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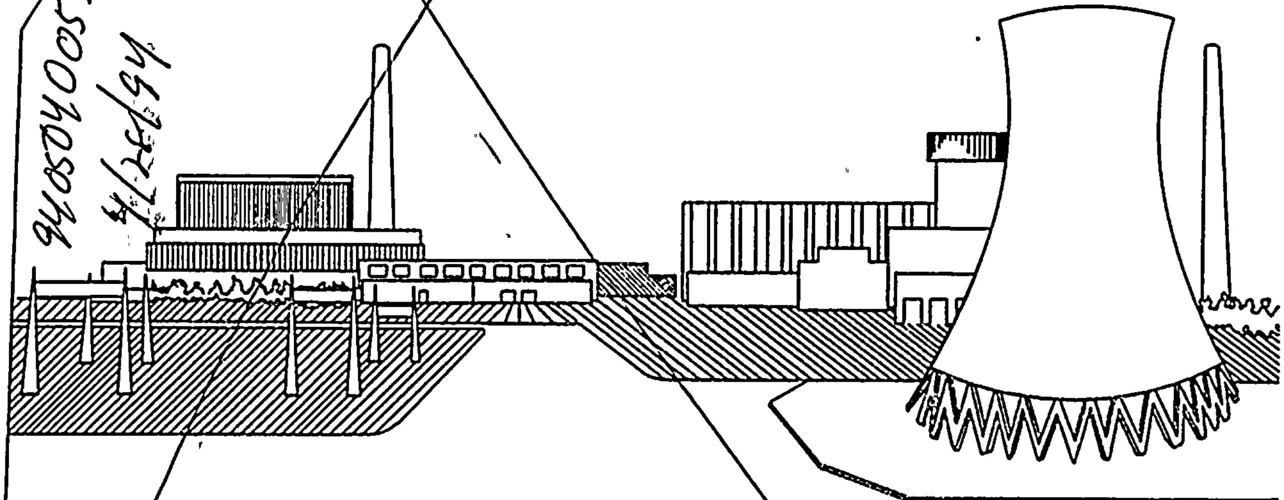
NINE MILE POINT NUCLEAR STATION — UNIT 2



NIAGARA MOHAWK

Superseded pages Per
Rev. 6 USAR

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2.5-174a	A15	2.5-194i	A14	T 2.5-8	A00
2.5-174b	A15	2.5-194j	A14	T 2.5-9 Sh 1	A00
2.5-174c	A15	2.5-194k	A26	T 2.5-9 Sh 2	A00
2.5-174d	A15	2.5-194L	A26	T 2.5-10	A00
2.5-175	A11	2.5-195	A14	T 2.5-11	A00
2.5-176	A11	2.5-196	A00	T 2.5-12 Sh 1	A00
2.5-176a	A05	2.5-197	A00	T 2.5-12 Sh 2	A00
2.5-176b	A05	2.5-198	A00	T 2.5-13	A00
2.5-177	A05	2.5-199	A00	T 2.5-14	A00
2.5-177a	A18	2.5-200	A00	T 2.5-15	A00
2.5-177b	R00	2.5-201	A00	T 2.5-16 Sh 1	A00
2.5-177c	R00	2.5-202	A00	T 2.5-16 Sh 2	A00
2.5-177d	R00	2.5-203	A00	T 2.5-17	A00
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T 2.5-23 Sh 2	A00	F 2.5-1	A00	F 2.5-46	A00
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T 15.2-1	R03	15.3-12	A00	T 15.4-6	A00
T 15.2-2	A00	15.3-13	A00	T 15.4-7	A00
T 15.2-3	A00	15.3-14	A00	T 15.4-8	A27
T 15.2-4	A00	T 15.3-1	A22	T 15.4-9	A22
T 15.2-5	A28	T 15.3-2	A10	T 15.4-10 Sh 1	A12
T 15.2-6	A00	T 15.3-3	A10	T 15.4-10 Sh 2	A12
T 15.2-7 Sh 1	A00	T 15.3-4	A22	T 15.4-11	A12
T 15.2-7 Sh 2	A00	T 15.3-5	A10	T 15.4-12	A12
T 15.2-8	A00	F 15.3-1	A00	T 15.4-13	A12
T 15.2-9	A22	F 15.3-2	A00	T 15.4-14	A12
T 15.2-10 Sh 1	A00	F 15.3-3	A00	T 15.4-15	A12
T 15.2-10 Sh 2	A00	F 15.3-4	A00	T 15.4-16	A12
T 15.2-11	A22	F 15.3-5	A00	F 15.4-1	A22
T 15.2-12	A27			F 15.4-2	A22
T 15.2-13	R03	15.4-1	R03	F 15.4-3	A00
F 15.2-1	A00	15.4-2	A22	F 15.4-4	A00
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F 15.2-5	A00	15.4-4b	A28	F 15.4-8	A00
F 15.2-6	A00	15.4-5	A00		
F 15.2-7	A00	15.4-6	A00	15.5-1	A00
F 15.2-8	A00	15.4-7	A22	15.5-2	R03
F 15.2-9	A10	15.4-8	A00	15.5-3	A00
F 15.2-10	A22	15.4-9	A00	T 15.5-1	A00
F 15.2-11	A22	15.4-10	R03	F 15.5-1	A00
15.2 Notes Sh 1	A22	15.4-11	A00		
15.2 Notes Sh 2	A22	15.4-12	A22	15.6-1	A12
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F 15.2-13	A00	15.4-14	A00	15.6-3	R00
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		15.4-17	A12	15.6-8	R00
15.3-1	A00	15.4-18	R00	15.6-8a	R03
15.3-2	A00	15.4-18a	R00	15.6-8b	A12
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15.3-4	A22	15.4-19	A12	15.6-10	A00
15.3-5	A00	15.4-20	A00	15.6-11	R00
15.3-6	A00	15.4-21	A28	15.6-12	R04
15.3-7	A22	T 15.4-1	A00	15.6-12a	R00
15.3-8	A27	T 15.4-2	A00	15.6-12a.1	R00
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15.6-12c	R03	T 15.6-17	A13	T 15.7-17	R01
15.6-12d	A18	T 15.6-18	A13		
15.6-13	R01	T 15.6-19	A13	15.8-1	R03
15.6-13a	A13	T 15.6-20	A13	15.8-2	R03
15.6-13b	A13	T 15.6-21	A22	15.8-3	A18
15.6-14	R03			15.8-4	R03
15.6-14a	A10	15.7-1	A00	15.8-5	R03
15.6-14b	A10	15.7-2	A25		
15.6-15	A00	15.7-3	A23	App 15A	A00
15.6-16	A25	15.7-4	A19	15A-i	A00
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15.6-18	A18	15.7-6	R03	15A-iii	A00
T 15.6-1 Sh 1	R00	15.7-7	A00	15A-iv	A00
T 15.6-1 Sh 2	R01	15.7-8	A00	15A-v	A00
T 15.6-2	A00	15.7-9	A00	15A-vi	A00
T 15.6-3	A12	15.7-10	A00	15A-vii	A00
T 15.6-4	R01	15.7-11	A00	15A-viii	A00
T 15.6-5	R02	15.7-12	R04	15A-1	A00
T 15.6-6 Sh 1	R00	15.7-13	R04	15A-2	A00
T 15.6-6 Sh 2	R00	15.7-13a	R04	15A-3	A00
T 15.6-7	A12	15.7-13b	R00	15A-4	A00
T 15.6-8	R00	15.7-14	A12	15A-5	A00
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15A-32	A00	F 15A-7	A00	F 15A-53	R04
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15A-34	A00	F 15A-9	A00	App 15B	A26
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15A-38	A00	F 15A-13	A00	15B-iv	R03
15A-39	A00	F 15A-14	A28	15B.1-1	R04
15A-40	R03	F 15A-15	R04	15B.1-2	R03
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15A-43	A00	F 15A-18	A00	15B.2-3	A26
15A-44	A00	F 15A-19	A00	15B.2-4	A26
15A-45	A00	F 15A-20	A28	15B.2-5	A26
15A-46	A00	F 15A-21	A00	15B.2-6	A26
15A-47	A00	F 15A-22	A00	15B.3-1	R03
15A-48	A00	F 15A-23	A00	15B.3-2	R03
15A-49	A00	F 15A-24	R03	15B.3-3	R03
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15A-51	A00	F 15A-26	A00	15B.3-4a	R03
15A-52	A00	F 15A-27	A00	15B.3-4b	R03
15A-53	A00	F 15A-28	A00	15B.3-5	R03
15A-54	A00	F 15A-29	A00	15B.3-6	A26
15A-55	A00	F 15A-30	A00	15B.3-7	A26
15A-56	A00	F 15A-31	A00	15B.3-8	R03
15A-57	A00	F 15A-32	R04	15B.3-9	R03
15A-58	A00	F 15A-33	A00	15B.3-10	R03
15A-59	A00	F 15A-34	A00	15B.3-11	R03
T 15A-1	R04	F 15A-35	A00	15B.3-12	R03
T 15A-2 Sh 1	A00	F 15A-36	A00	15B.3-13	R04
T 15A-2 Sh 2	A00	F 15A-37	A00	15B.3-13a	R04
T 15A-3	A00	F 15A-38	A00	15B.3-13b	R03
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T 15A-9	A00	F 15A-44	A00	15B.3-19	R03
T 15A-10	A00	F 15A-45	A00	15B.4-1	A26
T 15A-11	R04	F 15A-46	A00	15B.4-2	A26
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15B.6-3	R03	T 15D-2	R03	15G-ii	R03
15B.7-1	A26	T 15D-3	R03	15G-iii	R03
15B.7-2	R03	T 15D-4	R03	15G-1	R03
15B.8-1	A26	T 15D-5	R03	15G-2	R03
15B.8-2	R03	F 15D-1	R03	15G-3	R03
		F 15D-2	R03	15G-4	R03
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15C-i	A26	F 15D-4	R03	15G-6	R03
15C-ii	R03			15G-7	R03
15C-iii	R03	App 15E	A26	15G-8	R03
15C-1	R04	15E-i	R00	15G-9	R03
15C-2	R03	15E-ii	R00	15G-10	R03
15C-3	R03	15E-iii	A26	15G-11	R03
15C-4	R03	15E.1-1	A26	15G-12	R03
15C-5	R03	15E.1-2	R00	T 15G-1	R03
15C-6	R03	15E.2-1	R00	T 15G-2	R03
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15C-8	A26	15E.3-1	R00	T 15G-4	R03
15C-9	R03	15E.3-2	A26	T 15G-5	R03
15C-10	A26	15E.3-3	R00	T 15G-6	R03
15C-11	R03	15E.3-4	R01	T 15G-7	R03
15C-12	R03	15E.3-5	R00	T 15G-8	R03
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15C-16	R03	15E.3-9	A26	T 15G-11 Sh 2	R03
15C-17	R03	15E.3-10	R00	T 15G-12	R03
15C-18	R03	15E.3-11	R00	T 15G-13	R03
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15D-iii	R03	15E.3-18	A26	F 15G-6	R03
15D-1	R04	15E.4-1	A26	F 15G-7	R03
15D-1a	R04	15E.4-2	A26	F 15G-8	R03
15D-1b	R04	15E.4-3	R00	F 15G-9	R03
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15H-7	R00				
15H-8	R00				
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17-iii	A15	17.1-20	A00	17.1-40b	A05
17-iv	R00	17.1-21	A00	17.1-41	A09
17-v	A24	17.1-22	A28	17.1-42	A00
		17.1-22a	A28	17.1-43	A05
17.0-1	A07	17.1-22b	A09	17.1-43a	A05
17.0-2	A00	17.1-23	R00	17.1-43b	A05
17.0-3	A15	17.1-23a	A05	17.1-44	A05
T 17.0-1	A07	17.1-23b	A05	17.1-44a	A05
		17.1-24	A05	17.1-44b	A05
17.1-1	R00	17.1-24a	A05	17.1-45	A09
17.1-1a	A26	17.1-24b	A01	17.1-46	A00
17.1-1b	A20	17.1-25	A00	17.1-47	A28
17.1-2	A28	17.1-26	A05	17.1-48	A28
17.1-2a	R00	17.1-27	A09	17.1-49	A05
17.1-2b	A09	17.1-27a	A09	17.1-50	A01
17.1-3	A28	17.1-27b	A09	17.1-50a	A01
17.1-4	A28	17.1-28	A00	17.1-50b	A01
17.1-4a	A28	17.1-29	A00	17.1-51	A00
17.1-4b	A28	17.1-30	A05	17.1-52	A28
17.1-5	A28	17.1-30a	A05	17.1-53	A28
17.1-5a	R00	17.1-30b	A01	T 17.1-1 Sh 1	A00
17.1-5b	A05	17.1-31	A09	T 17.1-1 Sh 2	A00
17.1-6	A28	17.1-32	A09	T 17.1-1 Sh 3	A00
17.1-7	R00	17.1-32a	A09	T 17.1-1 Sh 4	A00
17.1-8	A28	17.1-32b	A09	T 17.1-1 Sh 5	A00
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17.1-10	A28	17.1-33a	A05	F 17.1-2	R00
17.1-10a	A13	17.1-33b	A05		
17.1-10b	A13	17.1-34	A09	17.2-1	R05
17.1-11	R00	17.1-34a	R01		
17.1-12	A00	17.1-34b	A09		
17.1-13	A09	17.1-35	A28		
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17.1-13b	A05	17.1-35b	A15		
17.1-14	A28	17.1-36	A09		
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17.1-15a	A09	17.1-36b	A05		
17.1-15b	A05	17.1-37	A00		
17.1-16	A05	17.1-38	A28		
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17.1-16b	A05	17.1-38b	A15		
17.1-17	A05	17.1-39	A05		
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18-ib	R03	F 18.2-3	R03		
18-ii	R03	F 18.2-4	R03		
18-iii	R03	F 18.2-5	R03		
18-iv	R03	F 18.2-6 Sh 1	R03		
		F 18.2-6 Sh 2	R03		
18.1-1	R03	F 18.2-7	R03		
18.1-1a	R03	F 18.2-8	R03		
18.1-1b	R03				
18.1-2	R03				
18.2-1	R03				
18.2-2	R03				
18.2-3	R03				
18.2-4	R03				
18.2-5	R03				
18.2-6	R03				
18.2-7	R03				
18.2-8	R03				
18.2-9	R03				
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A-iii	R03	A.15.0-4	R04	F A.15.2-1	R04
A-iv	R04	A.15.0-5	R03	F A.15.2-2	R04
A-iva	R04	A.15.0-6	R04		
A-ivb	R04	A.15.0-7	R04	A.15.4-1	R04
A-v	R04	T A.15.0-1 Sh 1	R04	A.15.4-2	R04
A-vi	R04	T A.15.0-1 Sh 2	R04	A.15.4-3	R04
A-vii	R04	T A.15.0-2	R04	A.15.4-4	R05
		T A.15.0-3	R04	A.15.4-5	R04
A.0-1	R04	T A.15.0-4 Sh 1	R04	A.15.4-6	R04
A.0-2	R04	T A.15.0-4 Sh 2	R04	A.15.4-7	R04
A.0-3	R04	T A.15.0-4 Sh 3	R04	A.15.4-8	R04
A.0-4	R04	F A.15.0-1	R04	A.15.4-9	R04
F A.0-1	R04	F A.15.0-2	R04	T A.15.4-1	R03
				T A.15.4-2	R04
A.4.1-1	R04	A.15.1-1	R04	T A.15.4-3	R04
A.4.2-1	R03	A.15.1-2	R04	T A.15.4-4	R04
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A.5.2-2	R04	T A.15.1-1	R04	A.15C-1	R04
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T A.6-2	R04	A.15.2-6	R04		
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F A.6-2	R03	A.15.2-8	R04		
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9B.12-4	R00
9B.12-5	R00
9B.12-6	R00
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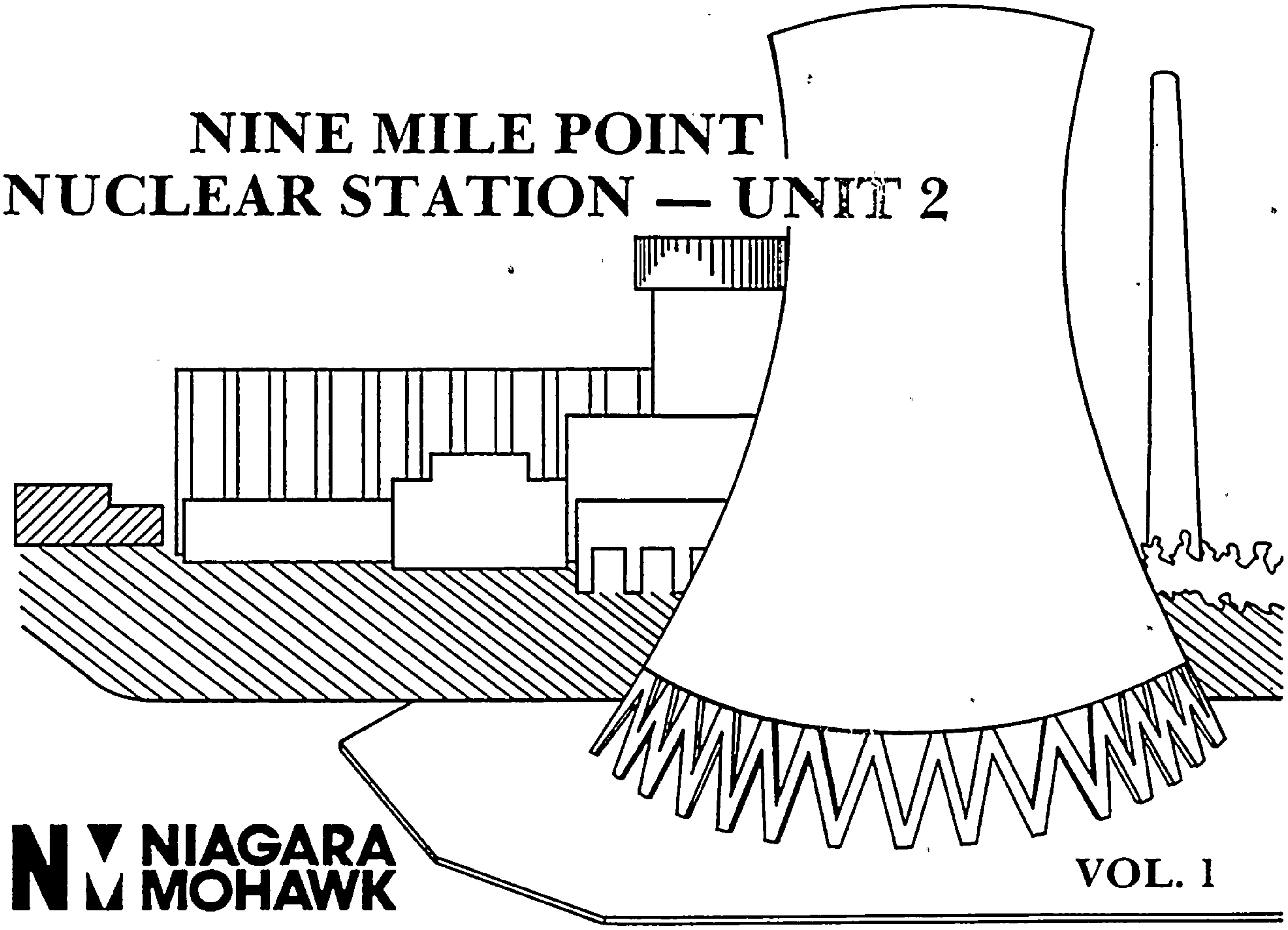
UPDATED SAFETY ANALYSIS REPORT

NINE MILE POINT NUCLEAR STATION — UNIT 2



NIAGARA MOHAWK

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CHAPTER 1

INTRODUCTION AND GENERAL DESCRIPTION OF THE PLANT

1.1 INTRODUCTION

This Final Safety Analysis Report (FSAR) is submitted by the Niagara Mohawk Power Corporation (Applicant) and its co-owners (Central Hudson Gas and Electric Corporation, Long Island Lighting Company, New York State Electric and Gas Corporation, and Rochester Gas and Electric Corporation) in support of the application for a Class 103 operating license for the nuclear power station designated Nine Mile Point Nuclear Station - Unit 2 (Unit 2).

Unit 2 is located on a 364-ha (900-acre) site owned by Niagara Mohawk Power Corporation (NMPC), and is situated on the southeast shore of Lake Ontario, Oswego County, NY, approximately 10 km (6.2 mi) northeast of the city of Oswego. Unit 2 and support facilities occupy about 18.2 ha (45 acres), and share the site with the existing Nine Mile Point Nuclear Station - Unit 1 (Unit 1) (Docket No. 50-220) which has been in commercial operation since 1969. The Nine Mile Point site is adjacent to the James A. FitzPatrick Nuclear Power Plant owned by the New York Power Authority (NYPA); Unit 2 is located 274 m (900 ft) east of Unit 1 and about 716 m (2,350 ft) west of the James A. FitzPatrick Plant.

Unit 2 employs a nuclear steam supply system (NSSS) consisting of a single-cycle, forced circulating boiling water reactor (BWR). The plant-rated core thermal power level (Figure 1.1-1) is 3,323 MWt corresponding to a net electrical output of 1,080 MWe, and design thermal power of 3,463 MWt corresponding to a gross electrical output of 1,202 MWe. The thermal power used for the plant transient and loss-of-coolant accident (LOCA) analyses is 3,463 MWt. All safety systems have been designed for a thermal power of 3,489 MWt. The NSSS supplier is General Electric Company-Nuclear Energy Operations (GE-NEO). The balance of the plant is designed and constructed by Stone & Webster Engineering Corporation (SWEC). Other plants designed by SWEC that are similar in concept are currently under review by the Nuclear Regulatory Commission (NRC). These are the Shoreham Nuclear Power Station, Brookhaven, Long Island, NY, and the River Bend Station, St. Francisville, LA.

The containment design employs the BWR Mark II concept of over-under pressure suppression with multiple downcomers connecting the reactor drywell to the water-filled pressure suppression chamber. The primary containment is a steel-lined, reinforced concrete enclosure housing the reactor and the suppression pool.

The reactor building completely encloses the primary containment. The structure provides secondary containment when the primary containment is closed and in service, and provides primary containment when the primary containment is open, as during refueling. The reactor building houses the refueling and reactor servicing equipment, new and spent fuel storage facilities, and other reactor auxiliary and service equipment. The primary purpose of the reactor building is to minimize ground-level release of airborne radioactive material.

The outer wall of the reactor building is reinforced concrete up to the crane rail level above the refueling floor. Above the crane rail level, the superstructure is a steel frame using metal wall panels with sealed joints. Access to the building is through airlocks.

The power generation complex includes several contiguous buildings: the reactor building with two auxiliary bays, the control building, the turbine building, and the radwaste building. Other buildings, such as the security facility, are also located in the general plant area. A screenwell for the circulating and service water systems is located approximately 107 m (350 ft) northwest of the centerline of the reactor building.

Condenser cooling for Unit 2 is provided from a counterflow, natural-draft, hyperbolic, concrete cooling tower located approximately 330 m (1,000 ft) south of the centerline of the reactor building. The ultimate heat sink for emergency core cooling is Lake Ontario. Below grade and north of the screenwell building, there are two concrete tunnels that convey the service water intake, service water discharge, and cooling tower blowdown. A safety-related intake pipe is enclosed in each tunnel. The intake pipes extend from the intake shaft approximately 396 m (1,300 ft) northward under Lake Ontario to the submerged intake structures. One tunnel also contains the discharge pipe which extends approximately 550 m (1,800 ft) to the discharge diffuser.

Radionuclides are emitted to the atmosphere from two locations at Unit 2. These are the stack and the combined vent for the radwaste and reactor buildings. Liquid radwaste is stored for decay or concentrated to a solid waste for controlled disposal at regulated storage sites.

The shielding design and plant layout are based on extensive experience of NMPC and SWEC in controlling radiological exposures to as low as reasonably achievable (ALARA) levels. Estimated radiological doses for normal operations and postulated accidents are all fractional parts of the doses listed in federal radiological guidelines for siting and operation of nuclear power plants. Environmental impacts are described in the separate Environmental Report - Operating License Stage being submitted for Unit 2.

This FSAR is written in accordance with Regulatory Guide 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants. The content of this FSAR addresses applicable regulatory guides issued up to July 1982.

The Unit 2 Preliminary Safety Analysis Report (PSAR) was submitted on June 15, 1972 (Docket No. 50-410). The NRC Safety Evaluation Report (SER) was issued on June 15, 1973, and the construction permit (CPPR-112) was issued on June 24, 1974. Environmental impacts related to the plant are discussed in the Environmental Report - Construction Permit Stage submitted on June 15, 1972. The NRC Final Environmental Statement was issued in June 1973.

The approximate schedule for Unit 2 fuel loading and commercial operation is as follows:

Fuel loading	March 1986
Commercial operation	October 1986

1.2 GENERAL PLANT DESCRIPTION

1.2.1 Principal Design Criteria

The principal architectural and engineering criteria for the design, construction, and operation of Unit 2 are summarized in this Section. There are two ways of considering principal design criteria: on a classification-by-classification basis, or on a system-by-system (system group) basis. Safety analyses generally utilize the information formatted in the classification-by-classification approach but system descriptions are more easily understood through the system-by-system method. This section uses both methods for summarizing the principal design criteria.

1.2.1.1 General Criteria

Some of the criteria are generally applicable to more than one classification or more than one system group. These general criteria are as follows:

1. Unit 2 is designed, fabricated, and erected to produce electric power in a safe and reliable manner. Unit design generally conforms with applicable codes and regulations. Exceptions are evaluated and justified. The General Design Criteria (GDC) of 10CFR50 Appendix A have been satisfied in the Unit 2 design.

2. Unit 2 is designed, fabricated, and erected to operate in such a way that the release of radioactive materials to the environment is limited to less than the limit and guideline values of applicable federal regulations pertaining to the release of radioactive materials for normal operations and abnormal events.

3. Unit 2 is designed to support a GE BWR and NSSS to produce steam for direct use in a turbine generator unit. This design incorporates features typical of many other BWR plants.

4. Certain portions of the plant are designed to withstand extreme natural phenomena such as earthquakes, flooding, or tornadoes, and unnatural phenomena such as fire, flooding from in-plant leakage, internally- or externally-generated missiles, and others.

5. The reactor core and reactivity control systems are designed so that control rod action is capable of bringing the core to subcritical condition and maintaining it, even with the rod of highest reactivity worth fully withdrawn and unavailable for insertion.

6. Design margins for the nuclear safety systems and engineered safeguards are conservative.

7. Nuclear safety systems are designed to respond to abnormal operation transients to preclude fuel damage. Any fission products released to the environs via normal discharge paths for radioactive material will not exceed the limits of 10CFR50 Appendix I.

8. Nuclear safety systems and engineered safeguards are designed to assure that no damage to the reactor coolant pressure boundary (RCPB) results from internal pressures caused by abnormal operational transients or accidents.

9. Where positive, precise action is immediately required in response to accidents, such action is automatic and requires no decision or manipulation of controls by Station operations personnel.

10. Essential safety actions are carried out by safety-related equipment of sufficient redundancy and independence so that no single failure of active components can prevent required actions. Any single failure within the safety-related protection system shall not prevent proper protective action at the system level when required.

11. Provisions have been made for control of the components of nuclear safety systems and engineered safeguards from the control room.

12. Nuclear safety systems and engineered safeguards are designed to permit demonstration of their functional performance requirements.

13. Unit 2 features essential to the mitigation of accident consequences are designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed.

1.2.1.2 Power Generation Design Criteria

Principal power generation design criteria are as follows:

1. Fuel cladding is designed to retain integrity as a radioactive material barrier throughout the full operational range of the plant and any abnormal operation transient. The fuel cladding is designed to accommodate, without loss of integrity, pressures generated by fission gases released from fuel material throughout the design life of the fuel.

2. Heat removal systems are provided in sufficient capacity and redundancy to remove heat generated in the reactor core for the full range of normal operating conditions from plant shutdown to design power, and

also for any abnormal operational transient. The capacity of such systems is adequate to prevent fuel cladding damage.

3. Backup heat removal systems are provided to remove decay heat generated in the core under circumstances where normal operational heat removal systems have become inoperative. The capacity of such systems is adequate to prevent fuel cladding damage.

4. Control equipment is provided to allow the reactor to respond automatically to major or minor load changes and other abnormal operational transients, including bringing the reactor to a hot shutdown condition.

5. Reactor power level is manually controlled.

6. Control of the NSSS, including the reactor, is possible from a single location.

7. NSSS and reactor controls, including alarms, are arranged to allow the operator to rapidly assess the condition of the nuclear system and locate process system malfunctions.

8. Fuel handling and storage facilities are designed to maintain adequate shielding, cooling, and water quality for spent fuel and to prevent inadvertent criticality.

9. Interlocks or other automatic equipment are provided as backup to procedural controls to avoid conditions requiring the functioning of nuclear safety systems or engineered safeguards.

1.2.1.2.1 Safety Design Criteria

1. The unit is designed, fabricated, and erected to operate so that the release of radioactive materials to the environment is significantly less than the requirements of 10CFR20 or 10CFR100. Those portions of the nuclear system that form part of the RCPB are designed to retain integrity as a radioactive barrier following abnormal operational transients or an accident event.

2. The reactor core is designed so its nuclear characteristics do not contribute to a divergent power transient.

3. The NSSS and supporting systems are designed so that there is no tendency for divergent oscillation of any operating characteristic, considering the interaction of the nuclear system with other appropriate unit systems.

4. Gaseous, liquid, and solid waste disposal facilities are designed so the discharge of radioactive effluents and offsite shipment of radioactive materials can be made in accordance with applicable regulations.

5. Design of the radwaste system provides means by which Unit 2 operators can be alerted when limits on the release of radioactive material are approached.

6. Sufficient indications are provided to determine that the reactor is operating within the limits of applicable regulations in any mode of unit operations.

7. Adequate radiation shielding is provided and access control patterns are established to allow a properly trained operating staff to control radiation doses to a level that is as low as reasonably achievable (ALARA) during normal operation.

8. Essential safety actions are designed to be carried out by equipment of sufficient redundance and independence that no single failure can prevent the required actions. Any single failure within the protection system shall not prevent proper action at the system level when required.

9. Provisions are made for control of the components of nuclear safety systems from the control room.

10. Nuclear safety systems are designed to demonstrate functional performance requirements.

11. The design of nuclear safety systems includes design allowances for environmental disturbances at the site such as earthquakes, floods, high winds, storms and other disturbances such as fire and flooding from leakage of fluid systems, internally- and externally-generated missiles, and others.

12. Standby electrical power sources are of sufficient capacity to power all necessary nuclear safety systems requiring electrical power. Standby ac and dc power sources are provided to remove decay heat when the offsite power supply is not available.

13. Engineered safeguards are designed to assure that no damage to the RCPB results from internal pressures caused by an accident or abnormal transient.

14. A primary containment is provided that completely encloses the reactor vessel. The primary containment uses the pressure suppression concept.

15. The primary containment is designed to retain integrity as a radioactive material barrier during and following accidents that release radioactive material into the primary containment volume.

16. It is possible to test primary containment integrity and leak-tightness at periodic intervals.

17. A reactor building is provided that completely encloses both the primary containment and the fuel storage areas. The secondary containment includes a method for controlling release of radioactive materials from the barrier and includes a capability for filtering radioactive materials within the barrier.

18. The reactor building is designed to act as a radioactive material barrier, if required, when the primary containment is open for expected operational purposes.

19. The primary containment and reactor building, in conjunction with other engineered safeguards, limits radiological effects of accidents resulting in the release of radioactive material to the primary containment volume to significantly less than the requirements of 10CFR100.

20. Provisions are made for removing energy from within the primary containment to maintain the integrity of the primary containment system following accidents that release energy to the primary containment.

21. Piping that penetrates the primary containment structure and serves as a path for the uncontrolled release of radioactive material to the environs is automatically isolated whenever such potential for radioactive material release exists. Such isolation is effected in time to limit radiological effects to significantly less than the requirements of 10CFR100.

22. The emergency core cooling system (ECCS) is provided to limit fuel cladding temperature to 2,200°F as a result of a LOCA.

23. The ECCS provides for continuity of core cooling over the complete range of postulated break sizes in the RCPB.

24. The ECCS is diverse, reliable, and redundant.

25. Operation of the ECCS is initiated automatically when required, regardless of the availability of offsite power.

26. The main control room is shielded against radiation to permit continued occupancy under accident conditions.

27. In the event that the main control room becomes uninhabitable, it is possible to bring the reactor from power range operation to a cold shutdown condition by manipulating local controls and equipment available outside the main control room.

28. Backup reactor shutdown capability is provided independent of normal reactivity control provisions. This backup system shuts down the reactor from any normal operating condition and maintains the shutdown condition.

1.2.1.3 System-by-System Approach

The principal architectural and engineering criteria for design are summarized below on a system-by-system or system group basis. The system-by-system presentation facilitates understanding of the actual design of any one system. Only the most restrictive of any related criteria are stated for a system. Where the most restrictive criterion is classified as a power generation consideration, less restrictive safety criteria may not be stated in the system-by-system presentation. However, the actual design of a system must reflect all criteria that pertain to it.

1.2.1.3.1 Nuclear System Criteria

Principal design criteria for the reactor, ECCS, RCPB, and reactivity control systems are as follows:

1. The nuclear system is designed to support a GE BWR rated at 3,323 MWt.

2. Fuel cladding is designed to retain integrity as a radioactive material barrier throughout the design power range. Fuel cladding is designed to accommodate, without loss of integrity, the pressures generated by the fission gases released from fuel material throughout the design life of the fuel.

3. Fuel cladding, in conjunction with other unit systems, is designed to retain integrity throughout any abnormal operational transient.

4. Those portions of the nuclear system that form part of the RCPB are designed to retain integrity as a radioactive material barrier following abnormal operational transients and accidents.

5. Heat removal systems including the ECCS and makeup water supplies are provided in sufficient capacity, redundancy, and operational adequacy to remove heat

generated in the reactor core for the full range of normal operational conditions from unit shutdown to design power and for any abnormal operational transient or accident. The capacity of such systems is adequate to prevent fuel cladding damage.

6. The reactor core and reactivity control system is designed to ensure that control rod action is capable of bringing the core subcritical and maintaining it thus, even with the rod of highest reactivity worth fully withdrawn and unavailable for insertion. An alternate reactivity control system is provided should the control rods or drive system become inoperable. The alternate system is capable of shutting down the reactor and maintaining it subcritical.

7. The reactor core is designed so that its nuclear characteristics do not contribute to a divergent power transient.

8. The nuclear system is designed so there is no tendency for divergent oscillation of any operating characteristic, considering the interaction of the nuclear system with other appropriate unit systems.

1.2.1.3.2 Power Conversion Systems Criteria

Components of the power conversion systems are designed to perform these basic objectives:

1. Reliably produce electrical power from the steam supplied by the reactor, condense the steam into water, and return the water to the reactor as heated feedwater, with a major portion of its gaseous and particulate impurities removed.

2. Assure that any fission products or radioactivity associated with the steam and condensate during normal operation or accident conditions are safely contained within the system or are released under controlled conditions in accordance with waste disposal regulations.

1.2.1.3.3 Electrical Power Systems Design Criteria

Sufficient preferred and standby ac and dc power sources are provided to attain prompt shutdown and continued maintenance of the unit in a safe condition under all credible circumstances. Power sources are adequate to accomplish all required engineered safeguard functions under postulated design basis accident (DBA) conditions.

1.2.1.3.4 Radwaste System Design Criteria

1. Radwaste systems are designed to limit release of radioactive materials from the unit during normal operation to significantly less than the requirements of 10CFR20, and within the guidelines of Appendix I to 10CFR50.

2. Gaseous, liquid, and solid waste disposal systems are designed so that offsite shipments will be in accordance with applicable regulations, including 10CFR20, 10CFR71, and 49CFR171 through 179, as appropriate.

3. The design provides means by which unit operations personnel can be alerted whenever operational limits on the release of radioactive material are approached.

1.2.1.3.5 Auxiliary Systems Design Criteria

1. Auxiliary systems are provided to support the NSSS and power generation system to provide for maintenance of the plant environment, in-plant radiation and airborne contamination control, compressed air supplies, sealing steam, etc.

2. Essential auxiliary systems are designed to function during accident conditions.

1.2.1.3.6 Shielding and Access Control Design Criteria

1. Radiation shielding is provided and access control patterns are established to allow a properly trained operating staff to control radiation doses within the limits of published regulations during normal operation. Shielding design, equipment layout, and zoning have been performed to ensure radiation doses are maintained ALARA.

2. The main control room is shielded against radiation so that in conjunction with the control room air conditioning system occupancy is allowed under accident conditions.

1.2.1.3.7 Nuclear Safety Systems and Engineered Safeguards Design Criteria

Principal design criteria for nuclear safety systems and engineered safeguards are as follows:

1. These criteria correspond to 10CFR50 General Design Criteria 1 through 64 as described in Section 3.1.2.

2. Standby ac and dc power sources are designed to have sufficient capacity to power all necessary nuclear safety systems and engineered safeguards requiring electrical power.

3. Standby ac power sources are provided to allow reactor shutdown and removal of decay heat when offsite power is not available.

4. In the event that the main control room is uninhabitable, it is possible to bring the reactor from power range operation to a cold shutdown condition by use of the shutdown room or manipulating local controls and equipment that are available outside the control room.

5. Backup reactor shutdown capability is provided independently of normal reactivity control provisions. This backup system has the capability to shut down the reactor from any normal or upset operating condition and subsequently to maintain the shutdown condition.

1.2.1.3.8 Process Control System Design Criteria

Principal design criteria for process control systems are listed as follows:

NSSS Process Control Design Criteria

1. Control equipment is provided to allow the reactor to respond automatically to load changes within design limits.

2. Controls are provided to manually control reactor power level.

3. Control of the nuclear system is possible from a single location.

4. Nuclear systems process controls and alarms are arranged to allow the operator to rapidly assess the condition of the nuclear system and to locate process system malfunctions.

5. Interlocks or other automatic equipment are provided as a backup to procedural controls to avoid conditions requiring actuation of nuclear safety systems or engineered safeguards.

