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B.2 DESIGN PARAMETERS AND PLANT COMPARISONS

The design parameters of the Point Beach Nuclear Plant were initially provided in a comparison with H. B. Robinson, Indian Point 2, and Ginna Station. This comparison provided an informational reference of similar aspects of similar plants constructed during the same time period to demonstrate that the technology employed at Point Beach was proven in multiple applications.

The design parameters listed were considered valid at the time of license issuance, and have been retained for historical context in the following Table B.2-1 and paragraphs. Note: The information provided is not currently reflective of the PBNP design specifications, and therefore should not be used as the basis of any Safety Evaluation without prior verification against current information provided elsewhere in the FSAR.

In 2003 a measurement uncertainty recapture power uprate was performed increasing the rated thermal power level to 1540 MWt. The tables of this section have not been updated since this Appendix is historical.

Design Highlights

The design of each Point Beach unit is based upon proven concepts which have been developed and successfully applied in the construction of pressurized water reactor systems. In subsequent paragraphs, a few of the design features are listed which represent slight variation or extrapolations from units presently operating such as San Onofre and Connecticut-Yankee.

POWER LEVEL - The license application power level of 1518.5 MWt is smaller than the capability of the Prairie Island plant and larger than the capability of the Ginna plant. This level is a reasonable increase over power levels of pressurized water reactors now operating.

REACTOR COOLANT LOOPS - The Reactor Coolant System for each Point Beach unit consists of two loops, the same as the Prairie Island and Ginna Units.

PEAK SPECIFIC POWER - Based on the design hot channel factors, operation at a primary heat output of 1518.5 MWt corresponds to a peak specific power of 16.0 kw/ft. This design rating is slightly lower than that licensed in Ginna (16.5 kw/ft) as well as that of Prairie Island (17.4 kw/ft). The maximum overpower condition is 17.9 kw/ft (112%) compared to 19.6 kw/ft (118%) for Prairie Island and 18.5 kw/ft for Ginna.

FUEL ASSEMBLY DESIGN - The fuel assembly design incorporates the rod cluster control concept in a canless assembly utilizing a spring clip grid to provide support for the 14 x 14 array of fuel rods. This concept incorporates the advantages of the Yankee canless fuel assembly and the Saxton spring clip grid with the rod cluster control scheme. Extensive out-of-pile tests have been performed on this concept and operating experience is available from the San Onofre and Connecticut-Yankee plants.

ENGINEERED SAFETY FEATURES - The engineered safety features provided are similar to those provided for the Connecticut-Yankee plant, augmented by borated water injection accumulators. There is a safety injection system of the Connecticut-Yankee type which can be operated from emergency on-site diesel power. The system design is such that it can be tested while the plant is at power. There is air recirculation cooling for post-loss-of-coolant conditions which utilizes the normal ventilation fans. A containment spray system provides cool, chemically-treated, borated water spray into the containment atmosphere for additional cooling capacity, and provides a means of rapidly reducing the concentration of airborne halogen fission products in the containment atmosphere.

EMERGENCY POWER - In addition to the multiple ties to outside sources for emergency power, four diesel generator units are provided as backup power supplies for the case of loss of all outside power. Each generator is capable of operating sufficient safety injection and containment cooling equipment to ensure an acceptable post-loss-of-coolant pressure transient in the affected unit, and safe shutdown of the other unit.

NET LOAD REJECTION - Each of the Point Beach units is designed to accept loss of 50% of external load without a reactor or turbine trip. This is accomplished by an automatic control system which dumps steam to the condenser and atmosphere as a short term supplemental load to provide time for the reactor control system to reduce the reactor output without exceeding acceptable core and coolant conditions. No unique or unproven features are required in the reactor control system to accomplish this.

