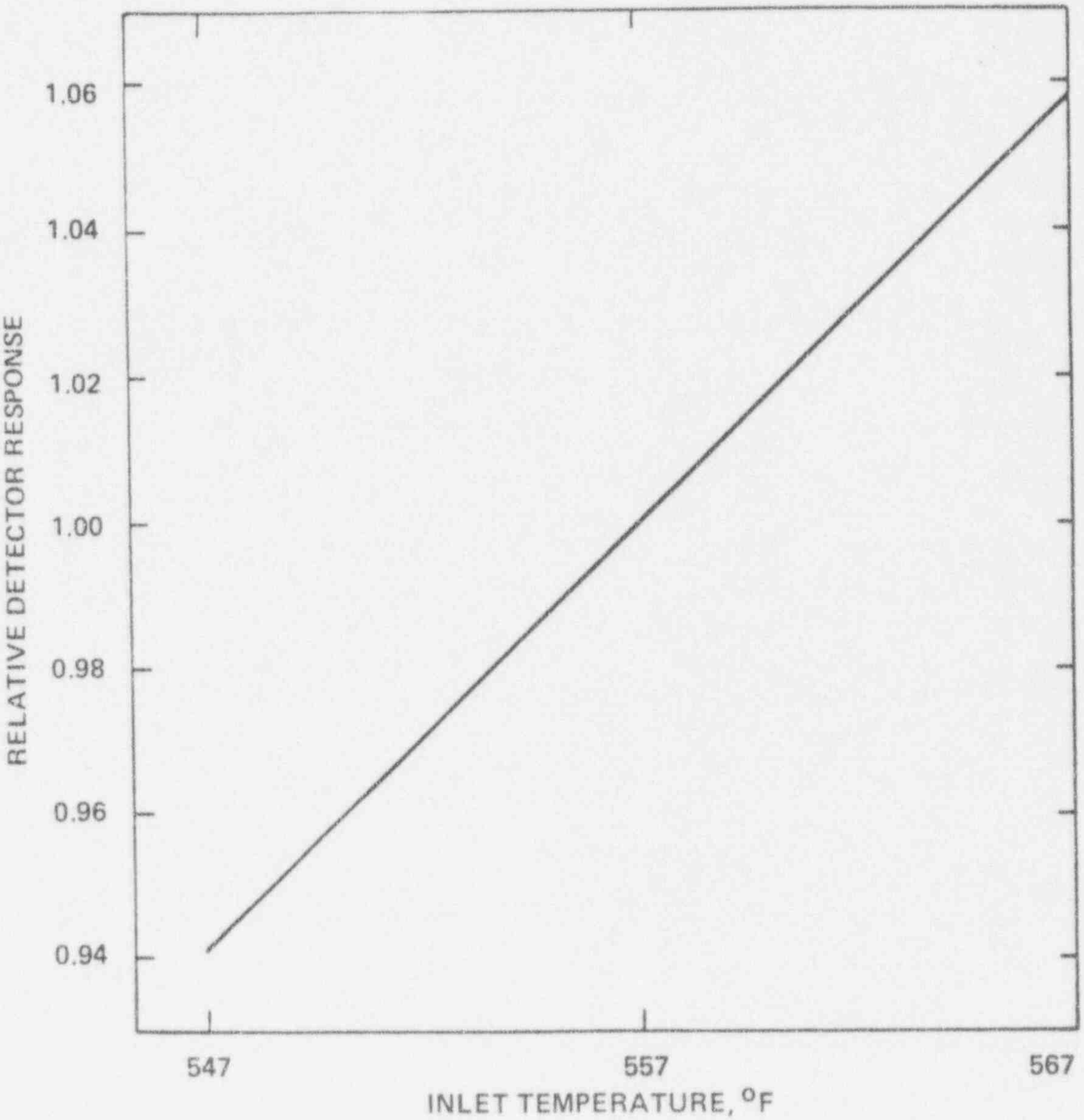

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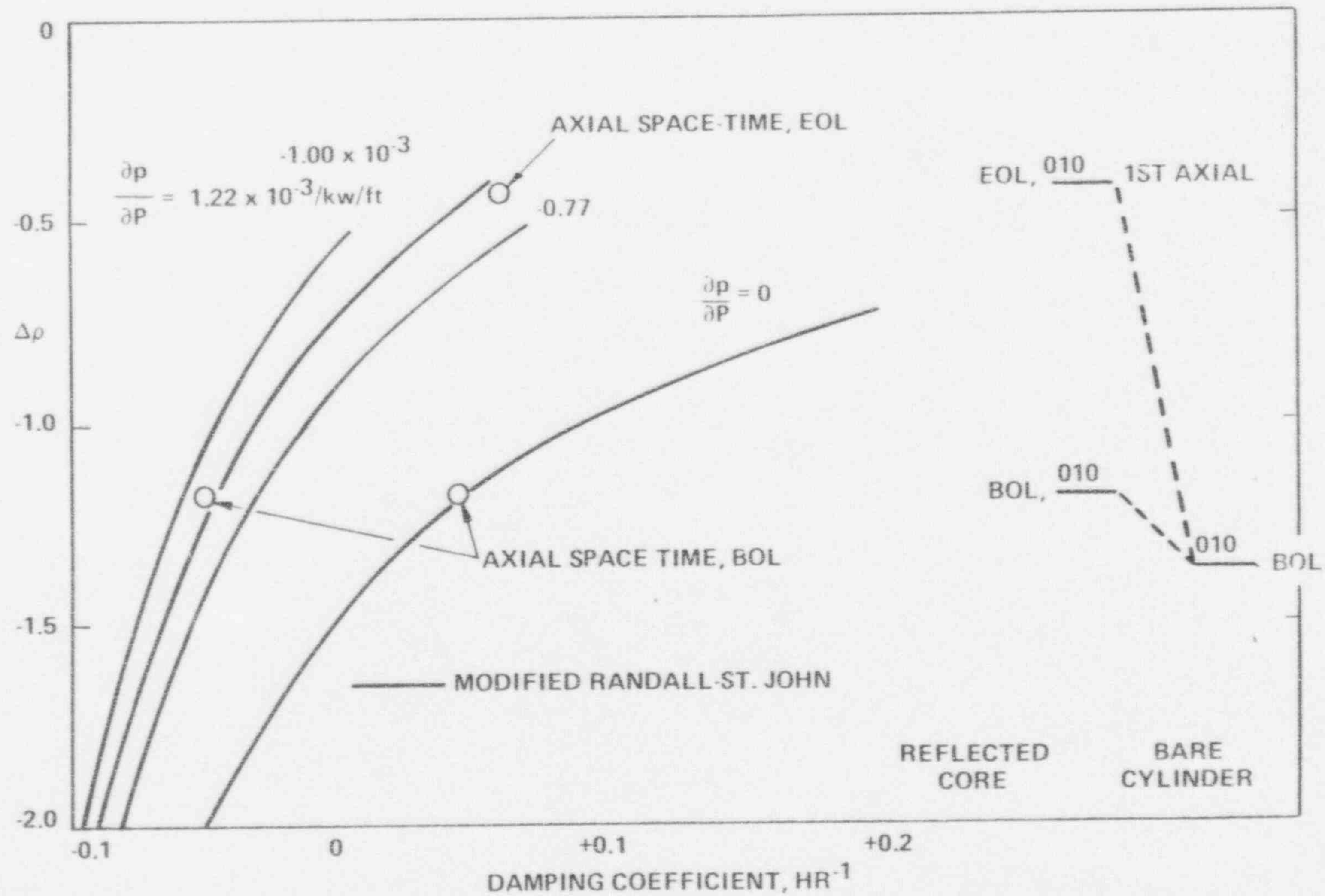
DECALIBRATION EFFECT