Power Conversion Systems Process Control Design Criteria

1. Control equipment is provided to control reactor pressure throughout its operating range.

2. The turbine is capable of responding automatically to minor changes in load.

3. The feedwater system is controlled to maintain the water level in the reactor vessel at the optimum level range.

4. Control of the power conversion equipment is possible from one location.

5. Interlocks or other automatic equipment are provided as a backup to procedural controls to avoid conditions requiring the actuation of nuclear safety systems or engineered safeguards.

Electrical Power System Process Control Design Criteria

1. The safety-related (Class 1E) electrical power system is designed as a three-division system, any two out of three divisions being adequate to safely shut down the unit.

2. The protection system is designed to detect and isolate faulted equipment from the system with a minimum of disturbance in the event of any fault in the system.

3. In the event of a loss of offsite power (LOOP), the protection system isolates the emergency buses from the offsite system and initiates the starting of the standby ac power sources.

4. In the event of a LOOP and LOCA, the protection system isolates the emergency buses from the offsite system and the normal electrical system, and the standby diesel generators are started and sequentially loaded by a programmed control system to energize all safety-related loads.

5. All electrically-operated breakers are controllable from the control room.

6. Metering for generators, transformers, and other essential circuits is available in the control room.

1.2.2 Site Description

1.2.2.1 Site Characteristics: Site Location and Size

The project site comprises approximately 364 ha (900 acres) and is located on the south shore of Lake Ontario in the town of Scriba, Oswego County, NY, on land owned by NMPC. Unit 2 shares the site with existing Unit 1; Unit 2 and support facilities occupy about 18.2 ha (45 acres) of the total site acreage. The James A. FitzPatrick plant, owned by the NYPA, is located east of

the project site. The centerline-to-centerline distance between Unit 2 and the FitzPatrick plant reactors is about 716 m (2,350 ft). The distance between the Unit 1 and Unit 2 reactor centerlines is about 274 m (900 ft). The site plan is shown on Figure 1.2-1. All activities at the site are under the direct control of NMPC[1].

1.2.2.2 Access to the Site

The protected area of the site is isolated from the surrounding area by fencing. Access to the site is controlled at the gate of the main entrance to the plant by security personnel. All other gates are kept locked[2].

1.2.2.3 Description of the Site and Environs

Most of the land immediately to the south and west of the site is pasture or inactive farmland. For the region west, south, and east of the site, the country is characterized by rolling terrain rising gently up from Lake Ontario which lies immediately to the north of the site.

Within an approximate 8-km (5-mi) radius of Unit 2, the 1980 population was 3,468. The population for this same area is projected to be 5,301 in 1990 and 7,213 in 2010. The nearest dwellings are on Lakeview Road approximately 1.6 km (1 mi) from the Station. The Ontario Bible Conference operates a summer camp on the lakefront adjacent to the western boundary of the site[3].

Oswego, which is the nearest city, is located about 10 km (6.2 mi) southwest of the site and had a 1980 population of 19,793. The nearest population center with a population in excess of 25,000 is the city of Syracuse, approximately 53 km (32.8 mi) southeast of the site. Buffalo is approximately 217 km (135 mi) west of the site. Figure 2.1-1 shows the location of the site relative to the larger cities in New York State which are within the 80-km (50-mi) radius of the site.

1.2.3 Structures and Equipment

The buildings and structures essential to the safe operation and shutdown of the plant are designed to withstand extreme environmental and abnormal loading conditions. The structures and/or portions thereof so designated are designed to provide the protection as required from tornadoes, missiles, earthquakes, pipe whip, and internal or external flooding. Additional discussions of design considerations are found in Chapter 3.

Locations and orientation of the structures are shown on Figures 1.2-1 and 1.2-2. The general arrangement of personnel access between structures is shown on Figures 1.2-3 through 1.2-5. The general arrangement of the major structures and equipment is shown on Figures 1.2-6 through 1.2-40.

The principal structures located at the site are listed below along with the brief description of the major equipment within each structure.

The primary containment structure (Figures 1.2-6 through 1.2-12) houses the reactor pressure vessel, reactor recirculation pumps and motors, drywell cooling system unit coolers, safety relief valves, accumulators, and other equipment.

The reactor building and auxiliary bays (Figures 1.2-6 through 1.2-12) enclose the primary containment structure. These structures house the remaining portions of the NSSS, refueling and fuel storage equipment, control rod drive (CRD) hydraulic units, equipment for the reactor water cleanup system (RWCU), equipment for the standby liquid control system (SLCS), equipment for the reactor building closed loop cooling water system (RBCLCW), and other equipment.

The radwaste building (Figures 1.2-13 and 1.2-14) houses primarily the tanks and equipment associated with the liquid and solid radwaste systems.

The control building (Figures 1.2-15 and 1.2-16) houses the main control room, standby switchgear, batteries and associated instrumentation, cables, and equipment.

The diesel generator building (Figures 1.2-17 and 1.2-18) houses three standby diesel generators, diesel oil storage tanks, and associated controls and instrumentation.

The turbine building including heater bay (Figures 1.2-19 through 1.2-25) houses the turbine generator, condensers, moisture separator reheater, condensate demineralizer system, feedwater heaters, steam jet air ejectors (SJAEs), reactor feed pumps, turbine building closed loop cooling water (TBCLCW) system, and miscellaneous tanks and equipment to support the power conversion system and other related systems.

The screenwell building (Figures 1.2-26 through 1.2-28) houses the circulating water pumps and the service water pumps with associated equipment and instrumentation.

The intake and discharge tunnels and intake structures (Figures 1.2-29 and 1.2-30) are used for transporting service water to and from the lake. The intake tunnels also supply makeup water for the circulating water system.

The main stack (Figure 1.2-31) is used to provide elevated release of gases from the offgas, standby gas treatment, and other systems.

The offgas regeneration and condensate demineralizer rooms (Figures 1.2-19 through 1.2-25) house the catalytic recombiners, offgas filter, condensate demineralizer, and related equipment.

The normal switchgear building (Figures 1.2-32 and 1.2-33) houses the normal switchgear and associated equipment.

The auxiliary boiler building (Figure 1.2-34), located north of the screenwell building, houses the electric boilers and accessories to supply steam to the plant during shutdown.

The standby gas treatment building and railroad access area (Figures 1.2-35 and 1.2-36) house the standby gas treatment filters and associated equipment and allow access for spent fuel shipping.

The condensate storage tank building (Figure 1.2-37) houses the condensate storage tanks and associated equipment.

The natural-draft cooling tower (Figures 1.2-38 and 1.2-39) provides the normal heat sink for heat transferred to the circulating water system from the main condensers.

The auxiliary service building (Figures 1.2-7 and 1.2-8), adjacent to the reactor building, houses the HVAC room and decontamination and shower facilities for personnel.

The decontamination area (Figures 1.2-19 through 1.2-21, 1.2-23, and 1.2-24), immediately south of the radwaste building, provides the facility for decontamination of large tools and equipment, and a sample room. It also houses clean steam reboilers and related equipment.

The hydrogen storage area (for hydrogen cooling of the turbine generator, Figure 1.2-40) is located west of the offgas area. The hydrogen storage bottles are mounted on concrete pads and are in a fenced area.

1.2.4 Nuclear Steam Supply System (NSSS)

The nuclear system includes a direct-cycle, forced circulation, GE BWR that produces steam for direct use in the steam turbine. A heat balance showing the major parameters of the nuclear system for the warranted power condition is shown on Figure 1.1-1.

The NSSS is further discussed in Chapters 4 and 5.

1.2.4.1 Reactor Core and Control Rods

The reactor fuel and core design are described in Section 2 of Reference 5 and Section 1 of Reference 6.

Experience has shown that the control rods are not susceptible to distortion and have an average life expectancy many times the residence time of a fuel loading.

1.2.4.2 Reactor Vessel and Internals

The reactor vessel contains the core and supporting structures; the steam separators and dryers; the jet pumps; the control rod guide tubes; the distribution lines for the feedwater, core sprays, and standby liquid control; the in-core instrumentation; and other components. The main connections to the vessel include the steam lines, coolant recirculation lines, feedwater lines, CRD and in-core nuclear instrument housings, core spray lines, residual heat removal lines, standby liquid control line, core differential pressure line, jet pump pressure-sensing lines, and water level instrumentation.

The reactor vessel is designed and fabricated in accordance with applicable codes for a pressure of 1,250 psig. The nominal operating pressure in the steam space above the separators is 1,020 psia. The vessel is fabricated of low-alloy steel and is clad internally with stainless steel (except for the top head nozzles and nozzle weld zones which are unclad).

The reactor core is cooled by demineralized water that enters the lower portion of the core and boils as it flows upward around the fuel rods. The steam leaving the core is dried by steam separators and dryers located in the upper portion of the reactor pressure vessel (RPV). The steam is then directed to the turbine through the main steam lines. Each steam line has two isolation valves in series, one on either side of the primary containment barrier.

1.2.4.3 Reactor Recirculation System

The reactor recirculation system consists of two recirculation pump loops external to the RPV. These loops provide the piping path for the driving flow of water to the RPV jet pumps. Each external loop contains one high-capacity motor-driven recirculation pump, two motor-operated maintenance valves, and one hydraulically-operated flow control valve. The variable position hydraulic flow control valve operates in conjunction with a low-frequency motor generator (MG) set to control reactor power level through the effects of coolant flow rate on moderator void content.

The jet pumps are RPV internals. They provide a continuous internal circulation path for the major portion of the core coolant flow. The jet pumps are located in the annular region between the core shroud and the vessel inner wall. Any recirculation line break still allows core flooding to approximately two-thirds of the core height, the level of the inlet of the jet pumps.

1.2.4.4 Residual Heat Removal System

The residual heat removal (RHR) system is a system of pumps, heat exchangers, and piping that fulfills the following functions:

1. Removes decay and sensible heat during and after plant shutdown.

2. Injects water into the RPV following a LOCA to reflood the core independently of other core cooling systems.

3. Removes heat from the primary containment following a LOCA, to limit the increase in primary containment pressure. This is accomplished by cooling and recirculating the suppression pool water and by spraying the drywell and suppression pool air spaces with suppression pool water.

1.2.4.5 Reactor Water Cleanup System

The RWCU system recirculates a portion of reactor coolant through a filter demineralizer to remove particulate and dissolved impurities from the reactor coolant. It also removes excess coolant from the reactor system under controlled conditions.

1.2.4.6 Nuclear Leak Detection System

The nuclear leak detection and monitoring system consists of temperature, pressure, flow, and fission-product sensors with associated instrumentation and alarms. This system detects and annunciates leakage in the following systems:

1. Main steam lines.

2. Reactor water cleanup (RWCU) system.

3. Residual heat removal (RHR) system.

4. Reactor core isolation cooling (RCIC) system.

5. Feedwater system.

6. Emergency core cooling systems (ECCS).

7. Miscellaneous systems.

Small leaks generally are detected by monitoring area temperatures, radiation levels, and drain sump fillup and pumpout rates. Large leaks are also detected by changes in reactor water level, primary containment pressure and changes in flow rates in process lines.

1.2.5 Electrical, Instrumentation, and Control Systems

1.2.5.1 Electrical Power System

The plant electrical power system consists of the unit generator, the switchyard, and the unit auxiliary power distribution system.

The unit generator is connected directly to the generator step-up transformers and the normal station service transformer through isolated phase bus duct. The generator step-up transformers step up the output of the unit generator from 25 kV to a nominal 345-kV transmission system voltage. The normal station service transformer steps down the unit generator voltage from 25 to 13.8 kV and provides an onsite (ac) power source to the unit auxiliary power distribution system.

The switchyard has two separate and independent sections: the 345-kV switchyard and the 115-kV switchyard. The output of the generator step-up transformers is connected to the 345-kV switchyard which connects the unit generator to the outgoing transmission system. The 115-kV switchyard receives power from two separate offsite power sources through two physically and electrically independent incoming circuits. The two circuits feed two separate reserve Station service transformers and an auxiliary boiler transformer. The reserve Station service transformers step down the offsite power from 115 to 13.8 and 4.16 kV, and provide two independent offsite power sources for the unit auxiliary power distribution system. The auxiliary boiler transformer steps down the offsite power from 115 to 13.8 and 4.16 kV. Its 13.8-kV winding supplies power to the auxiliary boiler and associated equipment; the 4.16-kV tertiary winding provides a backup source for the emergency 4.16-kV buses.

The unit auxiliary power distribution system feeds all unit auxiliary loads through 13.8-kV switchgear, 4.16-kV switchgear, 600-V load centers, 600-V motor control centers, and various ac and dc distribution panels. The system is divided into nuclear nonsafety-related and nuclear safety-related systems. The nuclear nonsafety-related auxiliary power distribution system feeds all non-Class 1E unit auxiliary loads. Under normal plant operating conditions, it is energized from the normal Station service transformer. During startup and normal shutdown conditions, it is energized from offsite power sources through reserve Station service transformers. A normal 125-V dc system, consisting of batteries, battery chargers, and distribution panels, provides a reliable source of power for protection, control, and instrumentation loads and dc motors under normal and emergency conditions of the plant. A ±24-V dc system provides a reliable source for the neutron monitoring system.

The nuclear safety-related auxiliary power distribution system supplies all Class 1E unit auxiliary loads. This system is divided into three independent divisions. Division I and Division II are independent redundant divisions and supply all nuclear safety-related auxiliary loads except the high pressure core spray (HPCS) system. The HPCS system and related equipment are supplied by Division III. All three divisions are normally energized from the offsite power sources through reserve Station service transformers. The auxiliary boiler transformer can be connected manually to act as a backup source for either the Division I or Division II supply.

Each of the three divisions of the nuclear safety-related auxiliary power distribution systems has its own independent standby diesel generator. In the event of a LOCA and/or LOOP, each division is energized from its own standby diesel generator. A 125-V emergency dc power system feeds all safety-related dc protection, control, and instrumentation loads and safety-related dc motors under normal operation of the plant as well as during emergency conditions. The system is divided into three independent divisions each consisting of its own battery, primary and backup battery chargers, switchgear, motor control centers, and distribution panels. Each division feeds the dc loads associated with the corresponding divisions of the nuclear safety-related auxiliary power distribution system.

Chapter 8 describes the electrical power system in detail.

1.2.5.2 Nuclear System Process Control and Instrumentation

Reactor Manual Control System

The reactor manual control system (RMCS) provides the means by which control rods are positioned from the control room for power control. The system operates valves in each hydraulic control unit to change control rod position. One control rod can be manipulated at a time. The RMCS includes the logic that restricts abnormal control rod movement (rod block) under certain conditions as a backup to procedural controls.

Recirculation Flow Control System

During normal power operation, a variable position discharge valve is used to control flow. Adjusting this valve changes the coolant flow rate through the core and thereby changes the core power level.

Neutron Monitoring System

The neutron monitoring system (NMS) is a system of in-core neutron detectors and out-of-core electronic monitoring equipment. The system provides indication of neutron flux, which can be correlated to thermal power level for the entire range of flux conditions that can exist in the core. The source range monitors (SRMs) and the intermediate range monitors (IRMs) provide flux level indications during reactor startup and low-power operation. The local power range monitors (LPRMs) and average power range monitors (APRMs) allow assessment of local and overall flux conditions during power range operation. The traversing in-core probe (TIP) system provides a means to calibrate the individual LPRM sensors. The NMS provides inputs to the RMCS to initiate rod blocks if preset flux limits are exceeded, and inputs to the reactor protection system to initiate a scram if other limits are exceeded.

Refueling Interlocks

A system of interlocks that restrict movement of refueling equipment and control rods when the reactor is in the refueling and startup modes is provided to prevent an inadvertent criticality during refueling operations. The interlocks back up procedural controls that have the same objective. The interlocks affect the refueling platform, refueling platform hoists, fuel grapple, and control rods.

Reactor Vessel Instrumentation

In addition to instrumentation for the nuclear safety systems and engineered safety features, instrumentation is provided to monitor and transmit information that can be used to assess conditions existing inside the reactor vessel and the physical condition of the vessel itself. This instrumentation monitors reactor vessel pressure, water level, coolant temperature, reactor core differential pressure, coolant flow rates, and reactor vessel head inner seal ring leakage.

Process Computer System

An online process computer is provided to monitor and log process variables and to make certain analytical computations. The nuclear measurement analysis and control rod worth minimizer (NUMAC RWM) prevents rod withdrawal under low-power conditions if the rod to be withdrawn is not in accordance with a preplanned pattern. The effect of the rod block is to limit the reactivity worth of the control rods by enforcing adherence to the preplanned rod pattern.

Chapter 7 describes these systems in detail.

1.2.5.3 Power Conversion Systems Process Control and Instrumentation

Pressure Regulator and Turbine Generator Control

The pressure regulator maintains control of the turbine control and turbine bypass valves to allow proper generator and reactor response to system load-demand changes while maintaining the nuclear system pressure essentially constant.

The turbine generator speed-load controls can initiate rapid closure of the turbine control valves (rapid opening of the turbine bypass valves) to prevent turbine overspeed on loss of the generator electric load.

Feedwater Control System

The feedwater control system automatically controls the flow of feedwater into the RPV to maintain the water within the vessel at predetermined levels. A three-element control system (main steam

flow rate, feedwater flow rate, and reactor vessel water level) is used to accomplish this function.

Chapter 10 describes the system in greater detail.

1.2.6 Radioactive Waste System

The disposal of radioactive wastes from the site is managed by waste systems designed to meet all applicable regulatory requirements, including 10CFR20, 10CFR50, 10CFR61, GDC 60, and Regulatory Guide (RG) 1.21.

There are three interrelated radioactive waste treatment systems: radioactive liquid waste, radioactive gaseous waste, and radioactive solid waste. These systems are described in Chapter 11.

The radioactive liquid waste (LWS) system collects and processes radioactive waste liquids generated during plant operation and refueling, either for recycle within the plant or for discharge offsite. The process operations available to treat the liquid wastes are filtration, evaporation, demineralization, and decantation. Process descriptions and flow charts illustrate the number and sequence of processing steps to be applied to each type of liquid waste.

Gaseous radwaste from the main condenser is processed through a recombiner to remove hydrogen, after which the gas is cooled, then dried. Waste gas is then passed through a charcoal bed and filter system, which holds up radioactive components and removes particulate matter before release. Contaminated drywell and building ventilation exhausts are processed by the standby gas treatment and ventilation systems.

The radioactive solid waste system provides holdup, packaging, and storage facilities for eventual offsite shipment and ultimate disposal of solid radioactive waste material. The process operations consist of volume reduction and solidification with asphalt of radioactive wastes such as LWS evaporator concentrates, spent resins, and filter sludges. Dry radioactive wastes such as contaminated paper, clothing, and tools are compacted and packaged. After processing, the solid waste materials are stored for additional decay and then shipped offsite for appropriate disposal. Shielding, as required during the processing and shipment of the solid wastes, is included in the planned operation of the solid waste system.

A description and flow diagram of the processing and handling sequences for the solid wastes generated onsite is provided in Chapter 11.

1.2.7 Fuel Handling and Storage Systems

1.2.7.1 New Fuel Storage

The new fuel facility is designed to prevent inadvertent criticality and load buckling of the new fuel assemblies. Both the new fuel storage vault and storage racks are designed to comply with Category I requirements.

The new fuel storage vault is designed with sufficient drainage to preclude flooding. The vault is also equipped with a monitoring system to warn of radiation level increases above normal operating conditions.

The design of the new fuel storage racks limits $k_{eff} \leq 0.90$ in the dry condition and $k_{eff} \leq 0.95$ in a flooded condition.

1.2.7.2 Spent Fuel Storage

The spent fuel storage racks are designed to maintain spent fuel in a space geometry that prevents criticality in normal and abnormal conditions. The racks are capable of withstanding maximum uplift forces generated without effect on the subcritical array. The design of the spent fuel racks will limit $k_{eff} \leq 0.95$ in normal and abnormal storage conditions. There is sufficient shielding, a cooling system, and radiation monitoring to prevent overheating and excessive personnel exposure. The spent fuel storage pool and racks are corrosion resistant, and adhere to Category I requirements.

1.2.7.3 Fuel Handling System

The fuel handling equipment includes the following:

1. Fuel inspection stand.
2. Fuel preparation machine.
3. 125-ton crane.
4. Refueling platform.
5. General purpose grapple.
6. Jib cranes.
7. Other related tools for fuel and reactor servicing.

All equipment conforms to applicable codes and standards.

1.2.7.4 Spent Fuel Pool Cooling and Cleanup System

The spent fuel cooling and cleanup (SFC) system provides removal of decay heat from the stored spent fuel and maintains specified

water temperature, purity, clarity, and level. This process prevents the spent fuel from overheating and the buildup of excessive radioactive materials in the cooling water, thereby minimizing radiation levels.

The system includes two heat exchangers, each of which is capable of removing the full decay heat from a normal refueling offload of spent fuel. A cross-connection to the RHR system provides additional emergency backup cooling and cooling during a full core offload.

Chapter 9 gives further details of the fuel handling and storage system.

1.2.8 Power Conversion System

Chapter 10 provides a detailed discussion of the following equipment systems.

1.2.8.1 Turbine Generator

The turbine is a 1,800-rpm tandem-compound, six-flow, single-stage reheat unit with an electrohydraulic governor control. The turbine generator has an emergency trip system for turbine overspeed. The output of the turbine generator is 1,165,663 kWe at turbine guarantee conditions with 2.0 in Hg abs backpressure and 0 percent makeup.

The generator is a direct driven, three-phase, 60-Hz, 25,000-V, 1,800-rpm hydrogen inner-cooled, synchronous generator rated at 1,348,400 kVA at 0.90 power factor, 0.58 short-circuit ratio at maximum hydrogen pressure of 75 psig.

1.2.8.2 Main Steam System

The main steam system delivers steam from the nuclear boiler system through four 26-/28-in OD steam lines to the turbine generator, turbine bypass valves, SJAEs, offgas preheaters, steam seal evaporator, and radwaste steam reboiler.

1.2.8.3 Main Condenser

The main condenser maintains 2.0 in Hg abs when operating at turbine guarantee conditions with 67.97°F circulating water inlet temperature. The condenser includes provisions for accepting steam bypassed around the turbine generator. Deaeration of condensate is accomplished in the condenser.

1.2.8.4 Main Condenser Air Removal System

The main condenser air removal system, using air ejectors for normal operation and vacuum hogging pumps for startup, evacuates gases from the main turbine and condenser during plant startup and maintains the condenser essentially free of gases during

operation. This system handles all inleakage of noncondensable gases through the turbine seals, condensate, feedwater, and steam systems, and noncondensables that are generated in the reactor by disassociation of water.

1.2.8.5 Turbine Gland Sealing System

The turbine gland sealing system provides mildly radioactive steam to the seals of the turbine throttle valve stem glands and the turbine shaft glands. The sealing steam is supplied by a clean steam reboiler using condensate. The unit auxiliary boiler provides an auxiliary steam supply for startup and when reactor heating steam is not available. The steam packing exhauster collects and condenses the air and steam mixture and discharges the air and other noncondensables to the plant exhaust duct to the atmosphere, using a motor-driven exhauster.

1.2.8.6 Steam Bypass System and Pressure Control System

A turbine bypass system is provided which passes steam directly to the main condenser under control of the pressure regulator. Steam is bypassed to the condenser whenever the reactor steaming rate exceeds the load passed to the turbine generator. The capacity of the turbine bypass system is 25 percent of the turbine rated steam flow. The pressure regulation system provides main turbine control valve and bypass valve flow demands to maintain a nearly constant reactor pressure during normal plant operation. It also provides demands to the recirculation system to adjust power levels by changing reactor recirculation flow rates.

1.2.8.7 Circulating Water System

The circulating water system provides the condenser with a continuous supply of cooling water. The circulating water system is a pumped closed loop system utilizing an air-cooled natural-draft cooling tower as a heat sink. Six one-sixth capacity circulating water pumps are provided to pump cooling water from the cooling tower basin through the main condenser and back to the top of the cooling tower. Makeup water is provided from Lake Ontario by the service water system.

1.2.8.8 Condensate and Feedwater Systems

The condensate and feedwater systems supply condensate from the condenser hotwell to the RPV. The condensate is pumped by two of the three condensate pumps through the full flow condensate demineralizer system, the intercooler of the air ejectors, and the steam packing exhauster to the condensate booster pumps. The condensate booster pumps pump the flow through three strings consisting of two drain coolers and five stages of low pressure heaters each. In addition, three heater drain pumps provide approximately one-third of the feedwater flow requirements. The last low-pressure heaters discharge to the suction of three

parallel motor-driven reactor feedwater pumps. The discharge of the reactor feedwater pumps passes through three one-third capacity parallel heaters and into the RPV. The feedwater flow is controlled by varying the feedwater flow control valve position.

1.2.8.9 Condensate Demineralizer System

A full flow, deep bed condensate demineralizer system complete with regeneration facilities, instrumentation, and semiautomatic controls is designed to ensure a constant supply of high-quality water to the reactor.

1.2.9 Nuclear Safety Systems and Engineered Safety Features

Chapters 3, 4, 5, 6, 7, 9, and 10 give further details for the following equipment and systems.

1.2.9.1 Reactor Protection System (RPS)

The RPS initiates a rapid, automatic shutdown (scram) of the reactor. It acts in time to prevent fuel cladding damage and any nuclear system process barrier damage following abnormal operational transients. The RPS overrides all operator actions and process controls and is based on a fail-safe design philosophy that allows appropriate protective action even if a single failure occurs.

1.2.9.2 Neutron Monitoring System (NMS)

The NMS is a system of in-core neutron detectors and out-of-core electronic monitoring equipment. The system provides indication of neutron flux, which can be correlated to thermal power level for the entire range of flux conditions that can exist in the core. The SRMs and the IRMs provide flux level indications during reactor startup and low power operation. The LPRMs and APRMs allow assessment of local and overall flux conditions during power range operation. The TIP system provides a means to calibrate the individual LPRM sensors. The NMS provides inputs to the RMCS to initiate rod blocks if preset flux limits are exceeded, and inputs to the RPS to initiate a scram if other limits are exceeded.

1.2.9.3 Control Rod Drive (CRD) System

When a scram is initiated by the RPS, the CRD system inserts negative reactivity necessary to shut down the reactor. Each control rod is individually controlled by a hydraulic control unit (HCU). When a scram signal is received, high-pressure water stored in an accumulator in the HCU or reactor pressure forces the control rod into the core.

1.2.9.4 Control Rod Drive Housing Supports

CRD housing supports are located underneath the reactor vessel near the control rod housings. The supports limit the travel of a control rod in the event that a control rod housing is ruptured. The supports prevent a nuclear excursion as a result of a housing failure and thus protect the fuel barrier.

1.2.9.5 Control Rod Velocity Limiter

A control rod velocity limiter is attached to each control rod to limit the velocity at which a control rod can fall out of the core should it become detached from the CRD. This action limits the rate of reactivity insertion resulting from a rod drop accident. The limiters contain no moving parts.

1.2.9.6 Nuclear System Pressure Relief System

A pressure relief system consisting of safety/relief valves (SRVs) mounted on the main steam lines is provided to prevent excessive pressure inside the nuclear system from operational transients or accidents. The SRV discharge steam is directed to the suppression pool within the primary containment.

1.2.9.7 Reactor Core Isolation Cooling (RCIC) System

The RCIC system provides makeup water to the RPV when the vessel is isolated. The RCIC system uses a steam-driven turbine-pump unit and automatically operates to maintain adequate water level in the RPV for events defined in Section 5.4.6.1.

1.2.9.8 Emergency Core Cooling Systems (ECCS)

Four ECCSs are provided to maintain fuel cladding below the temperature limit in 10CFR50.46 in the event of a breach in the RCPB that results in a loss of reactor coolant. The systems are as follows:

High Pressure Core Spray (HPCS)

The HPCS system provides and maintains an adequate coolant inventory inside the RPV to limit fuel cladding temperatures in the event of breaks in the RCPB. The system is initiated by either high pressure in the drywell or low water level in the vessel. It operates independently of all other systems over the entire range of pressure differences from greater than normal operating pressure to zero. The HPCS cooling decreases vessel pressure to enable the low pressure cooling systems to function. The HPCS system pump motor is powered by an onsite diesel generator if offsite power is not available. The system may also be used as a backup for the RCIC system.

Automatic Depressurization System (ADS)

The ADS rapidly reduces RPV pressure in a LOCA situation in which the HPCS system fails to maintain the RPV water level. The depressurization provided by the system enables the low-pressure ECCS to deliver cooling water to the RPV. The ADS uses some of the relief valves that are part of the nuclear system pressure relief system. The automatic relief valves are arranged to open on conditions indicating both that a break in the RCPB has occurred and that the HPCS system is not delivering sufficient cooling water to the RPV to maintain the water level above a preselected value. Setpoints are discussed in Section 5.2.2. The ADS is not activated unless either the low pressure core spray (LPCS) or low pressure coolant injection (LPCI) pumps are operating. This is to ensure that adequate coolant is available to maintain reactor water level after the depressurization.

Low Pressure Core Spray (LPCS)

The LPCS system consists of one independent pump and valves and piping to deliver cooling water to a spray sparger over the core. The system is actuated by either low water level in the reactor vessel or high pressure in the drywell, but water is delivered to the core only after RPV pressure is reduced. This system provides the capability to cool the fuel by spraying water into each fuel channel. The LPCS loop functioning in conjunction with the ADS or HPCS can provide sufficient fuel cladding cooling following a LOCA.

Low Pressure Coolant Injection (LPCI)

LPCI is an operating mode of the RHR system, but is discussed here because the LPCI mode acts as an engineered safety feature in conjunction with the other ECCSs. LPCI uses the pump loops of the RHR to inject cooling water into the RPV. LPCI is actuated by either low water level in the reactor vessel or high pressure in the drywell, but water is delivered to the core only after RPV pressure is reduced. LPCI operation provides the capability of core reflooding, following a LOCA, in time to maintain the fuel cladding below the prescribed temperature limit.

1.2.9.9 Containment Systems

Primary Containment

The primary containment is a Mark II design that incorporates a drywell pressure suppression system and utilizes a large reservoir of water to function as a heat sink to absorb energy.

1. Functional Design The primary containment is a steel-lined reinforced concrete structure. It consists of a conical drywell chamber above a cylindrical suppression pool chamber separated by a drywell floor. This floor contains a piping system which would direct

drywell steam into the suppression chamber reservoir in the event of a LOCA.

2. Heat Removal The containment heat removal system is summarized in Section 1.2.9.14.

3. Containment Spray The containment spray system consists of two redundant subsystems, each with its own full-capacity spray header. Each subsystem is supplied from a separate redundant RHR subsystem.

4. Combustible Gas Control The containment combustible gas control system is summarized in Section 6.2.5.

1.2.9.10 Containment and Reactor Vessel Isolation Control System

The primary containment and RPV isolation control system automatically initiates closure of isolation valves to close off all process lines that are potential leakage paths for radioactive material to the environs. This action is taken upon indication of a breach in the RCPB.

1.2.9.11 Main Steam Isolation Valves (MSIV)

Although all pipelines that both penetrate the primary containment and offer a potential release path for radioactive material have redundant isolation capabilities, the main steam lines, because of their large size and large mass flow rates, are given special isolation consideration. Automatic isolation valves are provided in each main steam line (MSIVs). Each is closed by spring force and pneumatic force and opened by pneumatic force. These valves fulfill the following objectives:

1. Prevent excessive damage to the fuel barrier by limiting the loss of reactor coolant from the RPV resulting from either a major leak from the steam piping outside the primary containment or a malfunction of the pressure control system resulting in excessive steam flow from the RPV.

2. Limit the release of radioactive materials by isolating the RCPB in case of a gross release of radioactive materials from the fuel to the reactor cooling water and steam.

3. Limit the release of radioactive materials by closing the containment barrier in case of a major leak from the nuclear system inside the containment.

1.2.9.12 Main Steam Flow Restrictors

A venturi-type flow restrictor is installed in each steam line. These devices limit the loss of coolant from the RPV before the

MSIVs are closed in case of a main steam line break outside the primary containment.

1.2.9.13 Main Steam Radiation Monitoring System

The main steam radiation monitoring system consists of four gamma radiation monitors located externally to the main steam lines just outside the primary containment in the main steam tunnel. The monitors are designed to detect a gross release of fission products from the fuel. On detection of high radiation, the trip signals generated by the monitors are used by the RPS to initiate a reactor scram and to close the MSIVs.

1.2.9.14 Residual Heat Removal (RHR) System

The RHR system is placed in operation to limit the temperature of the water in the suppression pool and of the atmospheres in the drywell and suppression chamber following a design basis LOCA, to control the pool temperature during normal operation of the SRVs and the RCIC system, and to reduce the pool temperature following an isolation transient. In the containment cooling mode of operation, the RHR main system pumps take suction from the suppression pool and pump the water through the RHR heat exchangers where cooling takes place by transferring heat to the service water. The coolant is then discharged back to the suppression pool, the drywell spray header, the suppression chamber spray header, or the RPV.

1.2.9.15 Ventilation Exhaust Radiation Monitoring System

Permanently-installed process and area radiation monitors provide indications and alarms on airborne radiation in the reactor building ventilation system, drywell atmosphere, fuel storage and refueling areas, and control room atmosphere. Additionally, connections are provided in the reactor building ventilation system ductwork for continuous airborne monitors (CAMs).

1.2.9.16 Standby Gas Treatment System (SGTS)

The SGTS processes exhaust air from various plant systems to limit the release of airborne radioactivity, maintaining offsite dose rates below the specified limits. During a DBA, the SGTS is automatically actuated. When high radiation levels are sensed in any exhaust system connected to the SGTS it is automatically placed in operation.

The SGTS consists of two identical, parallel but physically separated air filter train assemblies. Each assembly is capable of handling the maximum design air flow rate.

1.2.9.17 Safety-Related Electrical Power Systems

A standby power supply system is provided for the operation of emergency systems and engineered safety features (ESFs) during

and following the shutdown of the reactor when the preferred power supply is not available. The standby power supply system consists of three standby diesel generators. One generator is dedicated to each of the three divisions of the safety-related electric power distribution system feeding each Class 1E load group. Any two of the three standby diesel generators have sufficient capacity to start and supply all needed ESFs and emergency shutdown loads in case of a LOCA and/or LOOP. The standby diesel generator fuel oil storage tanks are sized to hold a 7-day supply of fuel oil based on the engine running continuously at full load. A LOCA and/or LOOP signal initiates start of the standby diesel generators and the generators pick up the loads in a programmed sequence. Standby diesel generators are independent and feed separate load groups through separate, physically- and electrically-isolated distribution systems.

Failure of any one unit will not jeopardize the capability of the remaining standby diesel generators to start and run the required shutdown system and ESF loads.

A 125-V emergency dc power system feeds all safety-related dc protection, control and instrumentation loads, and safety-related dc motors under normal operation of the plant as well as during emergency conditions. The system is divided into three redundant divisions, each consisting of its own battery, primary and backup battery chargers, switchgears/motor control centers, and distribution panels. Each division feeds dc loads associated with corresponding divisions of the safety-related electric power distribution system. Batteries and battery chargers are redundant and feed separate load groups through separate and isolated distribution systems, and failure of any one unit will not jeopardize the capability of remaining units to feed associated loads.

1.2.9.18 Standby Liquid Control System (SLCS)

Although not intended to provide prompt reactor shutdown, as the control rods are, the SLCS provides a redundant, independent, and alternate way to bring the nuclear fission reaction to subcriticality and to maintain subcriticality as the reactor cools. The system makes possible an orderly and safe shutdown in the event that not enough control rods can be inserted into the reactor core to accomplish shutdown in the normal manner. The system is sized to counteract the positive reactivity effect from rated power to the cold shutdown condition.

1.2.9.19 Safe Shutdown from Outside the Control Room

In the event that the control room becomes inaccessible, the reactor can be brought from power range operation to cold shutdown conditions by use of necessary controls located in the remote shutdown room.

1.2.9.20 Main Control Room Heating, Ventilating and Air Conditioning (HVAC) System

The main control room HVAC system provides and maintains an environment necessary for the safety and comfort of control room personnel during shutdown of the plant and in the event of a LOCA.

1.2.9.21 Redundant Reactivity Control System (RRCS)

The RRCS determines if there is an existing transient that exceeds certain RPV pressure and water level parameters and immediately activates anticipated transient without scram (ATWS) prevention equipment. If the logic has determined that a controlled shutdown is not occurring, the RRCS activates ATWS mitigation equipment.

1.2.10 Cooling Water and Auxiliary Systems

Chapter 9 provides a detailed discussion of these systems.

1.2.10.1 Reactor Building Closed Loop Cooling Water (RBCLCW) System

The RBCLCW system is a demineralized water, closed-circuit heat transfer system that consists of three 50-percent capacity pumps and heat exchangers, along with appropriate controls and instrumentation to ensure adequate cooling capacity for reactor plant auxiliary systems and components during normal plant operations. Heat removed from components by the RBCLCW system is transferred to the service water system.

1.2.10.2 Turbine Building Closed Loop Cooling Water (TBCLCW) System

The TBCLCW system is a demineralized water, closed-circuit heat transfer system that consists of three 50-percent capacity pumps and heat exchangers, along with appropriate controls and instrumentation to ensure adequate cooling capacity for the turbine building and radwaste building auxiliary systems and components during normal plant operation. Heat removed from components by the TBCLCW water system is transferred to the service water system.

1.2.10.3 Service Water System

The service water system provides cooling water to various essential and nonessential components throughout the plant. Essential components are serviced by two 100-percent redundant subsystems. The nonessential components will be automatically isolated upon receipt of a LOCA signal coincident with a LOOP. The service water pumps take their suction from Lake Ontario via the screenwell complex and intake tunnels. After passing through

the system, the discharge is returned to the lake and to the circulating water system as makeup.

1.2.10.4 Ultimate Heat Sink

The ultimate heat sink is Lake Ontario, which provides water to the intake/discharge tunnels and is available at the screenwell building for plant use.

1.2.10.5 Plant Chilled Water System

The plant chilled water system consists of two subsystems, one serving the turbine, normal switchgear, and radwaste buildings, and one serving the control building. The subsystems provide space cooling by distributing chilled water through cooling coils located in air moving units. The control building subsystem is an essential system designed to provide chilled water for cooling during all modes of plant operation. Each subsystem contains mechanical refrigeration water chillers, circulation pumps, piping valves, cooling coils, accessories, instrumentation, and controls.