Table B.2-1 COMPARISON OF DESIGN PARAMETERS
(See General Note)

Thermal and Hydraulic Parameters	PBNP U1/U2 Final Report	Robinson 2 Final Report	Indian Point 2 Final Report	R.E. Ginna Final Report	Reference Line No.
Total Primary Heat Output, MWt	1518.5	2200	2758	1300	1
Total Core Heat Output, Btu/hr	$5181x10^6$	$7479x10^6$	$9413x10^6$	$4437x10^6$	2
Heat Generated in Fuel,%	97.4	97.4	97.4	97.4	3
Maximum Thermal Overpower	12%	12%	12%	12%	4
System Pressure, Nominal, psia	2250	2250	2250	2250	5
System Pressure, Minimum Steady State, psia	2220	2220	2220	2220	6
Hot Channel Factors					
Heat Flux, F_q	2.32	3.23	3.23	3.38	7
Enthalpy Rise, $F_{\Delta H}$	1.60	1.77	1.77	1.77	8
DNB Ratio at Nominal Conditions	2.11	1.81	2.00	2.15	9
Minimum DNBR for Design Transients	1.30	1.30	1.30	1.30	10
Coolant Flow					
Total Flow Rate, lb/hr	$66.7x10^6$	$101.5x10^6$	$136.3x10^6$	$67.3x10^6$	11
Effective Flow Rate for Heat Transfer, lb/hr	$63.6x10^6$	$97.0x10^6$	$130x10^6$	$64.3x10^6$	12
Effective Flow Area for Heat Transfer, ft^2	27.0	41.8	51.4	27.0	13
Average Velocity Along Fuel Rods, ft/sec	15.0	14.3	15.4	14.7	14

Table B.2-1 COMPARISON OF DESIGN PARAMETERS
(See General Note)

Thermal and Hydraulic Parameters	PBNP U1/U2 Final Report	Robinson 2 Final Report	Indian Point 2 Final Report	R.E. Ginna Final Report	Reference Line No.
Average Mass Velocity, lb/hr-ft^2	2.37×10^6	2.32×10^6	2.53×10^6	2.38×10^6	15
Coolant Temperatures, °F					
Nominal Inlet, °F	552.5	546.2	543	551.9	16
Maximum Inlet Due to Instrumentation					
Error and Deadband, °F	556.5	550.2	547	555.9	17
Average Rise in Vessel, °F	57.6	55.9	53.0	49.5	18
Average Rise in Core, °F	60.0	58.3	55.5	52	19
Average in Core, °F	582.5	575.4	571.0	578.0	20
Average in Vessel, °F	581.3	574.2	569.5	577.0	21
Nominal Outlet of Hot Channel, °F	642.9	642	633.5	634.0	22
Average Film Coefficient, Btu/hr-ft^2-F	5600	5400	5790	5590	23
Average Film Temperature Difference, °F	31.0	31.8	30.3	26.9	24
Heat Transfer at 100% Power					
Active Heat Transfer Surface Area, ft^2	28,715	42,460	52,200	28,715	25
Average Heat Flux, Btu/hr-ft^2	175,800	171,600	175,600	150,500	26
Maximum Heat Flux, Btu/hr-ft^2	491,000	554,200	567,300	508,700	27
Average Thermal Output, kw/ft	5.7	5.5	5.7	4.88	28
Maximum Thermal Output, kw/ft	16.0	17.9	18.4	16.52	29

Table B.2-1 COMPARISON OF DESIGN PARAMETERS

(See General Note)

Thermal and Hydraulic Parameters	PBNP U1/U2 Final Report	Robinson 2 Final Report	Indian Point 2 Final Report	R.E. Ginna Final Report	Reference Line No.
Maximum Clad Surface Temperature at					
Nominal Pressure, °F	657	657	657	657	30
Fuel Central Temperature, °F					
Maximum at 100% Power	3750	4030	4090	3880	31
Maximum at Overpower	4000	4300	4380	4100	32
Thermal Output, kw/ft at Maximum Overpower	17.9	20.0	20.6	18.5	33
CORE MECHANICAL DESIGN PARAMETERS					
Fuel Assemblies					
Design	RCC Canless 14x14	RCC Canless 15x15	RCC Canless 15x15	RCC Can-less 14x14	34
Rod Pitch, in.	0.556	0.563	0.563	0.556	35
Overall Dimensions, in.	7.763x7.763	8.426x8.426	8.426x8.426	7.763x7.763	36
Fuel Assemblies					
Fuel Weight (as UO_2), pounds	118,729	176,200	216,000	120,782	37
Total Weight, pounds	154,519	226,200	276,000	152,895	38
Number of Grids per Assembly	7	7	9	9	39