SYSTEM 80+™	TYPICAL TEMPERATURE ~~DEFECT~~ vs REACTOR INLET TEMPERATURE	Figure 4.3-58



SYSTEM 80+™

DAMPING COEFFICIENT vs REACTIVITY DIFFERENCE BETWEEN FUNDAMENTAL AND EXCITED STATE

Figure 4.3-63

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rod with the minimum DNBR. Diversion crossflow and turbulent interchannel mixing are not input as factors on subchannel enthalpy rise but are explicitly treated in the TORC and CETOP analytical models.

Uncertainties in the power distribution factors are discussed in Section 4.4.2.9.4.

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Statistical Combination of Uncertainties (SCU) methods, as described in References 8, were used to statistically combine the uncertainties of the thermal hydraulic code input parameters (system parameters). This SCU methodology with plant-specific data is statistically combined with CE-1 CHF correlation statistics at the 95/95 confidence/probability level to yield an increased DNBR limit. This limit is approximately 1.24 when the following uncertainties are combined:

a) uncertainty in the inlet flow distribution;
b) systematic variation on fuel rod pitch;
c) systematic variation on fuel clad OD;
d) engineering enthalpy rise factor;
e) engineering heat flux factor;
f) penalty on DNBR (minimum) due to fuel rod bowing; and,
g) statistics associated with the NRC-approved 1.19 DNBR limit (Reference 2).

Also included in the MDNBR limit is the penalty due to the CHF correlation uncertainty and a 0.01 penalty for the HID grids, as well as penalties imposed by NRC to account for CHF correlation "prediction uncertainty" and TORC code uncertainty. The 1.24 DNBR limit is used in safety analysis, CPC trip setpoints and COLSS power operating limit calculations in conjunction with a CETOP model based on a nominal geometry.

4.4.2.2.2.1 Power Distribution Factors

A. Rod Radial Power Factor

The rod radial power factor is the ratio of the average power per unit length produced by a particular fuel rod to the average power per unit length produced by the average powered fuel rod in the core. The maximum rod radial power factor is the ratio of the average power per unit length produced by the highest powered rod in the core to the average power per unit length produced by the average powered fuel rod in the core. Radial power distributions are dependent upon a variety of parameters (e.g., control rod insertion, power level, fuel exposure). The core wide and hot assembly radial power distributions used for a typical DNB analysis are shown in Figures 4.4-1 and 4.4-2.

4.4.2.2.3 Fuel Densification Effect on DNBR

The perturbation in local heat flux due to fuel densification is given in Table 4.4-1. As shown in CENPD-207(2) (see Section 4.4.4.1), much larger local heat flux variations have no significant adverse effect on DNB. Therefore, no specific allowance is made or required for the effect on DNBR of local heat flux variations due to fuel densification.

4.4.2.3 Linear Heat Generation Rate

The core average and maximum fuel rod linear heat generation rates are given in Table 4.4-1. The maximum fuel rod linear heat generation rate is determined by multiplying the core average fuel rod linear heat generation rate by the product of the nuclear power factor, the engineering factor on linear heat rate, and the ratio of the hot to the average fuel rod energy deposition fractions. The effects of fuel densification are not included in the maximum fuel rod linear heat generation rate presented in Table 4.4-1; although, to determine the maximum local linear heat generation rate including the effect of gaps occurring between the fuel pellets, the augmentation factor is applied.

4.4.2.4 Void Fraction Distribution

The core average void fraction and the maximum void fraction are calculated using the Maurer method (10). The void fractions discussed below are values for the reactor operating conditions and engineering factors given in Table 4.4-1, for the radial power distribution in Figure 4.4-1 and 4.4-2, and for the 1.26 peaked axial power distribution in Figure 4.4-3. For these conditions, only subcooled boiling occurs in the core.

0.5

The core average void fraction is essentially zero. The local maximum void fraction is 0.~~[illegible]~~% and occurs at the exit of the subchannel adjacent to the rod with the minimum DNBR. The average exit void fractions and qualities in different regions of the core are shown in Figure 4.4-4 for the core radial power distribution shown in Figure 4.4-1. The axial distribution of void fraction and quality in the subchannel adjacent to the rod with the minimum DNBR is shown in Figure 4.4-5. The average void fraction in that subchannel is less than 0.1%. B

4.4.2.5 Core Coolant Flow Distribution

The core inlet flow distribution is required as input to the TORC thermal margin code (refer to Section 4.4.4.5.2). The inlet flow distribution for 4-loop operation was determined from a System 80

h. The same fuel rod energy deposition fraction is used for the hot rod as for the average rod. The hotter the rod, the lower is the actual value of energy deposition fraction with respect to that for the average rod. A lower energy deposition fraction reduces the hot rod heat flux and thereby increases its DNBR. The use of the average rod energy deposition fraction for the hot rod is therefore conservative. See Section 4.3 for a discussion of the calculation of the energy deposition fractions.

2. Uncertainty in the analytical model:

The ability of the TORC code to predict accurately subchannel local conditions in rod bundles is described in Reference 6. The ability of the code to predict accurately the core wide coolant conditions is described in Reference 13. However, an allowance for TORC code uncertainty is included in the Statistical Combination of Uncertainties analysis as discussed in Section 4.4.2.9.5.

3. Uncertainty in the DNB correlation:

The uncertainty in the DNB correlation is determined by a statistical analysis of DNB test data. A value of 1.20 has been shown to provide a 95% probability with 95% confidence that DNB will not occur on a fuel rod having that minimum DNBR (Reference 3).

4.4.2.9.5 Statistical Combination of Uncertainties (SCU)

Use of a 1.24 MDNBR limit with a best-estimate design CETOP model will ensure, with at least 95% probability and 95% confidence, that the hot pin will not experience a departure from nucleate boiling. The 1.24 MDNBR limit includes explicit allowances for system parameter uncertainties, CHF correlation uncertainty, rod bow, the NRC penalties for the TORC code uncertainty and CHF correlation "prediction uncertainty," and a 0.01 penalty for the HID grids.