1.2.10.6 Heating, Ventilating, and Air Conditioning (HVAC) Systems

Individual HVAC systems are provided throughout the plant to maintain indoor temperature and humidity design conditions as required for optimum performance of plant equipment and, where applicable, for human comfort.

1.2.10.7 Process Sampling

The process sampling system consists of the reactor plant, turbine plant, and radwaste sampling subsystems. These subsystems are composed of the necessary piping, valves, coolers, instrumentation, and analyzers to draw and analyze samples of various plant process streams. Representative samples are taken automatically and/or manually for online and laboratory analyses of pH, conductivity, turbidity, oxygen, hydrogen, gaseous activity, fission product activity, dissolved gas concentration, and various metallic concentrations (such as copper, iron, and silica).

1.2.10.8 Condensate Makeup and Drawoff System

The condensate makeup and drawoff system consists of two storage tanks, piping, and instrumentation. It receives drawoff water from and supplies makeup water to the main condenser and fuel pool, and provides makeup of reactor coolant inventory for the RCIC and HPCS systems. Water in the condensate storage tanks is replenished from the makeup water treatment system.

1.2.10.9 Water Treatment and Makeup Water Systems

The water treatment system is composed of a water filter, an activated carbon filter, an acid exchanger unit, a forced-draft degasifier, a base anion exchanger, and a mixed-bed ion exchange demineralizer. The system purifies lake water supplied from the service water system and delivers it to the demineralized water storage tanks.

The makeup water system provides demineralized makeup water from the storage tanks for the power conversion system and the TBCLCW and RBCLCW systems. In addition, the makeup water system satisfies various miscellaneous plant requirements for demineralized water, including the suppression pool and the spent fuel pool.

1.2.10.10 Domestic Water and Sanitary Drains and Disposal Systems

Domestic Water System

Domestic water for drinking and to satisfy the flow and pressure requirements of all installed plumbing fixtures is supplied from an existing city main. Water quality is in accordance with applicable standards promulgated by the State of New York.

Sanitary Drains and Disposal System

Raw sanitary waste from Unit 2 is directed to the Nine Mile Point Unit 1 sanitary waste treatment plant. This plant conforms to all applicable local, state, and federal discharge limitations.

1.2.10.11 Compressed Air Systems

The compressed air systems are composed of a service and instrument air system and a breathing air system.

Service and Instrument Air System

Three air compressors, each discharging through an aftercooler, a filter, and an air receiver into a common header, supply all the service and instrument air required by the plant. The service air is taken directly from the common header. The instrument air is passed through one of two 100-percent capacity air dryers and then through one of two 100-percent capacity air filters before delivery to the various plant instruments.

Breathing Air System

A single air compressor discharges through an aftercooler to a receiver. Air from this receiver is filtered to comply with OSHA requirements for breathing air.

1.2.10.12 Auxiliary Steam System

An auxiliary steam system furnishes a separate and independent steam supply. Process steam is generated in high-voltage, electrode boilers and distributed throughout the plant by an auxiliary steam header. Auxiliary steam is required for main turbine shaft sealing steam during startup and for the radwaste building during plant shutdown.

1.2.10.13 Standby Diesel Generator Fuel Oil Storage and Transfer System

The standby diesel generator fuel oil storage and transfer system supplies fuel oil for operation of the standby diesel generators. This system is an essential system and is capable of supplying fuel oil during all modes of plant operation, including a LOOP coincident with a LOCA.

1.2.10.14 Fire Protection System

The fire protection system consists of fire hydrants, hose stations, and automatic sprinkler and deluge systems. Where required, automatic or manually-actuated carbon dioxide (CO_2), foam, or Halon fire suppression systems are provided. Automatic fire detectors are provided in selected areas. Portable fire extinguishers and fire hose reels are located throughout the plant.

1.2.10.15 Communication Systems

The Station communication systems are designed to provide reliable communication between all essential areas of the Station and to locations remote from the Station during normal and emergency conditions and under maximum potential noise levels. This is achieved through five different communication systems as follows:

1. A dial telephone system for voice communication between selected office areas and selected locations inside and outside the Station. The dial telephone system is connected to the telephone tie system for offsite communication, including communication with local law enforcement authorities and the local fire department.

2. A portable radio communication system for communication between Station personnel and NMPC personnel located outside the Station in case the dial telephone system between the Station and the points outside the Station becomes inoperable. The radios are powered by rechargeable batteries and are independent of any electrical system of the plant.

3. A page party/public address (PP/PA) communication system with five party channels and one page channel

for communication between all buildings and locations within the plant, even under extremely noisy conditions. This is also used for the emergency alarm and evacuation system and is powered from an uninterruptible power supply (UPS).

4. A separate maintenance and calibration communication (M/CC) system for use in areas requiring communication for testing, instrument calibration, maintenance, and for use during construction and startup.

5. One of the channels of the M/CC system is capable of being operated as a sound-powered communication (SPC) system. This system can be used in case of a total loss of electric power to the PP/PA and M/CC systems. This system requires no plant electric power and works through M/CC system wiring.

The design of the communication systems permits routine surveillance and testing without disrupting normal communication facilities. The paging system is electrically supervised, permitting immediate corrective action to be taken if the line becomes inoperative.

1.2.10.16 Lighting Systems

The Station lighting systems are designed to provide adequate lighting in all necessary areas of the Station during both emergency and normal operating conditions. This is achieved through the following lighting subsystems:

1. Normal Station lighting system.

2. Emergency lighting system.

3. Essential lighting system.

4. Egress lighting system.

5. 8-hour battery-pack lighting system.

The normal Station lighting system provides adequate lighting in all areas of the Station under normal operating conditions. This is fed from the Station normal 600-V load centers, through the main lighting distribution panels, dry-type transformers, and subdistribution panels, except that panel 2LAR-PNL200 is normally fed from offsite power sources in a way similar to the emergency lighting system described below.

The emergency lighting system provides adequate lighting required for operating the safety-related equipment during emergency conditions in the control room, diesel generator rooms, emergency switchgear areas, and relay and computer room. This is treated as a Class 1E system except for the lighting fixtures. The

lighting fixtures are seismically supported. The emergency lighting system is divided into three separate divisions corresponding to Divisions I, II, and III of the plant emergency ac distribution system and is fed from the corresponding Class 1E load centers/motor control centers through the main lighting distribution panels, dry-type distribution transformers, and subdistribution panels. In a case of a loss of offsite power (LOOP), the emergency lighting system is automatically connected to the emergency diesel generators. The emergency lighting system fixtures are constantly energized. The essential lighting system provides partial lighting for certain critical areas of the Station requiring continuous lighting, such as the control room, relay and computer room, standby diesel generator rooms, emergency switchgear rooms, service water pump room, and for passageways to and from areas where safety-related equipment is located, with the exception of those areas and passageways where 8-hr battery-pack lighting is provided to meet the requirements of 10CFR50, Appendix R. In these areas and passageways, access and egress lighting is provided by the 8-hr battery-pack lighting in the event of loss of normal lighting. The essential lighting system receives power from the Station normal UPS system and is fed through main and subdistribution panels. The essential lighting fixtures powered by the UPS system are constantly energized.

The egress lighting system provides adequate lighting for all egress signs inside the plant. This is designed as a separate system specifically for the inside building egress emergency conditions in accordance with OSHA requirements. The egress lighting system receives power from the Station normal UPS system as part of the essential lighting distribution system. The egress lighting fixtures are constantly energized.

The 8-hr battery-pack lighting provides illumination in all areas required for operation of any safe shutdown equipment and in access and egress routes thereto in case of a fire. The 8-hr battery-pack lighting also provides required illumination for access/egress to certain areas of the plant if the normal lighting in these areas is not available.

Fluorescent, incandescent, and high-pressure sodium lamps are used for Station lighting. Fluorescent lamps are used for all offices and for most of the operating areas such as the control room, relay and computer room, and emergency and normal switchgear rooms. Incandescent lamps are used in the primary containment, primary containment access hatches, main steam tunnel, new fuel storage vault, spent fuel pool filter room, and in all other areas where lighting is infrequently required. High-pressure sodium lamps are used for all high bay lighting such as in the turbine building and reactor building general areas.

1.2.11 References

1. Niagara Mohawk Power Corporation. Preliminary Safety Analysis Report, Volume I, Nine Mile Point Nuclear Station - Unit 2, June 1972.

2. Niagara Mohawk Power Corporation. Environmental Report (Construction Permit Stage), Nine Mile Point Nuclear Station - Unit 2, June 1972.

3. Oswego County Planning Board. Preliminary Land Use Plan, August 1976.

4. Gulf & Western Topical Report. G&W-FSD 2538, Nuclear Main Steam Isolation Valve Systems, January 1979.

5. General Electric Licensing Topical Report. General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A (latest approved revision).

6. General Electric Licensing Topical Report. General Electric Standard Application for Reactor Fuel (U.S. Supplement), NEDE-24011-P-A-US (latest approved revision).

1.3 COMPARISON TABLES

1.3.1 Comparison with Similar Facility Designs

This section highlights the principal design features of Unit 2 and compares its major features with other BWR facilities. The design of Unit 2 is based on proven technology attained during the development, design, construction, and operation of BWRs of similar types. The data, performance characteristics, and other information presented here represent the current design.

Tables 1.3-1 through 1.3-7 compare Unit 2 with Washington Public Power Supply System (WPPSS) 2, Zimmer 1, and La Salle County Station 1 and 2 design characteristics for the following:

1. Nuclear steam supply system (NSSS).
2. Engineered safety features.
3. Containment design.
4. Electrical power systems.
5. Radioactive waste management systems.
6. Power conversion systems.
7. Structural design.

These comparisons were considered valid at the time the operating license was issued.

1.3.2 Comparison of Final and Preliminary Design Information

Significant changes in procedures or materials used in the design of Unit 2 since the Unit 2 PSAR are listed in Table 1.3-8 for NSSS and Table 1.3-9 for the balance of plant. Each change is cross-referenced to the primary FSAR section that discusses the item or system. Each of these changes was reviewed and approved in accordance with administrative procedures and either it does not represent a change to the principal architectural and engineering criteria for the design, or the NRC has been notified previously of the change.

1.3.3 Reference

1. General Electric Licensing Topical Report. General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A (latest approved revision).

TABLE 1.3-1

COMPARISON OF NUCLEAR STEAM SUPPLY SYSTEM DESIGN CHARACTERISTICS

	Nine Mile Point Unit 2	WPPSS Unit 2	Zimmer Unit 1	La Salle Units 1, 2	
THERMAL AND HYDRAULIC DESIGN (Section 4.4)					
Rated power, MWt	3,323	3,323	2,436	3,293	
Design power, MWt (ECCS design basis)	3,463	3,468	2,550	3,434	
Steam flow rate, millions lb/hr	14.263	14.295	10.477	14.166	
Core coolant flow rate, millions lb/hr	108.5	108.5	78.5	106.5	
Feedwater flow rate, millions lb/hr	14.564	14.256	10.477	14.127	
System pressure, nominal in steam dome, psia	1,020	1,020	1,020	1,020	
Average power density, kW/l	49.15	49.15	50.51	50.0	21
Minimum critical power flux ratio (MCPR)	1.24	1.24	1.24	1.28	
Coolant enthalpy at core inlet, Btu/lb	527.5	527.6	527.4	527.1	
Core max exit voids within assemblies	76.2	79	75	76	
Core average exit quality, % steam	13.10	13.5	13.2	13.2	
Feedwater temperature, °F	420	420	420	420	
Design Power Peaking Factor					
Maximum relative assembly power	1.40	1.40	1.40	1.40	
Axial peaking factor	1.40	1.4	1.4	1.40	

Nuclear Design (First Core)

See [4].

TABLE 1.3-1 (Cont)

	Nine Mile Point Unit 2	WPPSS Unit 2	Zimmer Unit 1	La Salle Units 1, 2	
CORE MECHANICAL DESIGN (Sections 4.2 and 7.6)					
Fuel Assembly					21
See (4).					
Fuel Rods					21
See (4).					
Fuel Pellets					21
See (4).					
Fuel Channel					21
See (4).					
Core Assembly					
See (4).					21

TABLE 1.3-1 (Cont)

	Nine Mile Point Unit 2	WPPSS Unit 2	Zimmer Unit 1	La Salle Units 1, 2
Reactor Control System				
Method of varying reactor power	Movable control rods; variable forced coolant flow			
No. movable control rods	185	185	137	185
Type of control roddrives	Bottom entry; locking piston			
Shape of movable control rods	Cruciform	Cruciform	Cruciform	Cruciform
Pitch of movable control rods	12.0	12.0	12.0	12.0
Control material in movable rods	B_4C granules compacted in SS tubes			
Type of temporary reactivity control for initial core	Burnable poison; gadolinia-urania fuel rods			
Incore Neutron Instrumentation				
Local power range monitors (LPRM)				
Total LPRM detectors	172	172	124	172
No. of incore LPRM penetrations	43	43	31	43
No. of LPRM detectors per penetration	4	4	4	4
Range	Approximately 1% power to 125% power			
Average power range monitors (APRM)				
No. detectors	6(a)	6(a)	6(a)	6(a)
Range	Approximately 1% power to 125% power			
Source range monitors (SRM)				
No. detectors	4	4	4	4
Range	Source to 0.001% power			
Intermediate range monitors (IRM)				
No. detectors	8	8	8	8
Range	0.001-10% power	0.001-10% power	0.001-10% power	0.001-10% power
No. flux-mapping neutron detectors	5	5	4	5
No. and type of incore neutron sources	7Sb-Be	7Sb-Be	5Sb-Be	7Sb-Be
REACTOR VESSEL DESIGN (Section 5.3)				
Material	Low-alloy steel/ stainless clad	Carbon steel/ stainless clad	Low-alloy steel/stainless	Carbon steel/ stainless clad
Design pressure, psig	1,250	1,250	1,250	1,250
Design temperature, °F	575	575	575	575
Inside diameter, ft-in	20-11	20-11	18-2	20-11
Inside height, ft-in	72-5	72-11	69-10	72-11

TABLE 1.3-1 (Cont)

	Nine Mile Point Unit 2	WPPSS Unit 2	Zimmer Unit 1	La Salle Units 1, 2
Minimum base metal thickness (cylindrical section), in	6.1875	6.75	5.375	6.75
Minimum cladding thickness, in	1/8	1/8	1/8	1/8
REACTOR COOLANT RECIRCULATION DESIGN				
(Sections 5.1, 5.2, and 5.4)				
No. recirculation loops	2	2	2	2
Design pressure				
Inlet leg, psig	1,250	1,250	1,250	1,250
Outlet leg, psig	1,650(2) 1,550(3)	1,650(2) 1,550(3)	1,675(2) 1,575(3)	1,650(2) 1,550(3)
Design temperature, °F	575	575	575	575
Pipe diameter, in	24	24	20	24
Pipe material, AISI	316K	304/316	304/316	304/316
Recirculation pump flow rate, gpm	47,200	47,250	32,500	47,250
No. jet pumps in reactor	20	20	20	20
MAIN STEAM LINES				
(Section 5.4)				
No. steam lines	4	4	4	4
Design pressure, psig	1,250	1,250	1,250	1,250
Design temp, °F	575	575	575	575
Pipe diameter, in	26/28	26	24	26
Pipe material	Carbon steel	Carbon steel	Carbon steel	Carbon steel

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(1)Channels of monitors from LPRM detectors
(2)Pump and discharge piping to and including the discharge block valve.
(3)Discharge piping from discharge block valve to vessel.
(4)General Electric Licensing Report. General Electric Standard Application for Reactor Fuel. NEDE-24011-P-A (latest approved revision). 21

TABLE 1.3-2

COMPARISON OF ENGINEERED SAFETY FEATURES
DESIGN CHARACTERISTICS

	Nine Mile Point Unit 2	WPPSS Unit 2	Zimmer Unit 1	La Salle Units 1, 2
EMERGENCY CORE COOLING SYSTEMS (Systems sized on design power) (Section 6.3)				
Low Pressure Core Spray System				
No. loops	1	1	1	1
Flow rate, gpm	6,350 @ 128 psid	6,250 @ 122 psid	4,625 @ 119 psid	6,250 @ 122 psid
High Pressure Core Spray System				
No. loops	1	1	1	1
Flow rate, gpm	1,550 @ 1,130 psid 6,350 @ 200 psid	1,650 @ 1,110 psid 6,250 @ 200 psid	1,330 @ 1,110 psid 4,725 @ 200 psid	1,650 @ 1,110 psid 6,250 @ 200 psid
Automatic Depressurization System				
No. systems	1	1	1	1
No. relief valves	7	7	6	7
Low Pressure Coolant Injection[(1)]				
No. LPCI systems	1	1	1	1
No. pumps	3	3	3	3
Flow rate, gpm/pump	7,450 @ 26 psid	7,450 @ 20 psid	5,050 @ 20 psid	7,067 @ 20 psid
AUXILIARY SYSTEMS				
Residual Heat Removal System (Section 5.4.7)				
No. Loops	2	2	2	3
No. Pumps	2	2	2	3
Flow rate, gpm/pump[(2)]	7,450	7,450	5,050	7,450
Duty, millions Btu/hr/heat exchanger[(3)]	41.6	41.6	30.8	46.6
No. heat exchangers	2	2	2	2
Primary containment cooling mode flow rate, gpm[(4)]	7,450	7,450	5,050	8,400

TABLE 1.3-2 (Cont)

	Nine Mile Point Unit 2	WPPSS Unit 2	Zimmer Unit 1	La Salle Units 1, 2
Service Water System (Section 9.2.1)				
Flow rate, gpm/RHR heat exchanger	7,400	7,400	5,000	7,400
No. pumps	6	4	4[4]	4
Reactor Core Isolation Cooling System (Section 5.4.6)				
Flow rate, gpm	600 @ 1,173 psia reactor pressure	600 @ 1,120 psid	400 @ 1,120 psid	600 @ 1,120 psia reactor pressure
Fuel Pool Cooling and Cleanup System (Section 9.1.3)				
Capacity, millions Btu/hr	15.0	7.6	6.9	8.0
Standby Gas Treatment System (Section 9.3.5)				
Charcoal bed design, lb charcoal	2 independent trains 1,360 lb/train total 2,720 lb total	2 trains	2 trains, 2,000 lb/train 4,000 lb/train	2 independent trains, 3,700 lb/train 7,400 total lb
Design efficiencies, %				
Elemental iodine	99.0	99.0	99.0	90.0
Organic iodine	99.0	99.0	99.0	90.0
0.3u particles	99.97	99.97	99.0	99.97
System flow, cfm	4,000[5] scfm/train	4,000 scfm/train	2,300 scfm/train	4,000 scfm/train

[1] A mode of residual heat removal system.
[2] Capacity during reactor flooding made with two or three pumps running.
[3] Heat exchanger duty at 20 hr following reactor shutdown.
[4] Includes HPCS service water pumps.
[5] Nominal system flow rate.

TABLE 1.3-3

COMPARISON OF CONTAINMENT DESIGN CHARACTERISTICS

	Nine Mile Point Unit 2	WPPSS Unit 2	Zimmer Unit 1	La Salle Units 1, 2
Primary Containment[1] (Section 3.8)				21
Type	Over & under pressure suppression Mark II	Over & under pressure suppression Mark II	Over & under pressure suppression Mark II	Over & under pressure suppression Mark II
Construction	Reinforced concrete steel liner	Steel free-standing	Concrete pre-stressed steel liner	Concrete post-tensioned steel liner
Drywell	Frustum of cone, upper portion	Frustum of cone, upper portion	Frustum of cone, upper portion	Frustum of cone, upper portion
Pressure suppression chamber	Cylindrical lower portion	Cylindrical lower portion with elliptical bottom	Cylindrical lower portion	Cylindrical lower portion
Pressure suppression chamber - internal design pressure, psig	45	45	45	45
Pressure suppression chamber - external design pressure, psig	4.7	2	2	5
Drywell - internal design pressure, psig	45	45	45	45
Drywell - external design pressure, psig	4.7	2	2	5
Drywell free volume, ft^3	303,418	200,540[2]	180,000[3]	221,518
Pressure suppression chamber free volume (min), ft^3	192,028	144,184[4]	93,000	166,400 21
Pressure suppression pool water volume, ft^3	154,794[5]	112,197	102,120	109,096

TABLE 1.3-3 (Cont)

	NMP Unit 2	WPPSS Unit 2	Zimmer Unit 1	La Salle Units 1, 2
Submergence of vent pipe below suppression pool surface, ft	9.5 min 11.0 max	11.67 min 12.00 max	10	12
Design environmental temperature of drywell, °F	340	340	340	340
Design environmental temperature of pressure suppression chamber, °F	270	275	275	275
Downcomer vent pipe pressure loss factor	1.37(6)	1.9	2.17	1.9
Break area/total vent area	0.0108	0.0105	0.008	0.0105
Calculated maximum pressure after blowdown to drywell, psig	39.7	34.7	40.4	34
Calculated maximum pressure in suppression chamber, psig	34.0	28.0	35.6	28
Calculated maximum initial pressure suppression pool temperature rise, °F	50	35	35	50
Leakage rate, % free volume/day at 45 psig and 340°F	1.1 @ 200°F	0.5 @ 200°F	0.635	0.5
Reactor Building (Sections 3.8.4, 6.2)				
Type	Controlled leakage, elevated release(7)	Controlled leakage, elevated release	Controlled leakage, elevated release	Controlled leakage, elevated release
Construction				
Lower levels	Reinforced concrete	Reinforced concrete	Reinforced concrete	Reinforced concrete
Upper levels	Steel superstructure and siding	Steel superstructure and siding	Steel superstructure and siding	Steel superstructure and siding
Roof	Steel decking	Steel decking	Steel decking	Steel decking

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TABLE 1.3-3 (Cont)

	Nine Mile Point Unit 2	WPPSS Unit 2	Zimmer Unit 1	La Salle Units 1, 2
Internal design pressure, psig	0.25	0.25	0.25	0.25
Design inleakage rate, % free volume/day at 0.25 in H_2O	100	100	100	100

(1) Where applicable, containment parameters are based on design power.
(2) Maximum water in suppression pool.
(3) Includes the vent volume.
(4) Maximum value.
(5) At high water level.
(6) Includes entrance and pipe friction.
(7) For accident conditions.

TABLE 1.3-4

COMPARISON OF ELECTRICAL POWER SYSTEM DESIGN CHARACTERISTICS

	Nine Mile Point Unit 2	WPPSS Unit 2	Zimmer Unit 1	La Salle Units 1, 2
Offsite Power System (Section 8.2)				
Outgoing lines (No.-rating)	1-345-kV	1-500-kV	3-345-kV	2-345-kV (per unit)
Incoming lines (No.-rating)	2-115-kV	1-230-kV 1-115-kV	1-69-kV 1-345-kV	2-345-kV (per unit)
Onsite ac Power System (Section 8.3.1)				
Normal station service transformers	1	2	1 (unit auxiliary)	1 per unit
Reserve station service transformers	3[1]	2	2	1 (system aux)
Standby diesel generators	3[2]	3[2]	3	3[3]
4,160-V ESF buses	3[2]	3[2]	3	3
ESF buses	3-600-V[2]	3-480-V[2]	5-480-V	4-480-V
dc Power Supply (Section 8.3.2)				
Batteries (No.-volts)	6-125-V[4] 4±24-V	4-24-V 5-125-V[4] 1-250-V	3-125-V 1-250-V	3-125-V 1-250-V
Buses (No.-volts)	6-125-V[4] 2±24-V	2-24-V 5-125-V[4] 1-250-V	3-125-V 1-250-V	3-125-V 1-250-V

[1] Includes one auxiliary boiler transformer.
[2] Includes an HPCS diesel generator.
[3] Five total for 2 units. One serves either unit.
[4] HPCS battery and bus included.

TABLE 1.3-5

COMPARISON OF RADIOACTIVE WASTE MANAGEMENT
DESIGN CHARACTERISTICS

	Nine Mile Point Unit 2	WPPSS Unit 2	Zimmer Unit 1	La Salle Units 1,2
Gaseous Radwaste (Section 11.3)				
Design basis, noble gases, uci/sec	100,000 after 30 min decay	100,000 after 30 min decay	100,000 after 30 min decay	100,000 after 30 min decay
Process treatment	Recombiner ambient charcoal	Low temperature charcoal	Chilled charcoal	Recombiner ambient charcoal
No. beds	8	8	5	8
Design condenser in-leakage, cfm	30	30	12.5	21
Release point, height above ground, ft	430 (stack) 187 (vent)	230	172	370
Liquid Radwaste* (Section 11.2)				
Treatment of:				
Floor drains	F or E, F, D returned to condensate storage, concentrates to radwaste solidification	F, D returned to condensate storage	F, E returned to condensate storage	E, D returned to condensate storage
Equipment drains	F, D returned to condensate storage	F, D returned to condensate storage	F, D returned to condensate storage	F, D returned to condensate storage
Chemical waste	E, F, D returned to condensate storage, concentrates to radwaste solidification	N, E, D returned to condensate storage	E, D concentrates to solid radwaste, distillate recycled	E, D concentrates to solid radwaste, distillate recycled

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TABLE 1.3-5 (Cont)

	Nine Mile Point Unit 2	WPPSS Unit 2	Zimmer Unit 1	La Salle Units 1,2
Liquid Radwaste (Section 11.2)				
Laundry waste	*	F, Chemical addition F, E, sent to circulating water discharge	R, discharged	R, discharged

*Laundry will be processed offsite at Nine Mile Point Unit 1.

KEY: D = Demineralizer
E = Evaporator or concentrator
F = Filter
N = Neutralized
R = Reverse osmosis

TABLE 1.3-6

COMPARISON OF POWER CONVERSION SYSTEM DESIGN CHARACTERISTICS

	Nine Mile Point Unit 2	WPPSS Unit 2	Zimmer* Unit 1	La Salle Units 1, 2
Design power, MWt	3,463	3,468	2,550	3,434
Design power, MWe, gross	1,202	1,205	883	1,122
Generator speed, RPM	1,800	1,800	1,800	1,800
Design steam flow, lb/hr	14.3×10^6	15.0×10^6	11.0×10^6	14.2×10^6
Turbine inlet pressure, psia	965	970	965	965 | 28
Turbine Bypass System (Section 10.4.4)				
Capacity, percent of turbine design steam flow	25	25	25	25
Main Condenser (Section 10.4.1)				
Heat removal capacity, Btu/hr	$7,830 \times 10^6$	$7,702 \times 10^6$	$7,053 \times 10^6$	$7,609 \times 10^6$
Circulating Water System (Section 10.4.5)				
No. Pumps	6	8	3	3
Flow rate, gpm/pump	105,000	82,000	150,000	210,000
Condensate and Feedwater Systems (Section 10.4.7)				
Design flow rate, lb/hr	14.917×10^6	14.260×10^6	10.971×10^6	14.127×10^6
No. condensate pumps	3 running	3 running	3	3 plus 1 spare
No. condensate booster pumps	3 running	3 running	3	3 plus 1 spare
No. feedwater pumps	2 running 1 standby	2 running	2	3
Condensate pump drive	ac power	ac power	ac power	ac power
Condensate booster pump drive	ac power	ac power	ac power	ac power
Feedwater pump drive	ac power	Turbine	Turbine	Turbine 2 Motor 1

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*Indicates parameters at rated power.

TABLE 1.3-7

COMPARISON OF STRUCTURAL DESIGN CHARACTERISTICS

	Nine Mile Point Unit 2	WPPSS Unit 2	Zimmer Unit 1	La Salle Units 1,2
Elevated Release Point (Section 11.3.3)				
Type	Stack, vent	Vent	Vent	Vent
Construction	Stack - Reinforced concrete Vent - steel	Steel	Steel	Steel
Height (above ground), ft	430 (stack) 187 (vent)	200	172	370
Seismic Design (Section 3.7)				
Operating basis earthquake				
Horizontal, g	0.075	0.125	0.10	0.10
Vertical, g	0.075		0.07	0.07
Safe shutdown earthquake				
Horizontal, g	0.15	0.250	0.20	0.20
Vertical, g	0.15		0.14	0.14
Wind Design (Section 3.3)				
Maximum sustained, mph	90	100	90	90
Tornado				
Rotational, mph	290	300	300	300
Translational, mph	70	60	60	60
Total, mph	360	360	360	360

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TABLE 1.3-8

COMPARISON OF FINAL AND PRELIMINARY DESIGN INFORMATION
FOR THE NSSS SCOPE OF SUPPLY

Item	Change	Reason For Change	FSAR Reference	
Control rod drive position	Changed to 11 wire probe and solid state.	Improved reliability and increased frequency of checking actual rod position	7.7.1	
Recirculation pump and motor	The flow rate and horsepower required have been reduced; voltage has changed from 4,160 V to 13,200 V.	Detailed system design	5.4	
Recirculation flow measurement	The recirculation flow measurement design was changed from a flow element to an elbow-tap type.	To improve flow measurement accuracy	5.4	21
Recirculation system	The pressure interlock for RHR injection was changed.	IEEE-279 requirements	7.3.1, 7.1	
Feedwater and recirculation nozzle safe ends and thermal sleeves	Material/design change. Piping changed to type 316K from type 304.	Mitigate IGSCC	5.3	21
Nuclear fuel	The number of fuel pins in each fuel bundle has been changed from 7 x 7 to 8 x 8.	Improved fuel performance by increasing safety margins	4.2	
Nuclear boiler	a. A turbine building high temperature trip for MSIVs was added.	Improve leak detection capability	7.3	
	b. Delete REVAB system.	GE Mark II suppression pool dynamics test program showed REVAB undesirable	5.2, 5.4, 8.3.1	
Main steam line isolation	A main condenser low vacuum initiation of the main steam line isolation was added.	NRC requirement	7.3.1	
Main steam line drain system	A main steam line drain system was improved.	Prevent accumulation of condensate in an idle line outboard of MSIV	5.1	
Feedwater sparger	The thermal sleeve was changed to provide welded design of sparger to nozzle.	To eliminate vibration and cracking	5.3	
RCIC steam supply	A warmup bypass line and valve were added.	Permits pressurizing and prewarming of the steam supply line downstream to the turbine during reactor vessel heatup	5.4	
Control rod drive	Alternate rod injection and scram discharge	To reduce potential for failure to scram	4.6.1	

TABLE 1.3-8 (Cont)

Item	Change	Reason For Change	FSAR Reference
system	volume modifications were implemented.		
RCIC vacuum breaker system	A vacuum breaker system was added to the RCIC turbine exhaust line into the suppression pool.	To prevent backup of water in the pipe and consequential high dynamic pipe loads and reactions	5.4
Automatic depressurization system (ADS)	The interlocks on the automatic depressurization system were revised.	To meet IEEE-279 requirements	7.3.1
RPV stabilizer support	The RPV stabilizer's configuration was changed by adding a top plate.	Provides a better seismic and alignment capability	5.3
Level instrumentation	The RPV level instrumentation was revised to eliminate Yarway columns and replace them with a conventional condensing chamber type; also, separation and redundancy features were added.	Improve ECCS separation in accordance with IEEE-279 and improve reliability	7.3.1
Leak detection system	The leak detection system was revised to upgrade the capability and incorporate the requirements of IEEE-279.	To meet IEEE-279 requirements	7.1
Reactor vibration monitoring	A confirmatory vibration monitoring test was added.	NRC requirement	14.2
Redundant reactivity control system	Added redundant reactivity control system to mitigate ATWS events.	To comply with NRC ruling on ATWS	7.6, 9.3.5, 15.8

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TABLE 1.3-9

COMPARISON OF FINAL AND PRELIMINARY DESIGN INFORMATION
FOR THE BALANCE OF PLANT

Item	Change	Reason for Change	FSAR Reference
Reactor building	Addition of auxiliary bays	Provide room to allow segregation of ECCS	6.2.3, 3.2.1, 3.8.4
Summary description of structures	Additional buildings included in the unit	Auxiliary bays, railroad access lock, condensate storage tank building, and demineralized water and waste neutralizer tank storage building added	1.2, 3.2.1
Primary containment cooling	a. Power electric motor components through normal 4-kV switchgear	Containment cooling is not a nuclear safety-related system	8.3
	b. Revised arrangement and number of unit coolers	Improve air distribution based on operating experience	9.4.9
Standby gas treatment (SGT) system	Add ASME Class 2 isolation valve between containment purge and SGT system	To isolate containment purge (Class 4 outside containment) from Class 2 SGT system	6.5.1
Reactor building ventilation system	a. Locate supply fans in SGT building	Provides a more efficient isolation of the reactor building	9.4.2
	b. Normal exhaust system to consist of two sets of two fans	Improve system design for better air movement and optimum fan performance	9.4.2
	c. Change from valves to zero-leakage dampers	Reduce seismic load and closure time and compact valve design	9.4.2
	d. Eliminate mixing box	Credit taken for turbulent mixing during emergency operation	9.4.2
Primary containment ventilation	All containment purge air is passed through the SGT system and vented out the stack	To enhance safety of plant ventilation design	6.5.1

TABLE 1.3-9 (Cont)

Item	Change	Reason for Change	FSAR Reference	
High density spent fuel storage	Changed spent fuel rack configuration to high density storage design	To increase onsite storage capacity of spent fuel	9.1	
Primary shield wall	Additional provisions added to supplement the commitment to use AISC Steel Construction Manual welding requirements	AISC standards do not account for certain weld configurations necessary to achieve proper erection, hence additional standards used for those weld configurations	3.8.3	
Primary containment liner	Referenced ASME code and construction for design of liner also added changes in material selection	To agree with specification requirements for incorporation of design improvements	3.8.1	18
Primary containment drywell floor seal	Eliminated flexible floor seal and revised floor connection to containment wall and reactor pedestal	Rigid design selected in order to improve maintainability and reliability and to eliminate testing of flexible seal	6.2.1, 3.8.3	18
Primary containment	Changed shape of primary containment to two conical frustums		3.8.1	
Downcomer piping and drywell floor	Drywell floor thickness increased and sloped cross section incorporated. Downcomers extended. Downcomer anchor design altered to two support plates anchored onto top and bottom of drywell slab and welded to downcomers at each face. Replaces two welded seal rings	Slab thickness increased for insulation, slope added to improve floor drainage. Downcomer extension maintains required clearance. Anchor redesign accounts for insulation slab and higher suppression pool hydrodynamic loads	3.8.3	
Drywell floor design criteria	Drywell floor designed for two independent cases; for 10 psid upward and 25 psid downward loads	To agree more accurately with calculated conditions	3.8.3	

TABLE 1.3-9 (Cont)

Item	Change	Reason for Change	FSAR Reference
Leakage detection - drywell air cooler drains	Flowmeters on drains to measure leakage condensation on individual air-coolers eliminated. Gaseous activity monitors added	Concern for reliability of flowmeters mounted in the wetwell due to environmental conditions Questionable operability and accuracy of drywell-located meter(s)	5.2.5
Main steam isolation valve (MSIV) leakage control system	Deleted system	System not required	1.8, 6.2.3.2.3, 15.6.5 | 28
Equipment and floor drainage	Route drywell equipment drainage and low conductivity secondary containment drainage to radwaste system	Drain line to condenser forms constant air leak. Resolve by routing to radwaste system	9.3.3
Containment influent isolation valves	Change from motor-operated stop check to motor-operated gate valve farthest from containment	Commercial availability of valves that meet design criteria	6.2.4
Fire protection-water system	a. Allow tie valve of one of two interplant connection lines to be operated by remote control at Nine Mile Point Unit 1 control room	Enhance fire protection at both plants on site. Selective interconnection by either plant to allow pumps to be used for pressure maintenance and fire fighting	9.5.1
	b. Upgrade fire protection water system pipe support hangers to Seismic Category I in safety-related areas and in the vicinity of safety-related equipment	Compliance with NRC Branch Technical Position (BTP) APCSB 9.5-1	9.5.1, 3.9.3, 3.7.3
Recombiner system	Upgrade from Safety Class 3 to Safety Class 2	Compliance with Regulatory Guides 1.7 and 1.26	6.2.5

TABLE 1.3-9 (Cont)

Item	Change	Reason for Change	FSAR Reference
Circulating water system	Change from once-through lake water system to closed loop with natural-draft cooling tower	EPA Effluent Guidelines*, which became effective after the last PSAR supplement, subjected the approval of once-through condenser cooling to a time-consuming development, review, and approval of a demonstration that no harm will occur to the aquatic community of the receiving water body. Based on scheduling impacts, close cycle cooling was incorporated to conform to best technology economically available as required by these guidelines	10.4.5
Service water system	Number of pumps changed from 4 to 6 and size of pumps changed from 8,000 gpm and 450 hp to 10,000 gpm and 600 hp	Addition of new heat loads required increase in system capacity	9.2.1
Main steam line	Standards for fabrication, inspection, and quality assurance revised to include turbine stop valves, main turbine bypass lines to and including bypass valves, and main branch lines of 2 1/2-in diameter or larger to and including the first valve on each branch	Conformance with Regulatory Guide 1.29	5.2.1, 5.4.9
Pipe break criteria	Revised criteria for pipe break location, orientation, dynamic force pipe whip	Requirements of Regulatory Guide 1.46 and report Pipe Rupture Criteria for NMP2	3.6
Structural design criteria - seismic	a. Revised criteria for accelerations in analysis for OBE and SSE	Conformance with Regulatory Guides 1.60 and 1.61.	3.7.1
	b. Added fixed-base to models of Seismic Category I structures	Foundation media of all major Seismic Category I structures are founded on rock. Allows use of fixed base for dynamic analysis	3.7.1

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TABLE 1.3-9 (Cont)