Table B.2-1 COMPARISON OF DESIGN PARAMETERS
(See General Note)

Thermal and Hydraulic Parameters	PBNP U1/U2 Final Report	Robinson 2 Final Report	Indian Point 2 Final Report	R.E. Ginna Final Report	Reference Line No.
Fuel Rods					
Number	21,659	32,028	39,372	21,659	40
Outside Diameter, in.	0.422	0.422	0.422	0.422	41
Diametral Gap, in.	0.0065	0.0065	0.0065	0.0065	42
Clad Thickness, in.	0.0243	0.0243	0.0243	0.0243	43
Clad Material	Zircaloy	Zircaloy	Zircaloy	Zircaloy	44
Fuel Pellets					
Material	UO_2 Sintered	UO_2 Sintered	UO_2 Sintered	UO_2 Sintered	45
Density (% of Theoretical)	Unit 1 94-92-91 Unit 2 94-93-92	94-92-91	94-92-91	94-92-91-93	46
Diameter, in.	0.3669	0.3669	0.3669	0.3669	47
Length, in.	0.6000	0.6000	0.6000	0.6000	48
Rod Cluster Control Assemblies					
Neutron Absorber	5% Cd-15% In-80% Ag.	5% Cd-15% In-80% Ag.	5% Cd-15% In-80% Ag.	5% Cd-15% In-80% Ag.	49

Table B.2-1 COMPARISON OF DESIGN PARAMETERS
(See General Note)

Thermal and Hydraulic Parameters	PBNP U1/U2 Final Report	Robinson 2 Final Report	Indian Point 2 Final Report	R.E. Ginna Final Report	Reference Line No.
Cladding Material	Type 304 SS-Cold Wrkd.	Type 304 SS-Cold Wrkd.	Type 304 SS-Cold Wrkd.	Type 304 SS-Cold Wrkd.	50
Rod Cluster Control Assemblies					
Clad Thickness, in.	0.019	0.019	0.019	0.019	51
Number of Clusters	33	53	53	29	52
Number of Control Rods per Cluster	16	20	20	16	53
Core Structure					
Core Barrel I.D./O.D., in.	109.0/112.5	133.875/ 137.875	148.0/152.5	109.0/112.5	54
Thermal Shield I.D./O.D., in.	115.3/122.5		158.5/164.0	115.3/122.5	55
Structural Characteristics					
Fuel Weight (as UO2), lbs.	118,729	176,200	216,000	120,130	56
Clad Weight, lbs.	24,260	36,300	44,600	22,440	57
Core Diameter, in. (Equivalent)	96.5	119.5	132.5	96.5	58
Core Height, in. (Active Fuel)	144	144	144	144	59

Table B.2-1 COMPARISON OF DESIGN PARAMETERS
(See General Note)

Thermal and Hydraulic Parameters	PBNP U1/U2 Final Report	Robinson 2 Final Report	Indian Point 2 Final Report	R.E. Ginna Final Report	Reference Line No.
Reflector Thickness and Composition					
Top - Water plus Steel, in.	10	10	10	10	60
Bottom - Water plus Steel, in.	10	10	10	10	61
Side - Water plus Steel, in.	15	15	15	15	62
H_20/U, (Cold Volume Ratio)	4.20	4.18	4.18	4.08	63
Number of Fuel Assemblies	121	157	193	121	64
UO2 Rods per Assembly	179	204	204	179	65
<u>Performance Characteristics</u>					
Loading Technique	3 region, non-uniform	3 region, non-uniform	3 region, non-uniform	3 region, non-uniform	66
Fuel Discharge Burnup, MWD/MTU					
Average First Cycle	15,100	14,500	14,200	~14,900	67
Equilibrium Region Average	33,000	33,000	24,700	~24,400	68
Feed Enrichments, w/o					
Region 1	2.27	1.85	2.2	2.44	69
Region 2	3.03	2.55	2.7	2.78	70
Region 3	3.40	3.10	3.2	3.48	71
Equilibrium	3.40	3.10			

Table B.2-1 COMPARISON OF DESIGN PARAMETERS
(See General Note)