Several conservatisms are included in the SCU methodology (References 8 AND 26). The significant conservatisms include:

1. Combination of system parameter probability distribution functions at the 95% confidence level to yield a resultant MDNBR at >95% confidence.

2. Use of pessimistic system parameter probability distribution functions.

Pressure	1745 to 2425 psia
Inlet temperature	333 to 631°F
Local coolant quality	-0.27 to +0.20
Local mass velocity	0.81×10^6 to 3.07×10^6 lb/h-ft^2

It was found that the mean and standard deviation for the ratio of measured to predicted DNB heat fluxes were 1.229 and 0.125, respectively, for the 369 DNB data within the parameter ranges noted above.

Testing was also conducted with rod bundles representative of the 16 x 16 fuel assembly to determine the effect on DNB of local perturbations in heat flux. Results are presented in CENPD-207(2) for two nonuniform axial power rod bundles which were similar except that one test bundle had a heat flux spike (23% higher heat flux for a 4-inch length) at the location where DNB was anticipated. The results show that there is no significant adverse effect on DNB due to that flux spike. Therefore, it is concluded that no allowance is required for the effect on DNB of local heat flux perturbations less severe than that tested.

One important factor in the prediction of DNB and local coolant conditions is the treatment of turbulent interchannel mixing. The effect of turbulent interchange on enthalpy rise in the subchannels of 16 x 16 fuel assemblies with standard spacer grids is calculated in the TORC code by

ADD

$$Pe = \frac{W'}{\bar{G}\,\bar{D}_e} = 0.0035$$

where:

Pe = inverse Peclet number.

ADD

W′ = turbulent interchange between adjacent subchannels, lb/h-ft.

$\bar{D}_e$ = average equivalent diameter of the adjacent subchannels, ft.

$\bar{G}$ = average mass velocity of the adjacent subchannels, lb/h-ft^2.

The value of 0.0035 for the inverse Peclet number for use with the 16 x 16 fuel assembly with standard spacer grids was originally chosen based on cold water dye mixing tests conducted for the 14 x 14 assembly and for a "prototype" of the Palisades

AMENDMENT V

24. Currin, H. B., et al, "HYDNA-Digital Computer Program for Hydrodynamic Transients in a Pressure Tube Reactor or a Closed Channel Core", Report CVNA-77, 1961.

25. "Additional Thermal-Hydraulic Information on Combustion Engineering 3390 MWTH Reactor Cores", CENPD-12, February, 1971.

26. "Modified Statistical Combination of Uncertainties", CEN-356(V)-P-A, Revision 1-P-A, Combustion Engineering, May 1988.

ADD REFERENCE

DELETE LINE

TABLE 4.4-1
(Sheet 1 of 2)

THERMAL AND HYDRAULIC PARAMETERS

Reactor Parameters	System 80+	System 80	Waterford-3
Core Average Characteristics at Full Power:			
Total core heat output, MWt	3,914	3,800	3,390
Total core heat output, million Btu/h	13,360	12,970	11,570
Average fuel rod energy deposition fraction	0.974	0.974	0.974
Hot fuel rod energy deposition fraction	0.971	0.971	0.971
Primary system pressure, psia	2,250	2,250	2,250
Reactor inlet coolant temperature, °F	556	565	553
Reactor outlet coolant temperature, °F	615	~~624~~ 621	611
Core exit average coolant temperature, °F	617	623	~~613~~ 612
Average core enthalpy rise, Btu/lbm	83	82	~~81~~ 80
Design minimum primary coolant flow rate, gpm	444,650	445,600	396,000
Design maximum core bypass flow, % of primary	3.0	3.0	~~3.5~~ 2.6
Design minimum core flow rate, gpm	431,300	432,200	~~382,000~~ 385,700
Hydraulic diameter of nominal subchannel, in.	0.471	0.471	0.471
Core flow area, ft^2	60.8	60.8	54.7
Core avg mass velocity, million $lbm/h\text{-}ft^2$	2.65	2.62	~~2.61~~ 2.64
Core avg coolant velocity, ft/s	16.7	~~16.7~~ 16.8	~~16.3~~ 16.5
Core avg fuel rod heat flux, $Btu/h\text{-}ft^2$	183,300	184,800[a)]	182,100
Total heat transfer area, ft^2	70,960	68,320[a)]	61,860

[a)] Corrected values for System 80 design

TABLE 4.4-1 (Cont'd)
(Sheet 2 of 2)
THERMAL AND HYDRAULIC PARAMETERS

Reactor Parameters	System 80+	System 80	Waterford-3
Average fuel rod linear heat rate kW/ft	~~5.37~~ 5.36	~~5.42~~ 5.41	~~5.34~~ 5.33
Power density, kW/liter	98.4	95.5	94.9
No. of active fuel rods	56,876	54,764	49,580
Power Distribution Factors:			
Rod radial power factor	1.55	1.55	1.55
Nuclear power factor	2.28	2.28	2.28
Total heat flux factor	~~2.35~~ 2.34	~~2.35~~ 2.34	~~2.35~~ 2.34
Engineering Factors:			
Engineering heat flux factor	1.03	1.03	1.03
Engineering enthalpy rise factor	1.03	1.03	1.03
Pitch, Bowing, and Clad Diameter Enthalpy Rise	1.05	1.05	1.05
Engineering factor on linear heat rate	1.03	1.03	1.03
Characteristics of Rod and Channel with Minimum DNBR:			
Maximum fuel rod heat flux, Btu/h-ft^2	429,100 ~~432,200~~	432,700 ~~434,300[a)]~~	426,300 ~~427,900~~
Maximum fuel rod linear heat rate, kW/ft	~~12.7~~ 12.6	12.7	12.5
UO_2 maximum steady state temperature, °F	3,179	3,205[a)]	3,180
Outlet temperature, °F	644.1	645.7[a)]	642
Outlet enthalpy, Btu/lbm	684.3	687.1[a)]	680
Minimum DNBR at nominal conditions (CE-1 correlation)	2.00	1.98[a)]	2.07