Item	Change	Reason for Change	FSAR Reference
	c. Seismic loading will include the effects of one vertical and two horizontal response accelerations simultaneously	NRC required this in Safety Evaluation Report (SER)	3.7.2
Structural design criteria	a. Change tornado model to adopt characteristics given in Regulatory Guide 1.76	Compliance with Regulatory Guide 1.76	3.3.2
	b. Add the criteria to design safety-related structures for time-dependent site foundation rock movement	Conformance with the geologic criteria developed by Dames & Moore, geologic consultants for the site	3.7.1, 3.7.2
Primary containment load analysis	Reduced amount of restraint material for piping and adopt requirements of ASME III-1974 NB-3225	Reduce cost of restraint	3.6
Reinforcing steel-splices	a. Change procedure for visual inspection of Cadweld splices	Apply sampling inspection criteria in place of 100 percent inspection	1.8.1.10
	b. Add DYWIDAG threadbar splices	Provides alternate to other rebar splicing systems; adds flexibility to construction	3.8.4 18
Equipment classifications - gaseous radwaste	Components reclassified as Quality Group D	Site boundary accident dose calculations indicate that system meets Group D criteria	3.2.2, 11.3.3
Quality assurance program	Revised engineer's QA program to adopt SWEC Standard Quality Assurance Program (SWSQAP)	To use improved QA program derived from SWEC Topical Report SWSQAP 1-74 Revision dated 8/1/75	17.1; Table 1.8-1, Regulatory Guides 1.28, 1.39, 1.74, 1.88, and 1.144

TABLE 1.3-9 (Cont)

COMPARISON OF FINAL AND PRELIMINARY DESIGN INFORMATION
FOR THE BALANCE OF PLANT

Item	Change	Reason for Change	FSAR Reference
Internal containment coatings	Change QA requirements from conformance with proposed ANSI standard N101.5.7 to ANSI N101.4, Regulatory Guide 1.54, and Chapter 10 of ANSI N512-74	To comply with new standards and regulatory guide	17.1; Table 1.8-1, Regulatory Guide 1.54
Materials for construction - reinforcing steel	a. Deleted requirements for independent chemical analysis of each heat of controlled chemistry bars by the engineers	Vendor has acceptable Category I QA program	17.1; Table 1.8-1, Regulatory Guide 1.94
	b. Changed ASTM A615-Gr.40 to A615-Gr60	Vendor discontinued manufacturing certain size reinforcing bars in Gr. 40	3.8.4
	c. Used high-strength reinforced steel in fuel pool beams and other miscellaneous areas of reactor building	To minimize congestion in these areas	3.8.4
Materials for construction - concrete	a. Revised requirements for aggregates	Provided revised QA program requirements which conform to ANSI N45.2.5	17.1; Table 1.8-1, Regulatory Guide 1.94
	b. Minimum density revised downward for concrete used as biological shielding	Radiation protection calculations show that average concrete density exceeds required value for radiological protection	12.2.2, 3.8.4 | 18
	c. Added provisions for using heavy density fill material (HDFM) in the biological shield wall	To provide the necessary radiation protection and shielding capabilites	3.8.3
	d. Porous concrete under reactor building mat uses calcium aluminate cement with no free calcium oxide or calcium carbonate	Prevent lime buildup in mat drainage pipes	3.8.4, 3.8.5

TABLE 1.3-9 (Cont)

Item	Change	Reason for Change	FSAR Reference
Waste solidification system	System uses asphalt to incapsulate waste prior to shipment offsite	To meet future burial site restrictions on disposal of solidified waste	11.4
SRV discharge device	Changed from "ramshead" to "T-quencher"	Improve thermal performance of suppression pool during SRV discharge	App. 6A
115-kV switchyard	a. Added a bus section with necessary circuit switchers and isolation switches to feed the auxiliary boiler transformer	To provide necessary 115-kV feed to the auxiliary boiler transformer	8.2.1
	b. Outgoing cables to reserve transformers and auxiliary boiler transformer are routed overhead.	Since the switchyard has been moved close to the plant, this arrangement will improve reliability and maintainability of 115-kV offsite power	
Auxiliary boiler transformer and switchgear	Added auxiliary boiler transformer and associated 13.8-kV switchgear	To provide a separate power feed to the auxiliary boilers for startup and shutdown. Also, the 4.16-kV tertiary winding of the auxiliary boiler transformer provides a backup feed to the redundant Class 1E buses	8.2.1, 8.3.1
Reserve transformers	Added tertiary windings to the reserve transformers	To feed emergency buses to improve reliability of the Class 1E system	8.2.1, 8.3.1
Normal and emergency ac distribution system	Revised distribution systems at 13.8-kV, 4.16-kV, and 600-V levels to include more 13.8-kV, 4.16-kV, and 600-V buses, to have offsite power as normal source to Class 1E buses and to have all load centers and MCCs double ended	The new system design improves separation of non-Class 1E and Class 1E buses, improves reliability, availability, and maintainability of the plant ac distribution system	8.3.1
Normal station service transformer	Increased in size from 75 to 100 MVA	Increase in the normal station auxiliary loads	8.2.1, 8.3.1

TABLE 1.3-9 (Cont)

Item	Change	Reason for Change	FSAR Reference
Auxiliary stepdown transformer	Decreased in size from 12/16/20 MVA to 8.5/10.6/11.9 MVA	Since Class 1E loads are fed from reserve transformers, the load on this transformer was reduced	8.3.1
Normal load centers	Normal load center transformers (13.8 kV/600 V) increased in size from 1,000 kVA to 1,500/2,025 kVA and all normal load centers made double ended with two/three bus sections on 600-V buses	Increased 600-V loads, allowance for future load growth, greater availability and maintainability of loads	8.3.1
Emergency load centers	Division I and II emergency load center transformers (4.16 kV/600 V) increased in size from 1,000 kVA to 1,500/2,025 kVA	Increase in 600-V emergency loads, allowance for future load growth	8.3.1
Stub buses	Two 4.16-kV buses, 1,000/1,350-kVA load centers and MCCs were added to power loads needed to shut the plant down after a loss of offsite power and a unit trip. These buses can be manually connected to be powered from the standby diesel generators	This arrangement allows for an orderly shutdown of the plant upon loss of offsite power and unit trip	8.3.1
Reactor recirculation system power supply	a. Added two Class 1E circuit breakers (Division I and II) in series to each recirculation pump	To assure trip of recirculation pump upon ATWS	8.3.1
	b. Added MG set and circuit breaker to each reactor recirculation pump to power the pumps during low plant power levels	MG sets run the recirculation pumps on 15-Hz power at 25% speed to preclude cavitation	

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TABLE 1.3-9 (Cont)

Item	Change	Reason for Change	FSAR Reference
Degraded voltage relaying	Added additional set of undervoltage relays on Class 1E buses	To detect slow degradation of voltage 4.16-kV Class 1E buses, to trip offsite power and power Class 1E buses from onsite diesel should offsite power supply voltage degrade to a point where operation of Class 1E equipment is endangered	8.3.1
120-V ac Class 1E instrument power system	Added Class 1E uninterruptible power supplies	To ensure availability of power to the Class 1E instrument buses that should not have power interruptions	8.3.1
Diesel generators	The Division I and II diesel generators increased in size from 2,850 kW to 4,400 kW; Division III HPCS diesel generator increased from 2,500 kW to 2,600 kW	Increase in Class 1E loads	8.3.1
250-V dc system	Deleted 250-V dc system and assigned all associated loads to 125-V dc system	To reduce the number of normal batteries requiring maintenance	8.3.2
Class 1E dc distribution system	Eliminated ties between safety-related 125-V dc systems and each Class 1E dc system is provided with a backup battery charger	Ensures separation and independence of redundant Class 1E systems and improves reliability	8.3.2
Cables	Non-Class 1E cables are colored black in lieu of white	Regulatory guidance requires color coding	8.3.1, 8.3.2
Separation	Separation in all tray, conduit, and other raceway systems revised to meet Regulatory Guide 1.75	Issuance of Regulatory Guide 1.75	8.3.1, 8.3.2

TABLE 1.3-9 (Cont)

Item	Change	Reason for Change	FSAR Reference	
Plant communication	a. The number of party lines increased from three to five thereby increasing the total page and party communication channels from four to six	To increase reliability and availability of the plant communication system	9.5.2	
	b. Added page line supervisory system			
	c. Added redundant communication paths in large areas			
Plant lighting system	a. Mercury vapor lighting fixtures changed to high pressure sodium	To upgrade the lighting system	9.5.3	
	b. Feed to the essential lighting subsystem changed to uninterruptible power supply	To increase reliability of the essential lighting system		
	c. 8-hr battery pack lighting added	To meet the requirements of 10CFR50, Appendix R		23
Outgoing transmission lines	Changed from 765 kV to 345 kV	Economic advantage	8.1	
RCIC initiation	Trip of main turbine on an initiation of the RCIC turbine	To prevent water induction	5.4.6	
Remote shutdown room	Remote shutdown room separated into two separate rooms with 3-hr rated barrier	To meet the requirements of 10CFR50, Appendix R		23
Disconnect panels	Disconnect panels 2CES*PNL415 and 2CES*PNL416 added	To isolate required circuits from the control room in the event of a control room fire (10CFR50, Appendix R)		

TABLE 1.3-9 (Cont)

Item	Change	Reason for Change	FSAR Reference
PSI of main steam lines	Up to turbine stop valves, including branch connection lines in main steam, main steam bypass lines 2 1/2-in diameter and larger, and up to and including the first stop valve in each line will be in accordance with ASME Section XI, 1980 Edition Winter 1980 Addenda, in lieu of that described in PSAR Section H.2.26.	To be consistent with overall inservice inspection program	5.2.4.8

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1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

1.4.1 Applicant

NMPC is a private, investor-owned utility involved in the generation, distribution, and selling of electrical power for residential, commercial, and industrial use. The utility has grown to its present status through the merger of many small companies, the oldest of which dates back to 1823. It presently serves over one million electrical customers in a 24,000 sq mi area of upstate New York.

NMPC is a utility experienced in the design and operation of generating facilities in both the conventional and nuclear fields. It operates 81 hydroelectric, 6 fossil-fueled, and 1 nuclear power unit, and has an installed capacity of over 3.5 million kW. It has been actively involved in the nuclear generating field since 1953. As a result of this experience, NMPC designed and is now operating Unit 1, a 1,850-MWt BWR nominally generating 610 MWe.

A sizable engineering staff in the corporate office in Syracuse, NY, consists of groups of disciplines required in the design and operation of generating stations. These groups are available to the station staff for support and consultation as required.

On September 22, 1975, NMPC entered into an agreement with four electric utilities, whereby each of the utilities would own, as tenants in common, proportional interests in Unit 2. The names of these utilities and their proportional interests in Unit 2 are as follows:

Niagara Mohawk Power Corporation	41 percent
Central Hudson Gas & Electric Corporation	9 percent
Long Island Lighting Company	18 percent
New York State Electric & Gas Corporation	18 percent
Rochester Gas and Electric Corporation	14 percent

Under the terms of the agreement, the participants will share the electrical output and pay construction and operating costs according to their respective shares in Unit 2. The NRC approved the utilities' co-ownership in an amendment to the Unit 2 Construction Permit in October 1978[1]. NMPC has the responsibility for licensing, design, procurement, construction, operation, and all related functions with respect to Unit 2.

1.4.2 Architect-Engineer

Stone & Webster Engineering Corporation (SWEC) is a Massachusetts corporation with offices in Boston, MA; Cherry Hill, NJ;

Denver, CO; New York, NY; and Houston, TX. In addition to its project-dedicated staff, SWEC has utilized its own staff of specialists in various engineering disciplines to ensure that Unit 2 is designed in accordance with industry codes and standards and meets the requirements of the applicable federal, state, and local regulations for commercial nuclear power plants.

In addition to commercial nuclear power projects, SWEC has engaged in engineering, design, and construction of chemical refineries, hydroelectric stations, and fossil fuel power plants. It has participated in the design and construction of fossil fuel plants with a total capacity in excess of 41,000,000 kW. SWEC has been actively engaged in nuclear engineering and construction of nuclear power plants since 1954, with an accumulated experience in excess of 20,000,000 kW reactor thermal power. It has participated in the design and/or construction of the following nuclear power stations, all of which are operating or have operated successfully:

1. Shippingport Atomic Power Plant of Duquesne Light Company and ERDA.

2. Army Package Power Reactor (APPR, also known as SM-1).

3. Yankee Nuclear Power Station of Yankee Atomic Electric Company.

4. Carolinas-Virginia Tube Reactor of the Carolinas-Virginia Nuclear Power Associates, Inc.

5. Haddam Neck Plant of Connecticut Yankee Atomic Power Company.

6. Nine Mile Point Nuclear Station - Unit 1 of Niagara Mohawk Power Corporation.

7. Maine Yankee Atomic Power Station of Maine Yankee Atomic Power Company.

8. Surry Power Station Units 1 and 2 of Virginia Electric and Power Company.

9. James A. FitzPatrick Nuclear Power Plant - Unit 1 of the Power Authority of the State of New York.

10. North Anna Power Station Units 1 and 2 of Virginia Electric and Power Company.

11. Beaver Valley Power Station Unit 1 of Duquesne Light Company.

In addition to Unit 2, SWEC has under design or construction at this time the following nuclear power stations:

1. Beaver Valley Power Station Unit 2, Duquesne Light Company.

2. Millstone Nuclear Power Unit 3, Northeast Utilities Service Company.

3. River Bend Station Unit 1, Gulf States Utilities Company.

4. Shoreham Nuclear Power Station Unit 1, Long Island Lighting Company.

5. Enrico Fermi Unit 2, Detroit Edison Company.

In addition, SWEC is providing construction management services for the Demonstration Liquid Metal Fast Breeder Reactor Plant (Clinch River Project) and for the Gas Centrifuge Uranium Enrichment Plant by the U.S. Department of Energy.

1.4.3 Nuclear Steam Supply System (NSSS)

GE has been awarded contracts to design, fabricate, and deliver the direct-cycle boiling water NSSS, to fabricate the first core of nuclear fuel, and to provide technical direction of installation and startup of this equipment. GE has engaged in the development, design, construction, and operation of BWRs since 1955. Thus, GE has substantial experience, knowledge, and capability to design, manufacture, and furnish technical assistance for the installation and startup of reactors. Table 1.4-1 lists over 70 GE reactors completed, under construction, or on order.

1.4.4 Turbine Generator Supplier

GE-LSTG (large steam turbine generator) has been awarded the contract for designing, fabricating, and delivering the turbine generator. GE I&SE (installation and service engineering) will provide technical assistance for installation and startup of this equipment. GE has a history in the application of turbine generators in nuclear power stations that dates back to the inception of nuclear facilities for the production of electrical power. GE has furnished the turbine generator units for 21 of the BWR plants that were on-line by 1977 and has scheduled orders to supply about 30 additional turbine generator units for use in BWR units. GE has also supplied or is scheduled to supply turbine generator units for approximately 60 pressurized water reactors (PWRs). GE's nonnuclear turbine experience is extensive. The ratings of these units range from 210 MW to over 1,100 MW. Thus, GE is technically qualified to design, fabricate, and deliver the turbine generator unit and to provide technical assistance for its installation and startup.

1.4.5 Technical Consultants

Dames & Moore

The independent consulting firm of Dames & Moore was employed in consultation for the preparation of the sections relating to hydrology, geology, and seismology. Having performed environmental studies for approximately 40 nuclear power plant sites, Dames & Moore is active in the field of environmental engineering related to nuclear power plant construction.

Listed below are some of the nuclear power plants for which Dames & Moore has performed environmental studies:

Plant	Company
Donald C. Cook	American Electric Power Company
Pilgrim	Boston Edison Company
Nine Mile Point Unit 1	Niagara Mohawk Power Corporation
Zion	Commonwealth Edison Company
Quad Cities	Commonwealth & Iowa - Illinois Gas & Electric Company
Midland	Consumers Power Company
Turkey Point	Florida Power & Light Corporation
Duane Arnold	Iowa Electric Light & Power Company
Peach Bottom	Philadelphia Electric Company
San Onofre	Southern California Edison Company
Salem	Public Service Electric and Gas Company
Seabrook	Public Service Company of New Hampshire
Wm. H. Zimmer	Cincinnati Gas & Electric Company; Columbus & Southern Ohio Electric Company; The Dayton Power and Light Company

Meteorological Evaluation Services, Inc. (MES)

The independent consulting firm of Meteorological Evaluation Services, Inc., (MES) was employed in the preparation of the meteorological section. MES is active in the field of environmental assessment, meteorological monitoring, and environmental studies for nuclear power plants.

Listed below are some of the nuclear power plants for which MES has performed environmental studies:

Plant	Company
Donald C. Cook	American Electric Power Service Corporation
Nine Mile Point Unit 1	Niagara Mohawk Power Corporation
James A. FitzPatrick	Power Authority of the State of New York
Limerick Generating Station Units 1 and 2	Philadelphia Electric Company
Peach Bottom Atomic Power Station Units 2 and 3	Philadelphia Electric Company
Salem Nuclear Generating Station Units 1 and 2	Public Service Electric & Gas Company
Hope Creek Nuclear Generating Stations Units 1 and 2	Public Service Electric & Gas Company

Lawler, Matusky & Skelly, Consulting Engineers (LMS)

The environmental engineering consulting firm of Lawler, Matusky & Skelly Engineers (LMS) was employed for the preparation of the sections relating to aquatic biology, physical hydrology, lake circulation, temperature, and discharge evaluation. LMS has completed over 60 major projects that involved specialized environmental science and engineering services in such areas as circulating water system design, power plant siting, thermal discharge, biological surveys, impact assessment, mathematical modeling of biological populations, and regulatory aspects of nuclear, fossil-fueled, and hydroelectric power plant operations.

LMS has conducted hydrodynamic, physicochemical, and biological investigations for over 20 major electric utilities and utility associations throughout the United States. The firm has conducted studies at 12 nuclear sites for 9 utilities.

Listed below are some of the nuclear power plants for which LMS has performed environmental studies:

Plant	Company
Brunswick	Carolina Power & Light Company
Indian Point Units 2 and 3	Consolidated Edison Company
Davids Island (proposed)	
Midland Units 1 and 2	Consumers Power Company
Oyster Creek	Jersey Central Power & Light Company
Nine Mile Point Units 1 and 2	Niagara Mohawk Power Corporation
Fort Calhoun	Omaha Public Power District
Pickering	Ontario Hydro
James A. FitzPatrick	Power Authority of the State of New York
San Onofre	Southern California Edison Company

1.4.6 Reference

1. U.S. NRC Construction Permit No. CPPR-112, Amendment No. 1, NRC Docket No. 50-410. Nine Mile Point Nuclear Station - Unit 2, Niagara Mohawk Power Corporation, October 27, 1978.

TABLE 1.4-1

COMMERCIAL NUCLEAR REACTORS COMPLETED, UNDER CONSTRUCTION, OR IN DESIGN BY GENERAL ELECTRIC

Station	Utility	Rating (MWe)	Year of Order	Year of Commercial Operation	
Dresden 1	Commonwealth Edison	207	1955	1960	
Humboldt Bay	Pacific G&E	63	1958	1963	
Kahl	Germany	15	1958	1961	
Garigliano	Italy	150	1959	1964	
Big Rock Point	Consumers Power	63	1959	1962	21
JPDR	Japan	11	1960	1963	
KRB	Germany	237	1962	1967	
Tarapur 1	India	200	1962	1969	21
Tarapur 2	India	200	1962	1969	
GKN	Holland	55	1963	1969	
Oyster Creek	JCP&L	620	1963	1969	
Nine Mile Point 1	Niagara Mohawk	610	1963	1969	
Dresden 2	Commonwealth Edison	794	1965	1970	
Pilgrim	Boston Edison	670	1965	1972	21
Millstone 1	NUSCO	660	1965	1970	
Tsuruga	Japan	410	1965	1970	
Santa Maria de Garona	Spain	440	1965	1971	
Fukushima 1	Japan	439	1966	1971	
BKW KKM	Switzerland	320	1966	1972	21
Dresden 3	Commonwealth Edison	794	1966	1971	
Monticello	Northern States Power	536	1966	1971	
Quad Cities 1	Commonwealth Edison	789	1966	1972	
Browns Ferry 1	TVA	1,067	1966	1974	
Browns Ferry 2	TVA	1,067	1966	1975	
Quad Cities 2	Commonwealth Edison	789	1966	1972	
Vermont Yankee	Vermont Yankee	514	1966	1972	
Peach Bottom 2	Philadelphia Electric	1,065	1966	1974	
Peach Bottom 3	Philadelphia Electric	1,065	1966	1974	
James A. FitzPatrick	PASNY	821	1968	1975	
Shoreham	LILCO	809	1967		21
Cooper	Nebraska PPD	778	1967	1974	
Browns Ferry 3	TVA	1,067	1967	1977	
Limerick 1	Philadelphia Electric	1,055	1969		
Hatch 1	Georgia	810	1967	1975	21
Fukushima 2	Japan	760	1967	1974	
Brunswick 1	Carolina P&L	790	1968	1977	
Brunswick 2	Carolina P&L	790	1968	1975	
Arnold	Iowa ELP	545	1968	1974	
Fermi 2	Detroit Edison	1,100	1968		
Hope Creek 1	PSE&G	1,070	1969		21

TABLE 1.4-1 (Cont)

Station	Utility	Rating (MWe)	Year of Order	Year of Commercial Operation
Chinshan 1	Taiwan	604	1969	1978
Caorso 1	Italy	875	1969	1981
Hatch 2	Georgia Power	820	1970	1979
La Salle 1	Commonwealth Edison	1,078	1970	1982
La Salle 2	Commonwealth Edison	1,078	1970	1984
Susquehanna 1	Pennsylvania P&L	1,050	1967	1983
Susquehanna 2	Pennsylvania P&L	1,050	1968	1985
Chinshan 2	Taiwan	604	1970	1979
WNP 2	WPPSS	1,100	1971	1984
Nine Mile Point 2	Niagara Mohawk	1,080	1971	
Grand Gulf 1	MSEI/MP&L	1,250	1972	1985
Grand Gulf 2	MSEI/MP&L	1,250	1973	
Kaiseraugst	Switzerland	925	1971	
Fukushima 6	Japan	1,067	1971	1979
Tokai 2	Japan	1,056	1971	1978
River Bend 1	Gulf States Utilities	940	1972	
Perry 1	Cleveland Electric Illum.	1,205	1971	
Perry 2	Cleveland Electric Illum.	1,205	1971	
Laguna Verde 1	Mexico	654	1972	
Leibstadt	Switzerland	942	1972	1984
Kuosheng 1	Taiwan	948	1972	1981
Kuosheng 2	Taiwan	948	1972	1983
Clinton 1	Illinois Power	933	1973	
Laguna Verde 2	Mexico	654	1973	
Cofrentes	Spain	975	1973	1985
Alto Lazio 1	Italy	982	1974	
Alto Lazio 2	Italy	982	1974	
Valdecaballeros	Spain	975	1974	

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1.5 REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION

1.5.1 Current BWR Development Programs

1.5.1.1 Instrumentation for Vibration Detection

Vibration testing for reactor internals is performed for each GE BWR product line. At the time of issue of RG 1.20, test programs for compliance were instituted. The first BWR plant of each size and within each product line is considered a prototype design and is instrumented and subjected to both cold and hot, two-phase flow testing to demonstrate that flow-induced vibrations similar to those expected during operation do not cause damage. Subsequent plants that have internals similar to those of the prototypes are tested in compliance with the requirements of RG 1.20 to confirm the adequacy of the design with respect to vibration. Further discussion is presented in Section 3.9.2B.

1.5.1.2 Core Spray Distribution

The design basis for core spray distribution for BWR 5 plants is described in NEDO-10846 and NEDO-20566-3[1,2]. Other loss-of-coolant accident (LOCA) programs jointly sponsored by GE/NRC/EPRI show that the core spray systems' introduction of core spray water into the upper plenum results in a pool of water in the upper plenum. This provides a water downflow into all fuel bundles. When this water inventory in the upper plenum subcools, the countercurrent flow limiting at the upper tieplate breaks down, the water flows through the core and refloods the core at an earlier time than currently calculated. Fuel bundle heat transfer consistent with system performance during the time from rated core spray to core reflood has been shown to be greater than the values prescribed by 10CFR50 Appendix K. This behavior has been verified by overseas testing and reported at the Ninth Water Reactor Safety Research Information Meeting, October 26-30, 1981[3].

1.5.1.3 Core Spray and Core Flooding Heat Transfer Effectiveness

Due to the incorporation of an 8x8 fuel rod array with unheated "water rods," tests have been conducted to demonstrate the effectiveness of ECCS in the new geometry. These tests are regarded as confirmatory only, since the geometry change is very slight and the water rods provide an additional heat sink for the central fuel rods of the bundle which improves heat transfer effectiveness.

There are two distinct programs involving the core spray performance evaluation. The core spray distribution adequacy has been verified and Licensing Topical Reports NEDO-10846 and NEDO-20566-3 have been submitted[1,2]. The other program concerns the testing of core spray and core flooding heat transfer effectiveness. The results of testing with stainless steel cladding were reported in Licensing Topical Report NEDO-10801[4].

The results of testing using Zircaloy cladding were reported in Licensing Topical Report NEDO-20231[5].

1.5.1.4 Verification of Pressure Suppression Design

The Mark II Pressure Suppression Test Program was initiated in the fall of 1975 to investigate suppression pool dynamic phenomena. Phase I blowdown tests were completed late in 1975. These tests utilized a single 24-in diameter (590-mm ID) downcomer that vented into a 7-ft (2.13-m) inside diameter tank, representative of a single downcomer/pool cell in a typical Mark II suppression pool. The objective of this phase of testing was to quantify pool dynamics phenomena, particularly on pool swell. In addition, data were recorded for the following, which at the time were considered of secondary importance: chugging, condensation oscillations, lateral loads, wetwell pressurization, diaphragm floor loading, and pool temperature distribution. Primary variables that were simulated are: break size, initial vent submergence, and wetwell air space configuration, i.e., vented or closed wetwell.

The Phase II tests were generally similar to the Phase I tests, except a 20-in diameter (489-mm ID) downcomer was used. The Phase I and II tests thus bound the range of vent to pool area ratios of all Mark II containments. Although the test objectives were similar during Phases I and II, some changes were made in the Phase II test matrix after review of the Phase I data. For example, since the Phase I test had shown that wetwell configuration was the variable that had the most pronounced effect on pool dynamics, the decision was made to concentrate the testing effort on the closed wetwell configuration, which is characteristic of the Mark II containment.

In place of the open wetwell tests, additional blowdowns were included in the Phase II test matrix in order to investigate the effect of saturated liquid versus saturated steam breaks and the effect of downcomer bracing configuration.

As was the case for the Phase I tests, the primary Phase II variables were simulated break size and initial vent submergence.

The Phase III tests investigated the pool temperature sensitivity of pool swell and of the load associated with the chugging phenomenon. Only a single break size and vent submergence were tested, with pool temperature alone being a variable. A significant number of blowdowns were performed to yield a statistically significant data set.

1.5.1.5 Boiling Transition Testing

Since the formulation of the 1966 Hench-Levy design limit lines for use in BWR thermal design, GE has continued to perform extensive steady-state and transient boiling test programs. Prior to 1974, over 14,000 data points had been obtained from

test assemblies having various axial heat flux profiles and rod-to-rod power distributions, covering prototypical aspects of reactor operating conditions. Among these, 2,100 data points were full-scale simulation of 7x7 and 8x8 BWR fuel assemblies performed in the ATLAS test facility. A new boiling transition correlation (GEXL) was developed and applied to GE BWR thermal design. Detailed information is provided in the approved Licensing Topical Report, NEDO-10958A[6].

Since the implementation of the GEXL correlation on design in 1974, GE has continued to conduct full-scale 8x8 assembly boiling transition tests. These tests have accumulated over 1,600 data points after GE thermal analysis basis (GETAB) introduction, to extend the data base and assure applicability to new 8x8 fuel designs such as the two-water-rod design for BWR 2 through BWR 6. It has been shown that the 8x8 GEXL correlation with the appropriate R-factors can predict boiling transition critical power data for the two-water-rod assemblies, with an accuracy typical of the GEXL correlation predictability for other 8x8 designs as described in NEDO-10958A[6].

1.5.2 Geotechnical Investigations

This section presents a summary of programs in geology, seismology, and geotechnical engineering that were undertaken to ensure conservative final design parameters for plant structures, systems, and components.

In September 1976, several small low-angle thrust faults and associated low-amplitude folds were discovered in the bedrock during excavation of plant structures such as the heater bay. Additionally, high-angle faults with associated buckles were discovered in the excavation for the cooling tower structure and in the drainage ditch. Excavations in April 1979 revealed small zones of bedding plane slip and folding in the circulating water trenches. Small low-angle thrust faults were observed in the radwaste excavation and in each lake-water tunnel in late 1979 and early 1980.

Each of these geologic conditions was studied in detail and reports were developed and submitted to the NRC to describe their safety significance. The detailed studies included a description of the origin, age, and significance of these features. As part of the then ongoing evaluation, the reports were submitted to the NRC for review and meetings were held to respond to NRC questions.

Geologic investigations at the site indicated that conditions of in situ stress, rock lithology and anisotropy, and groundwater fluctuations are elements that had to be considered in the design. To further support the geologic investigations, a geologic monitoring program was initiated. This program is currently collecting data and will be completed by January 1985.

The results of the monitoring program will be submitted as amendments to the FSAR.

A further description of the features and their significance is provided in Section 2.5.

1.5.3 References

1. BWR Core Spray Distribution, NEDO-10846, April 1973.

2. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50 - Appendix K - Effect of Steam Environment on BWR Core Spray Distribution, NEDO-20566-3, April 1977.

3. Sozzi, G. L. and Lee, L. S. BWR Blowdown/Emergency Core Cooling Integral Test Program Final Results from the Two Loop Test Apparatus (TLTA), October 26, 1981.

4. Modeling the BWR/6 Loss-of-Coolant Accident: Core Spray and Bottom Flooding Heat Transfer Effectiveness, NEDO-10801, March 1973.

5. Emergency Core Cooling Tests of an Internally Pressurized Zircaloy Clad, 8x8 Simulated BWR Fuel Bundle, NEDO-20231, December 1973.

6. General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, NEDO-10958A, January 1977.

1.6 MATERIAL INCORPORATED BY REFERENCE

Table 1.6-1 is a list of all topical reports and any other report or document which are incorporated in whole or in part by reference in this FSAR and have been filed with the NRC.

Additional documents referenced in this FSAR are listed at the end of the sections in which they have been referenced.

TABLE 1.6-1

REFERENCED REPORTS FOR THE NSSS SCOPE OF SUPPLY

Report Number	Title	Referenced in Section
A. GE Reports		
APED-4378	Maximum Flow Rate of a Single Component Two-Phase Mixture (October 25, 1963)	6.2
APED-5458	Effectiveness of Core Standby Cooling Systems for General Electric Boiling Water Reactors (March 1968)	5.4
APED-5460	Design and Performance of General Electric BWR Jet Pumps (July 1968)	3.9, 6B
APED-5555	Impact Testing on Collet Assembly for Control Rod Drive Mechanism 7RDB144A (November 1967)	4.6
APED-5696	Tornado Protection for Spent Fuel Storage Pool (November 1968)	3.3.2
APED-5706	In-Core Neutron Monitoring System for General Electric Boiling Water Reactors (November 1968, Revised May 1969)	6A, 7.6
APED-5750	Design and Performance of General Electric Boiling Water Reactor Main Steam Isolation Valves (March 1969)	3.9B, 5.4 28
APED-5756	Analytical Methods for Evaluating the Radiological Aspects of General Electric Boiling Water Reactors (March 1969)	15.4

TABLE 1.6-1 (Cont)

Report Number	Title	Referenced in Section
GEAP-5620	Failure Behavior in ASTM A106B Pipes Containing Axial Through-Wall Flaws	5.2

TABLE 1.6-1 (Cont)

Report Number	Title	Referenced in Section	
NEDC-30088	Responses to NRC Post-Implementation Review Criteria for Post-Accident Sampling System	1.10	10
NEDE-10313	PDA-Pipe Dynamic Analysis Program for Pipe Rupture Movement (Proprietary Filing)	3.6.3B	10
NEDE-20566	Analytical Model for Loss-of-Coolant Analysis in accordance with 10CFR50 Appendix K (Proprietary Document)	3.9.6B, 6.3	
NEDE-20944-P-1	BWR/4 and BWR/5 Fuel Design (October 1976) (Amendment 1, January 1977) (only BWR/4 and 5)	4.2, 4.3, and 4.6	
NEDE-21078	Test Results Employed by GE for BWR Containment and Vertical Vent Loads	6A	
NEDE-21175-3-P	BWR Fuel Channel Mechanical Design and Deflection (September 1976)	6A	10
NEDE-21471-P	Analytical Model for Estimating Drag Forces on Rigid Submerged Structures Caused by LOCA and Safety Valve Ramshead Air Discharge	3.9, 3A.13.1	10
NEDE-21544-P	Mark II Pressure Suppression Containment Systems: An Analytical Model of the Pool Swell Phenomenon (December 1976)	6A	
NEDE-21730	Mark II Pressure Suppression Containment Systems - Loads on Submerged Structures, An Application Memorandum (December 1977)	3A.13.1	10

TABLE 1.6-1 (Cont)

Report Number	Title	Referenced in Section
NEDE-21821-02	Boiling Water Reactor Feedwater Nozzle/Sparger Final Report (August 1979)	5.3
NEDE-22178-PA	Mark II Containment Drywell to Wetwell Vacuum Breaker Models	6.2
NEDE-23014	Hex 01 User's Manual (July 1976)	15.2
NEDO-24010-P	Technical Basis for the Use of the Square Root of the Sum of Squares (SRSS) Method for Combining Dynamic Loads for Mark II Plants (Supplement 1, October 1978) (Supplement 2, December 1978) (Supplement 3, August 1979)	6A
NEDE-24011-P-A NEDE-24011-P-A-US	General Electric Standard Application for Reactor Fuel Including United States Supplement (latest approved revision)	4.1, 4.2 4.3, 4.4 5.2, 6.3 15.3, 15.4
NEDE-24057-P NEDE-2-P-24075 NEDO-24075-1-P	Assessment of Reactor Internals Vibration in BWR/4 and BWR/5 Plants (Class III) (November 1977) and Amendment 1 (December 1978), Amendment 2 (June 1979)	3.9, 6B
NEDE-24222	Assessment of BWR Mitigation of ATWS (Vol. I, II) (May, December 1979)	15.8
NEDE-24302-P	Generic Chugging Load Definition Report	6A, 3A.26.3
NEDE-24326-1-P	Environmental Qualification Program Class IV (January 1983)	3.11
NEDE-24822-P	Mark II Improved Chugging Methodology, Class III (May 1980)	3A.12.1
NEDE-24988-P	Analysis of Generic BWR Safety/Relief Valve Operability Test Results	6A

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TABLE 1.6-1 (Cont)

Report Number	Title	Referenced in Section
NEDO-10173	Current State of Knowledge, High Performance BWR Zircaloy-Clad UO_2 Fuel (May 1973)	11.1
NEDO-10349	Analysis of Anticipated Transients Without Scram (March 1971)	15.8
NEDO-10466-A	Power Generation Control Complex Design Criteria and Safety Evaluation	9.5 10
NEDO-10505	Experience with BWR Fuel Through September 1971 (May 1972)	11.1
NEDO-10602	Testing of Improved Jet Pumps for the BWR/6 Nuclear System (June 1972)	3.9
NEDO-10739	Methods for Calculating Safe Test Intervals and Allowable Repair Times for Engineered Safeguard Systems (January 1973)	6.3, 15A
NEDO-10801	Modeling the BWR/6 Loss-of-Coolant Accident: Core Spray and Bottom Flooding Heat Transfer Effectiveness (March 1973)	1.5
NEDO-10802	Analytical Methods of Plant Transient Evaluations for General Electric Boiling Water Reactor (February 1973)	15.0, 15.1
NEDO-10846	BWR Core Spray Distribution (April 1973)	1.5
NEDO-10871	Technical Derivation of BWR 1971 Design Basis Radioactive Material Source Terms (March 1975)	11.1

TABLE 1.6-1 (Cont)

Report Number	Title	Referenced in Section	
NEDO-10899	Chloride Control in BWR Coolants (June 1973)	5.2	
NEDO-10905	HPCS Power Supply Topical Report	6A	
NEDO-10958-A	General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation, and Design Application (January 1977)	1.5, 4.4.2, 16.1, 15.0	10
NEDO-20231	Emergency Core Cooling Tests of an Internally Pressurized, Zircaloy-Clad, 8X8 Simulated BWR Fuel Bundle (December 1973)	1.5	
NEDO-20533	The General Electric Mark III Pressure Suppression Containment System Analytical Model (June 1974)	3B.4, 15.2	
NEDO-20566-3	General Electric Company Model for Loss-of-Coolant Analysis in Accordance with 10CFR50 Appendix K - Effect of Steam Environment on BWR Core Spray Distribution (April 1977)	1.5, 6.3	10
NEDO-20626	Studies of BWR Designs for Mitigation of Anticipated Transients Without Scrams (October 1974)	15.8	
NEDO-20626-1	Studies of BWR Designs for Mitigation of Anticipated Transients Without Scrams (June 1975)	15.8	
NEDO-20626-2	Studies of BWR Designs for Mitigation of Anticipated Transients Without Scrams (July 1975)	15.8	

TABLE 1.6-1 (Cont)

Report Number	Title	Referenced in Section	
NEDO-20913	Lattice Physics Methods	9.1	
NEDO-20922	Experience with BWR Fuel Through September 1974 (June 1975)	11.1	
NEDO-20944	BWR/4 and BWR/5 Fuel Design (October 1976)	4.2, 4.3, 4.6	
NEDO-21061-3	Mark II Containment Dynamic Forcing Functions Information Report (June 1978), also Revision 4 (November 1981)	3.9.6B, 6A, 3A.13.3	
NEDO-21064-4	Mark II Containment Dynamic Forcing Functions	6A	
NEDO-21143-1	Radiological Accident Evaluation - The CONCACO3 Code (December 1981)	15.6	
NEDO-21159	Airborne Release from BWRs for Environment Impact Evaluations (March 1976)	11.1	
NEDO-21231	Banked Position Withdrawal Sequence	15.4	
NEDO-21506	Stability and Dynamic Performance of the General Electric Boiling Water Reactor (January 1977)	4.1.5	
NEDO-21660	Experience with BWR Fuel Through December 1976 (July 1977)	11.1	
NEDO-21778	Transient Pressure Rises Affecting Fracture Toughness Requirements for Boiling Water Reactors (December 1978)	5.3	
NEDO-21985	Functional Capability Criteria for Essential Mark II Piping (September 1978)	6A.9.1.1.2, 3.9.3A	19

TABLE 1.6-1 (Cont)

Report Number	Title	Referenced in Section
NEDO-24057	Assessment of Reactor Internals Vibration in BWR/4 and BWR/5 Plants (Class I) (November 1977)	3.9.6B

TABLE 1.6-1 (Cont)

Report Number	Title	Referenced in Section
NEDO-23538	Users Manual for CRPLS01 Computer Program (December 1976)	4.1.5
NEDO-24154-P	Qualification of One-Dimensional Core Transient Model for BWR	15.0, 15.1
NEDO-24210	Analysis of WRC Problem (August 1979)	3.9.6B
NEDO-24548	Annulus Pressurization Load Adequacy Evaluation (January 1979)	6.2
B. Other Reports		
BHR/DER 70-1	Radiological Surveillance Studies at a Boiling Water Nuclear Power Reactor (March 1970)	11.1
SWECO-7703	Missile-Barrier Interaction, (September 1977)	3.5.3
WPC-VRS-001	Radwaste Volume Reduction and Solidification System (May 1978)	11.4.7
WPC-VRS-002	10CFR61 Waste Form Conformance Program for Solidified Waste Products Produced by a Wastechem Corporation Volume Reduction and Solidification System (August 1987)	11.4.7
SWECO 8101	Models used in LOCTVs - A Computer Code to Determine Pressure and Temperature Response of Vapor Suppression Containments Following a Loss-of-Coolant Accident (1981)	6.2.1
EN-136	Stone & Webster Engineering Corporation Shallow Water Wave Generation (SWWAVE) Proprietary User's Manual (January 1977)	2.4.15

TABLE 1.6-1 (Cont)

Report Number	Title	Referenced in Section
ORNL-NISC-22	Missile Generation and Protection in Light Water-Cooled Power Reactor Plants (September 1968)	3.5.4
GEN-02-02	Final Report Pipe Rupture Analysis of Recirculation System for 1965 Standard Plant Design	3.6.3B
RP-8A	Radiation Shielding Design and Analysis Approach for Light Water Reactor Power Plant (May 1975)	12.3

1.7 DRAWINGS AND OTHER DETAILED INFORMATION

The drawings listed in this section are provided to assist the NRC in the FSAR review.