Thermal and Hydraulic Parameters	PBNP U1/U2 Final Report	Robinson 2 Final Report	Indian Point 2 Final Report	R.E. Ginna Final Report	Reference Line No.
<u>Control Characteristics</u>					
Effective Multiplication (Beginning of Life)					
Cold, No Power, Clean	1.211	1.180	1.257	1.188	72
Hot, No Power, Clean	1.167	1.38	1.999	1.137	73
Hot, Fuel Power, Xe and Sm Equilibrium	1.113	1.077	1.152	1.080	74
Rod Cluster Control Assemblies					
Material	5% Cd-15% In-80% Ag.	5% Cd-15% In-80% Ag.	5% Cd-15% In-80% Ag.	5% Cd-15% In-80% Ag.	75
Number of RCC Assemblies	37	53	53	33	76
Number of Absorbers per RCC Assembly	16	20	20	16	77
Total Rod Worth	See Table 3.2.1-3	See Table 3.2.1-3	See Table 3.2.1-3	6.8%	78
Boron Concentrations					
To shut reactor down with no rods inserted,					
clean (k_{eff} = .99) Cold/hot	1598 ppm/ 1676 ppm	1250 ppm/ 1210 ppm	1480 ppm/ 1370 ppm	1160 ppm/ 820 ppm	79
To control at power with no rods inserted,					
clean/equilibrium xenon and samarium	1465 ppm/ 1007 ppm	1000 ppm/920 ppm	1200 ppm/ 780 ppm	1310 ppm/ 890 ppm	80

Table B.2-1 COMPARISON OF DESIGN PARAMETERS
(See General Note)

Thermal and Hydraulic Parameters	PBNP U1/U2 Final Report	Robinson 2 Final Report	Indian Point 2 Final Report	R.E. Ginna Final Report	Reference Line No.
Boron Worth, Hot	1% δk/k/130 ppm	7.3 δk/k	1% δk/k/89 ppm	1% δk/k/120 ppm	81
Boron Worth, Cold	1% δk/k/98 ppm	5.6 δk/k	1% δk/k/72 ppm	1% δk/k/90 ppm	82
<u>Kinetic Characteristics</u>					
Moderator Temperature Coefficient (δk/k/^{o}F)	$+0.3x10^{-4}$ to $-3.5x10^{-4}$	$+0.3x10^{-4}$ to $-3.5x10^{-4}$	$-0.3x10^{-4}$ to $-3.0x10^{-4}$	$+0.5x10^{-4}$ to $-3.5x10^{-4}$	83
Moderator Pressure Coefficient (δk/k/psi)	$-0.3x10^{-6}$ to $3.5x10^{-6}$	$-0.3x10^{-6}$ to $3.5x10^{-6}$	$+0.3x10^{-6}$ to $+3.0x10^{-6}$	$-0.5x10^{-6}$ to $3.5x10^{-6}$	84
Moderator Void Coefficient	-0.10 to -0.30 δk/k/g/cm^3	$+0.5x10^{-3}$ to $-2.5x10^{-3}$ δk/k/% void	+0.03 to -0.30 δk/k/g/cm^{-3}	-0.10 to -0.30 δk/k/g/cm^3	85
Doppler Coefficient (δk/k/^{o}F)	$-1x10^{-5}$ to $-1.6x10^{-5}$	$-1x10^{-5}$ to $-1.6x10^{-5}$	$-1.1x10^{-5}$ to $+1.8x10^{-5}$	$-1.1x10^{-5}$ to $1.8x10^{-5}$	86

Table B.2-1 COMPARISON OF DESIGN PARAMETERS
(See General Note)

Thermal and Hydraulic Parameters	PBNP U1/U2 Final Report	Robinson 2 Final Report	Indian Point 2 Final Report	R.E. Ginna Final Report	Reference Line No.
<u>REACTOR COOLANT SYSTEM - CODE REQUIREMENTS</u>					
Reactor Vessel	ASME III Class A	ASME III Class A	ASME III Class A	ASME III Class A	87
Steam Generator					
Tube Side	ASME III Class A	ASME III Class A	ASME III Class A	ASME III Class A	88
Shell Side	ASME III Class C*	ASME III Class C*	ASME III Class C*	ASME III Class C*	89
Pressurizer	ASME III Class A	ASME III Class A	ASME III Class A	ASME III Class A	90
Pressurizer Relief Tank	ASME III Class C	ASME III Class C	ASME III Class C	ASME III Class C	91
Pressurizer Safety Valves	ASME III	ASME III	ASME III	ASME III	92
Reactor Coolant Piping	USAS B31.1	USAS B31.1	USAS B31.1	USAS B31.1	93

*The shell side of the steam generator conforms to the requirements for Class A vessels and is so stamped as permitted under the rules of Section III.