[a)] Based on updated System 80 flow distribution

TABLE 4.4-8

REACTOR COOLANT SYSTEM GEOMETRY

Component	Flow Path Length (ft)	Top Elevation[d] (ft)	Bottom Elevation[d] (ft)	Minimum Flow Area (ft^2)	Volume (ft^3)
Hot Leg (ea)	14.10	2.38	- 1.75	9.62	135.64
Suction Leg (ea)	24.22	0.58	- 9.97	4.91	118.91 ~~118.92~~
Discharge Leg (ea)	19.31	1.25	- 1.25	4.91	94.8 ~~94.82~~
Pressurizer		(e)			2400
Liquid Level (full power)		(e)	(e)	50.07[a]	1200
Surge Line	~~120.96~~ 105.0	(e)	1.75	0.56	58.75 ~~70~~
Steam Generator					
Inlet Nozzle	3.07	3.90	- 0.48	9.62	26.81 ~~30.42~~
Outlet Nozzle (ea)	2.79	2.41	- 1.19	4.91	13.49
Inlet Plenum	4.74[b]	6.48	- 0.10	19.07	423.4
Outlet Plenum	4.74[b]	6.48	- 0.10	9.74	423.4
Tubes (Active & Inactive)	63.9	40.94	6.48	0.002[c]	2072.8
Reactor Vessel					
Inlet Nozzle (ea)	3.7	1.4	- 1.5	4.9	21.7
Downcomer	21.4	11.7	-22.6	33.8	1157.1
Lower Plenum	3.2	-20.5	-25.9	32.5	430.2
Lower Support Structure & Inactive Core	2.8	-17.7	-20.5	44.4	250.0
Active Core	12.5	~~- 5.3~~ -5.1	~~-17.8~~ -17.6	60.8	817.2
Upper Inactive Core	~~2.8~~ 2.6	- 2.5	~~- 5.3~~ -5.1	46.3	251.1
Outlet Plenum	5.7	2.1	- 2.4	26.6	459.4
Core Shroud Bypass	15.9	- 2.7	-19.6	0.1	240.6
CEA Shroud Assembly & Tie Tubes	17.9	15.6	- 3.5	0.4	1352.5
UGS, CEA Shroud Annulus	10.6	12.7	2.1	1.6	226.0
Top Head	3.2	19.9	12.7	7.8	422.6
Outlet Nozzle (ea)	4.0	1.7	- 1.8	9.6	32.2

(a) For the cylinder.
(b) Represents a geometrical rather than an actual flow path length.
(c) Flow path area per tube.
(d) Reactor Vessel nozzle centerline is the reference elevation. It has an elevation of 0.0 ft.
(e) See Section 5.4.

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TABLE 1.9-1

INDEX OF SYSTEM, STRUCTURE OR COMPONENT INTERFACE REQUIREMENTS FOR SYSTEM 80+

System, Structure or Component	Section
Buildings/Structures	
Administration Building	1.2.1.4.1.1
Personnel Access Portal	1.2.1.4.1.2
Switchyard	8.2
Warehouse	1.2.1.4.1.3
Emergency Operations Facility	13.3.3.2
Bulk Gas Storage	9.5.10.1.2
Station Service Water Pump Structure	9.2.1.1.4
Ultimate Heat Sink, Including SSWS Intake/Discharge	9.2.5.1.3
Potable and Sanitary Water System Structure	9.2.4
Systems	
Condenser Circulating Water System, including Normal Heat Sink, Pump Structure, Intake and Discharge, and Turbine Building Service Water System	10.4.5.1, 9.2.10.2.1
Offsite Power System, including Switchyard	8.1
Potable and Sanitary Water Systems, including Sewage Treatment	9.2.4.1
Security System	13.6.1
Service Water Pump Structure Ventilation System	9.4.8.1.2
Layout and Equipment for the Laboratory Facilities	13.3.3.4
Layout and Equipment for the Onsite Decontamination Facilities	13.3.3.6
Components	
Component Cooling Water Heat Exchanger Materials	9.2.2
Condenser Materials Specification	10.3.6.2,10.4.1.2
TOXIC GAS MONITORS	9.4.1.1, FIG. 9.4-2
FILTERED WATER SOURCE	9.2.3.2, 9.2.4
COMMUNICATIONS (OFF-SITE)	9.5.2.2.5

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& Tom Wambach
& Jerry Wilson

1.9 SYSTEM 80+ STANDARD DESIGN INTERFACES

This section provides a listing of the interfaces ~~requirements~~ as used in 10 CFR 52.47(a). The System 80+™ Standard Design includes an essentially complete nuclear plant, except for structures, systems and components which require site-specific design. These structures, systems and components are not included in the System 80+™ design certification and shall be provided by the applicant (owner/operator) during site specific engineering. To ensure that the design of these items is compatible with the System 80+™ Standard Design, interface requirements must be satisfied by the applicant. In general, interface requirements for applicant-supplied structures, systems and components which are related to a specific mechanical or electrical system are covered in the appropriate CESSAR-DC chapter (The word "shall" is used to identify interface requirements included in descriptive text). Table 1.9-1 provides an index of all sections in CESSAR-DC containing interface requirements.

Site specific assumptions on which the System 80+™ Standard Design is based are presented in Section 1.2.1, Principal Site Characteristics, and Chapter 2.0, Site Envelope Characteristics and the applicant (owner/operator) shall verify that the chosen site is enveloped by the characteristics given in Sections 1.2.1 and 2.0. These site-specific characteristics must be compatible with the System 80+ design envelopes, but they are not considered interface requirements as used in 10 CFR 52.47(a).

Interface requirements which have sufficient significance to safety are specified in the Certified Design Material.

ATTACHMENT 3

pump and its reactor coolant pump support loads. Each RAT has the capability of supplying power directly (i.e., not through any bus supplying non-Class 1E loads) to its respective Class 1E buses.

UAT power feeders, and instrumentation and control circuits are separated from the RAT's power feeders, and instrumentation and control circuits

Power feeders, and instrumentation and control circuits for the UMT and its switching station are separated from power feeders, and instrumentation and control circuits for the RATs and their switching station.

EPDS medium voltage switchgear, low voltage switchgear and their respective transformers, MCCs, and MCC feeder and load circuit breakers are sized to supply their load requirements. EPDS medium voltage switchgear, low voltage switchgear and their respective transformers, and MCCs are rated to withstand fault currents for the time required to clear the fault from its power source.

The GCB, medium voltage switchgear, low voltage switchgear, and MCC feeder and load circuit breakers are rated to interrupt fault currents.

EPDS interrupting devices (circuit breakers and fuses) are coordinated so that the circuit interrupter closest to the fault is designed to open before other devices.

Instrumentation and control power for Class 1E Divisional medium voltage switchgear and low voltage switchgear is supplied from the Class 1E DC Power System in the same Division.

The GCB is equipped with redundant trip coils supplied from separate non-Class 1E DC power systems.

EPDS cables and buses are sized to supply their load requirements. EPDS cables and buses are rated to withstand fault currents for the time required to clear the fault from its power source.

For the EPDS, Class 1E power is supplied by two independent Class 1E Divisions. Independence is maintained between Class 1E Divisions, and between Class 1E Divisions and non-Class 1E equipment.

Class 1E medium voltage switchgear, low voltage switchgear, and MCCs are identified according to their Class 1E Division/Channel. Class 1E medium voltage switchgear, low voltage switchgear, and MCCs are located in Seismic Category I structures and in their respective Division areas.

The backup pressurizer heaters, emergency lighting, RCP seal injection pump and RCP seal injection pump room ventilation fan, are the only electrical loads classified as non-Class 1E which are directly connectable to the Class 1E buses.