1.7.1 Electrical, Instrumentation, and Control Drawings

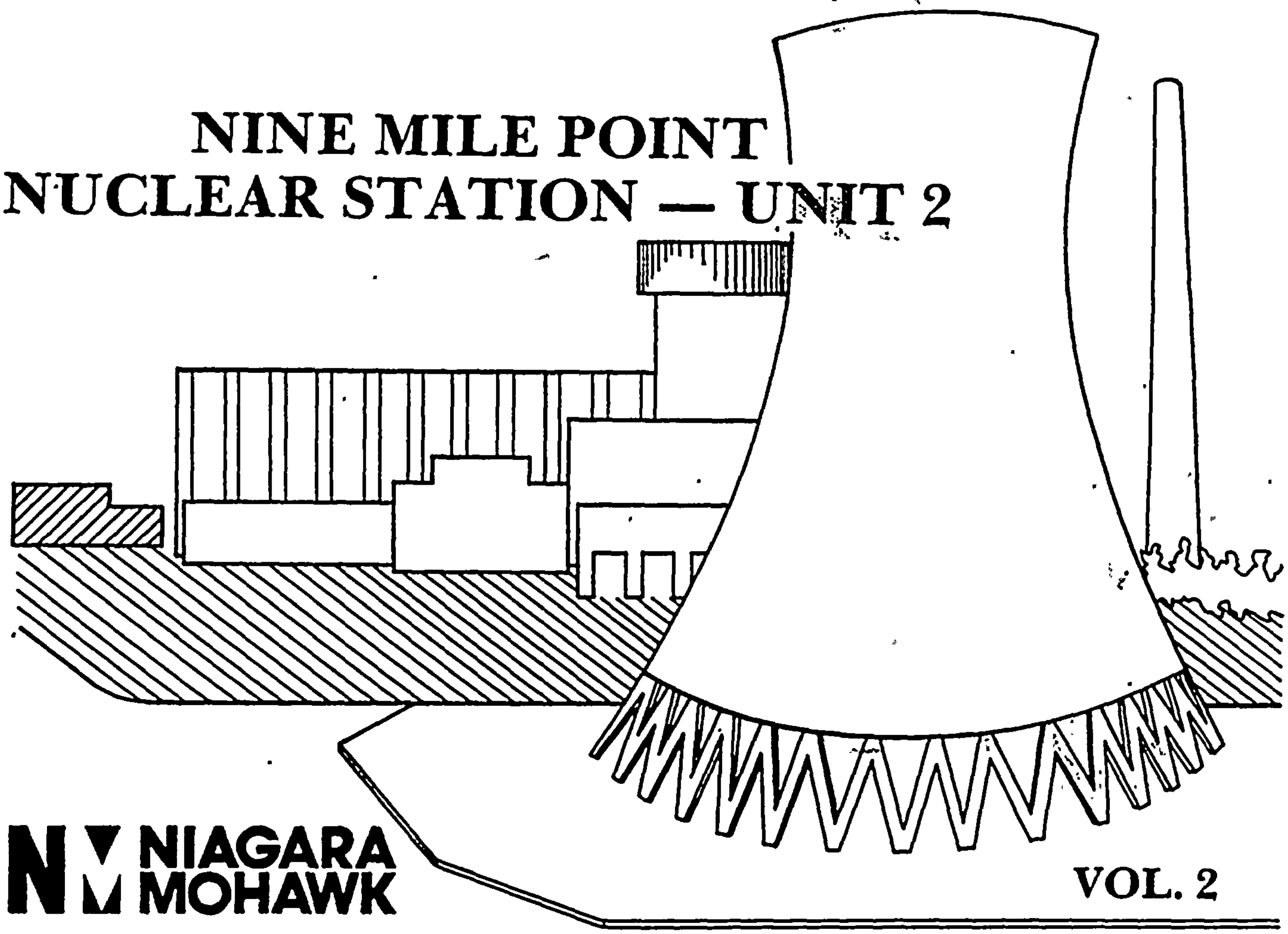
Table 1.7-1 contains a list of safety-related electrical, instrumentation, and control drawings which are separate from the FSAR but incorporated by reference. This table includes drawings that are considered necessary to evaluate the safety-related features in Chapters 7 and 8.

1.7.2 Piping and Instrumentation Diagrams (P&ID)

Table 1.7-2 contains a list of system piping and instrumentation diagrams provided in the FSAR. P&ID symbols used on NMPC diagrams are shown on Figure 1.7-1; those used on GE diagrams are shown on Figure 1.7-2. Figure 1.7-3 shows logic symbols used on NMPC logic diagrams. Figure 1.7-4 shows symbols used on NMPC electrical one-line diagrams.

UPDATED SAFETY ANALYSIS REPORT

NINE MILE POINT NUCLEAR STATION — UNIT 2



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1.8 CONFORMANCE TO NRC REGULATORY GUIDES

Tables 1.8-1 and 1.8-2 indicate the extent of compliance with all applicable NRC Regulatory Guides and the revision number of those guides. A reference to the FSAR section(s) in which the applicable design features are described is also provided.

Where the design differs from the Regulatory Guides, alternative methods of providing an equivalent level of safety have been utilized. These differences are discussed in Tables 1.8-1 and 1.8-2, or reference is made to the appropriate FSAR section(s) in which they are discussed. Regulatory guides and revisions to existing regulatory guides issued through July 1982 are addressed.

TABLE 1.8-1

CONFORMANCE WITH DIVISION 1 NRC REGULATORY GUIDES

Regulatory Guide 1.1, Revision 0 (November 1970)

Net Positive Suction Head for Emergency Cooling and Containment Heat Removal System Pumps

FSAR Section 6.3.2.2

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

The physical location of the RHR, LPCS, and HPCS pumps in relation to the minimum suppression pool water level is such that the required NPSH is maintained on these pumps under the conditions of zero psig containment pressure and 212°F suppression pool water temperature for all operating modes. Adequate NPSH is verified by system calculations.

Regulatory Guide 1.2, Revision 0 (November 1970)

Thermal Shock to Reactor Pressure Vessels

FSAR Section 5.3.3

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

Regulatory Guide 1.3, Revision 2 (June 1974)

Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors

FSAR Sections 3.11.5, 15.6.5

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

RG 1.3, together with TID 14844 models, has been used to arrive at the estimated site boundary, 2-hr dose, and low population zone (LPZ) 30-day dose for a LOCA.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.3, Revision 2 (June 1974) (cont'd.)

However, the meteorological assumptions were based upon Murphy and Campe and RG 1.145, as discussed in Section 2.3.4.3.

Regulatory Guide 1.4, Revision 2 (June 1974)

Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors

Position

RG 1.4 applies to PWR plants and therefore is not applicable to Unit 2.

Regulatory Guide 1.5, Revision 0 (March 1971)

Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors

FSAR Section 15.6.3

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

The assumptions in RG 1.5 have been used to arrive at the site boundary and LPZ doses for the steam line break accident. The coolant activity level given by the NRC in the Standard Technical Specification for GE BWRs, NUREG-0123, is used as a reference point for the accident analysis and is given as the final coolant activity level for the Unit 2 Technical Specifications.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.6, Revision 0 (March 1971)

Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems

FSAR Sections 8.3.1.2, 8.3.2.2

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide for Division I, II, and III diesels.

Regulatory Guide 1.7, Revision 2 (November 1978)

Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident

FSAR Sections 6.1.1.2, 6.2.5

Position

Unit 2 complies with the Regulatory Position (Paragraph C) of this guide.

Conformance with this guide is achieved by redundant recombiner systems.

The Unit 2 design includes the recombiners and an inerted containment which is described in Section 6.2.5.2.3.

Regulatory Guide 1.8, Revision 1-R (May 1977)

Personnel Selection and Training

FSAR Section Chapter 13

Position

NMPC complies with RG 1.8 Revision 1-R (May 1977) in meeting the criteria for the selection and training of nuclear power plant personnel as contained in ANSI N18.1.

This is delineated in Nine Mile Point Nuclear Station Site Administration Procedures which will be followed while staffing Unit 2.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.9, Revision 2 (December 1979)

Selection of Diesel Generator Set Capacity for Standby Power Supplies

FSAR Section 8.3.1.2

Position

The criteria used in the selection, design, qualification, and testing of the Unit 2 Division I and II standby diesel generators comply with the Regulatory Position (Paragraph C) of this guide.

The Division III diesel generator (HPCS) conforms to Regulatory Positions C.1, C.2, and C.3 of this guide. The HPCS diesel generator performance is considered an acceptable departure from literal conformance to Regulatory Position C.4.

Environmental qualification of the Division III diesel generator is discussed in Section 3.11; testability is discussed in RG 1.108.

Regulatory Guide 1.10, Revision 1 (January 1973)*

Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures

FSAR Section None

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide, with the following alternative approaches:

1. Paragraph C.2 Visual inspection of mechanical splices prior to forming will be performed on each splice by the Cadwelder and on a random basis (20 percent) by field quality control (FQC).

2. Paragraph C.3b The locations of all mechanical splices for reinforcing bars are shown on permanent plant records that are kept for the plant lifetime.

* This regulatory guide was withdrawn on July 8, 1981, by the NRC. It has been superseded by RG 1.136 Revision 2 (June 1981).

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.10, Revision 1 (January 1973)* (cont'd.)

3. Paragraph C.4a Reinforcing bars with a radius of curvature of 60 ft 0 in or greater are tested at the sampling frequency specified in Paragraph C.4a. Reinforcing bars with a radius of curvature of less than 60 ft are tested using only sister splices with the following frequency for each splicing crew:

    1 sister splice for the first 10 production splices.

    4 sister splices for the next 90 production splices.

    3 sister splices for the next and subsequent units of 100 production splices.

    If any sister splice used for tensile testing fails to equal or exceed 125 percent of the minimum yield strength specified for the reinforcing bar, or the average tensile strength of each group of 15 consecutive samples fails to equal or exceed the guaranteed minimum tensile strength of the reinforcing bar, the individual Cadwelder is stopped and the procedure in Section C.5 of the regulatory guide is followed.

4. Paragraph C.5a If any completed mechanical splice fails to pass the individual inspection, and the rate of splices that fail the visual inspection does not exceed 1 for each 15 consecutive observed splices, the sampling program is started anew without requalifying the crew. If the failure rate exceeds 1 in 15 the crew is requalified.

    For tests failing the second criterion of two or more splices for any six additional samples, it is considered that the failure rate pertains to the total output of all splicers, and the previous 100 splices are evaluated accordingly.

Conformance to this guide is ensured through a purchase specification.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.11, Revision 0 (March 1971)

Instrument Lines Penetrating Primary Reactor Containment

FSAR Sections 6.2.4, 7.1.2.3

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

Instrument lines penetrating the containment are designed in accordance with this regulatory guide.

Regulatory Guide 1.12, Revision 2 (July 1981)

Instrumentation for Earthquakes

FSAR Section 3.7.4.1A

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

The technical requirements of the regulatory guide are implemented in the seismic portions of the procurement specification for seismic instrumentation.

Regulatory Guide 1.13, Revision 1 (December 1975) (For Comment)

Spent Fuel Storage Facility Design Basis

FSAR Sections 9.1.2, 9.1.3, 9.1.4

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

The design of the spent fuel storage facility incorporates the guidance listed in the regulatory position to assure that the fuel storage facility maintains the capability to perform its safety functions. An analysis of tornado protection for spent fuel storage is documented in a GE report entitled, Tornado Protection for Spent Fuel Storage Pool, APED-5696, November 1968.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.14, Revision 1 (August 1975) (For Comment)

Reactor Coolant Pump Flywheel Integrity

Position

This regulatory guide is not applicable to BWRs.

Regulatory Guide 1.15, Revision 1 (December 1972)*

Testing of Reinforcing Bars for Category I Concrete Structures

FSAR Sections 3.8.3, 3.8.4, 3.8.5

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide. Conformance to this regulatory guide is ensured through the purchase specification for reinforcing bars.

* This regulatory guide was withdrawn on July 8, 1981, by the NRC.

Regulatory Guide 1.16, Revision 4 (August 1975) (For Comment)

Reporting of Operating Information - Appendix A Technical Specifications

FSAR Sections Chapter 13, Technical Specifications

Position

NMPC complies with RG 1.16 Revision 4 (August 1975) in meeting the criteria of reporting of operating information.

This is delineated in the Nine Mile Point Nuclear Station Site Administration Procedure that is incorporated in the Technical Specifications for Unit 2.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.17, Revision 0 (June 1973)

Protection of Nuclear Power Plants Against Industrial Sabotage

FSAR Section 13.6

Position

The Unit 2 project designs, procures, and installs Unit 2 plant equipment and structures in accordance with this regulatory guide.

The Unit 2 project ensures that compliance is achieved by the following methods:

1. The Unit 2 project Security Design Review Committee reviews the design and arrangement of security-related plant equipment and structures for conformance with the position outlined above.

2. The Unit 2 project controls accessibility to Unit 2 security-related materials.

Tests and operability checks will be performed as required by this regulatory guide.

Regulatory Guide 1.18, Revision 1 (December 1972)*

Structural Acceptance Tests for Concrete Primary Reactor Containments

FSAR Section 3.8.1

Position

The Unit 2 project will comply with the Regulatory Position (Paragraph C) of this guide.

On February 10, 1976, NMPC transmitted the report, Primary Containment Structural Acceptance Test, to the NRC. This report describes the structural acceptance test in accordance with Section 13.6.3.38.2 of the PSAR.

* This regulatory guide was withdrawn on July 8, 1981, by the NRC. It has been superseded by RG 1.136 Revision 2 (June 1981).

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.18, Revision 1 (December 1972)* (cont'd.)

Conformance with this guide is provided by a test procedure outlined in Section 3.8.1.7.1.

Regulatory Guide 1.19, Revision 1 (August 1972)*

Nondestructive Examination of Primary Containment Liner Welds

FSAR Section 3.8.2

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) with the following alternate approaches:

1. Nondestructive test methods (Paragraph C.2.a) and acceptance standards (Paragraphs C.7.a, C.7.b, C.7.d) are in accordance with the effective ASME Section III addenda which correspond to the referenced Subarticle NE-5120.

2. Leak-tightness testing of leak chase system channel-to-liner welds will be in accordance with Paragraph C.1.d or methods of equal or greater sensitivity, such as halide leak detector testing or pressure testing with bubble solutions.

3. Liner seam welds that are not accessible for radiography after construction (Paragraph C.1.b) may be examined by the liquid penetrant method, magnetic particle method, or ultrasonic method.

4. The following are exceptions to Paragraph C.3:

   a. An exception is taken regarding the qualification of welders who welded the CRD insert and withdrawal piping instrumentation, and temperature monitoring penetration adaptor to sleeve welds. These welders were initially qualified to the rules of ASME IX except that their qualification for small-diameter pipe employed incorrect bend diameters. The welders were subsequently qualified to the full provisions of ASME IX.

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* This regulatory guide was withdrawn on July 8, 1981, by the NRC. It has been superseded by RG 1.136 Revision 2 (June 1981).

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.19, Revision 1 (August 1972)* (cont'd.)

b. An exception is taken regarding the qualification of the thermit welding process used to connect copper electrical grounding cables to the concrete side of thickened areas of the liner. The adequacy of this procedure was demonstrated by sectioning and metallurgically examining sample welds.

5. An exception to Paragraph C.2 is taken regarding the documentation reports for the nondestructive examination (NDE) of certain hatch and airlock welds. NDE reports for some welds in the personnel airlock, escape airlock, equipment hatches, and suppression chamber access hatch contain conflicting dates and, consequently, do not provide complete evidence of NDE. Subsequent review of documentation, engineering evaluation, and satisfactory pressure and leak tests demonstrates that the welds will maintain their structural integrity and leak-tightness.

Regulatory Guide 1.20, Revision 2 (May 1976)

Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing

FSAR Sections 3.9.2.4B, 14.2.7, 14.2.12

Position

NSSS analysis, design, and/or equipment utilized in this facility is in compliance with the intent of the subject regulatory guide through the incorporation of the alternate approach discussed in Section 3.9.2.4B.

Regulatory Guide 1.21, Revision 1 (June 1974)

Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants

FSAR Section 11.5.1

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.21, Revision 1 (June 1974) (cont'd.)

NMPC will submit semiannual reports specifying effluent release information to the NRC as required by 10CFR50.36a(a).(2), Technical Specifications on Effluents from Nuclear Power Reactors.

Regulatory Guide 1.22, Revision 0 (February 1972)

Periodic Testing of Protection System Actuation Functions

FSAR Sections 7.2.2.3, 7.3.2.1, 7.4.2.1, 7.6.2.5

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide as discussed in applicable portions of Chapter 7.

Regulatory Guide 1.23, Revision 0 (February 1972)

Onsite Meteorological Program

FSAR Section 2.3.3

Position

NMPC complies with RG 1.23 (Safety Guide 23) (February 1972). However, the wind speed sensor in use at the 200-ft level until July 1982 did not meet the requirements of the guide. The new sensor installed after July 1982 has the starting speed and accuracy specified by the regulatory guide.

The severe weather conditions encountered at Nine Mile Point lead to the choice of the very rugged wind speed sensor installed in November 1972. It had a starting speed of about 2.6 mph and would continue to operate with speeds of 1 to 1.5 mph. The wind speed accuracy is ±1.0 mph above 10 mph as opposed to the RG 1.23 Criterion I 0.5 mph for all wind speeds. More sensitive wind speed sensors available at that time were prone to icing and physical damage from high wind speeds.

The meteorological instrumentation installed after July 1982 meets the accuracy requirements outlined in Section C.4 of Proposed Revision 1 to RG 1.23 (September 1980). The operational

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.23, Revision 0 (February 1972) (cont'd.)

meteorological instrumentation system accuracies are listed in Table 2.3-5A.

Regulatory Guide 1.24, Revision 0 (March 1972)

Assumptions Used for Evaluating the Potential Radiological Consequences of Pressurized Water Reactor Radioactive Gas Storage Tank Failure

Position

RG 1.24 applies to PWR plants and, therefore, is not applicable to the Unit 2 project.

Regulatory Guide 1.25, Revision 0 (March 1972)

Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors

FSAR Section 15.7.4.6

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described in Section 15.7.4.5.

Regulatory Guide 1.26, Revision 3 (February 1976)

Quality Group Classification and Standards for Water-, Steam-, and Radioactive Waste-Containing Components of Nuclear Power Plants

FSAR Sections 3.2, 5.4.8.1, 9.2.2, 9.4.4, 9.5.6, 9.5.7, and 9.5.8

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described in Tables 3.2-1 and 3.2-2.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.27, Revision 2 (January 1976)

Ultimate Heat Sink for Nuclear Power Plants

FSAR Section 9.2.5

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

The Unit 2 ultimate heat sink design was reviewed and approved by the NRC (then the AEC) during the PSAR review. Since that time, the circulating water system design was revised to incorporate a natural-draft cooling tower, and modifications were made to the service water system which were later approved by the NRC.

Regulatory Guide 1.28, Revision 2 (February 1979)

Quality Assurance Program Requirements (Design and Construction)

FSAR Section Chapter 17, Appendix B (QA Topical Report)

Position*

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

* This commitment is modified at the time of the QA Topical Report implementation. At that time, the QATR (see Appendix B) supersedes this commitment.

Regulatory Guide 1.29, Revision 3 (September 1978)

Seismic Design Classification

FSAR Sections 3.2.1, 5.4.8.1, 7.1.2.3, 8.3.1.2, 9.2.2, 9.4.4, 9.5.6, 9.5.7, 9.5.8, and 10.3.3

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described in Tables 3.2-1 and 3.2-2.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.30, Revision 0 (August 1972)

Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment

FSAR Sections 3.11 and 7.2, Chapter 17, Appendix B (QA Topical Report)

Position*

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

The Unit 2 Quality Assurance Program complies with RG 1.30 as described in Appendix VII of the Quality Assurance Manual for the project during construction. Unit 2 also complies with this regulatory guide as described in Chapter 17 of the FSAR.

RG 1.30 Revision 0 endorses IEEE-336-1971. Unit 2 Specification E061A, Electrical Installation, invokes IEEE-336-1977, which is more conservative than IEEE-336-1971.

Section 3 of IEEE-336 addresses the requirements for preinstallation verification of material and equipment. It also states that "it is not intended to duplicate inspections but rather to verify that items are in satisfactory condition for installation." Preinstallation verification includes the following:

1. Identification of materials and equipment.

2. Availability of procedures, instruction manuals, and special work instructions.

3. Review of records of storage and preventive maintenance measures.

4. Visual examination of materials and equipment to ensure physical integrity.

All these required verifications are addressed by the SWEC QA program for receipt, storage, and preventive maintenance inspections. These inspections meet the intent of IEEE-336, Section 3; therefore, additional preinstallation verification is not done for the following components and materials (all equipment, however, is subject to preinstallation verification):

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* This commitment is modified at the time of the QA Topical Report implementation. At that time, the QATR (see Appendix B) supersedes this commitment.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.30, Revision 0 (August 1972) (cont'd.)

1. Balance-of-plant electrical components and materials such as terminal blocks, fuses, connectors, lugs, mounting hardware, etc.

2. PGCC electrical components and materials that are shipped separately from the main panels by GE, e.g., relays, meters, switches, connectors, lugs, mounting hardware, etc.

The above components and materials are subject to in-process installation inspection and final installation inspections.

Regulatory Guide 1.31, Revision 3 (April 1978)

Control of Ferrite Content in Stainless Steel Weld Metal

FSAR Sections 4.5.1.2, 4.5.2.4, and 5.2.3.4

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approaches described below and in Section 5.2.3.4.

Procurement and erection specifications referencing ASME Section III include the requirements as delineated in RG 1.31 Revision 3.

As an alternative method of conformance, specifications for which all work has been essentially completed or is at a stage where cost of alteration would be prohibitive require compliance with the Interim NRC Position (BTP MTEB 5-1) on RG 1.31 Revision 1, except for:

1. Welding of austenitic stainless steel castings that contain a minimum of 5 percent delta ferrite.

2. Full penetration welds of 1/4-in thickness or less.

3. Welds in pipe of 2-in nominal diameter or less.

4. Welds made between other than austenitic stainless steel to austenitic stainless steel.

5. Fillet welds having a throat dimension of 3/8 in or less.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.32, Revision 2 (February 1977)

Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants

FSAR Sections 8.3.1.2 and 8.3.2.2

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described below and in Section 8.3.1.

Conformance with this regulatory guide is ensured through the criteria used in the design of the offsite and the safety-related onsite power systems and in sizing the battery chargers.

Regulatory Guide 1.33, Revision 2 (February 1978)

Quality Assurance Program Requirements (Operation)

FSAR Section Chapter 17

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

Regulatory Guide 1.34 (December 28, 1972)

Control of Electroslag Weld Properties

FSAR Sections 5.2.3.3; 5.2.3.4, and 5.3.1.4

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

Purchase and/or erection specifications include the requirements as delineated in the guide. Presently, no electroslag welding is anticipated for Unit 2 safety-related components.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.35, Revision 2 (January 1976)

In-service Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures

Position

This regulatory guide is not applicable to Unit 2.

Regulatory Guide 1.36, Revision 0 (February 1973)

Nonmetallic Thermal Insulation for Austenitic Stainless Steel

FSAR Sections 4.5.2.4, 5.2.3.2, and 6.1.1

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described below and in Section 5.2.3.2.

Nonmetallic thermal insulation for austenitic stainless steel, including filler material for encapsulated insulation, complies with RG 1.36, issued February 23, 1973, with the exception of packaging and shipping requirements of Paragraph C.1 of this guide. In lieu of controlled packaging and shipping, receipt inspection and tests are required by specification. This consists of visual inspection for physical or water damage to all cartons. Damaged cartons are segregated. The potentially contaminated insulation is not accepted unless randomly selected samples from each carton are shown to be acceptable after being resubjected to the production test outlined in RG 1.36.

Purchase and/or erection specifications include the requirements as delineated above. No nonmetallic insulation will be in direct contact with safety-related austenitic stainless steel fluid systems within the primary containment on the Unit 2 project.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.37, Revision 0 (March 16, 1973)

Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants -

FSAR Sections 4.5.1.4, 4.5.2.4, 6.1.1, 17.2, Appendix B (QA Topical Report)

Position*

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approaches described below.

1. Paragraph C.3 The water quality for final flushes of fluid systems and associated components is at least equivalent to the quality of the operating system water, except for the oxygen content.

2. Paragraph C.4 Expendable materials, i.e., inks and related products, temperature indicating sticks, tapes, gummed labels, wrapping materials (other than polyethylene), water-soluble dam materials, lubricants, NDT penetrant materials, and couplants that contact stainless steel or nickel alloy surfaces are in accordance with the Unit 2 Position for RG 1.38 Revision 2.

3. Due to seasonal conditions, freshwater from Lake Ontario will have an allowable upper pH limit of 8.5.

4. Upgraded piping systems and components constructed of carbon steel materials will meet Class B cleanness requirements except for final flushing/cleaning which may exhibit rust staining in accordance with Class C cleanness requirements.

The quality assurance requirements of RG 1.37 have been addressed in Appendix VII of the Quality Assurance Program Manual and Section 17 for the Unit 2 project.

Erection specifications and procedures for Category I fluid systems and associated components include the requirements of the guide as delineated above.

* This commitment is modified at the time of the QA Topical Report implementation. At that time, the QATR (see Appendix B) supersedes this commitment.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.38, Revision 2 (May 1977)

Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants

FSAR Section Chapter 17, Appendix B (QA Topical Report)

Position*

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

SWEC and NMPC QA program satisfies the QA requirements of RG 1.38 (Unit 2 QA Program Manual Appendix VII and Section 17).

* This commitment is modified at the time of the QA Topical Report implementation. At that time, the QATR (see Appendix B) supersedes this commitment.

Regulatory Guide 1.39, Revision 2 (September 1977)

Housekeeping Requirements for Water-Cooled Nuclear Power Plants

FSAR Section Chapter 17, Appendix B (QA Topical Report)

Position*

The Unit 2 project complies with the requirements of the Regulatory Position (Paragraph C) of this guide.

Erection and installation specifications establish the requirements and the QA provisions to ensure compliance with this guide. Additionally, the requirements are implemented by site administrative procedures.

* This commitment is modified at the time of the QA Topical Report implementation. At that time, the QATR (see Appendix B) supersedes this commitment.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.40, Revision 0 (March 16, 1973)

Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants

FSAR Sections 3.11.2 and 7.1.2.3

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

The Unit 2 design does not include any Class 1E, continuous-duty motors inside the primary containment.

Regulatory Guide 1.41 (March 1973)

Preoperational Testing of Redundant Onsite Electrical Power Systems to Verify Proper Load Group Assignments

FSAR Section Chapter 14

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

The requirements of this guide will be implemented through the preoperational test program.

Regulatory Guide 1.42, Revision 1 (March 1974)

Interim Licensing Policy on As-Low-As Practicable for Gaseous Radioiodine Releases from Light-Water-Cooled Nuclear Power Reactors

Position

This regulatory guide was withdrawn on March 18, 1976, by the NRC. Refer to the position statement for RG 1.111.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.43 (May 1973)

Control of Stainless Steel Cladding of Low-Alloy Steel Components

FSAR Sections 5.2.3.3, 5.3.1.4

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

Regulatory Guide 1.44, Revision 0 (May 1973)

Control of the Use of Sensitized Stainless Steel

FSAR Sections 4.5.2.4, 5.2.3.4, and 6.1.1.1

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) through the alternate approaches listed below:

BOP

1. Paragraph C.4 When components whose normal operating temperature is in excess of 200°F are subjected to welding after solution heat treating, the material is low carbon grades, or alternatively, the welds are resolution annealed. The only exception is in the inlet fitting in the SLCS explosive valve, which is Type 304 stainless steel.

2. Paragraph C.6 Also, materials that contain more than 0.03 percent carbon and have a normal operating temperature in excess of 200°F will be subjected to an intergranular corrosion test of the heat-affected zone (HAZ). The ASTM A708 standard may be used in lieu of the A262 standard to perform the HAZ intergranular corrosion test, except that the radius of the bend specimen is as specified in ASME Section IX, with the weld-base metal interface at the centerline of the bend.

3. Requirements for maintaining cleanliness, solution annealing, sensitization testing, resolution annealing, and the use of low carbon grades of material in accordance with the guide and the modifications set forth above will be identified in the specifications and procedures.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.44, Revision 0 (May 1973) (cont'd.)

NSSS

All wrought austenitic stainless steel was purchased in the solution heat-treated condition. Heating above 800°F was prohibited (except for welding) unless the stainless steel was subsequently solution annealed. For Type 304 steel with carbon content in excess of 0.035 percent carbon, purchase specifications restricted the maximum weld heat input to 110,000 Joules/in, and the weld interpass temperature to 350°F maximum. Welding was performed in accordance with Section IX of the ASME Boiler and Pressure Vessel Code. These controls were employed to avoid severe sensitization and to comply with the intent of RG 1.44.

Regulatory Guide 1.45 (May 1973)

Reactor Coolant Pressure Boundary Leakage Detection Systems

FSAR Sections 5.2.5.1, 5.2.5.9, and 11.5

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approaches described below and Section 5.2.5.9.

The interpretation given to Regulatory Position C.5 for the sensitivity and response time of each leakage detection system is consistent with equipment capabilities available in the industry.

Regulatory Guide 1.46 (May 1973)

Protection Against Pipe Whip Inside Containment

FSAR Sections 3.6.2A, 3.6.2B, and 9.2.2

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approaches described in Sections 3.6.2A and 3.6.2B, with exceptions as permitted by NRC Generic Letter 87-11, dated June 19, 1987.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.47, Revision 0 (May 1973)

Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems

FSAR Sections 7.1.2, 7.4.2, 7.6.2, and 8.1.7

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

All systems that initiate reactor trip or respond to reactor accidents, such as the core standby cooling system, and the containment isolation system, are designed to conform to this position.

Regulatory Guide 1.48 (May 1973)

Design Limits and Loading Combinations for Seismic Category I Fluid System Components

FSAR Sections 3.9.3.1A, 3.9.3B, and Table 3.9B-3

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approaches described below.

BOP

Complies with the Regulatory Position of this guide.

NSSS

Complies through the alternate approach described in Table 3.9B-3.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.49, Revision 1 (December 1973)

Power Levels of Nuclear Power Plants

FSAR Sections 5.2.2, 15.0.3, Appendix A

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide for overpressure and transient analyses which do not impact the minimum critical power ratio (MCPR). These analyses will be performed at 102% of core power level.

Alternate Position

For pressurization events that could establish the operating limit MCPR, the Unit 2 project will implement an alternate assessment to Regulatory Positions C.2 and C.3. These events will be analyzed assuming a core power level of 100 percent of nuclear boiler rated power. This alternate assessment will be used when the GEMINI methodology of the computer code ODYN is used, as approved by the NRC (reference letter from G. C. Lainas (NRC) to J. S. Charnley (GE), "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A, Supplement to Amendment 11," dated March 22, 1986).

Regulatory Guide 1.50, Revision 0 (May 1973)

Control of Preheat Temperature for Welding of Low-Alloy Steel

FSAR Sections 5.2.3.3 and 6.1.1

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approaches described below:

BOP

1. In cases where it is impractical to maintain preheat until a postweld heat treatment has been performed, a temperature of 300°F or the applicable preheat temperature (whichever is higher) will be maintained for 2 hr/in of thickness in lieu of Paragraph C.2.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.50, Revision 0 (May 1973) (cont'd.)

2. The project interprets "low-alloy steels" to include those steels listed as low-alloy steels in mandatory Appendix I to ASME Section III.

3. Paragraph C.1.a of the regulatory guide requires that a minimum preheat be specified in the procedure qualification, while none is required by ASME Section III. The project therefore specifies the recommended preheats of nonmandatory Appendix D to ASME Section III.

NSSS

1. The use of low-alloy steel is restricted to the reactor pressure vessel. Other ferritic components in the RCPB are fabricated from carbon steel materials.

2. Preheat temperatures employed for welding of low-alloy steel meet or exceed the recommendations of ASME Section III, Subsection NA. Components were either held for an extended time at preheat temperature to assure removal of hydrogen, or preheat was maintained until postweld heat treatment. The minimum preheat and maximum interpass temperatures were specified and monitored.

3. All pressure-retaining welds were nondestructively examined by radiographic methods. In addition, a supplemental ultrasonic examination was performed.

Regulatory Guide 1.51 (May 1973)

In-service Inspection of ASME Code Class 2 and 3 Nuclear Power Plant Components

Position

This regulatory guide was withdrawn by the NRC on July 21, 1981.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.52, Revision 2 (March 1978)

Design, Testing, and Maintenance Criteria for Engineered Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Unit of Light-Water-Cooled Nuclear Power Plants

FSAR Sections 6.5.1 and 9.4.1

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approaches described below:

1. Paragraph C.2.a Demisters will be provided only where entrained water droplets could be present. The control building supply air system special filter train (Section 9.4.1) does not require a demister because:

    a. Air entering the filter train is a mixture of recirculated and outdoor air, with sufficient physical separation between the outdoor air louver and filter train to prevent moisture carryover.

    b. The entering air side of the filter train is provided with an electric heating coil, limiting the relative humidity of entering air to 70 percent.

2. Paragraph C.2.d Filter mounting frames and ducts located outside the containment are not designed to withstand DBA pressure surges.

3. Paragraph C.2.h The following exceptions are taken to the requirement that "all instrumentation and equipment controls should be designed to IEEE-279-1971":

    a. All instruments and equipment controls that sense or process one or more variables and that act to accomplish the protective function are designed in accordance with IEEE-279. These include sensors, signal conditioners, logic, and actuation device control circuitry. (The protective function with which the subject guide is concerned is atmospheric cleanup to mitigate accident doses.)

    b. In addition, a very limited class of analog indicators may be designed in accordance with selected applicable paragraphs of IEEE-279. The basis for selecting specific indicators to be so designed is their significance to safety. All

Regulatory Guide 1.52, Revision 2 (March 1978) (cont'd.)

paragraphs of IEEE-279 are applicable, except 4.12, 4.13, 4.15, 4.16, and 4.17. For this limited class of indicators, redundant analog channels are provided, one of which is recorded. The systems are designed to operate before and after, but not necessarily during, a SSE.

c. Annunciator functions are incorporated into overall system design. Annunciators for the ESF filter systems (Sections 6.5.1 and 9.4.1) are not safety related; therefore, they are not designed in accordance with IEEE-279.

4. Paragraph C.2.j ESF atmosphere cleanup systems are designed to be removed as a minimum number of segmented sections. Individual filter components will be removed prior to cutting the housing into segmented sections.

5. Paragraph C.2.1 ESF atmospheric cleanup filter housing will be designed and tested in accordance with the requirement of this paragraph. Exception is taken to the leak testing of ductwork, specifically to Section 4.12 of ANSI-N509-1980, for the two ESF atmospheric cleanup systems as follows:

Control Room Filtration Units

HVAC systems (i.e., ductwork, dampers, fans, etc.) within the control room pressure boundary (El 288'-6" and 306'-0" of the control building) will be leak tested to minimize air leakage to a reasonably achievable level. Air cleaning effectiveness, duct and housing quality requirements, and health physics requirements leak rates are not applicable to HVAC systems within this pressure boundary.

Elevations 288'-6" and 306'-0" of the control building are maintained at a positive pressure relative to their surroundings, thus precluding infiltration of potentially-contaminated air. Air in the suction side of the filtration unit fan is at a negative pressure relative to the pressure boundary, thus ensuring in-leakage and filtration of air. Air leaving the filtration units is clean and ultimately will be discharged into the pressure boundary environment. Thus, leakage of air from ductwork, dampers, fans, etc., downstream of the filters will be clean air and is not of concern.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.52, Revision 2 (March 1978) (cont'd.)

The control room ventilation system will satisfy the intent of air-cleaning effectiveness, duct and housing quality, and health physics requirements, by verifying through tests that the airflows are balanced, positive pressure is achieved, and filter effectiveness is maintained.