<u>PRINCIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT SYSTEM</u>					
Reactor Primary Heat Output, MWt	1518.5	2200	2758	1300	94

Table B.2-1 COMPARISON OF DESIGN PARAMETERS
(See General Note)

Thermal and Hydraulic Parameters	PBNP U1/U2 Final Report	Robinson 2 Final Report	Indian Point 2 Final Report	R.E. Ginna Final Report	Reference Line No.
Reactor Primary Heat Output, Btu/hr	5181×10^6	7508×10^6	9413×10^6	4437×10^6	95
Operating Pressure, psig	2235	2235	2235	2235	96
Reactor Inlet Temperature	552.5	546.2	543	551.9	97
Reactor Outlet Temperature	610.1	602.1	596.0	601.4	98
Number of Loops	2	3	4	2	99
Design Pressure, psig	2485	2485	2485	2485	100
Design Temperature, ^oF	650	650	650	650	101
Hydrostatic Test Pressure (Cold), psig	3110	3110	3110	3110	102
Coolant Volume, including pressurizer, cu. ft.	6450	9088	12,600	6245	103
Total Reactor Flow, gpm	178,000	268,500	358,800	180,000	104
Material	SA-302 Grade B, low alloy steel, internally clad with austenitic SS	SA-302 Grade B, low alloy steel, internally clad with austenitic SS	SA-302 Grade B, low alloy steel, internally clad with austenitic SS	SA-302 Grade B, low alloy steel, internally clad with austenitic SS	105
Design Pressure, psig	2485	2485	2485	2485	106
Design Temperature, ^oF	650	650	650	650	107
Operating Pressure, psig	2235	2235	2235	2235	108

Table B.2-1 COMPARISON OF DESIGN PARAMETERS
(See General Note)

Thermal and Hydraulic Parameters	PBNP U1/U2 Final Report	Robinson 2 Final Report	Indian Point 2 Final Report	R.E. Ginna Final Report	Reference Line No.
Inside Diameter of Shell, in.	132	155.5	173	132	109
Outside Diameter Across Nozzles, in.	224-1/16	236	262-7/16	219-5/16	110
Overall Height of Vessel & Enclosure Head, ft-in.	39-0	41-6	43' 9-11/16"	39' 1-5/16"	111
Minimum Clad Thickness, in.	5/32	5/32	5/32	5/32	112
PRINCIPAL DESIGN PARAMETERS OF THE STEAM GENERATORS					
Number of Units	2	3	4	2	113
Type	Vertical U-tube with interal-moisture separator	Vertical U-tube with integral-moisture separator	Vertical U-tube with integral-moisture separator	Vertical U-tube with integral-moisture separator	114
Tube Material	Inconel	Inconel	Inconel	Inconel	115
Shell Material	Carbon Steel	Carbon Steel	Carbon Steel	Carbon Steel	116
Tube Side Design Pressure, psig	2485	2485	2485	2485	117
Tube Side Design Temperature, °F	650	650	650	650	118
Tube Side Design Flow, lb/hr	33.35×10^6	33.93×10^6	34.07×10^6	33.63×10^6	119
Shell Side Design Pressure, psig	1085	1085	1085	1085	120
Shell Side Design Temperature, °F	556	556	556	556	121

Table B.2-1 COMPARISON OF DESIGN PARAMETERS
(See General Note)