2.6.3 AC INSTRUMENTATION AND CONTROL POWER SYSTEM AND DC POWER SYSTEM

DESIGN DESCRIPTION

The AC Instrumentation and Control (I&C) Power System and DC Power System consist of Class 1E and non-Class 1E power systems. The non-Class 1E AC I&C Power System and DC Power System have non-Class 1E batteries, inverters, electrical distribution panels, and battery chargers. The non-Class 1E AC I&C Power System and DC Power System provide power to non-Class 1E equipment.

The Class 1E AC Instrumentation and Control (I&C) Power System (also referred to as the Vital AC I&C Power System) and the Class 1E DC Power System (also referred to as the Vital DC Power System) consist of Class 1E uninterruptible power supplies, their respective alternating current (AC) and direct current (DC) distribution centers, along with power, instrumentation and control cables to the distribution system loads. The Class 1E AC I&C Power System and the Class 1E DC Power System include the protection equipment provided to protect the AC and DC distribution equipment.

The Basic Configuration of the Class 1E AC Instrumentation and Control Power System and Class 1E DC Power System is as shown on Figures 2.6.3-1 and 2.6.3-2.

<u>Class 1E AC Instrumentation and Control Power System</u>

The Class 1E AC I&C Power System consists of two Division (Division I and II) and four Channel (A, B, C, D) uninterruptible power supplies, with their respective distribution panels.

Each Class 1E AC I&C power supply is a constant voltage constant frequency inverter power supply unit, which in normal operating mode receives Class 1E direct current (DC) power from its respective Class 1E DC distribution center. Each Class 1E inverter power supply unit also has capability to transfer from its respective Class 1E DC distribution center to an alternate source of alternating current (AC) power to directly supply the Class 1E AC I&C Power System loads. This alternate power source is a voltage regulating device which is supplied power from the same AC power source as the battery charger associated with the Class 1E DC distribution center servicing the inverter power supply unit.

Each Class 1E inverter power supply unit is synchronized, in both frequency and phase, with its alternate power supply and maintains continuity of power during transfer from the inverter to the alternate power supply.

Each Class 1E inverter power supply unit is sized to provide power to its respective distribution center loads.

Class 1E inverter power supply units and their respective distribution centers are identified according to their Class 1E Division/Channel and are located in Seismic Category I structures and in their respective Division/Channel areas.

Independence is provided between Class 1E Divisions. Independence is provided between Class 1E Channels. Independence is provided between Class 1E Divisions/Channels and non-Class 1E equipment.

Class 1E AC I&C Power System distribution panels and their circuit breakers, disconnect switches and fuses are sized to supply their load requirements. Distribution panels and disconnect switches are rated to withstand fault currents for the time required to clear the fault from its power source. Circuit breakers and fuses are rated to interrupt fault currents.

Class 1E AC I&C Power System interrupting devices (circuit breakers and fuses) are coordinated so that the circuit interrupter closest to the fault opens before other devices.

Class 1E AC I&C Power System cables are sized to supply their load requirements and are rated to withstand fault currents for the time required to clear the fault from its power source.

The Class 1E AC I&C Power System supplies an operating voltage at the terminals of the Class 1E equipment which is within the equipment's voltage tolerance limits.

Class 1E AC I&C Power System cables and raceways are identified according to their Class 1E Division/Channel. Class 1E cables are routed in Seismic Category I structures and in their respective Division or Channel raceways.

Class 1E equipment is classified as Seismic Category I.

Class 1E DC Power System

The Class 1E DC Power System consists of two Divisional (Division I and II) and four Channel (A, B, C, D) batteries (2 Channel batteries per Division) with their respective DC electrical distribution panels and battery chargers. The Class 1E DC distribution system provides DC power to Class 1E DC equipment and instrumentation and control circuits.

Each Class 1E battery is sized to supply its Design Basis Accident (DBA) loads, at the end-of-installed-life, for a minimum of 2 hours without recharging.

Each Class 1E battery charger is sized to supply its respective Class 1E Division/Channel steady-state loads while charging its respective Class 1E battery.

Manual interlocked transfer capability exists within a Division between Class 1E DC distribution centers.

The Class 1E batteries, battery chargers and respective MCCs, DC distribution panels, disconnect switches, circuit breakers, and fuses are sized to supply their load requirements. The Class 1E batteries, battery chargers and respective MCCs, DC distribution panels, and disconnect switches are rated to withstand fault currents for the time required to clear the fault from its power source.

Class 1E DC Power System circuit breakers and fuses are rated to interrupt fault currents.

Class 1E DC Power System electrical distribution system circuit interrupting devices (circuit breakers and fuses) are coordinated so that the circuit interrupter closest to the fault is designed to open before other devices.

Class 1E DC Power System electrical distribution system cables are sized to supply their load requirements and are rated to withstand fault currents for the time required to clear the fault from its power source.

The Class 1E DC Power System electrical distribution system supplies an operating voltage at the terminals of the Class 1E equipment which is within the equipment's voltage tolerance limits.

Each Class 1E battery is located in a Seismic Category I structure and in its respective Division/Channel battery room.

Class 1E DC Power System distribution panels and MCCs are identified according to their Class 1E Division/Channel. Class 1E cables are routed in Seismic Category I structures and in their respective Division/Channel raceways.

Independence is provided between Class 1E Divisions. Independence is provided between Class 1E Channels. Independence is provided between Class 1E Divisions/Channels and non-Class 1E equipment.

The Class 1E DC Power System has the following alarms and displays in the main control room (MCR):

1) Alarms for battery ground detection.

2) Parameter displays for battery voltage and amperes.

3) Status indication for battery circuit breaker/disconnect position.

Class 1E equipment is classified as Seismic Category I.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.6.3-1 specifies the inspections, tests, analyses and associated acceptance criteria for the AC Instrumentation and Control Power System and DC Power System.

For the AC Instrumentation and Control Power System and DC Power System, independence is provided between Class 1E Divisions. Independence is provided between Class 1E Channels. Independence is provided between Class 1E Divisions/Channels and non-Class 1E equipment.

The Containment Equipment Hatch Trolley, the Reactor Cavity Flood Valves, the Holdup Volume Flood Valves, and the Hydrogen Ignitors are the only electrical loads classified as non-Class 1E which are directly connectable to buses supplied Class 1E Power.

2.10 TECHNICAL SUPPORT CENTER AND OPERATIONS SUPPORT CENTER

Design Description

The Technical Support Center (TSC) performs a non-safety-related function and is located adjacent to the main control room (MCR) in the nuclear annex. The TSC provides facilities for management and technical support to plant operations during emergency conditions.

The TSC is located less than or equal to two minutes walking time from the MCR.

The TSC has floor space of at least 75 square feet per person for a minimum of 25 persons.

The TSC has radiation detection equipment for monitoring radiation levels within the TSC when the TSC is in use.