Standby Gas Treatment System (SGTS)

The SGTS consists of filter trains, fans, piping, and valves. ASME III piping and valves are used in this system in lieu of ductwork and dampers. A portion of the fan discharge section is through a concrete tunnel that travels underground to the main stack. The major portion of the system (i.e., filter trains, fans, piping, and valves) is located in a clean interspace (standby gas treatment building).

Air-cleaning effectiveness, duct (pipe) and housing quality, and health physics requirement leak rates will be used for all pipe under positive pressure that discharges into the plant stack.

Air-cleaning effectiveness, duct (pipe) and housing quality, and health physics requirement leak rates are not applicable to the suction side of the filtration unit fans since air in the suction side is at a negative pressure relative to the surroundings, thus ensuring in-leakage and filtration of air.

6. Paragraph C.3.d Deleted.

7. Paragraph C.3.e Filter and adsorber mounting frames are constructed and designed in accordance with the recommendations of Section 4.3 of ERDA 76-21, except for the frame tolerance guidelines in Table 4.2. The tolerances selected for HEPA and adsorber mountings are sufficient to satisfy the bank leak test criteria of Paragraphs C.5.c and C.5.d of RG 1.52 Revision 2.

8. Paragraph C.3.h Exception is taken to the recommendations of Section 4.5.8 of ERDA 76-21 relative to drain sizes and arrangement. Manual valves, in addition to water seals and traps, will be provided to control the discharge of the fire sprinkler flow (see Figure 1.8-1).

9. Paragraph C.3.i Exception is taken to the requirement that the absorption unit should be designed for a

Regulatory Guide 1.52, Revision 2 (March 1978) (cont'd.)

maximum loading of 2.5 mg of total iodine per gram of activated carbon. RG 1.52 Revision 1 states that "the absorption unit should have the capacity of loading 2.5 mg of total iodine (radioactive plus stable) per gram of activated carbon." The absorption unit provided has a loading capacity of 10.0 mg of total iodine per gram of activated carbon.

10. Paragraph C.3.k Exception is taken to the requirement for humidity control to below 70 percent relative humidity for low flow air bleed cooling.

    Each filter train is physically separated, and the common connection between the filter trains is provided with redundant high temperature sensors and isolation valves to maintain equipment integrity in one filter train upon detection of high temperature.

11. Paragraph C.3.1 System resistances will be determined in accordance with Section 5.7.1 of ANSI N509-1976 except that fan inlet and outlet losses will not be calculated in accordance with AMCA 201, but will be estimated and documented accordingly.

    Exception is taken to balancing techniques defined in Section 5.7.3 of ANSI N509-1976. The acceptable amplitude of vibration, peak to peak, in any plane measured on the shaft adjacent to the bearings, corresponds to a vibration velocity of 0.1 in/sec at the rated speed using the displacement values given in AMCA Publication 801. The displacement criteria using maximum vibration velocity method in accordance with ANSI N509-1976 are not required by the later ANSI N509-1980. The acceptable criteria of the later standard are similar to that of AMCA Publication 801. Documentation will not be furnished in accordance with Section 5.7.5 where AMCA certification ratings are submitted.

12. Paragraph C.3.n Exception is taken to Section 5.10.3.5 of ANSI N509-1976; ductwork which consists of sheet sections and is considered as a complete structure, will have a resonant frequency above 25 Hz.

13. Paragraph C.3.p Exception is taken to the provisions in Section 5.9 of ANSI N509-1976 of designing dampers to ANSI B31.1 and to using butterfly valves. Class B dampers may be designed and tested to meet the

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.52, Revision 2 (March 1978) (cont'd.)

verification of strength and leak-tightness necessary for use in a contaminated air stream. (NOTE: This exception does not pertain to containment penetrations.)

In addition, exceptions are taken to the following:

Minimum diameter of damper shaft length 24 in and under will be 1/2 in; and 3/4 in for shafts between 25 and 48 in.

14. Paragraph C.4.a Exception is taken to full compliance with Section 2.3.8 of ERDA 76-21; i.e., SWEC does not use any communication system, floor drains are as noted in Paragraph C.3.h above, decontamination areas and showers are not "nearby," filters are not used at duct inlets, and duct inspection hatches are not provided.

15. Paragraph C.4.d ESF atmosphere cleanup systems are tested once a month to ensure that all operating components are functional as required. Electric heaters within the filter system are automatically energized during the test to ensure that air stream relative humidity does not rise above 70 percent, thereby preventing the buildup of moisture on the adsorber and HEPA filters. The filter systems are scheduled to be tested a minimum of 10 hr/month; however, the test duration may be reduced if the buildup of moisture between tests proves to be insignificant, as substantiated by actual test data.

Regulatory Guide 1.53, Revision 0 (June 1973)

Application of the Single Failure Criterion to Nuclear Power Plant Protection Systems

FSAR Sections 7.2.2, 7.3.2, 7.6.2, and 9.2.2

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

All systems that initiate reactor trip or respond to reactor accidents, such as containment isolation system and core standby cooling systems, will be designed in conformance with this guide.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.54 (June 1973)

Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants

FSAR Sections 6.1.2, Chapter 17

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approaches described below.

NMPC and BOP

In lieu of the inspection in accordance with ANSI N5.9-1967, as defined in Section 6.2.4 of ANSI N101.4-1972, inspection will be in accordance with ANSI N512-1974, Section 10, Inspection for Shop and Field Work. Also, the documentation of prime coating of structural shapes used in pipe, conduit, and instrumentation support fabrication meets all aspects of the regulatory guide, except that the exact batch number of qualified paint used on these items may not be traceable to the specific support.

NSSS

ANSI N101.4-1972, in conjunction with ANSI N45.2-1971, provides an adequate basis for complying with quality assurance requirements for protective coatings applied to ferritic steels, aluminum, stainless steel, galvanized steel, concrete, or masonry.

Most NSSS equipment for this plant is coated with a prime coat of inorganic zinc. This coating was one of the first to be qualified under ANSI N101.2 for DBA, radiation, etc., in nuclear applications. Equipment specifications in place at the time of ordering equipment for this plant specified inorganic zinc.

There is a minimum amount of unqualified paint which is addressed in FSAR Section 6.1.2 and Table 6.1-3. Equipment tightly covered with thermal insulation is not included in this total since potential paint debris could not escape to the suppression pool during a LOCA.

The quality assurance requirements in the regulatory guide were not imposed on painted material and paint application since most NSSS equipment was ordered prior to issuance of the guide.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.55 (June 1973)*

Concrete Placement in Category I Structures

FSAR Sections 3.8.3, 3.8.4, 3.8.5

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

Conformance to this regulatory guide is ensured through purchase specifications.

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* This regulatory guide was withdrawn on July 8, 1981, by the NRC. It is superseded by RG 1.136 Revision 2 (June 1981).

Regulatory Guide 1.56, Revision 1 (July 1978) (For Comment)

Maintenance of Water Purity in Boiling Water Reactors

FSAR Section 5.2.3.2, Technical Specifications

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

The design of the Unit 2 condensate demineralizer system is in conformance with the regulatory guide.

The Unit 2 Technical Specifications will describe the operating requirements to conform with this guide.

Regulatory Guide 1.57, Revision 0 (June 1973)

Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components

Position

RG 1.57 applies to power plants with metal primary containments and, therefore, is not applicable to the Unit 2 project.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.58, Revision 1 (September 1980)

Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel

FSAR Section 14.2

Position

During the design and construction phase, startup and test personnel involved in testing met the requirements of RG 1.58 and ANSI N45.2.6-1978, with exceptions as discussed in Chapter 14.

Unit 2 plant personnel met the requirements of this regulatory guide as discussed in Chapter 13.

GE startup operations personnel supporting the startup test phase met the requirements of this regulatory guide as discussed in Table 14.2-403.

During the operations phase, the qualification of nuclear power plant inspection, examination, and testing personnel is stated in the NMPC QA Program requirements and is satisfied as specified in the Quality Assurance Program Topical Report (QATR) for Nine Mile Point Nuclear Station Units 1 and 2 - Operations Phase (see Appendix B).

Regulatory Guide 1.59, Revision 2 (August 1977)

Design Basis Floods for Nuclear Power Plants

FSAR Section 9.4.4, 14.2.7.3, Chapter 17

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide, with the following limitation:

No commitments for compliance are made or implied for the "to be issued" appendices.

The Unit 2 site has hardened protection from flooding by use of a lakefront revetment ditch.

Evaluation of the conditions (Paragraph C.1) resulting in the worst site-related flood probable at the Unit 2 site has been made in conformance with ANSI N170-1976/ANS 2.8. The combined events considered were:

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.59, Revision 2 (August 1977) (cont'd.)

1. Probable maximum surge and seiche with wind wave action and maximum controlled lake level.

2. Probable maximum precipitation and historical maximum lake level.

3. Probable maximum lake level and historical maximum precipitation.

The analysis showed that Unit 2 is designed to withstand these combined events with no safety impact.

Regulatory Guide 1.60, Revision 1 (December 1973)

Design Response Spectra for Seismic Design of Nuclear Power Plants

FSAR Sections 3.7A, 3.7B

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide as discussed below and Sections 3.7A and 3.7B.

The design response spectra has been used to generate the seismic data sheets for equipment loadings, for systems and component analysis, and for the structural responses of the various buildings.

Regulatory Guide 1.61, Revision 0 (October 1973)

Damping Values for Seismic Design of Nuclear Power Plants

FSAR Sections 3.7A, 3.7.1B

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

The damping values used for the design of Category I structures, systems, and components for Unit 2 meet the requirements of this guide.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.61, Revision 0 (October 1973) (cont'd.)

The damping values in the guide are used in conjunction with RG 1.60 for fully defining seismic design criteria.

Regulatory Guide 1.62, Revision 0 (October 1973)

Manual Initiation of Protective Actions

FSAR Sections 7.1.2, 7.2.2, 7.3.2, 7.4.2, 7.6.2, 8.1.3, 8.3.1

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

All systems that initiate reactor trip or respond to reactor accidents, such as core standby cooling systems and the containment isolation system, are designed to conform to this guide.

Regulatory Guide 1.63, Revision 2 (July 1978)

Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants

FSAR Sections 3.11, 7.1.2, 8.3.1

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide, except as noted as follows:

In lieu of the installation, inspection, and testing requirements in Section 8.1 of IEEE-317-1976, the electrical penetrations are installed, inspected, and tested in accordance with SWEC quality standards, QA and EA procedures relating to Category I installation of safety-related equipment, which conform to 10CFR50 Appendix B. It may be noted that the penetrations are attached (bolted) to the containment liner, which is a noncode-stamped component.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.64, Revision 2 (June 1976)

Quality Assurance Requirements for the Design of Nuclear Power Plants

FSAR Sections Chapter 17

Position

RG 1.64 Revision 2, dated June 1976, is not applicable to Unit 2. The regulatory guide revision specifies that the revision is applicable to evaluation of submittals in connection with construction permit application docketed after July 15, 1975. The Unit 2 construction permit application was docketed on June 15, 1972.

Regulatory Guide 1.65, Revision 0 (October 1973)

Materials and Inspections for Reactor Vessel Closure Studs

FSAR Section 5.3.1.7, Chapter 14

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described below.

The reactor pressure vessel closure studs are SA-540 Grade B23 or 24 (AISI 4340) and have a maximum ultimate tensile strength of 169 ksi. Additionally, GE has specified the bolting material must have Charpy V-notch impact properties of 45 ft-lbs minimum with 25 mil lateral expansion. NDE before and after threading is specified to be in accordance with Subsubarticle NB-2580, ASME Section III, which complies with Regulatory Position C.2. Subsequent to fabrication, the studs are manganese phosphate coated and are lubricated with a graphite/alcohol or a nickel powder base lubricant.

In relationship to Regulatory Position C.2.b, the bolting materials were ultrasonically examined after final heat treatment and prior to threading, as specified. The specified requirement for examination according to SA-388 was complied with. The procedures approved for use in practice were judged to insure comparable material quality and, moreover, were considered adequate on the basis of compliance with the applicable requirements of ASME Code, Paragraph NB-2583. Additionally, straight beam examination was performed on 100 percent of cylindrical surfaces, and from both ends of each stud using a 3/4

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.65, Revision 0 (October 1973) (cont'd.)

maximum diameter transducer. In addition to the code-required notch, the reference standard for the radial scan contained a 1/2-in diameter flat bottom hole with a depth of 10 percent of the thickness, and the end scan standard contained a 1/4-in diameter flat bottom hole 1/2 in deep. Also, angle beam examination was performed on the outer cylindrical surface in both a flat and circumferential direction. Surface examinations were performed on the studs and nuts after final heat treatment and threading, as specified in the guide, in accordance with Paragraph NB-2583 of the applicable ASME Code.

Radial scan calibration is based on a 1/2-in (12.7-mm) diameter flat bottom hold of a depth equal to 10 percent of the material thickness. Angle beam examination is performed on the outer cylindrical surface of nuts and washer in accordance with ASME SA-388 in both axial and circumferential directions. Any indication greater than the indication from the applicable calibration feature is unacceptable. A distance-amplitude correction curve in accordance with paragraph NB-2585 is used for the longitudinal wave examination.

In relationship to Regulatory Position C.3, GE practice allows exposure of stud bolting surfaces to high purity fill water; nuts and washers are dry-stored during refueling.

An in-service inspection is described in Section 14 of the FSAR.

Regulatory Guide 1.66, Revision 0 (October 1973)

Nondestructive Examination of Tubular Products

Position

This regulatory guide was withdrawn by the NRC on September 28, 1977.

Regulatory Guide 1.67, Revision 0 (October 1973)

Installation of Overpressure Protective Devices

FSAR Section 3.9.3A

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.67, Revision 0 (October 1973) (cont'd.)

For ASME Code Class 1, 2, and 3 piping analysis, transient analysis and dynamic loading are considered in the stress allowable limits.

Regulatory Guide 1.68, Revision 2 (August 1978)

Initial Test Programs for Water-Cooled Reactor Power Plants

FSAR Section Chapter 14

NMPC Position

Unit 2 complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described below.

The format of the test procedures may differ from that described in Appendix C of this guide. However, all required elements are included. A detailed test program is provided in Chapter 14 of the FSAR.

Regulatory Guide 1.68.1, Revision 1 (January 1977)

Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants

FSAR Sections Chapter 14

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

The condensate and feedwater systems will be preoperationally tested in accordance with RG 1.68. The reactor level control components will be calibrated and functionally checked to the maximum extent practical prior to fuel loading. During the startup phase, reactor level control will be dynamically tested at various power levels.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.68.2, Revision 1 (July 1978)

Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants

FSAR Section Chapter 14

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

Regulatory Guide 1.68.3, Revision 0 (April 1982)

Preoperational Testing of Instrument and Control Air Systems

FSAR Section Chapter 14

Position

The Unit 2 project complies with the intent of this regulatory guide by taking an alternative approach.

Unit 2 follows the guidelines in RG 1.80.

The Unit 2 instrument air system is not nuclear safety related. Portions of RG 1.68.3 are addressed since they are implemented by RG 1.68. Otherwise, RG 1.68.3 is not applicable to Unit 2 since its instrument air system is not safety related.

The implementing activities described in Positions C.1, 2, 3, 4, 5, 6, 9, and 10 of RG 1.68.3 are typical for the testing of instrument air systems. The project complies with the requirements of these regulatory positions.

Regulatory Positions C.8 and 11 of RG 1.68.3 are applicable to projects with a nuclear safety-related instrument air system; therefore, these positions are not applicable to Unit 2.

The instrument air system does interface with components that are part of nuclear safety-related systems. Tests will be conducted on such components in accordance with RG 1.68.3 to demonstrate that these components function as designed upon loss of their nonnuclear safety-related air supply.

To verify that loss of instrument air will not affect any safety-related system from performing its designated function, a loss-of-air-supply test will be performed on those portions of the instrument air system which interface with nuclear

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.68.3, Revision 0 (April 1982) (cont'd.)

safety-related systems. This test will demonstrate that air-controlled components supplied directly from the instrument air system will assume their design fail-safe positions upon loss of air supply. This test will also demonstrate that those components provided with safety-related air accumulators will properly isolate from the instrument air supply system while retaining sufficient stored air capacity to perform their designated safety functions.

Testing will include a verification that no crossties between service and instrument control air degrade the system. Sudden pressure drops in the entire system are not feasible because the design incorporates excess flow checks. However, all excess flow check valves will be verified to be operable.

Regulatory Guide 1.69, Revision 0 (December 1983)

Concrete Radiation Shields for Nuclear Power Plants

FSAR Sections 3.8.4, 12.3

Position

The Unit 2 project complies with the Regulatory Position (Section C) of this guide through the alternate approaches described below.

Coatings for concrete surfaces comply with the requirements of Paragraph C.3.g of RG 8.8 and ANSI N5.12, Protective Coatings (Paints) for the Nuclear Industry.

1. The increase in allowable stresses for earthquake forces normally permitted by ACI 318 will not be used.

2. Finishing and patching of concrete surfaces after removal of forms will conform to Chapter 9 of ACI 301 instead of to Section 8.7.5 of ANSI N101.6. It is not necessary or customary to complete this work within 96 hr after the placing of concrete.

3. Testing of shielding by a point source to determine the adequacy of the shields will not be performed. Radiation levels of the actual radiation sources are monitored on a periodic basis throughout the plant where personnel access is required, including during startup testing when radiation levels are measured throughout the plant for various power levels.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.69, Revision 0 (December 1983) (cont'd.)

Further, the density of the concrete shielding is continuously monitored by the quality control program during construction to ensure the adequacy of the protection.

Regulatory Guide 1.70, Revision 3 (November 1978)

Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants - LWR Edition

Position

The Unit 2 FSAR has been prepared following the format and content requirements contained in this regulatory guide, except in some cases where the content requirements were described or presented to conform to the Unit 2 established programs. The differences are not considered an exception, as the material is presented in a manner consistent with the intent of this regulatory guide.

Regulatory Guide 1.71 (December 1973)

Welder Qualification for Areas of Limited Accessibility

FSAR Sections 4.5.2.4, 5.2.3.3, 5.2.3.4, 6.1.1.1

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approaches described below and Sections 4.5.2 and 5.2.3. Welder qualification for areas of limited accessibility in Section III fabrications complies with RG 1.71 Revision 0 which was issued in December 1973. Paragraph C of the regulatory guide will be implemented as follows:

1. Low-alloy steels and high-alloy steels include steels listed as such in the table in Appendix I of the ASME Section III Code.

2. Applicable weld joint designs include full and partial penetration groove, fillet, and socket welds.

3. Section III applicability includes all subsections of ASME Section III.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.71 (December 1973) (cont'd.)

4. Limited accessibility will be determined only as it relates to the direction from the joint from which welding is to be performed, not to any direction from the joint. Limited accessibility is defined as:

   a. When the welder cannot see the entire weld joint without visual aid.

   b. When the welder can see the entire weld joint but cannot manipulate his electrode to achieve adequate fusion.

An acceptable alternative to this position is as follows:

In lieu of paragraphs C1 and C2a, all applicable full penetration welds of limited accessibility are volumetrically inspected to the requirements and standards of Section III, Class 1.

Regulatory Guide 1.72, Revision 2 (November 1978)

Spray Pond Piping Made From Fiberglass-Reinforced Thermosetting Resin

Position

RG 1.72 applies to power plants with spray ponds and therefore is not applicable to the Unit 2 project.

Regulatory Guide 1.73, Revision 0 (January 1974)

Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants

FSAR Sections 3.11, 7.1.2

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approaches described in Sections 3.11 and 7.1.2.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.74 (February 1974)

Quality Assurance Terms and Definitions

FSAR Section Chapter 17

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

Regulatory Guide 1.75, Revision 2 (September 1978)

Physical Independence of Electric Systems

FSAR Sections 7.1.2, 7.6.2, 8.3.1

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described below and in Section 7.6.2 and 8.3.1.

Regulatory Position C.1 states that "interrupting devices actuated only by fault current are not considered to be isolation devices within the context of this document." In the case of control and instrument circuits, a combination of two interrupting devices actuated by fault current have been used to isolate non-Class 1E circuits from Class 1E circuits. Both of these devices are Class 1E, and both of them are coordinated with the main breaker upstream so that a failure of a non-Class 1E device or circuit will not affect any Class 1E device or system. Any circuit breakers associated with this redundant protection will be tested during each refueling outage.

Regulatory Position C.9 requires that cable splices in raceways be prohibited. Splicing in electrical penetrations for termination is considered to be exempt from this requirement.

Regulatory Position C.10 requires that the cables be marked at 5-ft intervals. This is a typographical error as confirmed by the former Electrical, Instrument and Control Branch Chief of USNRC, T. A. Ippolito, on October 10, 1975, and the NRC Power Systems Branch Section Leader, R. G. FitzPatrick, on October 30, 1980. The correct distance is 15 ft, which has been followed in Unit 2. Additionally, the cable markings are inspected (100 percent) by Field Quality Control during installation. As of June 1984 more than 50 percent of all cables had been pulled and marked at 15-ft intervals. We believe that mixing the marking of

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.75, Revision 2 (September 1978) (cont'd.)

the cables is inappropriate and that marking at 15-ft intervals is sufficient to ensure separation of cables.

The minimum separation distance from 600 V or less nonsafety-related conduit to safety-related open cable trays and cable in free air for any service level is 1 in.

All cables used in Unit 2 are flame retardant. The cable trays are not filled above the side rails. The hazard, in this case, is limited to failure or faults internal to the nonsafety cables in rigid steel conduit. Unit 2 has determined by analysis that 1-in separation between the Class 1E cable tray and non-Class 1E conduit provides adequate protection for the Class 1E cables in the open ladder tray in the event of any failure of the non-Class 1E cables in conduit. This has been established by tests with 600 V levels, as explained later in this section.

Aluminum sheath cables (ALS) used for low-energy 120-V ac systems and 8-hr battery-pack lighting systems, are considered enclosed raceways. These cables have flame-retardant cross-linked polyethylene insulation, chlorosulphonated polyethylene jacket, and polypropylene fillers enclosed in a continuous, impervious aluminum sheath which provides adequate protection. As such, the minimum separation between these cables and Class 1E raceways is 1 in.

The minimum separation between any Class 1E raceway and any lighting cord for drops to the lighting fixtures shall be 1 in. These cords are of size 12 AWG and supply 120/208 V ac low energy in low-density applications. As such, 1-in separation provides adequate protection to the Class 1E circuits in the event of a fault in any lighting cord.

IEEE-384-1974, Section 5.1.1.2, allows lesser separation distances than those specified in Sections 5.1.3 and 5.1.4, if established by analysis. Various tests have indicated that the following minimum separation distances between redundant Class 1E cables and raceways, or between Class 1E and non-Class 1E cables and raceways, 600 V level and below, should be adequate to maintain independence of the redundant systems. NMPC also has verified these minimum separation distances by plant-specific tests (Wyle Test Report No. 47906-02, Electrical Separation Verification Testing).

Cable tray to cable tray	10 in horizontal or 10 in vertical
Cable tray to conduit	1 in

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.75, Revision 2 (September 1978) (cont'd.)

Cable in free air to conduit	1/2 in
Cable in free air to cable in free air	10 in vertical or 10 in horizontal
Cable in free air to cable tray	10 in vertical or 10 in horizontal
Wrapped cable to unwrapped cable	0 in
Conduit to conduit	1/2 in
Class 1E control/instrument cable to non-Class 1E control instrument cable inside control/instrument cabinets	1 in

Where the minimum separation distances specified in Sections 5.1.3 and 5.1.4 of IEEE-384-1974 cannot be maintained due to physical arrangements, the minimum separation distances specified above shall be maintained.

Where the minimum separation distances specified in this section cannot be maintained, enclosed raceways will be used; or a separation barrier will be installed.

A comparison of the Unit 2 design to the criteria contained in RG 1.75 and IEEE-384-1974 is shown in Appendix 7B for instrumentation and control systems within the PGCC and balance of plant.

Regulatory Guide 1.76, Revision C (April 1974)

Design Basis Tornado for Nuclear Power Plants

FSAR Section 3.3

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.76, Revision C (April 1974) (cont'd.)

All applicable structures, systems, or components important to safety are designed to withstand, or are enclosed in structures that will withstand, the six descriptive parameters given in Table I of this regulatory guide. The parameters of Region 1 are applicable to this facility.

Regulatory Guide 1.77 (May 1974)

Assumptions Used for Evaluating a Control Rod Ejection Accident for PWRs

Position

RG 1.77 applies to PWR plants and is not applicable to the Unit 2 project.

Regulatory Guide 1.78, Revision 0 (June 1974)

Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release

FSAR Section 6.4.2.3

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

Regulatory Guide 1.79, Revision 1 (September 1975)

Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors

Position

RG 1.79 applies to PWR plants and is not applicable to the Unit 2 project.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.80 (June 1974)

Preoperational Testing of Instrument Air Systems

FSAR Section 9.3.1

Position

This regulatory guide was withdrawn by the NRC on April 20, 1982. It has been superseded by RG 1.68.3 Revision 0 (April 1982).

Regulatory Guide 1.81, Revision 1 (January 1975)

Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants

Position

RG 1.81 applies to multi-unit nuclear power plants and is not applicable to the Unit 2 project because Unit 2 does not share any emergency or shutdown electric systems.

Regulatory Guide 1.82 (June 1974)

Sumps for Emergency Core Cooling and Containment Spray Systems

Position

RG 1.82 applies to PWR plants and is not applicable to the Unit 2 project.

Regulatory Guide 1.83, Revision 1 (July 1975)

In-service Inspection of PWR Steam Generator Tubes

Position

RG 1.83 applies to PWR plants and is not applicable to the Unit 2 project.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.84, Revision 22 (July 1984)

Design and Fabrication Code Case Acceptability - ASME Section III Division I

FSAR Section 5.2.1.2

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approaches described below.

RG 1.84 provides a list of ASME Design and Fabrication Code cases that have been generically approved by the regulatory staff. Code cases on this list may, for design purposes, be used until appropriately annulled. Annulled cases are considered "active" for equipment that has been contractually committed to fabrication prior to the annulment.

The various ASME Code cases that were applied to components in the RCPB are listed in Table 5.2-1. All Safety Class 2 and 3 equipment has been designed to ASME Code or ASME-approved Code cases. This provision, together with the quality control programs, provides adequate safety equipment functional assurances.

Regulatory Guide 1.85, Revision 22 (July 1984)

Materials Code Case Acceptability - ASME Section III Division I

FSAR Section 5.2.1.2

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approaches described below.

RG 1.85 provides a list of ASME design and fabrication code cases that have been generically approved by the regulatory staff. Code cases on this list may, for design purposes, be used until appropriately annulled. Annulled cases are considered "active" for equipment that has been contractually committed to fabrication prior to the annulment.

The various ASME Code cases that applied to components in the RCPB are listed in Table 5.2-1.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.85, Revision 22 (July 1984) (cont'd.)

All Safety Class 2 and 3 equipment has been designed to ASME Code or ASME-approved Code cases. This provision, together with the quality control programs, provides adequate safety equipment functional assurances.

Regulatory Guide 1.86, Revision 0 (June 1974)

Termination of Operating Licenses for Nuclear Reactors

FSAR Section General Application

NMPC Position

At the termination of operation of Unit 2, NMPC will meet the criteria set forth in this regulatory guide and will fully comply with its requirements.

Regulatory Guide 1.87, Revision 1 (June 1975)

Guidance for Construction of Class 1 Components in Elevated-Temperature Reactors

Position

RG 1.87 applies to HTGR plants and is not applicable to the Unit 2 project.

Regulatory Guide 1.88, Revision 2 (October 1976)

Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records

FSAR Section Chapter 17, Appendix B (QA Topical Report)

Position*

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide, except to change ANSI N45.2.9-1974 Section 5.6, Paragraph 3, to "Two hour minimum rated facility" in accordance with NFPA 232-1980. Implementation is as described below.

Unit 2 quality assurance records (and other required records) are stored in facilities designated as the Permanent Plant File and the Records Acceptance Center. In-process records are stored in

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.88, Revision 2 (October 1976)

controlled Intermediate Storage Facilities. Specific requirements for each include:

1. Permanent Plant File Complies to the above paragraph of this position statement.

2. Records Acceptance Center Complies with ANSI N45.2.9-1974 Section 5.3 to provide a mechanism to control records. The storage facility shall meet Section 5.6 except as follows:

   a. Structure has a minimum 2-hr fire rating.

   b. Doors, frames, and hardware have a 2-hr vault door.

   c. Electrical facilities shall be limited to ceiling lights, air-conditioning units, smoke detectors, and alarm circuits.

3. Intermediate Storage Facilities Complies with ANSI N45.2.9-1974 Section 5.3 to provide a mechanism to control records. Each intermediate storage facility shall be evaluated by a Fire Protection Engineer to fulfill NFPA 232-1980 requirements. NOTE: All intermediate storage facilities will be eliminated as contractor work is concluded.

The above controls and facilities are prepared to protect quality assurance records which take their physical form as radiographs, microfilm, and paper.

1. Special handling and environmental storage considerations must be maintained for radiographs.

2. Designated archive (silver halide only) microfilm requires environmental storage considerations.

3. Use of fire-retardant cabinets is applicable to paper storage only.

* This commitment is modified at the time of the QA Topical Report implementation. At that time, the QATR (see Appendix B) supersedes this commitment.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.88, Revision 2 (October 1976) (cont'd.)

Technical Justification

ANSI N45.2.9-1974 does not adequately define the storage facilities for in-process quality records or NFPA requirements for fire rating of the facility. NFPA 232-1980, 1-3, emphasizes, "To consult with an experienced and competent Fire Protection Engineer or Records Protection Consultant." This position is based upon his recommendations. The Unit 2 Records Management Plan establishes the program for turnover, collection, review, transfer, receipt, verification, permanent plant file entry, and retention of all Unit 2 records with implementing policy guidelines which specify the facility types.

Regulatory Guide 1.89, Revision 0 (November 1974)

Qualification of Class 1E Equipment for Nuclear Power Plants

FSAR Sections 3.11, 7.1.2

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described below.

BOP

Determination of the radiation dose used for qualification of Class 1E plant equipment takes into account design features such as the location of equipment within or outside the containment, fission product cleanup of the containment atmosphere by the containment spray system, local shielding, the time period required for equipment operation, and spatial location. These design features will be applied in a conservative manner to realistically determine the radiation doses to which the devices must be qualified in addition to the other environmental factors.

NSSS

All environmental and seismic qualification testing of Class 1E equipment within GE's scope of supply was in compliance with IEEE-323-1971 and IEEE-344-1971.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.90, Revision 1 (August 1977)

In-service Inspection of Prestressed Concrete Containment Structures with Grouted Tendons

Position

RG 1.90 applies to power plants with prestressed concrete containments and is not applicable to the Unit 2 project.

Regulatory Guide 1.91, Revision 1 (February 1978) (For Comment)

Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plant Sites

FSAR Section 2.2.3

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

Regulatory Guide 1.92, Revision 1 (February 1976)

Combining Modal Responses and Spatial Components in Seismic Response Analysis

FSAR Sections 3.7.3A, 3.7.3B

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described in Section 3.7.3A and 3.7.3B.

Regulatory Guide 1.93, Revision 0 (December 1974)

Availability of Electric Power Sources

FSAR Section 8.3; Technical Specifications

Position

The Unit 2 project complies with Regulatory Position (Paragraph C) of this guide.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.94, Revision 1 (April 1976)

Quality Assurance Requirements For Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants

FSAR Section Chapter 17, Appendix B (QA Topical Report)

Position*

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described below.

1. ANSI N45.2.5-1974 Section 5.3 Bolt holes generally will not be burned (oxygen cut). If holes must be burned, the following criteria will be followed: a) after cutting, the edges of the cut will be ground or reamed back a minimum of 1/32 in, and b) the final bolt hole dimensions will not exceed those given in the Specification for Structural Joints Using ASTM A325 or A490 bolts.

2. ANSI N45.2.5-1974 Section 5.4 For the Unit 2 project, the criterion established for correct bolt length is one thread extending beyond the face of the nut.

3. ANSI N45.2.5-1974 Section 5.5 All reinforcing bar splices made by arc welding, except those splices welded to metal embedments, will be selected on a random basis for radiography as specified in the Unit 2 PSAR Section 12.6.3, and inspected in accordance with AWS D12.1. Splices welded to metal embedments will be inspected in accordance with AWS 12.1. Additionally, sister splice testing will be done in accordance with Specification No. NMP2-S203C with the same frequency as specified for B-series sister splices when required by the engineers.

4. ANSI N45.2.5-1974 Section 6.2.2 Exceptions regarding mechanical splicing of QA Category I reinforcing bars can be found in Unit 2 Project Position 1.10.

* This commitment is modified at the time of the QA Topical Report implementation. At that time, the QATR (see Appendix B) supersedes this commitment.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.95, Revision 1 (January 1977)

Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release

FSAR Section 6.4

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

Regulatory Guide 1.96, Revision 1 (June 1976)

Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants

FSAR Sections 1.2.9.11, 5.4.5, 6.2.3.2.3, 15.6.5

Position

MSIV leakage, at the maximum rate allowed by the Technical Specifications, has been included in the secondary containment bypass leakage analysis (Section 6.2.3.2.3) and in the LOCA radiological consequence analysis (Section 15.6.5). These design-basis analyses demonstrate that the calculated exposures are within the guidelines of 10CFR100 and 10CFR50 Appendix A, General Design Criteria 19.

In addition, a qualitative comparison has been made between Unit 2 and the plant used as the basis for analyses presented in NUREG-1169, "Technical Findings Related to Generic Issue C-8; Boiling Water Reactor Main Steam Isolation Valve Leakage and Leakage Treatment Methods." This comparison demonstrated that the design features of Unit 2 are sufficiently similar to the NUREG-1169 base plant, such that the conclusions of NUREG-1169 are considered directly applicable to Unit 2. NUREG-1169 concluded that the overall risks from the accident sequences in which MSIV leakage could be a significant factor are low without a leakage control system, and alternate fission product handling techniques, which make use of the holdup volume of the main steam lines and condenser, produce significant reductions in offsite dose consequences. It is therefore concluded that a MSIV leakage control system is not required for Unit 2.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.97, Revision 3 (May 1983)

Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident

FSAR Section 7.1.2

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described below.

Type A	1.	Conformance is in accordance with BWROG report position on NRC RG 1.97 Revision 3, dated May 1983 (see FSAR Section 7.5.2.1).
Type B	1.	Neutron flux - Conformance is in accordance with BWROG report position on NRC RG 1.97 Revision 3, dated May 1983 (see FSAR Section 7.5.2.1).
	2.	Core thermocouples (also incorporates Type C) - See TMI Item II.F.1 in Section 1.10 (see FSAR Section 7.5.2.1).
Type C	1.	Drywell drain sumps level - See TMI Item II.F.1 in Section 1.10 (see FSAR Section 7.5.2.1).
Type D	1.	Suppression pool temperature - Meets intent of guide. See TMI Item II.F.1 in Section 1.10 (see FSAR Section 7.5.2.1).
	2.	Drywell atmosphere temperature - Meets intent of guide. See TMI Item II.F.1 in Section 1.10 (see FSAR Section 7.5.2.1).
	3.	Cooling water temperature to ESF components - Meets intent of guide. See TMI Item II.F.1 in Section 1.10 (see FSAR Section 7.5.2.1).

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.98, Revision 0 (March 1976) (For Comment)

Assumptions Used for Evaluating the Potential Radiological Consequences of a Radioactive Offgas System Failure in a Boiling Water Reactor

FSAR Section 15.7.1

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

Regulatory Guide 1.99, Revision 2 (May 1988)

Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials

FSAR Sections 5.3.1, 5.3.2

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described in Sections 5.3.1 and 5.3.2.

Regulatory Guide 1.100, Revision 1 (August 1977)

Seismic Qualification of Electric Equipment for Nuclear Power Plants

FSAR Sections 3.10A, 3.10B, 7.1, 8.3.

Position

The Unit 2 project addresses the Regulatory Position (Paragraph C) of this guide through the alternate approach described in Sections 3.10A and 3.10B.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.101, Revision 2 (October 1981)

Emergency Planning for Nuclear Power Plants

FSAR Section Site Emergency Plan

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described below.

The Emergency Plan has been written to comply with NUREG-0654.

Regulatory Guide 1.102, Revision 1 (September 1976)

Flood Protection for Nuclear Power Plants

FSAR Sections 2.4.2, 2.4.10, 3.4

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

Regulatory Guide 1.103, Revision 1 (October 1976)

Post-Tensioned Prestressing Systems for Concrete Reactor Vessels and Containments

Position

Not applicable to Unit 2.

Regulatory Guide 1.104 (February 1976)

Overhead Crane Handling Systems for Nuclear Power Plants

FSAR Section 9.1.4

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described in Section 9.1.4.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.104 (February 1976) (cont'd.)

The Unit 2 crane was compared to RG 1.104, and a report was submitted to the NRC, and preliminary approval was granted by the NRC in a letter dated August 22, 1977.

The polar crane will be tested in accordance with NUREG-0554 and NUREG-0612.

Regulatory Guide 1.105, Revision 2 (February 1986)

Instrument Setpoints

FSAR Section 7.1.2

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

Regulatory Guide 1.106, Revision 1 (March 1977)

Thermal Overload Protection for Electric Motors on Motor-Operated Valves

FSAR Section 8.3.1

Position

The Unit 2 project complies with Regulatory Position (Paragraph C) of this guide. Unit 2 utilizes Position 1, Method (b).

The thermal overload on all safety-related MOVs is bypassed by any automatic safety signal or manually by the operator holding the spring return control switch. Annunciation is provided in the control room for those overload control conditions. An example is shown on Drawing Nos. ESK-6CSL02 and ESK-6CSL03 in the drawing package.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.107, Revision 1 (February 1977)

Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures

Position

Not applicable to Unit 2.

Regulatory Guide 1.108, Revision 1 (August 1977)

Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants

FSAR Sections 8.3.1.2, 14.2

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide except as indicated below.

The IEEE-387-1977 (Section 3.7) definitions of Continuous Rating and Short-Time Rating are applied to the diesel generator continuous rating and 2-hr rating referred to in Section C.2.a.3 of the regulatory guide.