Thermal and Hydraulic Parameters	PBNP U1/U2 Final Report	Robinson 2 Final Report	Indian Point 2 Final Report	R.E. Ginna Final Report	Reference Line No.
Operating Pressure, Tube Side, Nominal, psig	2235	2235	2235	2235	122
Operating Pressure, Shell Side, Maximum, psi	1020	1020	1015.3	1020	123
Maximum Moisture at Outlet at Full Load, %	1/4	1/4	1/4	1/4	124
Hydrostatic Test Pressure, Tube Side (Cold), psig	3110	3110	3110	3110	125
PRINCIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT PUMPS					
Number of Units	2	3	4	2	126
Type	Vertical, single stage radial flow with bottom suction & horiz. disch.	Vertical, single stage radial flow with bottom suction & horiz. disch.	Vertical, single stage radial flow with bottom suction & horiz. disch.	Vertical, single stage radial flow with bottom suction & horiz. disch.	127
Design Pressure, psig	2485	2485	2485	2485	128
Design Temperature, ^{o}F	650	650	650	650	129
Operating Pressure, Nominal, psig	2235	2235	2235	2235	130
Suction Temperature, ^{o}F	551.5	546.5	556	551.9	131
Design Capacity, gpm	89,000	88,500	90,000	90,000	132
Design Head, ft.	259	261	252	252	133
Hydrostatic Test Pressure (Cold), psig	3110	3110	3110	3110	134

Table B.2-1 COMPARISON OF DESIGN PARAMETERS
(See General Note)

Thermal and Hydraulic Parameters	PBNP U1/U2 Final Report	Robinson 2 Final Report	Indian Point 2 Final Report	R.E. Ginna Final Report	Reference Line No.
Motor Type	AC induc. single speed air cooled	AC induc. single speed air cooled	AC induc. single speed air cooled	AC induc. single speed air cooled	135
Motor Rating (Nameplate)	6000 HP	6000 HP	6000 HP	5500 HP	136
Material	Austenitic SS	Austenitic SS	Austenitic SS	Austenitic SS	137
Hot Leg - I.D., in.	29	29	29	29	138
Cold Leg - I.D., in.	27-1/2	27-1/2	27-1/2	27-1/2	139
Between Pump and Steam Generator - I.D., in.	31	31	31	31	140
Design Pressure, psig	2485	2485	2485	2485	141

B.3 INITIAL PLANT DESIGN

Research and development (as defined in Section 50.2 of the Commission's regulations) was conducted regarding core design details and parameters, analytical methods for kinetics calculations, thermal shock and its effects on reactor vessel integrity, the safety injection (emergency core cooling) system, xenon stability and related control systems, containment spray additive effectiveness, and capability of reactor internals to resist blowdown forces.

Core Design

The nuclear design, including fuel configuration and enrichments, control rod pattern and worths, reactivity coefficients, and boron requirements are presented in Section 3.2 and the thermal-hydraulics design parameters are also in Section 3.2. Section 3.2 presents the fuel, fuel rod, fuel assembly, and control rod mechanical design. The core design incorporates fixed burnable poison rods (Reference 1) in the initial loading and, when necessary, in subsequent core reloads to ensure a negative moderator reactivity temperature coefficient at operating temperature. This improves reactor stability and lessens the consequences of a rod ejection or loss-of-coolant accident. The mechanical design is presented in Section 3.2.

Development Of Analytical Methods For Reactivity Transients From Rod Ejection Accidents

A control rod ejection accident is not considered credible since it would require the fracture of a control rod mechanism housing. Nevertheless, the reactivity and associated pressure and temperature transients for this accident have been analyzed. Rod ejection analyses for this plant were performed using the CHIC-KIN code(Reference 2), which uses a point reactor kinetics model and a single channel fuel and coolant description. The rod ejection analysis results are given in Section 14.2.6 of this report, together with a brief description of the CHIC-KIN code. These analyses show that the temperature and pressure transients associated with a rod ejection accident do not cause any consequential damage to the reactor coolant system. The consequences of a rod ejection accident are now lessened because the moderator coefficient of reactivity is always negative at operating conditions. In addition, the effects of rod ejection are inherently limited in this reactor, in which boric acid chemical shim is employed, since full-length control rods need only to be inserted sufficiently to handle load changes.

The initial cores contain fixed burnable poison rods. These, by allowing a reduction in the chemical shim concentration, ensure that the moderator coefficient of reactivity is always negative at operating temperature. The burnable poison rods, contain borosilicate glass. Critical experiments were conducted at the Westinghouse Reactor Evaluation Center using rods containing 12.8 w/o boron and Zircaloy clad UO_2 fuel rods, 2.27% enriched. These values are also typical of this plant's initial core. The experiments showed that standard analytical methods can be used to calculate the reactivity worth of the burnable poison rods. The design basis and critical experiments are described in Reference 1. In-core testing completed in the Saxton reactor showed satisfactory performance of these rods.