The TSC has means for voice communication to the MCR, to on-site emergency support facilities, and to off-site via dedicated or commercial telephone networks.[1]

Displays of the information from the discrete indication and alarm system (DIAS) and the data processing system (DPS) exist in the TSC or can be retrieved there.[2]

The Operations Support Center (OSC) performs a non-safety related function and is located in the nuclear island structures. The OSC provides an assembly area separate from the MCR and TSC where operations support personnel can assemble in an emergency.

The OSC has equipment for voice communication with the MCR and the TSC.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.10-1 specifies the inspections, tests, analysis, and associated acceptance criteria for the Technical Support Center, and Operations Support Center.

[1] Communication Systems are addressed in Section 2.7.25.

[2] Display information from the DIAS and DPS is addressed in Section 2.5.3.

TABLE 2.10-1

TECHNICAL SUPPORT CENTER
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1.a) The TSC is located less than or equal to two minutes walking time from the MCR.	1.a) A test of walking time from the TSC to the MCR will be performed.	1.a) The TSC can be reached in less than or equal to two minutes walking time from the MCR.
1.b) The TSC has floor space of at least 75 square feet per person for a minimum of 25 persons.	1.b) Inspection of the TSC will be performed.	1.b) Floor space of at least 1875 sq. ft. is provided in the TSC.
1.c) The TSC has radiation detection equipment for monitoring radiation levels within the TSC when the TSC is in use.	1.c) An inspection of the radioactivity detection equipment in the TSC will be performed.	1.c) Radiation detection equipment to monitor radiation levels within the TSC is available in the TSC.
1.d) The TSC has means for voice communications to the MCR, to on-site emergency support facilities, and to off-site via dedicated or commercial telephone networks.	1.d) An inspection of the TSC will be performed.	1.d) Communications equipment is installed, and voice transmission and reception are accomplished.
2. Displays of information from the DIAS and the DPS exist in the TSC or can be retrieved there.	2. Inspection for the existence or retrievability in the TSC of the information from the DIAS and the DPS will be performed.	2. Displays of information from the DIAS and the DPS exist in the TSC or can be retrieved there.
3. The OSC is located in the nuclear island structures.	3. Inspection of the location of the OSC will be performed.	3. The OSC is located in the nuclear island structures.
4. The OSC has equipment for voice communication with the MCR and the TSC.	4. Testing of the equipment for voice communication will be performed.	4 Communicati[illegible] equipment is installed and voice transmission and reception are accomplished.

2.4.6 CONTAINMENT SPRAY SYSTEM

Design Description

The Containment Spray System (CSS) is a safety-related system which removes heat and reduces the concentration of radionuclides released from the fuel from the Containment atmosphere and transfers the heat to the component cooling water system following events which increase Containment temperature and pressure. The CSS can also remove heat from the in-containment refueling water storage tank (IRWST).

The CSS is located in the reactor building subsphere and Containment.

The Basic Configuration of the CSS is as shown on Figure 2.4.6-1.

The CSS consists of two Divisions. Each CSS Division has a CSS pump, a CSS heat exchanger, valves, piping, controls and instrumentation.

Each CSS Division has the heat removal capacity to cool and depressurize the containment atmosphere, such that containment design temperature and pressure are not exceeded following a loss of coolant accident (LOCA) or a main steam line break (MSLB).

The CSS limits the maximum flow in each Division.

The CSS pump and the Shutdown Cooling System (SCS) pump in the same Division are connected by piping and valves such that the SCS pump in a Division can perform the pumping function of the CSS pump in that Division. The piping and valves in the cross-connect line between the SCS pump suction and the CSS pump suction permit flow in either direction.

A flow recirculation line around each CSS pump provides a minimum flow recirculation path.

The CSS pumps can be flow tested during plant operation.

The ASME Code Section III Class for the CSS pressure retaining components shown on Figure 2.4.6-1 is as depicted on the Figure.

The safety related equipment shown on Figure 2.4.6-1 is classified Seismic Category I.

CSS pressure retaining components shown on Figure 2.4.6-1, except the shell side of the heat exchangers, have a design pressure outside Containment of at least 900 psig.

Displays of the CSS instrumentation shown on Figure 2.4.6-1 exist in the main control room (MCR) or can be retrieved there. Controls exist in the MCR to start and stop the CSS pumps, and to open and close those remote-operated valves shown on Figure 2.4.6-1. CSS alarms shown on Figure 2.4.6-1 are provided in the MCR.

Water is supplied to each CSS pump at a pressure greater than the pump's required net positive suction head (NPSH).

The Class 1E loads shown on Figure 2.4.6-1 are powered from their respective Class 1E Division. The CSS pump motor and the SCS pump motor in each Division are powered from different Class 1E buses in that same Division.

Independence is provided between Class 1E Divisions and between Class 1E Divisions and non-Class 1E equipment in the CSS.

The two mechanical Divisions of the CSS are physically separated.

The CSS pumps are started upon receipt of a containment spray actuation signal (CSAS), except when the CSAS is aligned to the SCS pump in the same Division. The isolation valves to the CSS spray headers and nozzles are opened upon receipt of a containment spray actuation signal (CSAS).

Motor operated valves (MOVs) having an active safety function will open, or will close, or will open and also close under differential pressure or fluid flow conditions, and under temperature conditions.

Check valves shown on Figure 2.4.6-1 will open, or will close, or will open and also close under system pressure, fluid flow conditions, or temperature conditions.

Inspections, Tests, Analyses and Acceptance Criteria

Table 2.4.6-1 specifies the inspections, tests, analyses and associated acceptance criteria for the Containment Spray System.

An emergency containment spray backup function provides a means of supplying water to the containment spray header from a station AC independent external source.

2.7.5 STATION SERVICE WATER SYSTEM

Design Description

The Station Service Water System (SSWS), in conjunction with the ultimate heat sink (UHS), provides cooling water to remove heat from the component cooling water system (CCWS).

The Basic Configuration of the SSWS is as shown on Figure 2.7.5-1. The SSWS is a safety-related system as noted on the Figure.

The SSWS consists of two Divisions. Each SSWS Division receives heat from its corresponding CCWS Division through the component cooling water heat exchangers.

Each Division of the SSWS has two station service water pumps, two station service water strainers, piping, valves, controls, and instrumentation.

The SSWS pumps and strainers are located in the SSWS pump structure(s). Interconnecting piping runs between the SSWS pump structure(s) and the component cooling water heat exchanger structure.

The SSWS has the capacity to remove heat from the CCWS during operation, shutdown, refueling, and design basis accident conditions. Each Division has the heat dissipation capacity to achieve and maintain cold shutdown.

The ASME Code Section III Class for the SSWS pressure retaining components shown on Figure 2.7.5-1 is as depicted on the Figure.

The safety-related equipment shown on Figure 2.7.5-1 is classified Seismic Category I.