Section 3/4.8 of the Technical Specifications details the periodic testing requirements of the diesel generator units.

See Table 4.8.1.1.2-1 of the Technical Specifications for periodic testing requirements of diesel generator units in lieu of Section C.2.d of the regulatory guide. The Unit 2 Technical Specifications incorporate appropriate surveillance testing requirements that are necessary to minimize mechanical stress and wear on diesel engines.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.109, Revision 1 (October 1977)

Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I

FSAR Sections 11.2, 11.3

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

Regulatory Guide 1.110, Revision 0 (March 1976) (For Comment)

Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors

FSAR Section Chapter 11

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

Regulatory Guide 1.111, Revision 1 (July 1977)

Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors

FSAR Sections 2.3.5, 11.3, 11A

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described in Sections 11.3 and 11A.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.112, Revision 0-R (May 1977)

Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors

FSAR Sections 2.3.4, 11.2.3, 11.3.3, 15.7.1, 15.7.2, 15.7.3

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

Regulatory Guide 1.113, Revision 1 (April 1977)

Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I

FSAR Sections 2.4.12, 2.4.13, 11.2, 15.7.2

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

Regulatory Guide 1.114, Revision 1 (November 1976)

Guidance on Being Operator at the Controls of a Nuclear Power Plant

FSAR Section 13.5

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.115, Revision 1 (July 1977)

Protection Against Low-Trajectory Turbine Missiles

FSAR Section 3.5

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described below and in Section 3.5.1.3.

> Paragraph C4: The protection of essential systems located within the low-trajectory missile strike zone is acceptable if the probability of damage summed over all structural cubicles containing such systems is less than 10^{-7} per annum.

Regulatory Guide 1.116, Revision 0-R (May 1977)

Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems

FSAR Section Chapter 17, Appendix B (QA Topical Report)

Position*

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

* This commitment is modified at the time of the QA Topical Report implementation. At that time, the QATR (see Appendix B) supersedes this commitment.

Regulatory Guide 1.117, Revision 1 (April 1978)

Tornado Design Classification

FSAR Sections 3.3, 3.5.1

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.118, Revision 2 (June 1978)

Periodic Testing of Electric Power and Protection Systems

FSAR Sections 7.1.2, 8.1, 8.3.1

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described below.

Regulatory Position C.6

Circuit response time tests will normally be conducted by connecting jumper wires to terminals in the termination cabinets permanently wired to spare contacts inside PGCC panels. However, where spare contacts and terminals are not available for response time tests, the jumper wires will be attached to appropriate dedicated terminals.

Regulatory Guide 1.119 (June 1976)

Surveillance Program for New Fuel Assembly Designs

Position

This regulatory guide was withdrawn by the NRC on June 3, 1977.

Regulatory Guide 1.120, Revision 1 (November 1977) (For Comment)

Fire Protection Guidelines for Nuclear Power Plants

FSAR Sections 9.5.1, Appendix 9A, Chapter 13

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described below.

The Unit 2 design is in accordance with BTP 9.5-1 Revision 0 (1975), as described in FSAR Section 9.5.1 and Appendix 9A.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.121 (August 1976) (For Comment)

Bases for Plugging Degraded PWR Steam Generator Tubes

Position

RG 1.121 applies to PWR plants and therefore is not applicable to the Unit 2 project.

Regulatory Guide 1.122, Revision 1 (February 1978)

Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components

FSAR Section 3.7.2A

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

Regulatory Guide 1.123, Revision 1 (July 1977)

Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants

FSAR Section Chapter 17, Appendix B (QA Topical Report)

Position*

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described as follows:

> Certain standard catalog or nonengineered items may be processed without seller prequalification. This alternative method is described in Section 7, paragraphs 1.4.1, 1.4.2, 1.4.3, and 3.1.2 of the Quality Assurance Program for Unit 2.

* This commitment is modified at the time of the QA Topical Report implementation. At that time, the QATR (see Appendix B) supersedes this commitment.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.124, Revision 1 (January 1978)

Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports

FSAR Section 3.9.3A, 3.9.3B

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described as follows:

BOP

1. Regulatory Position C.4 The increase of the design limits for bolts in shear as specified in NF-3231.1(a) and (c) should be limited to 0.70 of the bolting material shear ultimate at temperature.

2. Regulatory Position C.8 Supports for the active components that are required only during an emergency or faulted plant condition, and that are subjected to loading combinations described in Regulatory Positions C.6 and C.7, should be designed within the design limits described in Regulatory Position C.5 or other justifiable design limits.

3. Regulatory Position C.5a Paragraph C.5.a should be revised as follows:

    The stress limits of XVII-2000 of Section III and Regulatory Position 3 of this guide should not be exceeded for component supports designed by the linear elastic analysis method. These stress limits may be increased according to the provisions of NF-3231.1(a) of Section III and Regulatory Position 4 of this guide, when effects resulting from constraint of free-end displacement and anchor motion are added to the loading combination.

The following is an addition to the regulatory guide as implemented on the Unit 2 project:

> The bending stress limit F resulting from tension and bending in structural members, as specified in Appendix XVII-2214 of ASME Section III, Division 1, should be the smaller value of 0.66 S_y or 0.55 S_u for compact sections, 0.75 S_y or 0.63 S_u for doubly symmetrical members with bending about the minor axis, and 0.6 S_y or 0.5 S_u for box-type flexural members and miscellaneous members.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.124, Revision 1 (January 1978) (cont'd.)

NSSS

RG 1.124 Revision 1 (January 1978) was issued after the docketing date for Unit 2. However, Unit 2 complies with the indicated item of the Regulatory Position, described as follows. The remaining design analysis criteria of the regulatory guide are adequately addressed by conservatisms in the existing ASME III Code.

1. Paragraph C.2 Ultimate strength temperature correlation of this guide was used in regions adjacent to pipe having high temperatures. Additionally, the critical buckling strength limits of ASME III Appendix XVII, Paragraph 2110(b), are observed.

Regulatory Guide 1.125, Revision 1 (October 1978)

Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants

FSAR Section 2.4

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

Regulatory Guide 1.126, Revision 1 (March 1978)

An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification

FSAR Section Chapter 5

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A (latest approved revision), and General Electric Standard Application for Reactor Fuel - United States Supplement, NEDE-24011-P-A-US (latest approved revision), are used to comply with the requirement of this regulatory guide.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.127, Revision 1 (March 1978)

Inspection of Water-Control Structures Associated with Nuclear Power Plants

Position

Not applicable to Unit 2.

Regulatory Guide 1.128, Revision 1 (October 1978)

Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants

FSAR Section 8.1, 8.3.2

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

Regulatory Guide 1.129, Revision 1 (February 1978)

Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants

FSAR Sections 8.1, 8.3.2

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

Regulatory Guide 1.130, Revision 1 (October 1978)

Design Limits and Loading Combinations for Class 1 Plate-and-Shell Type Component Supports

FSAR Section 3.9.3.B

RG 1.130 Revision 0 (July 1977) was issued after the docketing date for Unit 2 and work was in progress. However, Unit 2 complies with the indicated items of the Regulatory Position of this guide through the alternative approach, described as follows. The remaining design analysis criteria of this regulatory guide are adequately addressed by conservatisms in the existing ASME III Code.

Regulatory Guide 1.130, Revision 1 (October 1978) (cont'd.)

1. Paragraph C.2 Ultimate strength temperature correlation of this guide was used in regions adjacent to pipe having high temperatures.

2. Paragraph C.3 Regulatory Position C.4, with alternate conservative collapse criteria developed by the NSSS supplier for plates and shells, was used in lieu of Regulatory Position C.3.

Regulatory Guide 1.131, Revision 0 (August 1977) (For Comment)

Qualification Tests of Electrical Cables, Field Splices, and Connections for Light-Water-Cooled Nuclear Power Plants

FSAR Section 3.11

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described below.

BOP

Complies with this regulatory guide.

NSSS

This regulatory guide is not applicable for the GE scope of supply because GE-supplied cabling does not experience severe environmental conditions (control room environment) and is qualified as part of the PGCC floor section module.

Regulatory Guide 1.132, Revision 1 (March 1979)

FSAR Sections 2.5, 2F, 3.7.2A

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described below.

This guide was originally issued in September 1977, 6 yr after the field work for the foundation investigation was performed in

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.132, Revision 1 (March 1979) (cont'd.)

late 1971. The results were provided in a report* dated May 4, 1972. Although the investigation predates the guide, the work-performed was well documented and adequate to support the Construction Permit Application and complies with the intent of the guide.

For any aspect of the Unit 2 site, investigation performed after March 30, 1979, conforms to this regulatory guide.

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* Report: Foundation Investigation, Nine Mile Point Nuclear Station, Proposed Unit 2, Scriba, New York, Niagara Mohawk Power Corporation.

Regulatory Guide 1.133, Revision 1 (May 1981)

Loose-Part Detection Program for the Primary System of Light-Water Reactors

FSAR Section 4.4.6.1

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described below.

1. Paragraph C.1.q The audio/visual alarm capability is not qualified to remain functional following seismic events. As an alternative, plant operating procedures will require the operator to verify the operability of the LPMS following any detected seismic event. If inoperable, the operator will initiate any appropriate maintenance activities to restore the system to operability. The LPMS need not be qualified according to the requirements of RG 1.89.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.134, Revision 2 (April 1987)

Medical Evaluation of Licensed Personnel for Nuclear Power Plants

FSAR Section None

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

The medical requirements for personnel requiring operator licenses, as detailed in ANSI/ANS-3.4-1983 and endorsed by this guide, will be implemented.

Regulatory Guide 1.135 (September 1977) (For Comment)

Normal Water Level and Discharge at Nuclear Power Plants

FSAR Sections 2.4, 9.2.5

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

Regulatory Guide 1.136, Revision 2 (June 1981)

Materials for Concrete Containments

FSAR Sections 3.8.1, 3.8.2, 3.8.3, 3.8.4, 3.8.5

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described below.

RG 1.136 and ACI 359 (or ASME III - Division 2) were formally issued in 1977 and 1975, respectively, and subsequently reissued thereafter. The construction permit for this project following NRC evaluation of the Unit 2 PSAR was received in June 1974. The reissue of RG 1.136, as Revision 2 in June 1981, incorporates the recommendations of several regulatory guides under this regulatory guide with the intention of withdrawing RG 1.10, 1.15, 1.18, 1.19, 1.55, and 1.103. Due to the advanced stage of procurement, fabrication, and construction, it was not feasible to adopt the changes contained in Revision 2 of this guide. The

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.136, Revision 2 (June 1981) (cont'd.)

NRC Safety Evaluation Report (SER) was issued for this project, indicating acceptance to using ACI 318 for the design of concrete containment.

The design of the Unit 2 primary containment is in accordance with RG 1.10, 1.15, 1.18, 1.19, 1.55, and 1.103 (as previously described in lieu of RG 1.136).

Regulatory Guide 1.137, Revision 1 (October 1979)

Fuel-Oil Systems for Standby Diesel Generators

FSAR Section 9.5.4

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide except for the alternate approach described in response to Question F430.61, which is incorporated in the Unit 2 Technical Specifications.

Regulatory Guide 1.138, Revision 0 (April 1978) (For Comment)

Laboratory Investigations of Soils for Engineering Analysis and Design of Nuclear Power Plants

FSAR Sections 2.5, 2F, 3.7.2A

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described below.

The results of laboratory work performed for the foundation investigation were submitted in a report* dated 6 yr prior to the issuance of the guide. The work performed was well documented and adequate to support the Construction Permit Application and complies with the intent of the guide.

* Report: Foundation Investigation, Nine Mile Point Nuclear Station, Proposed Unit 2, Scriba, New York, Niagara Mohawk Power Corporation.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.138, Revision 0 (April 1978) (For Comment) (cont'd.)

Laboratory investigations performed after December 1, 1978, were in conformance with this regulatory guide.

Regulatory Guide 1.139, Revision 0 (May 1978) (For Comment)

Guidance for Residual Heat Removal

FSAR Sections 5.4.7, 6.3

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described below.

> Paragraph C.1.b states: "In demonstrating that the system can perform its function assuming a single failure, limited operator action outside the control room would be acceptable if suitably justified." The common RHR shutdown cooling suction line valves are in two divisions (Division I - the outside valve; Division II - the inside valve) to satisfy containment isolation criteria. In the event that the RHR shutdown suction line is not available during shutdown because of a single-valve failure (loss of a division of emergency power), either valve can be opened manually with limited operator action or by establishing an alternate shutdown cooling path.

Regulatory Guide 1.140, Revision 1 (October 1979)

Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants

FSAR Section 9.4.3

Position

Regulatory Guide 1.140 applies only to the radwaste building general area and equipment exhaust systems, each of which is designed to remove only particulate matter. Because charcoal adsorbers are not provided, the sections of the guide relating to adsorbers and iodine adsorption are not addressed. The air filtration units are nonsafety related; however, redundancy is provided for reliability and ease of maintenance.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.140, Revision 1 (October 1979) (cont'd.)

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approaches described below:

1. Paragraph c.2.b - Equipment exhaust system operating capacity is 47,800 cfm. HEPA filter banks are arranged five cells high and seven cells wide, for a total of 35 cells. Maximum air flow through each cell is 1,366 cfm. The filter bank configuration was dictated by spatial limitations. Each filter cell is rated by the manufacturer for an air flow of 1,430 cfm.

2. Paragraph c.2.f - Each HEPA filter casing and the enclosure in which it is mounted is leak tested in accordance with the guide. All ductwork under positive pressure is tested in accordance with procedures of the Associated Air Balance Council (AABC). Ductwork under negative pressure (suction side of fans) will not be leak tested because any leakage would be into the ductwork and would therefore be processed through filters.

3. Paragraph c.3.i - Displacement criteria based on normal industry practice will be used to determine vibration levels. The maximum vibration velocity method of ANSI N509-1980 imposes unrealistic requirements for certain operating speeds.

4. Paragraph c.3.1 - Exception is taken to Section 5.9 of ANSI N509-1980 as follows:

    a. Dampers are designed, fabricated, and installed to preclude the uncontrolled release of radioactivity, thereby satisfying the intent of Section 5.9.7. Leakage rates shall be derived from tests conducted in accordance with AMCA 500 and may be the result of project-specific tests or other tests conducted with dampers of the same design. The damper manufacturer is responsible for ensuring that all test reports are maintained on file and are readily available for verification.

    b. Shaft diameters are the damper manufacturer's standard sizes. The shaft diameter is 1/2 in on dampers less than 24 in in length and 3/4 in on dampers 24 in to 48 in in length.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.141, Revision 0 (April 1978) (For Comment)

Containment Isolation Provisions for Fluid Systems

FSAR Section 6.2.4

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described in Section 6.2.4.

Regulatory Guide 1.142, Revision 1 (October 1981)

Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)

FSAR Section 3.8.4A

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described below.

The RG 1.142 and ACI 349-76 (discussed in this regulatory guide) were originally issued in 1978 and 1976, respectively. RG 1.142 was subsequently revised and reissued in 1981. The construction permit for this project, following NRC evaluation of the Unit 2 PSAR, was received in June 1974. A supplement to ACI 349-76 was issued in 1979. Due to the advanced stage of design, procurement, fabrication, and construction, it is not feasible to assure compliance to this guide. The design of safety-related structures (other than concrete primary containment) meets or exceeds the requirements of ACI 318-71 or ACI 318-77, as explained in FSAR Section 3.8.4.2, which were accepted to provide an adequate basis for design of Category I structures. The NRC SER was issued for this project, indicating acceptance of ACI 318-71 for the design of concrete primary containment.

The design of the Unit 2 primary containment is in accordance with RG 1.10, 1.15, 1.18, 1.19, 1.55, and 1.103 (as previously described in lieu of RG 1.142).

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.143, Revision 1 (October 1979)

Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants

FSAR Section 15.7.1, 11.4

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described below.

A. Liquid Waste System

The fiberglass tanks purchased for the LWS have been designed in accordance with the National Bureau of Standards (NBS) Product Standard (PS) PS 15-69, Custom Contact-Molded Reinforced-Polyester Chemical-Resistant Process Equipment, as identified in NMP2 PSAR, Table C-10b.

NBS PS 15-69 provides the necessary design and fabrication requirements to ensure the integrity of the tanks without the additional cost of burst testing.

The RWCU phase separator tanks (2LWS-TK6A & 6B), which had been purchased as code-stamped ASME VIII vessels, had their code stamps removed because they were not rehydrotested after a nozzle was added to the top head of each of the two vessels in the field. The vessels still satisfy the intent of the requirements of this regulatory guide in that they are designed and fabricated to the requirements of ASME VIII (including the added nozzles) using materials which meet ASME VIII requirements, and the shop hydrotest established the integrity of the vessels before the nozzles were added. The nozzles were added near the top of the vessels' heads and the vessels see only atmospheric operating conditions (although designed for a nominal 15-psi design pressure). The new nozzles were added to allow improved operation of the vessels' level transmitters and are identical to the original level transmitter nozzles which were blind flanged and abandoned. Therefore, the added nozzles do not affect the proven integrity of the vessels in this application.

TABLE 1.8-1 (Cont'd.)

<u>Regulatory Guide 1.143, Revision 1 (October 1979)</u> (cont'd.)

B. Offgas System

The charcoal adsorbers of the offgas system are not designed to the seismic requirements of this regulatory guide.

Offsite dose calculations in accordance with Chapter 15.7.1 of the NMP2 FSAR show that release of gaseous activity due to failure of the charcoal adsorbers results in offsite doses less than 0.5 Rem to the whole body. In accordance with RG 1.29, this permits classification as nonseismic. At the time of design and procurement of the offgas system (July 1974), RG 1.29 Revision 1 established the seismic requirements for the radioactive waste processing systems.

C. Waste Solidification System

The waste solidification system complies with the requirements of NRC Branch Technical Position ETSB11.1 Revision 1 as outlined in Werner and Pfleiderer Corporation (WPC) Topical Report No. WPC-VRS-001 Revision 1, dated May 1978, with exceptions as discussed in Section 11.4.3. The waste sludge tank is designed, fabricated, examined, and tested (hydrotest at 1.5 times design) in accordance with the requirements of ASME Code Section VIII, Division 1, with no codes stamp. The WPC topical report lists API 620 or 650 for this tank.

The WPC Topical Report has been accepted by the NRC as satisfying the requirements of ETSB11.1 Revision 1, which are consistent with the requirements of RG 1.143. For the waste sludge tank, the requirements of ASME VIII are more stringent than the requirements of API 620 or 650.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.144, Revision 1 (September 1980)

Auditing of Quality Assurance Programs for Nuclear Power Plants

FSAR Section Chapter 17, Appendix B (QA Topical Report)

Position*

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described below.

The preaudit and postaudit conferences required by Sections 4.3.1 and 4.3.3 of ANSI N45.2.12-1977 may be fulfilled by a variety of communications such as telephone conversations.

* This commitment is modified at the time of the QA Topical Report implementation. At that time, the QATR (see Appendix B) supersedes this commitment.

Regulatory Guide 1.145, Revision 0 (August 1979) (For Comment)

Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants

FSAR Sections 2.3, 13.3, 15.7

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.146, Revision 0 (August 1980)

Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants

FSAR Section Chapter 17, Appendix B (QA Topical Report)

Position*

The Unit 2 project complies with Regulatory Position (Paragraph C) of this guide.

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* This commitment is modified at the time of the QA Topical Report implementation. At that time, the QATR (see Appendix B) supersedes this commitment.

Regulatory Guide 1.147, Revision 5 (August 1986)

In-service Inspection Code Case Acceptability - ASME Section XI Division I

FSAR Section 14

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described below.

In accordance with 10CFR50.55a(g)(2), the NMP2 In-service Inspection Program is based on the 1983 Edition of ASME Section XI, Summer 1983 Addenda. Code cases included in these programs are identified in the In-service Inspection Program Plan (First Ten Year).

Regulatory Guide 1.148 (March 1981)

Functional Specification for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants

FSAR Section 5.4

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide through the alternate approach described below.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.148 (March 1981) (cont'd.)

BOP

Other safety-related valve assemblies classified as Quality Group A, B, or C in RG 1.26 comply with the regulatory guides as described below.

a. Section C.2.a, Valve Application Characteristics

The frequency of use for each safety-related valve assembly is not specified. The normal (open/closed) position is not specified, except in the case of safety-related butterfly and solenoid valve assemblies.

b. Section C.2.b, Structural Requirements

The dynamic loading and the piping frequency response spectra are not specified. Potential water hammer is not considered when establishing the maximum differential pressure across a valve.

c. Section C.2.c, Operational Requirements

The safety-related function (open/close, remain-as-is) is not specified, except in the cases of ball, butterfly, and solenoid valve assemblies. Motor power requirements for valve assemblies are not specified.

NSSS

Fast-closing isolation valve assemblies classified as Quality Group D in RG 1.26 meet the requirements of ANSI B31.1.0, 1977. They also comply with RG 1.148, dated March 1981, with the following clarification:

a. Section C.2.a, Valve Application Characteristics

The frequency of use for each safety-related valve assembly is not specified. The normal (open/closed) position is not specified, except in the case of safety-related butterfly and solenoid valve assemblies.

b. Section C.2.b, Structural Requirements

The dynamic loading and the piping frequency response spectra are not specified. Potential water hammer is not considered when establishing the maximum differential pressure across a valve.

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.148 (March 1981) (cont'd.)

Main steam isolation valve assemblies comply with RG 1.148, dated March 1981, with the following clarification:

a. Section C.1.c.2, Applicability and Relationship with other Standards

   The functional specification does not reference the design specification.

Regulatory Guide 1.149 (April 1981)

Nuclear Power Plant Simulators for Use in Operator Training

FSAR Section Chapter 14

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

Regulatory Guide 1.150, Revision 1 (February 1983)

Ultrasonic Testing of Reactor Vessel Welds During Preservice and In-service Examination

FSAR Section PSI/ISI Plan

Position

The Preservice Inspection Plan for Unit 2 consists of three separate documents as follows:

1. Preservice Inspection Plan for Nuclear Piping Systems and the Reactor Pressure Vessel.

2. In-service Testing Plan for Pumps and Valves.

3. Preservice Inspection Plan for Nuclear Piping System and Component Supports.

Part 1 was submitted for the Staff's review by letter dated October 15, 1985, and included all NDE for items required by ASME Section XI for nuclear piping systems, and the RPV, its internals and safe ends. Also included in that submittal is a listing of PSI ultrasonic procedures and the PSI isometric drawings. Part 2

Regulatory Guide 1.150, Revision 1 (February 1983) (cont'd.)

was transmitted for the Staff's review by letter dated November 27, 1985, and Part 3 was submitted on December 17, 1985.

As discussed in Section 1.7 of the PSI Plan (Part 1), the required examinations for PSI shall be performed prior to initial startup of the plant. The applicable code for Unit 2 PSI is ASME Section XI, 1980 Edition through the Winter 1980 Addenda, as discussed in Section 1.1.1 of the PSI Plan (Part 1). Section 1.2.1 of the PSI Plan (Part 1) discusses the application of the examination selection criteria. Selected for volumetric examination were 7.5 percent of the welds in the RHR system, high-pressure core spray system, and low-pressure core spray system, normally excluded from preservice volumetric examination.

The degree of compliance with RG 1.150 Revision 1 is provided in Table 1.8-1a. This document presents the alternate method of compliance with the regulatory guide, how the compliance is achieved (under the column "Response"), and what documents or procedures reference the compliance implementation (under the column "Procedures and References"). The Degree of Compliance position is incorporated in the RPV examination procedures. A technique qualification shall be performed using the ultrasonic examination systems that will be employed during the automated PSI examinations of the RPV. Actual RPV and nozzle segments containing known size reflectors located in the ID surface shall be used. This qualification shall be witnessed by NMPC, its Agents, and the NMPC ANII, as a minimum. The qualification shall demonstrate that flaws of the maximum allowable limits are detectable.

Regulatory Guide 1.155 (August 1988)

Station Blackout

FSAR Section 8.3.1.5

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

Nine Mile Point Unit 2 is evaluated against the requirements of the Station Blackout Rule, 10CFR50.63, using the guidance contained in NUMARC 87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," NUMARC 87-00 Supplemental Questions/Answers, dated December 27, 1989, and NUMARC 87-00 Major Assumptions, dated

TABLE 1.8-1 (Cont'd.)

Regulatory Guide 1.155 (August 1988) (cont'd.)

December 27, 1989, except where RG 1.155 takes precedence. Table 1 of RG 1.155 provides a cross-reference between the regulatory guide and NUMARC 87-00. Any exceptions to the NUMARC guidance taken by NMPC are identified in the SBO documentation maintained by NMPC (see Letter No. NMP2L 1230, dated April 3, 1990, to NRC, TAC No. 68571).

TABLE 1.8-1a

COMPLIANCE WITH REGULATORY GUIDE 1.150

Appendix A Alternative Method	Response	Procedure and References
Ultrasonic examination of reactor vessel welds should be performed according to the requirements of Section XI of the ASME B&PV Code, as referenced in the Safety Analysis Report (SAR) and its amendments, supplemented by the following:		
1. INSPECTION SYSTEM PERFORMANCE CHECKS	Comply as stated in the following paragraphs.	
The conduct of a quality examination requires that the performance characteristics of the inspection system used be well defined and documented. This is particularly true for situations which require comparisons of examination results generated during successive examinations on the same components.		
A system comprises:		
a. a transducer		
b. a single channel instrument or each channel of a multichannel instrument		
c. a given cable type and length		
The checks described in paragraphs 1.1 and 1.2 should be made for any UT system used for inspection of reactor pressure vessel (RPV) welds.		
The field performance checks described in 1.2 (with the possible exception of 1.2.c) should be conducted on a basic calibration block that represents the thickness range to be examined.		

Appendix A Alternative Method	Response	Procedure and References
1.1 Pre-exam Performance Checks		
a. Frequency of checks	See response to Paragraph 1.1b.	
These checks should be verified within six months before reactor pressure vessel examinations performed during one outage. Pulse shape and noise suppression controls should remain at the same settings during calibration and examination.		
b. RF Waveform		
A record of the RF (radiofrequency) pulse waveform from a reference reflector should be obtained for each search unit used in the examination in a manner which will provide frequency amplitude information. At the highest amplitude portion of the beam, the RF return signal should be recorded before it has been rectified or conditioned for display. The reflector used in generating the RF return signal as well as the electronic system (i.e., the basic ultrasonic instrument, gating, and form of gated signal) should be documented. These records should be used for comparison with previous and future records.	A spectrum analysis and RF waveform of each search unit is performed within six months before the start of examinations using laboratory equipment.	NES document 85A280 delineates the procedure for obtaining this spectrum analysis and RF waveform. Spectrum analyses and RF waveforms are required in NES Document 83A1754, Paragraph 7.2. Test parameters and results are recorded on a Search Unit Qualification Record Sheet. These are available on site and are provided in the final report.
1.2 Field Performance Checks		
a. Frequency of Checks		
As a minimum, these checks should be verified onsite before and after examining all the welds that need to be examined in a reactor pressure vessel during one outage. Pulse shape and noise suppression controls should remain at the same setting	Comply with the requirements this paragraph.	Not applicable.

TABLE 1.8-1a (Cont)

Appendix A Alternative Method	Response	Procedure and References
during examination and calibration.		
b. Instrument Sensitivity during Linearity Checks The initial instrument sensitivity during the performance of 1.2.e should be such that it falls at the calibration sensitivity or at some point between the calibration sensitivity and the scanning sensitivity.	Verification of horizontal, screen height and amplitude-control linearity is performed before and every 90 days during an examination. There is no requirement in the referencing procedure to perform this on a basic-calibration block representing the thickness range. The test for amplitude-control linearity covers a range of instrument sensitivity but may not include the calibration or scanning sensitivity.	NES Document 80A9053 delineates the procedure for performing this instrument linearity. Instrument linearity is required in NES Document 83A1754, para. 7.1. The results of the instrument linearity are recorded on an Ultrasonic Instrument Linearity Record Sheet. These are available on site and are provided in the final report.
d. Screen Height Linearity Screen height linearity of the ultrasonic instrument should be determined according to the mandatory Appendix I to Article 4, Section V of the ASME Code or Appendix I to Section XI of the ASME Code.	Intermediate screen height linearity checks are performed during and at the end of examinations as an integral part of the calibration check. The check is performed with a Digital Transponding Ultrasonic Calibrator (DTUC), a type of electronic simulator. The screen-height linearity is correlated to the actual calibration at the calibration sensitivity.	The procedure to initially establish the screen height linearity presentation is stated in NES Document 83A1754, Paragraph 6.2. The procedure to check screen-height linearity is defined in NES Document 83A1754, Paragraphs 6.2.1 and 6.2.2. A record of these checks is made with UDRPS. The file numbers for these checks are recorded on a Raster Data Sheet which is part of the examination record. (As used herein, examination record consists of any data sheets generated as a result of the actual examinations.)
e. Amplitude Control Linearity Amplitude control should be determined according to the mandatory Appendix II of Article 4, Section V of the ASME Code of Appendix	Amplitude-control linearity is not verified during an intermediate check. The examination is performed and recorded with UDRPS at the calibration sensitivity. Once established,	Not applicable.

TABLE 1.8-1a (Cont)

Appendix A Alternative Method	Response	Procedure and References
I to Section XI of the ASME Code.	there is no need to change the calibration sensitivity during the course of the examination.	
c. RF Waveform A record of the RF (radiofrequency) pulse waveform from a reference reflector should be obtained and recorded in a manner that will permit extraction of frequency amplitude information. At the highest amplitude portion of the beam, the RF return signal should be recorded before it has been rectified or conditioned for display. This should be determined on the same reflector as that used in 1.1.b. above. This record should be retained for future reference.	RF waveforms are performed on site using the basic-calibration block before and after each examination setup. For this waveform, the RF-return signal is used before it has been conditioned or rectified.	NES Document 85A281 delineates the procedure for obtaining RF waveforms. RF waveforms are required by NES Document 83A1754, Paragraph 7.3. Critical parameters and results are recorded on a RF Waveform Record Sheet. These are available on site and are provided in the final report.
f. Angle Beam Profile Characterization The vertical beam profile should be determined for each search unit used during the exmamination by a procedure similar to that outlined in nonmandatory Appendix B-60, Article 4, Section V of the ASME	Vertical beam profile is determined in one of two ways. A direct plot of the hole in the actual calibration block or block of similar material and thickness may be made using a technique similar to Appendix B-60.	Vertical-beam profiles are required by NES Document 80A1754, Paragraph 7.5. Beam-profile information is available on site and is provided in the final report.
Code or Appendix I to Section XI of the ASME Code. Beam profile curves should be determined at different depths to cover the thicknesses of materials to be examined. Interpolation may be used to obtain beam profile correction for assessing flaws at intermediate depths for which beam profile has not been determined.	Alternatively, after calibration, an option available with UDRPS may be used to provide vertical beam profile information of each reflector. From this, actual beam profiles can be constructed as required for evaluation.	Alternative techniques are permitted by NES Document 83A1754, Paragraph 7.5.
Beam profile measurements should be made at the sensitivity required for	With either method, beam profiles are done at the sensitivity	This is a requirement of NES Document 83A1754, Paragraph 7.5.

Appendix A Alternative Method	Response	Procedure and References
sizing. For example, sizing to a 20 percent DAC criteria requires that the beam profile be determined at 20 percent DAC.	required for sizing.	
	Beam profiles are not performed during or at the end of examinations unless a significant change of search-unit performance is identified.	This is permitted by NES Document 83A1754, Paragraph 7.5.
2. CALIBRATION System calibration should be performed to establish the DAC curve and the sweep range calibration in accordance with Article 4, Section V of the ASME Code or Appendix I to Section XI. Calibration should be confirmed before and after each RPV examination, or each week in which the system is in use, whichever is less. Where possible, the same calibration block should be used for successive inservice examinations of the RPV.	Initial calibration is performed on site with the actual calibration block to establish sweep range and amplitude calibration. At this time, the calibration is correlated to a simulator block and DTUC. After initial calibration, checks using the simulator block and DTUC are performed before the start of examination (if the start exceeds 12 hours from the initial calibration) and every 12 hours during the examination. The final check after examinations are complete shall be performed on the actual calibration block. A check will be performed on the actual calibration block any time results from the simulator block or DTUC warrant.	The procedure for calibration delineated in NES Document 83A1749, Section 4. Calibration-check requirements are described in NES Document 83A1754, Paragraph 5.3. The procedure for correlating the calibration to the DTUC is delineated in NES Document 83A1754, Paragraph 6.1. A record of the calibration and calibration checks is made with UDRPS. The file number and data from the calibration are recorded on the Calibration Data Sheet. The file numbers for the calibration checks taken with UDRPS are recorded on the Raster Data Sheets. The Calibration Data Sheets and Raster Data Sheets are part of the examination record.
2.1 Calibration for Manual Scanning For manual sizing of flaws, static calibration may be used if sizing is performed using a static transducer. When signals are maximized during calibration, they should also be	Not applicable.	

TABLE 1.8-1a (Cont)

Appendix A Alternative Method	Response	Procedure and References
maximized during sizing. For manual scanning for the detection of flaws, reference hole detection should be shown at scanning speed and detection level set accordingly.		
2.2 Calibration for Mechanized Scanning		
When flaw detection is to be done by mechanized equipment, the calibration should be performed using the following guidelines.		
a. The DAC curve should be established using either a moving transducer mounted on the mechanism that will be used for examination of the component or a mechanism that duplicates the critical factors (e.g., transducer mounting, weight, pivot points, couplant) present in the scanning mechanism.	The DAC curve is established with the actual exam head to be used in conjunction with a scanning mechanism tht duplicates the scanner motion and critical factors of mounting, pivot points, and couplant. With examination heads riding on hard-contact points along the surface, weight is not a critical factor.	Standard department practice.
b. Calibration speed should be at or higher than the scanning speed, except when correction factors established in 2.2.d. are used.	Calibration speed is greater than or equal to scanning speed.	This is a requirement of NES Document 83A1754, Paragraph 4.2.2. The calibration speed is recorded on the Calibration Data Sheet. Scanning speed is recorded on the Automated Weld Scan Data Sheet. Both sheets are a part of the examination record.
c. The direction of transducer movement (forward or backward) during calibration to establish the DAC curve should be the same direction during scanning unless it can be shown that a change in scanning direction does not reduce flaw detection capability.	Dynamic calibration is performed in both directions (forward and backward) and the resultant-lower amplitude DAC is used for examination.	This is a requirement of NES Document 83A1754, Paragraph 7.4. The actual dynamic amplitudes used are recorded on the Calibration Data Sheet which is part of the examination record.

Appendix A Alternative Method	Response	Procedure and References
d. One of the following alternative guidelines should be followed to establish correction factors if static calibration is used. (1) Correction factors between dynamic and static response should be established using the basic calibration block or, (2) Correction factors should be established using models and taking scaling factors into consideration (assumed scaling relationship should be verified) or, (3) Correction factors should be established using full scale mockups.	Not applicable.	
2.3 Calibrtion Confirmation Calibration confirmation performed as mid-shift or interim confirmation between on-site calibrations should comply with stability requirements in T-433, Article 4, Section V of the ASME Code.	Due to the configuration of UDRPS, data recorded previous to a calibration check that does not meet the requirements of T-433 may be evaluated at a correspondingly adjusted evaluation criteria. This evaluation is acceptable at the discretion of the Level III. Once the condition has been identified, no further examination would be performed until the cause is determined and the necessary corrective action taken; i.e., a minimum of a calibration check on the calibration block and if required, recalibration.	This is permitted in NES Document 83A1754, Paragraph 5.4. Data sets so evaluated are identified on the Raster Analysis Sheet which is part of the examination record. A record of the calibration check and recalibration (if required) is made with UDRPS. The file numbers are recorded on the Raster Data Sheet and Calibration Data Sheet respectively. Both sheets are part of the examination record.
When an electronic simulator is used for on-site calibration confirmation after a Code-re-	Not applicable.	

TABLE 1.8-1a (Cont)

Appendix A Alternative Method	Response	Procedure and References
quired block calibration performed off-site, the following should also apply: a. Complete system performance should be maintained stable prior to off-site calibrations and on-site calibration confirmation by use of target reflectors. The target reflectors should be mounted with identical physical displacement in both the off-site calibration facilities and the on-site mechanized equipment. Each on-site periodic calibration should be preceded by complete system performance verification using a minimum of two (2) target reflectors separated by a distance representing 75 percent of maximum thickness to be examined. b. Written records of calibrations should be established for both target reflector responses and Code calibration block DAC curves for each transducer. These written records may be used to monitor drift since the original recorded calibration. c. Measures should be taken to ensure that the different variables such as temperature, vibration, and shock limits are minimized by controlling packaging, handling, and storage.		
2.4 Calibration Blocks Calibration blocks should comply with Appendix I to Section XI or Article 4, Section V of the ASME	Calibration blocks are supplied by and are the responsibility of the client. New blocks or modifications to existing blocks have the approval of the client.	Not applicable.

Appendix A Alternative Method	Response	Procedure and References
Code. When an alternative calibration block or a new conventional block is used, a ratio between the DAC curves obtained from the original block and from the new block should be noted (for reference) to provide for a meaningful comparison of previous and current data.		
The calibration side-drilled holes in the basic calibration block and the block surface should be protected so that their characteristics do not change during storage. These side-drilled holes or the block surface should not be modified in any way (e.g., by polishing) between successive examinations. If the block surface or the calibration reflector holes have been polished by any chemical or mechanical means, this fact should be recorded.	While blocks are in the custody of NES, the calibration surfaces and reflectors are protected from degradation.	Standard department practice.
3. EXAMINATION		
The scope and extent of the ultrasonic examinations should comply with IWA-2000, Section XI of the ASME Code.	The scope and extent of examinations complies with IWA-2000, Section XI.	Not applicable.
If electronic gating is used to define the examination volume within which indications are recorded, the start and stop control points should include the entire required thickness including the material near each surface. If a single gate is used, it should be capable of recording multiple indications appearing in the gate. Alternative means of recording may be used provided they do not reduce flaw detection and recording capability.	UDRPS does not use electronic gating. It records the rectified RF output of the UT system. UDRPS is configured to make a record of all indications in the full-examination volume.	Not applicable.