Safety Injection System (SI) Design

The design of the safety injection system includes nitrogen-pressurized accumulators to inject borated water into the reactor coolant system to rapidly and reliably reflood the core following a loss-of-coolant accident. Additional analyses have been performed to demonstrate that the accumulators, in conjunction with other components of the emergency core cooling system, can adequately cool the core for any pipe rupture. These analyses are presented in Section 14.3. The computer code, FLASH-R, used for the blowdown phase of the loss-of-coolant accident was modified to take into account the accumulator injection.

Research and development work has also been performed on the integrity of Zircaloy-clad fuel under conditions simulating those during a loss-of-coolant accident. Under the conservatively elevated temperatures predicted for the fuel rods during loss-of-coolant accident, the clad may burst due to a combination of fuel rod internal gas pressure and the reduction of clad strength with temperature. Burst cladding could block flow channels in the core, so that core cooling by the safety injection system would be insufficient to prevent fuel rod melting. Rod burst experiments have, therefore, been conducted on Zircaloy rods. The results of single-rod tests have been presented to the AEC in WCAP-7379-L Volume I (Westinghouse Proprietary) and Volume II. The results of multi-rod tests have been reported to the AEC in WCAP-7495-L.

Systems For Reactor Control During Xenon Instabilities

Extensive analytical work has been performed on reactor core stability(Reference 3, Reference 4, and Reference 5). These indicated that a core of this size may be unstable against axial power redistribution, but is nominally stable against transverse (denoted X-Y) power oscillations. The plant was, therefore, provided with instrumentation and control equipment which would allow the operator to detect and suppress the axial power oscillations.

The original plant design provided for part-length control rods to control axial power oscillations which could result from the potential of power spatial redistribution caused by instabilities in local xenon concentration. Initial plant operations established that part-length control rods were not necessary for control of axial power oscillation. The part-length control rods at Point Beach Nuclear Plant Units 1 and 2 were subsequently removed.

Control information for axial power oscillation suppression is obtained from four long ion chambers, each divided into an upper and lower section mounted vertically outside the core. Both calculation and experimental measurements at SENA, San Onofre, and Haddam Neck have shown that this out-of-core instrumentation represents in-core power distribution adequately for power distribution control(Reference 5).

The control strategy is based on the difference in output between the top and bottom sections of the long ion chambers. If the operator allows axial power imbalance to exceed operating limits, various levels of protection are invoked automatically. These include generation of alarms, turbine power cutback, blocking of control rod withdrawal, and reactor trip. This capability is described in Section 7.0.

Containment Spray Additive For Iodine Removal

Initially, sodium thiosulphate, $Na_2S_2O_3$, was proposed as the iodine removal additive to the boric acid containment spray, but an evaluation program led to the selection of sodium hydroxide, NaOH. The results of the evaluation program are detailed in Reference 6 and are summarized briefly below:

1. Chemical Characteristics

   The $Na_2S_2O_3$ solution was found to be oxidized by air at the post-accident temperatures in containment. NaOH was not unstable in this way.

2. Iodine Removal Characteristics

   The removal efficiency of the NaOH solution (at pH not less than 9.5) was comparable to that of the $Na_2S_2O_3$ solution.

3. Materials Compatibility

   Corrosion rates of copper and copper-alloy heat exchanger tubing were reduced by more than an order of magnitude compared with high pH $Na_2S_2O_3$ solution and were acceptably low (<0.01 mils/month at 100^0F) for the application. These tests showed that pitting or local corrosion did not occur.

4. Radiolysis

   The NaOH solution was radiolytically stable, and liberates significantly less net hydrogen than the unstable $Na_2S_2O_3$ solution.

Therefore, further testing has centered on the use of NaOH as the spray additive leading to the development of a technical basis for its inclusion in the plant engineered safety features as a means of "fixing" absorbed iodine, enhancing the natural rate of deposition of I_2, and thus lowering the calculated off-site thyroid dose resulting from a postulated release of fission products to the containment atmosphere.