The Class 1E loads shown on Figure 2.7.5-1 are powered from their respective Class 1E Division.

Independence is provided between Class 1E Divisions, and between Class 1E Divisions and non-Class 1E equipment, in the SSWS.

The two mechanical Divisions of the SSWS are physically separated.

Displays of the SSWS instrumentation shown on Figure 2.7.5-1 exist in the main control room (MCR) or can be retrieved there.

Controls exist in the MCR to start and stop the station service water pumps, and to open and close those power operated valves shown on Figure 2.7.5-1.

Motor operated valves (MOVs) having an active safety function will open, or will close, or will open and also close, under differential pressure or fluid flow conditions and under temperature conditions.

Check valves shown on Figure 2.7.5-1 will open, or will close, or will open and also close, under system pressure, fluid flow conditions, or temperature conditions.

Interface Requirements

The Ultimate Heat Sink (UHS) transfers heat from the SSWS to the environment during operation, shutdown, refueling, and design basis accident conditions. The Ultimate Heat Sink is capable of dissipating a heat load of at least 143.0 million BTU/hr during the initial phase of a design basis accident. The UHS is sized so that makeup water is not required for at least 30 days following a design basis accident. During this period of 30 days, the design basis temperatures of safety-related equipment are not exceeded.

Water is supplied to each SSWS pump at a net positive suction head (NPSH) greater than the pump's required NPSH.

The Station Service Water Pump Structure is classified Seismic Category I and provides physical barriers to maintain separation of SSWS mechanical Divisions.

The SSWS pump structure ventilation system is classified Seismic Category I, and its mechanical Divisions are separated by physical barriers.

The SSWS pump structure is designed such that an internal fire or internal flood on one side of the Divisional wall will not affect the function of the other Division.

Inspections, Tests, Analyses, and Acceptance Criteria

Table 2.7.5-1 specifies the inspections, tests, analyses and associated acceptance criteria for the Station Servic Water System.

3.1 PIPING DESIGN

Design Description

The requirements for piping design in this section apply to ASME Class 1, 2 and 3 piping that is classified as Seismic Category I unless otherwise noted.

Piping classified as Seismic Category I is required to withstand the effects of a safe shutdown earthquake (SSE), maintain dimensional stability, and remain functional. Seismic Category I piping, structures, systems and components assure: (1) the integrity of the reactor coolant pressure boundary, and (2) the capability to shut down the reactor and maintain it in a safe shutdown condition, or (3) the capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures.

Seismic Category I piping is designed to meet the requirements of the ASME Code, Section III.

Applicable piping loads due to pressure, gravity, thermal expansion, seismic excitation, wind, tornado, fluid transients, thermal stratification, missiles, and postulated pipe breaks are considered in the piping analyses. Analytical methods and load combinations used for analysis of piping systems will be referenced or specified in the ASME Code certified stress report. Computer programs used for piping system dynamic analysis shall be benchmarked.

The as-built ASME Code Section III piping will be reconciled with the piping design requirements described herein. The as-built reconciliation will be documented in the as-built piping report.

Piping systems are designed to reduce the potential for effects of erosion/corrosion, and to reduce the potential for waterhammer and steam hammer. Piping system supports for Seismic Category I and II piping systems are designed to meet the requirements of the ASME Code Section III, Subsection NF. Pipe loads applied to attached equipment are shown to be less than the equipment allowable loads.

and fabrication processes to ensure that the system i

For those piping systems using ferritic materials as permitted by the design specification, the material~~s~~ will be chosen ~~that are~~ not susceptible to brittle fracture under the expected service conditions. For those piping systems using austenitic stainless steel materials as permitted by the design specification, the material and fabrication process will be selected to reduce the possibility of cracking during service. Chemical, fabrication, handling, welding, and examination requirements that reduce the potential for cracking shall be employed.

SYSTEM 80+™

Piping systems classified as ASME Code Section III Class 1, 2, or 3 are designed to maintain dimensional stability and functional integrity under design loadings expected to be experienced during a 60 year design life.

Design of piping systems provides for clearances between adjacent piping, components, and other structures when the piping moves due to design static, dynamic, and thermal loadings.

The following piping systems are designed to meet leak-before-break (LBB) criteria:

> Reactor coolant system hot leg piping, reactor coolant pump (RCP) suction piping and RCP discharge piping,
>
> Surge line,
>
> Main steam lines inside containment from the steam generator to the anchor at the containment penetrations.
>
> Shutdown cooling lines inside containment from the reactor coolant system to the anchor at the containment penetration, and
>
> Direct vessel injection lines inside containment from the reactor vessel to the safety injection tank and the anchor at the containment penetration.

LBB acceptance criteria are established and LBB evaluations are performed for each piping system designed to meet LBB criteria. For each piping system qualified for LBB, the as-built piping and materials will be reconciled with the bases for the LBB acceptance criteria.

Structures, components, ~~equipment~~ and systems required for safe shutdown are protected from the dynamic effects of postulated pipe breaks in Seismic Category I and non-nuclear safety-related (NNS) piping systems where consideration of these dynamic effects is not eliminated by LBB. Design of features which protect these items consider, as applicable, pipe whip, water spray, jet impingement, flooding, compartment pressurization, and environmental conditions in the area where the piping is located.

Each postulated pipe crack and break shall be documented in a pipe break analysis report.

Structures, systems, and components that are required to be functional during and following an SSE are protected against the effects of spraying, flooding, pressure, and temperature due to postulated pipe breaks and cracks in Seismic Category I and NNS piping systems.

Inspections, Tests, Analyses and Acceptance Criteria

Table 3.1-1 specifies the inspections, tests, analyses, and associated acceptance criteria for the Piping Design.

TABLE 3.1-1

PIPING DESIGN
Inspections, Tests, Analyses, and Acceptance Criteria

and concludes that the as-built piping has been reconciled with the documents used for design.

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
1. The as-built piping is reconciled with the as-designed piping configurations.	1. A reconciliation analysis using the as designed and as-built information will be performed.	1. An as-built piping stress report exists. The as-built piping is reconciled with the piping design requirements described in the piping design description. For ASME Code Class piping, the as-built stress report includes the ASME Code Certified Stress Report and documentation of the results of the as-built reconciliation analysis.
2. Piping systems classified as ASME Code Section III Class 1, 2, or 3 are designed to maintain dimensional stability and functional integrity under design loadings expected to be experienced during a 60-year design life.	2. Inspection for the existence of ASME design reports will be performed.	2. ASME design reports for piping systems classified as ASME Code Section III Class 1, 2, or 3 exist and concludes that the design complies with the requirements of the ASME Code, Section III.
3. For each piping system qualified for LBB, the as-built piping and materials will be reconciled with the bases for the LBB acceptance criteria.	3. For each piping system qualified for LBB, an inspection of the LBB evaluation report will be performed.	3. A LBB evaluation report exists which documents that leak-before-break acceptance criteria are met by the as-built piping and piping materials.

TABLE 3.1-1 (Continued)

PIPING DESIGN
Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment (and environmental)	Inspections, Tests, Analyses	Acceptance Criteria (Seismic Category I)
4. Structures, components, and ~~[illegible]~~ systems required for safe shutdown are protected from the dynamic effects of postulated pipe breaks (and cracks) in Seismic Category I and non-nuclear safety-related (NNS) piping systems where consideration of these dynamic effects is not eliminated by LBB. Each postulated pipe crack and break shall be documented in a pipe break analysis report.	4. For piping systems with postulated pipe breaks, an inspection of the pipe break analysis report will be performed. An inspection of the as-built high (and moderate) energy pipe break mitigation features will be performed.	4. A pipe break analysis report exists and concludes that structures, systems, and components ~~classified as ASME Code Section III Class 1, 2, or 3~~ remain functional after postulated pipe breaks. The pipe break analysis report includes the results of inspections of high- and moderate-energy pipe break mitigation features (including spatial separation).

ATTACHMENT 4

14.2.12.1.40 Containment Spray System (CSS) Test

1.0 OBJECTIVE

1.1 To verify the proper operation of the Containment Spray System and the containment spray pumps.

2.0 PREREQUISITES

2.1 Construction activities on the systems to be tested are complete.

2.2 Plant systems required to support testing are operable and temporary systems are installed and operable.

2.3 Permanently installed instrumentation is operable and calibrated.

2.4 Test instrumentation is available and calibrated.

2.5 The emergency containment spray backup pumping device is operable.

3.0 TEST METHOD

3.1 Verify proper operation of each containment spray pump with minimum flow established.

3.2 Verify pump performance including head and flow characteristics for all design flow paths.

3.3 Verify, if applicable, proper operation, stroking speed, and position indication of control valves.

3.4 Verify by using service air that the Containment Spray header and nozzles are free of obstructions.

3.5 Verify the automatic operation of all components in response to a Containment Spray Actuation Signal.

3.6 Verify the interchangeability of the Shutdown Cooling pumps with the CSS pumps.

3.7 Verify adequate heat removal capability by the CSS heat exchangers.

3.8 Verify power-operated valves fail to the position specified in Section 6.5.2 and 6.3.2 upon loss of motive power.

3.9 Verify emergency containment spray backup pumping device connectability to the containment spray tee connection. Verify pumpin device performance including head and flow characteristic

4.0 DATA REQUIRED

4.1 Valve position indications.

4.2 Pump head versus flow characteristics.

4.3 Valve opening and closing time, where required.

4.4 Setpoints at which interlocks and alarms occur.

4.5 Position response of valves to loss of motive power.

5.0 ACCEPTANCE CRITERIA

5.1 The Containment Spray System and Containment Spray Pumps perform as described in Section 6.5.

5.2 The emergency containment spray backup pumping device performs as described in Section ~~[illegible]~~ 6.5.2.

ATTACHMENT 5

Report No.	Title	Date Issued	CESSAR-DC Chapter
~~CENPD-169~~	~~Combustion Engineering, Inc. "Assessment of the Accuracy of PWR Operating Limits as Determined by the Core Operating Limit Supervisory System"~~	~~July 1975~~	~~7~~
CENPD-183-A	Combustion Engineering, Inc. "C-E Methods for Loss of Flow Analysis"	June 1984	15
CENPD-188-A	HERMITE, A Multi-Dimensional Space-Time Kinetics Code for PWR Transients	July 1976	4,15
CENPD-190-A	Combustion Engineering, Inc. "C-E Method for Control Element Assembly Ejection Analysis"	January 1976	15
CEN-203-NP	Response to NRC Action Item II.K.3(30), Justification of Small-Break LOCA Methods	April 1982	Appendix A
CENPD-206-NP-A	Combustion Engineering, Inc. "TORC Code Verification and Simplified Modeling Method"	June 1981	4,15
CENPD-207-NP-A	Combustion Engineering, Inc. "Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids, Part 2, Non Uniform Axial Power Distribution"	December 1984	4
CEN-214(A)-NP	CETOP-D Code Structure and Modeling Methods for Arkansas Nuclear One, Unit 2	July 1982	4
CEN-227	Summary Report on the Operability of the Pressurizer Safety Relief Valves in CE Designed Plants	December 1982	Appendix A
CENPD-254-NP-A	"Post LOCA Long Term Cooling Evaluation Model"	June 1980	6
CENPD-255-A, Rev. 3	"Qualification of Combustion Engineering Class 1E Instrumentation"	October 1985	3
CEN-356(V)-P-A Rev. 1-P-A	"Modified Statistical Combination of Uncertainties"	May 1988	4,7

Report No.	Title	Date Issued	CESSAR-DC Chapter
2NPX80-IC-VP790-03, Rev 01	Human Factors Engineering Verification and Validation Plan for Nuplex 80+	~~September 1992~~ February 1994	18
NPX80-IC-DP790-01, Rev 03	Human Factors Program Plan for the System 80+ (TM) Standard Plant Design ~~LD-92-120~~	~~December 1992~~ February 1994	18
NPX-IC-DR-791-02 Rev 01	Human Factors Engineering Standards, Guidelines, and Bases for System 80+, ~~LD-92-069~~	~~May 1992~~ February 1994	18
LD-93-120	System 80+ Fire Hazards Assessment	R0 March 13, 1992 R1 April 5, 1993	9
NPX80-IC-RR790-02, Rev. ~~1~~ 02	Human Factors Evaluation and Allocation of System 80+ Functions~~, LD-93-056~~	March 1993 February 1994	18
NPX80-IC-DB-790-01, Rev. 01	Nuplex 80+ Advanced Control Complex Design Bases, ~~LD-92-102~~	~~September 1992~~ February 1994	18
NPX-TE-790-01	Nuplex 80+ Verification Analysis Report, LD-92-065	May 1992	18
NPX80-IC-DP790-02, Rev. 01	System 80+ Function & Task Analysis Report, LD-92-065	May 1989	18
NPX80-IC-DP790-03, Rev. 01	Functional Task Analysis Plan, LD-93-172	November 1993	18
NPX80-SQP-0101.0, Rev 02	Software Program Manual for NUPLEX 80+~~, LD-93-009~~	~~January 1993~~ February 1994	7
NPX80-IC-QP790-2	Nuplex 80+ Software Safety Plan Description, LD-93-009	January 1993	7
NPX-IC-QG790-00, Rev 01	Qualification Guidelines for Instrumentation and Controls Equipment for NUPLEX 80+, ~~LD-92-113~~	~~November 1992~~ February 1994	7
NPX80-QPS-0401.1	Requirements for The Supply of Commercial Digital Hardware and Software Components to be used in NUPLEX 80+ Safety Systems, LD-92-114	May 1992	7