Section 6 gives a further discussion of iodine removal by the containment spray system.

Blowdown Capability Of Reactor Internals

The forces exerted on reactor internals and the core following a loss-of-coolant accident are computed by employing the BLODWN-2 digital computer program developed for the space-time-dependent analysis of multi-loop PWR plants. This program and the models used are discussed in Section 14.3.3.

Reactor Vessel Thermal Shock

Research was performed prior to and following the issuance of the Point Beach Operating Licenses to determine the effect of the addition of cold water from the accumulators to the reactor pressure vessel. This research considered three failure modes: the ductile failure mode, the fatigue yielding mode and the brittle failure mode. Analysis of the ductile and fatigue modes determined that reactor vessel integrity is maintained following addition of the accumulator water. Extensive analysis of the brittle failure mode demonstrated adequate reactor vessel fracture toughness to prevent brittle failure for a period of several years of plant operation.

Subsequently, but before the end of the analyzed period, the NRC issued 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events." 10 CFR 50.61 contains "screening criteria" for material fracture toughness, such that, if the materials of construction of the reactor vessel for a nuclear power plant maintain fracture toughness in compliance with the screening criteria, the functional integrity of the reactor vessel is ensured. It has been demonstrated that Point Beach Units 1 and 2 have adequate fracture toughness to be in compliance with the screening criteria of 10 CFR 50.61 through the end of their Operating Licenses. Therefore, brittle failure of the Point Beach reactor vessels is not a credible failure mode.

Identification Of Contractors

The Licensee engaged or approved the engagement of the contractors identified below in connection with the design and construction of the Point Beach Nuclear Plant. However, irrespective of the contractual arrangements discussed below, Wisconsin Electric Power Company is the sole holder of the operating licenses and, as the Licensee, is responsible for the design, construction, and operation of the Point Beach Nuclear Plant.

Point Beach Nuclear Plant was designed and built by Westinghouse Electric Corporation as prime contractor for the Licensee. Westinghouse contracted to provide a complete, safe, and operable nuclear power unit ready for commercial service. The project was directed by Westinghouse from the offices of its Atomic Power Divisions in Pittsburgh, Pennsylvania, and by Westinghouse representatives at the plant site during construction and plant startup. Westinghouse engaged the engineering firm of Bechtel Corporation, San Francisco, California, to provide the design of the structures and non-nuclear portions of the plant and to prepare specifications for the purchase and construction thereof. The Licensee reviewed the designs and specifications prepared by Westinghouse and Bechtel to assure that the general plant arrangements, equipment, and operating provisions were satisfactory.

The plant was constructed under the general direction of Westinghouse through Bechtel as the general contractor who was responsible for the management of all site construction activities and who either performed or subcontracted the work of construction and equipment erection.

NUS Corporation, Washington, D.C., was engaged as consultants on general site studies and meteorology. The firm of Murray and Trettel, Inc. assisted on meteorology. The firm of Dames and Moore, Chicago, Illinois, was engaged as consultants on earth science and geology. The engineering firm of Sargent and Lundy, Chicago, Illinois, was engaged to design cooling water facilities.

In addition, specialists in environmental sciences have participated in developing information concerning the Point Beach site. Harza Engineering Company of Chicago, Illinois, provided assistance in hydrology and the firm of John A. Blume and Associates of San Francisco, California, provided assistance is assessing the seismic history of the sites and establishing the ground accelerations associated with the design earthquake.

Stone and Webster Engineering Corporation of Boston, Massachusetts, provided assistance in system planning and site studies.

The Licensee had qualified representatives at the site throughout construction and, with their own personnel and consultants, inspected major components and construction installations. The Licensee's initial operating force performed acceptance testing of all structures and equipment.

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4. McGaugh, J. D., "The Effect of Xenon Spatial Variations and the Moderator Coefficient on Core Stability," WCAP-2983 (August 1968).

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8. Westinghouse Customer Report, "Fracture Mechanics Evaluation of the Wisconsin Electric Power Company and Wisconsin Michigan Power Company Point Beach Nuclear Plant Unit 1 Reactor Vessel," WCAP-8742 (February 1977).