TABLE 1.8-1a (Cont)

Appendix A Alternative Method	Response	Procedure and References
Examination should be done with a minimum 25 percent scan overlap based on the transducer element size.	Examinations are performed with a minimum of 25 percent scan overlap.	This is a requirement of NES Document 83A1754, Paragraph 4.2.1. Actual indexes are recorded on the Automated Weld Scan Data Sheet which is part of the examination record.
3.1 Internal Surface The capability to effectively detect defects at the internal clad/base metal interface shall be considered acceptable if the examination procedure(s) or technique(s) meet the requirements of Section 6.0 of this document <u>and demonstrate</u> the following:	Comply as stated in the following paragraphs.	
a. Procedures for examination from the outer surface, or when using full vee from the inside surface, should include the use of the 2 percent notch which penetrates the internal (clad) surface of the calibration blocks, defined by Section XI, Appendix I, Figure I-3131 or Section V, Article 4, Figure T-434.1. Procedures for examination from the internal surface when not using the full vee should conform to Paragraph 3.1.b. below.	The response from the ID notch is recorded as part of the calibration.	This is a requirement of NES Document 83A1749, Paragraphs 4.2.3 and 4.2.4.
b. An alternate reflector, other than the 2 percent notch described above may be used provided: 1) that it is located at the clad/base metal interface or at an equivalent distance from the surface, 2) that it does not exceed the maximum allowable defect size, and 3) that equivalent or superior results can be demonstrated.	See 3.1.c and 3.2.	Not applicable.
c. The examination procedure(s) should	To meet the requirements of these	Not applicable.

TABLE 1.8-1a (Cont)

Appendix A Alternative Method	Response	Procedure and References
provide for volumetric examination of at least one inch of metal as measured perpendicular to the nominal location of the base metal-cladding interface.	two paragraphs, a tandem element 45 degrees shear-wave technique is used. It is capable of detecting an inner near surface flaw of maximum-allowable size.	
3.2 Scanning Weld-Metal Interface The beam angles used to scan welds should be based on the geometry of the weld/parent metal interface. Where feasible for welds such as those identified in Section T-441.4.2 of Article 4, Section V of the ASME Code, at least one angle should be such that the beam is perpendicular (±15 degrees to the perpendicular) to the weld/parent metal interface, or it should be demonstrated that unfavorably oriented planar flaws can be detected by the UT technique being used. If this is not feasible, use of alternative volumetric NDE techniques, as permitted by the ASME Code, should be considered.	The calibration block contains two slots machined in the end of the block. One is located at at the clad/base metal interface (CBMI) and the other is located in the base metal, one inch from the interface. The faces of both notches are of maximum allowable size for surface and subsurface planar flaws, respectively. Both faces are perpendicular to the ID surface, which is the worst-case orientation.	Not applicable.
	The response from the CBMI slot is recorded as part of the calibration for the 45 degrees tandem technique. This is also used for comparison when evaluating indications.	This is a requirement of NES Document 83A1749, Paragraph 4.2.3.
	The second slot is present for comparision when evaluating indications.	Standard operating practice.
4. BEAM PROFILE (Delete entire paragraph. This section included in Recommended Change 1.2.f., Angle Beam Profile Characterization.).	Not applicable.	

TABLE 1.8-1a (Cont)

Appendix A Alternative Method	Response	Procedure and References
5. SCANNING WELD-METAL INTERFACE (Delete entire paragraph. This section included in Recommended Change 3.2, Scanning Weld-Metal Interface.)	Not applicable.	
6. RECORDING AND SIZING The capability to detect, record, and size the flaws delineated by Section XI, IWB-3500 <u>should be demonstrated</u>. The measurement tolerance established should be applied when sizing flaws detected and recorded during scanning (see paragraph 7.a.).	The capability to detect, record, and size flaws is documented as part of the procedure-qualification record.	A procedure-qualification record is required by NES Document 80A9084. Additional sizing data is also provided. The procedure-qualification records are available on site and are provided in the final report.
6.1 Geometric Indications Indications determined to be from geometric sources need not be sized. Recording of these indications should be at 50 percent DAC. When indications are evaluated as geometric in origin, the basis for that determination should be described. After recording sufficient information to identify the origin of the geometric indication, further recording and evaluation are not required.	Comply with the requirements of this paragraph.	This is a requirement of NES Document 83A1749, Section 5. This information becomes part of the examination record.
6.2 Indications with Changing Metal Path	Comply as stated in the following paragraphs.	Recording requirements are defined in NES Document 83A1749, Section 5. Recordable indications are dispositioned on a Flaw Evaluation Sheet which is part of the examination record.
a. Indications that change metal path distances (indicating through-wall dimension) when scanned in accordance with the requirements	Flaws with changing-metal path that exceed the applicable-amplitude criteria are considered recordable regard-	Standard department practice.

TABLE 1.8-1a (Cont)

Appendix A Alternative Method	Response	Procedure and References
of ASME Section XI, for a distance greater than that recorded from the calibration reflector should be recorded.	less of whether the change in the flaw's metal-path distance is greater than that recorded for the calibration reflector.	
b. Reflectors which are at metal paths representing 25 percent and greater of the through-wall thickness of the vessel wall measured from the inner surface should be recorded in accordance with the requirments of ASME Section XI and characterized at 50 percent DAC.	Comply with the requirements of this paragraph.	
c. Reflectors which are within the inner 25 percent of the through-wall thickness should be recorded at 20 percent DAC. Characterization should be in accordance with the demonstrated methods under Paragraph 6.0. Where the indication is sized at 20 percent DAC, this size may be corrected by subtracting the beam width in the through-thickness direction obtained from the calibration hole (between 20 percent DAC points) which is at a depth similar to the flaw depth. If the indication exceeds 50 percent DAC, the length should be recorded by measuring the distance between 50 percent DAC levels. The determined size should be the larger of the two.	Comply with the requirements of this paragraph.	The sizing requirements of this paragraph are implicitly referenced in NES Document 83A1754, Paragraph 7.6.
6.3 Indications without Changing Metal Path	Comply as stated in the following paragraphs.	Recording requirements are defined in NES Document 83A1749, Section 5. Recordable indications are dispositioned on a Flaw Evaluation Sheet which is part of the examination record.

TABLE 1.8-1a (Cont)

Appendix A Alternative Method	Response	Procedure and References
a. Indications which do not change metal path distance when scanned in accordance with the requirements of ASME Section XI and are within the outer 75 percent of the through-wall dimension should be recorded when any continuous dimension exceeds one inch.	Comply with the requirements of this paragraph.	
b. If the indication falls within the inner 25 percent of the through-wall dimensions, it should be recorded at 20 percent DAC and evaluated at 50 percent DAC.	Comply with the requirements of this paragraph.	
c. Precautionary note: Indications lying parallel to welds may appear as nontraveling (without changing metal path) when scanned by parallel moving transducers whose beams are aimed normal to the weld, i.e. at 90°. Multiple scans, however, may reveal that these indications are traveling indications. If so, recording and sizing are to be done in accordance with Paragraph 6.2.	This paragraph applies to angle-beam examinations. The scanning orientation described is not applicable to the performance of the automated examinations.	Not applicable.
6.4 Additional Recording Criteria		
The following information should also be recorded for indications that are reportable according to this regulatory positon:		
a. Indications should be recorded at scan intervals no greater than one-fourth inch.	Comply with the requirements of these paragraphs.	The requirements of this paragraph are implicity referenced in NES Document 83A1754, paragraph 7.6.
b. The recorded information should include the indication travel (metal path distance) and the transducer position for 20 percent,		

Appendix A Alternative Method	Response	Procedure and References
(where applicable), 50 percent, and 100 percent DAC and the maximum amplitude of the signal.		
c. When multi-channel equipment is used in the examination system such that all examination displays are not available for simultaneous viewing, an electronic gating system should be used which will provide on-line reproducible recorded information, regarding metal path, amplitude and position of all indications exceeding a preset level. The preset level should be the minimum recording level required. To ensure that all recordable indications are recorded, a preferred method would incorporate multi-gates in each channel or a single gate for each channel with multi-indication recording capability.	During examination, UDRPS makes a simultaneous record of each UT channel. The record for each channel includes the full extent of all indications including depth and metal path, amplitude, and position. Due to its unique features, the recording of indications is not amplitude dependent. Therefore, there is no requirement to set a minimum recording level. Evaluation of the data record for indications is done off line in the near term before leaving site.	Not applicable.
7. REPORTING OF RESULTS	Comply with the requirements of this paragraph.	Records are maintained on site as stated throughout this document.
Records obtained while following the recommendations of regulatory positions 1.2, 3 and 6, along with discussions and explanations, if any, should be kept available at the site. If the size of an indication, as determined in regulatory positions 6.2 or 6.3, exceeds the allowable limits of Section XI of the ASME Code, the indications should be reported as abnormal degradation of reactor pressure boundary in accordance with the recommendation of regulatory position 2.a(3) of Regulatory Guide 1.16.	Indications determined to be reportable after evaluation are reported to the client.	This is a requirement of NES Document 83A1754. Paragraph 7.6.
Along with the report of ultrasonic examination test results, the following information should also be included:		

Appendix A Alternative Method	Response	Procedure and References
a. The best estimate of the tolerances in sizing the flaws at the sensitivity required in Section 6 and the basis for this estimate.	The best estimate of flaw-sizing tolerance is reported with the evaluation. See also response to 3.1c and 3.2.	This is a requirement of NES Document 83A1754, Paragraph 7.6. The sizing tolerance or error band is included on the Flaw Evaluation Sheet which is part of the examination record.
This estimate may be determined in part by the use of additional reflectors in the basic calibration block.		
b. A description of the technique used to qualify the effectiveness of the examination procedure, including as a minimum, material, section thickness, and reflectors.	The information required in this paragraph is furnished as part of the procedure-qualification record. See also response to 6.	Procedure-qualification records are available on site and are provided in the final report.
c. The best estimate of the portion of the volume required to be examined by the ASME Code that has not been effectively examined such as volumes of material near each surface because of near-field or other effects, volumes near interfaces between cladding and parent metal, volumes shadowed by laminar material defects, volumes shadowed by part geometry, volumes inaccessible to the transducer, volumes affected by electronic gating, and volumes near the surface opposite the transducer.[1]	Limitations in scan area are identified as they occur during the course of the examination included are sketches of equipment, fixtures or part geometry which contribute to the incomplete coverage. From the information provided. The volume of WRV not examined can be determined. Generally, the percentage of scan area missed is greater than the corresponding volume not examined.	This is a requirement of NES Document 83A1754, Paragraph 4.5. Limitatios are identified on an Examination Limitation Report Sheet which is part of the examination record.
Sketches and/or descriptions of the tools, fixtures, and component geometry which contribute to incomplete coverage should be included.	The WRV not examined due to near field, laminar, and similar conditions are identified.	This information is provided in the final report
d. Provide sketches of equipment (i.e., scanning mechanism and transducer holders) with reference points and necessary dimensions to allow a reviewer to follow	Sketches and/or drawings of the equipment with reference points are provided.	This information is available on site and is provided in the final report. A description of the reference system is in NES Document 83A1749, Paragraph 3.2

TABLE 1.8-1a (Cont)

Appendix A Alternative Method	Response	Procedure and References
the equipment's indication location scheme.		
e. When other volumetric techniques are used, a description of the techniques used should be included in the report.	Not applicable. Other volumetric techniques are not used.	

[1] It should be noted that the licensee is required to apply for relief from impractical ASME Code requirements according to Paragraph 50.55a of 10 CFR. If the licensee is committed to examine a weld in accordance with the inspection plan in the plant SAR, the licensee is required to file an amendment when the committments in the SAR cannot be met.

TABLE 1.8-2

CONFORMANCE TO DIVISION 8 NRC REGULATORY GUIDE

Regulatory Guide 8.1, Revision 0 (February 1973)

Radiation Symbol

FSAR Section None

Position

The Unit 2 project complies with Regulatory Position (Paragraph C) of this guide.

TABLE 1.8-2 (Cont)

Regulatory Guide 8.2, Revision 0 (February 1973)

Guide for Administrative Practices in Radiation Monitoring

FSAR Section None

Position

See Section 12.5.3 for an assessment of this Regulatory Guide.

TABLE 1.8-2 (Cont)

Regulatory Guide 8.3, Revision 0 (February 1973)

Film Badge Performance Criteria

FSAR Section None

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

TABLE 1.8-2 (Cont)

Regulatory Guide 8.4, Revision 0 (February 1973)

Direct-Reading and Indirect-Reading Pocket Dosimeters

FSAR Section None

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

TABLE 1.8-2 (Cont)

Regulatory Guide 8.5, Revision 1 (March 1981)

Criticality and Other Interior Evacuation Signals

FSAR Section None

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

TABLE 1.8-2 (Cont)

Regulatory Guide 8.6, Revision 0 (May 1983)

Standard Test Procedures for Geiger-Müller Counters

FSAR Section None 7

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

TABLE 1.8-2 (Cont)

Regulatory Guide 8.7, Revision 0 (May 1973)

Occupational Radiation Exposure Records Systems

FSAR Section None

Position

See Section 12.5.3 and Exhibit 12.1-2 for an assessment of this Regulatory Guide. 11

TABLE 1.8-2 (Cont)

Regulatory Guide 8.8, Revision 3 (June 1978) 28

Information Relevant to Ensuring That Occupational Radiation Exposure at Nuclear Power Stations Will Be as Low as is Reasonably Achievable

FSAR Sections 11, 12, and 13

Position

The Unit 2 project complies with this guide with the following clarifications:

Regarding Position C.2, the recommendations stated in this section of the regulatory guide were considered during the development of the design for Unit 2. The implementation of these recommended ALARA improvements is evidenced in the FSAR and plant drawings.

As part of the ongoing ALARA program, procedure(s) addressing the guidance of Position C.2 will be implemented.

Regulatory Position C.2.g.1 recommends that a radiation monitoring readout be available at the main access control point. The Unit 2 digital radiation monitoring system has a complete console readout in the radiation protection office. The purpose of this readout is for radiation protection personnel only to monitor radiation levels and respond to unusual occurrences. There is also readout capability in the Technical Support Center for monitoring during accident conditions.

The radiation protection office is located on elevation 306' at the main access point. Radiation protection personnel in the office could alert personnel entering the restricted area if radiologic conditions warranted.

Regulatory Position C.3.a.8.e recommends that the work permit state an estimated exposure time required to complete a task and the estimated dose anticipated from the exposure. A site procedure requires that this information be documented on the Radiation Work Permit Request form. 28

Regulatory Guide C.4.a.2 recommends that the counting room facility be equipped with a low-background alpha-beta proportional counter. The Unit 2 counting room will utilize an Nuclear Measurements Corporation PC-5 counter or its equivalent. This equipment is a gas flow proportional

TABLE 1.8-2 (Cont)

counter. It provides adequate sensitivity for nuclear power reactor applications. A description of the instrument is in Table 12.5-1. Calibration of the instrument is described in Section 12.5.2.2.1.

Regulatory Position C.4.b.2 recommends that portable high-range (0.1-500 R/hr) ion chambers be provided. Unit 2 will utilize 0-50 R/hr ion chambers (Eberline RO-2A or equivalent). An electronically quenched Geiger-Muller detector will be used for radiation fields up to 1,000 R/hr. 14

Regulatory Position C.4.c.2 recommends the use of a 0-200 mR personnel pocket dosimeter. Unit 2 will utilize 0-500 mR pocket dosimeters.

Regulatory Position C.4.c.5 recommends hand and foot monitors be used. Unit 2 will use Geoger-Muller type probes for personnel monitoring; however, these probes will not be in a fixed hand and foot configuration.

TABLE 1.8-2 (Cont)

Regulatory Guide 8.9, Revision 0 (September 1973)

Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program

FSAR Section None

Position

See Sections 12.5.3 and Exhibit 12.1-2 for an assessment of this Regulatory Guide.

TABLE 1.8-2 (Cont)

Regulatory Guide 8.10, Revision 1-R (May 1977)

Operating Philosophy for Maintaining
Occupational Radiation Exposures as
Low as Is Reasonably Achievable

FSAR Sections 12.1, 12.5.3.

Position

The Unit 2 project complies with this guide. 14

TABLE 1.8-2 (Cont)

Regulatory Guide 8.12, Revision 0 (December 1974)

Criticality Accident Alarm System

Position

This regulatory guide is based on the requirements of 10CFR70.24 and nuclear power plants need not comply with this regulation.

TABLE 1.8-2 (Cont)

Regulatory Guide 8.13, Revision 1 (November 1975)

Instruction Concerning Prenatal Radiation Exposure 7

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

TABLE 1.8-2 (Cont)

Regulatory Guide 8.14, Revision 1 (August 1977)

Personnel Neutron Dosimetry

Fsar Section None 7

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

TABLE 1.8-2 (Cont)

Regulatory Guide 8.15, Revision 0 (October 1976)

Acceptable Programs for Respiratory Protection

FSAR Section None 7

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

TABLE 1.8-2 (Cont)

Regulatory Guide 8.19, Revision 1 (June 1977) for comment

Occupational Radiation Dose Assessment
in Light-Water Reactor Nuclear Power Plants-
Design Stage Man-Rem Estimates

FSAR Section None

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

TABLE 1.8-2 (Cont)

Regulatory Guide 8.20, Revision 1 (September 1979)

Applications of Bioassay for I-125 and I-131

FSAR Section 7

Position

The Unit 2 project complies with the Regulatory Position (Paragraph C) of this guide.

TABLE 1.8-2 (Cont)

Regulatory Guide 8.26, Revision 0 (September 1980)

Applications of Bioassay for Fission and
Activation Projects

FSAR Section None

Position

See Sections 12.5.3 and Exhibit 12.1-2 for an assessment of this Regulatory Guide.

TABLE 1.8-2 (Cont)

Regulatory Guide 8.27, Revision 0 (March 1981)

Radiation Protection Training for Personnel
at Light-Water Cooled Nuclear Power Plants

FSAR Section None

Position

See Section 12.5.3 for an assessment of this Regulatory Guide.

TABLE 1.8-2 (Cont)

Regulatory Guide 8.28, Revision 0 (August 1981)

Audible Alarm Dosimeters

FSAR Section

Position

This regulatory guide is not applicable since Unit 2 does not use Audible Alarm dosimeters. Refer to Chapter 12 and Section 1.9, Table 1.9-72.

TABLE 1.8-2 (Cont)

Regulatory Guide 8.29, Revision 0 (July 1981)

Instruction Concerning Risks From
Occupational Radiation Exposure

FSAR Section None

Position

See Section 12.5.3 for an assessment of this Regulatory Guide.

1.9 STANDARD REVIEW PLAN CONFORMANCE TO ACCEPTANCE CRITERIA

Table 1.9-1 lists the Standard Review Plans, identifies those applicable to Unit 2, and indicates whether or not Unit 2 complies with the acceptance criteria. Where Unit 2 does not comply with the acceptance criteria, a table where there is a justification of the difference is referenced.

1.9 STANDARD REVIEW PLAN CONFORMANCE TO ACCEPTANCE CRITERIA

Table 1.9-1 lists the Standard Review Plans, identifies those applicable to Unit 2, and indicates whether or not Unit 2 complies with the acceptance criteria. Where Unit 2 does not comply with the acceptance criteria, an attachment where there is a justification of the difference is 1
referenced. The attachments follow Table 1.9-1.

ATTACHMENT 1.9-1

STANDARD REVIEW PLAN 2.1.3, REVISION 2
POPULATION DISTRIBUTION

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TABLE 1.9-1 (Cont)

SRP Number	Title	Revision	Conformance	Difference
9.1.3	Spent Fuel Pool Cooling and Cleanup System	1		Attachment 1.9-52A 5
9.1.4	Light Load Handling System (Related to Refueling)	2	X	
	BTP ASB 9-1	2	NA	NA
9.1.5	Overhead Heavy Load Handling Systems	0		Attachment 1.9-53
9.2.1	Station Service Water System	2	X	
9.2.2	Reactor Auxiliary Cooling Water Systems	1		Attachment 1.9-54
9.2.3	Demineralized Water Makeup Systems	2		Attachment 1.9-55
9.2.4	Potable and Sanitary Water System	2	X	
9.2.5	Ultimate Heat Sink	2	X	
	BTP ASB 9-2	2	X	
9.2.6	Condensate Storage Facilities	2	NA	NA
9.3.1	Compressed Air System	1	NA	NA
9.3.2	Process and Post-Accident Sampling System	2		Attachment 1.9-56
9.3.3	Equipment and Floor Drainage System	2	X	
9.3.4	Chemical and Volume Control System (PWR) (Including Boron Recovery System)	2	NA	NA
9.3.5	Standby Liquid Control System (BWR)	2	X	
9.4.1	Control Room Area Ventilation System	2		Attachment 1.9-57
9.4.2	Spent Fuel Pool Area Ventilation System	2		Attachment 1.9-58
9.4.3	Auxiliary and Radwaste Area Ventilation System	2		Attachment 1.9-59
9.4.4	Turbine Area Ventilation System	2	X	
9.4.5	Engineered Safety Feature Ventilation System	2		Attachment 1.9-60
9.5.1	Fire Protection Program	3		Attachment 1.9-61
	BTP CMEB 9.5.1	2		Attachment 1.9-61
9.5.2	Communications System	2	X	
9.5.3	Lighting Systems	2	X	
9.5.4	Emergency Diesel Engine Fuel Oil Storage and Transfer System	2	X	
9.5.5	Emergency Diesel Engine Cooling Water System	2	X	
9.5.6	Emergency Diesel Engine Starting System	2		Attachment 1.9-62
9.5.7	Emergency Diesel Engine Lubrication System	2	X	
9.5.8	Emergency Diesel Engine Combustion Air Intake and Exhaust System	2	X	
CHAPTER 10: STEAM AND POWER CONVERSION SYSTEM				
10.2	Turbine Generator	2	X	
10.2.3	Turbine Disk Integrity	1	X	
10.3	Main Steam Supply System	2		Attachment 1.9-63
10.3.6	Steam and Feedwater System Materials	2	X	
10.4.1	Main Condensers	2	X	
10.4.2	Main Condenser Evacuation System	2		Attachment 1.9-64

ATTACHMENT 1.9-2

STANDARD REVIEW PLAN 2.3.2, REVISION 2, JULY 1981
LOCAL METEOROLOGY

<u>Difference 1</u>

No topographic description is provided in FSAR Section 2.3.2. Effects of terrain modification and plant structures are also not discussed.

<u>Discussion</u> A description of the topography in the site region is provided in Section 2.5.1. The effects of terrain modification and plant structures on local meteorology are not significant and, therefore, are not expected to have any impact on plant operation. They are discussed in Section 2.3.5.

<u>Difference 2</u>

The wind sensors (Bendix Aerovanes) at the 61-m (200-ft), 30-m (100-ft), and 9-m (30-ft) levels of the onsite
meteorological tower did not meet the accuracy and starting 1
speed recommended in Regulatory Guide 1.23 for data collected until July 1982.

<u>Discussion</u> The severe weather conditions encountered on the shoreline of Lake Ontario required the choice of very rugged wind speed equipment prior to the tower's installation in late 1973. The Bendix Aerovane was chosen for its proven ability to withstand the climate of the region as opposed to measuring the infrequent calm hours.

The Bendix Aerovane has a starting speed of about 1.2 m/sec (2.6 mph) and continues to operate with speeds of 0.4 to 0.7 m/sec (1 to 1.5 mph). The wind speed accuracy is ±0.4 m/sec (±1.0 mph) above 4.5 m/sec (10 mph) as opposed to the Regulatory Guide 1.23 criterion of ±0.2 m/sec (±0.5 mph) for all wind speeds. More sensitive wind speed sensors available at that time were prone to icing and physical damage from high wind speeds.

Subsequent to July 1982, new instruments were installed to comply with the regulatory guide.

ATTACHMENT 1.9-3

STANDARD REVIEW PLAN 2.3.3, REVISION 2, JULY 1981
ONSITE METEOROLOGICAL MEASUREMENTS PROGRAMS

<u>Difference</u>

The wind sensors (Bendix Aerovanes) at the 61-m (200-ft), 1
30-m (100-ft), and 9-m (30-ft) levels of the onsite meteorological tower did not meet the accuracy and starting speed recommended in Regulatory Guide 1.23 for data collected before July 1982.

<u>Discussion</u> See the discussion of Difference 2 for SRP 2.3.2.

ATTACHMENT 1.9-4

STANDARD REVIEW PLAN 2.4.12, REVISION 2, JULY 1981
GROUNDWATER

Difference 1

Section 2.4.12 does not address groundwater.

Discussion The material required by SRP 2.4.12, Groundwater, can be found in FSAR Section 2.4.13. Except for the number change there are no differences noted.

ATTACHMENT 1.9-5

STANDARD REVIEW PLAN 2.4.13
ACCIDENTAL RELEASES OF LIQUID EFFLUENT
IN GROUND AND SURFACE WATERS

Difference 1

No chapter/section exists with the required SRP 2.4.13 title.

Discussion Section 2.4.13.3 addresses accidental effects and dilution modeling as required by this SRP acceptance criteria.

ATTACHMENT 1.9-6

STANDARD REVIEW PLAN 2.5.4, REVISION 2, JULY 1981
STABILITY OF SUBSURFACE MATERIALS AND FOUNDATIONS

Difference

The acceptance criteria for Section 2.5.4.8, Liquefaction
Potential, states that liquefaction potential assessments 1
using both deterministic and probabilistic approaches are
desireable. However, only the deterministic method was
used.

Discussion All major Category I structures for Unit 2 are founded on sound bedrock.

The deterministic approach discussed in Section 2.5.4.8 for the analysis of liquefaction potential under a few minor structures founded in soil backfill is considered adequate and does not require a supplementary probabilistic analysis.

ATTACHMENT 1.9-7

STANDARD REVIEW PLAN 2.5.5, REVISION 2, JULY 1981
STABILITY OF SLOPES

Difference

Only the deterministic method was used in the design and analysis of slopes. 1

Discussion The revised SRP promotes both deterministic and probabilistic approaches to slope design analysis, indicating that the latter method is desirable rather than mandatory, and that it may be employed by the NRC staff itself. To analyze and design the manmade slopes, which are discussed in FSAR Section 2.5.2.2, only the deterministic approach was utilized. It is considered adequate and does not require a supplementary probabilistic analysis.

ATTACHMENT 1.9-8

STANDARD REVIEW PLAN 3.2.1, REVISION 1
SEISMIC CLASSIFICATION 1

All differences between seismic classifications in Regulatory Guide 1.29, Revision 3, and the Unit 2 design are indicated in FSAR Table 3.2-1. A discussion of the differences is also included in the table.

Attachment 1.9-9

STANDARD REVIEW PLAN 3.2.2, REVISION 1
SYSTEM QUALITY GROUP CLASSIFICATION 1

All differences between System Quality Group Classification in Regulatory Guide 1.26, Revision 3, and the Unit 2 design are indicated in FSAR Table 3.2-1. A discussion of the differences is also included in the table.

ATTACHMENT 1.9-10

STANDARD REVIEW PLAN 3.3.1, REVISION 2
WIND LOADINGS

<u>Difference</u>

The acceptance criteria for Revision 1 of SRP 3.3.1 (NUREG-75/087) are allowed using either ASCE Paper No. 3269 or ANSI A58.1-1972 as the basis for wind design. The acceptance criteria for Revision 2 of SRP 3.3.1 (NUREG-0800) 1
considers ANSI A58.1 as the base document while permitting use of ASCE Paper No. 3269 only for cases which ANSI A58.1 does not cover.

Unit 2 is designed using ASCE Paper No. 3269 consistent with the PSAR commitment and with the state of the art available at the time of plant design.

<u>Discussion</u> A review of ANSI A58.1, for derivation of wind pressure for a typical structure, or parts and portions of a structure, indicates that the values thus derived are essentially identical to those derived using ASCE Paper No. 3269.

ATTACHMENT 1.9-11

STANDARD REVIEW PLAN 3.5.1.3, REVISION 2
TURBINE MISSILES

Difference

For Unit 2 the following was used in lieu of paragraph II.1:

> The protection of essential systems located within the low trajectory missile strike zone is acceptable if the probability of damage summed over all structural cubicles containing such systems is less than 10^{-7} per annum.

Discussion The purpose for using an annual damage probability of 10^{-7} and an annual turbine failure rate of 10^{-4}, rather than directly using the resulting factor of 10^{-3}, is to permit compliance with Section III.2 of SRP 3.5.1.3, Revision 2. This allows the turbine failure rate of 10^{-4} per annum to be subdivided as follows: 1

> $P = 6 \times 10^{-5}$ per turbine year for design speed failures
> $P = 4 \times 10^{-5}$ per turbine year for destructive overspeed failures

The reason for evaluating acceptability by summing probabilities over cubicles containing essential systems, rather than by summing over the essential systems themselves, is to simplify the analysis. It is impractical to evaluate the strike probability for each system, considering the complex routing and the possibility for minor layout changes during plant design. By performing the evaluation on the basis of cubicles containing essential systems, this difficulty is avoided while still ensuring that all essential systems are considered in the evaluation.

ATTACHMENT 1.9-12

STANDARD REVIEW PLAN 3.5.3, REVISION 1
BARRIER DESIGN PROCEDURES

Difference

The tornado missile spectrum in Table 2 of this SRP is not used.

Discussion

Unit 2 is designed to withstand the tornado-generated missiles of Spectrum A of SRP 3.5.1.4, Revision 2. 5

See Section 3.5.3 for further discussion.

ATTACHMENT 1.9-13

CONFORMANCE TO BTP ASB 3-1, REVISION 1 ATTACHED TO STANDARD REVIEW PLAN 3.6.1 PROTECTION AGAINST POSTULATED PIPING FAILURES IN FLUID SYSTEMS OUTSIDE CONTAINMENT

<u>Difference 1</u>

Section B.1.a.1 states that "even though portions of the main steam and feedwater lines meet the break exclusion requirements of B.1.6 of BTP MEB 3-1, they should be separated from essential equipment. In order for essential equipment to be properly separated, the essential equipment must be protected from the jet impingement and the environmental effects of an assumed longitudinal break of the main steam line and feedwater lines. Each assumed longitudinal break should have a cross sectional area of at least one square foot and should be postulated to occur at a location that has the greatest effect on essential equipment."

FSAR Section 3.6.2.1.5 states that "regardless of the fact that all conditions [for break exclusion piping] have been met, a crack is postulated in the main steam or feedwater piping in the main steam tunnel. The crack in the pipe, 1
equal in area to a single-ended pipe rupture, is considered a singular event. Pipe whip and jet impingement are not considered, and a single active failure is not taken as a concurrent event."

<u>Discussion</u> As a result of the issuance of SRPs 3.6.1 and 3.6.2 in 1975, the Unit 2 pipe rupture criteria were revised in a letter to the NRC on July 31, 1978 (Section 3.6, Reference 1) to show compliance to the latest requirements. Subsequently, it was recognized that there were additional concerns in the main steam tunnel. The Unit 2 plant was modified to incorporate the requirements outlined in a letter to the ASLAB from the NRC, dated October 4, 1978, concerning Carolina Power and Light's Shearon Harris plant. The NRC position was as follows:

<u>STEAM TUNNEL DESIGN FOR NUCLEAR PLANTS</u>

> We have revised our requirements for the design of nuclear power plants relating to postulated high energy line breaks outside the containment. Specifically the revision will require that the compartment between the containment and the turbine building, which houses the

ATTACHMENT 1.9-13 (Cont)

main steam lines and feedwater lines and the isolation valves for these lines, be designed to consider the pressure and environmental effects from an assumed break, equivalent to the flow area of a single ended pipe rupture in these lines. This revision will require that if this assumed break could cause the collapse of this compartment, then the collapse should not jeopardize the safe shutdown of the plant. Furthermore, it will require that essential equipment located within the compartment, or adjacent to the compartment, be designed to withstand the environmental effects resulting from the above break.

The results of a postulated pipe break in a high energy line (one in which either the pressure exceeds 275 psig or the temperature exceeds 200°F) are pipe whip, jet impingement, and the environmental effects of pressure, temperature, humidity and flooding. With the exception of certain break exclusion regions, pipe breaks are postulated at terminal ends and other points of relatively high stress and fatigue. Within the break exclusion region, which is limited to the containment penetration area, a combination of low stress and fatigue design coupled with augmented inservice
1 inspection is used to assure that no pipe breaks will occur due to the design loads. However, the stress is not normally low enough to reasonably preclude the possibility of a postulated pipe crack in the region. As such, it is prudent to require that the surrounding pipe tunnel be designed to withstand the effects of a postulated pipe crack. These effects are of an extreme environmental nature for equipment in the vicinity and include pressure, temperature, humidity and flooding. Because of the augmented inservice inspection and the low stress and fatigue design, it is reasonable to assume that a postulated pipe crack will be detected and repaired before it becomes through wall or, at the latest, in its initial phase of leakage. The flow area for postulated pipe cracks is conservatively selected as the cross sectional area of the pipe, however, to ensure that the largest possible crack is enveloped by the design.

Implementation of this position will ensure that essential equipment will not be arbitrarily housed in the main steam tunnel such that safety by separation is maintained.

ATTACHMENT 1.9-13 (Cont'd)

Difference 2

Section B.1.c states that "a program should be developed to ensure that the system stresses due to long-term changes in the system and its supports and restraints, such as due to pipe relaxation and differential settling, will not be adversely affected by the restraints. Details of the methods used to obtain these assurances should be submitted to the staff for review."

Discussion Clearances at pipe whip restraints were extensively reviewed by recording piping displacements at selected whip restraints during start-up testing. A comparison between predicted and measured pipe movements was made. It was concluded that piping systems would not experience any additional stress due to these long-term changes provided that:

1. the existing clearances are maintained, or
2. if the maximum pipe movements as predicted by reanalysis are changed, the new displacements at the restraint location are reviewed on a case-by-case basis to ensure that the intent of BTP MEB 3-1 is met.

Difference 3

Section B.2.d states that "piping classification as required by Regulatory Guide 1.26 should be maintained without change until beyond the outboard restraint. If the restraint is located at the isolation valve, a classification change at the valve interface is acceptable." For Unit 2, the piping classification change is made at the valve (not beyond the outboard restraint) in accordance with Regulatory Guide 1.26.

Discussion Although the classification change is made at the valve, the piping between the valve and the first restraint outside containment is B31.1 (for main steam and feedwater) and ASME Section III, Class 2 and 3 (for reactor core isolation cooling and reactor water cleanup, respectively). Additionally, the piping meets the stringent break exclusion requirements in Item B.1.b of BTP MEB 3-1. It is therefore concluded that this will not degrade the safety of the plant.

ATTACHMENT 1.9-14

STANDARD REVIEW PLAN 3.6.2A, REVISION 1, JULY 1981
BRANCH TECHNICAL POSITION MEB 3-1
DETERMINATION OF RUPTURE LOCATIONS AND DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING

<u>Difference 1</u>

Section B.1.e states that "with the exceptions of those portions of piping identified in B.1.b, leakage cracks should be postulated in ASME Code, Section III, Class 1 piping where the stress range by Eq. (10) of Paragraph NB-3653 exceeds 1.2 0 and in Class 2 or 3 or nonsafety class piping where the stress by the sum of Eq. (9) and (10) of Paragraph NC/ND 3652 exceeds 0.4. Nonsafety piping which has not been evaluated to obtain similar stress information shall have cracks postulated at locations that result in the most severe environmental consequence." For high energy piping in areas other than the containment penetration Unit 2 postulates breaks in accordance with Sections B.1.a and B.1.c.

<u>Discussion</u> By evaluating the effects of jet impingement,
pipe whip, environment, etc, for high energy piping systems
in accordance with Sections B.1.a and B.1.c, any event that
could adversely affect the safety of the plant will be 1
considered. Generally this is due to the following:

1. The criteria in Section B.1.a are invoked whenever possible to separate essential equipment from high energy systems. In this case, breaks are arbitrarily postulated and a stress criterion is meaningless.

2. When it is not possible to separate high energy piping from essential equipment, redundancy is provided or an evaluation is performed to ensure that the equipment will remain operable.

3. In areas in which high energy pipe is routed, a sufficient number of breaks will always be postulated such that the effects of jet impingement, pipe whip, environment, etc, which result, will envelop any intermediate or additional cracks.

The following discussion shows that for all areas of the plant, an additional criterion to postulate cracks is only repetitive and will not improve the safety of the plant.

ATTACHMENT 1.9-14 (Cont)

<u>High-energy piping not in the reactor building</u> High-energy pipe is not routed near systems, components, or structures essential to safe shutdown in areas other than the reactor building. For example, there is a significant amount of high-energy piping located in the turbine building; however, there is no essential equipment located there which could compromise the safety of the plant. This piping actually meets the criteria of Section B.1.a where breaks are arbitrarily postulated to ensure separation of high-energy piping and essential equipment. It is therefore concluded that this will not degrade the safety of the plant.

<u>High-energy piping in the reactor building (excluding primary containment and containment penetration areas)</u> Excluding the main steam tunnel piping, the only systems which qualify as high-energy piping in secondary containment are the RWCU, SLC, CRD, and RCIC systems. Routing of these systems has been controlled so that they are located in well-defined areas (i.e., RCIC pipe chase and turbine room, RWCU pump room, valve room, demineralizer room, heat exchanger room, and pipe chases). The walls of these compartments have been designed for jet impingement loads using a worst-case condition applied at any location. The compartments have also been evaluated for environmental, flood, pressure, etc, effects using the worst-case condition. Breaks in these areas are often arbitrarily postulated, and imposing additional criteria will not really enhance safety. It is therefore concluded that this will not degrade the safety of the plant.

Furthermore, the environmental effects of cracks in high-energy piping are enveloped by the effects of postulated cracks in moderate-energy systems. As discussed in
12 Appendix 3C, all safety-related equipment in the reactor building is reviewed to ensure either operability or functional redundancy, where required, under environmental effects of spray from a crack in the residual heat removal system piping. The resultant environment is equal to or is more severe than the effects of high-energy system piping cracks for both spray and flooding considerations. Therefore, a separate evaluation of high-energy crack environments is not required.

In the main steam tunnel, the effects of jet impingement will govern all cases assuming a minimum break criterion; therefore, this will not degrade the safety of the plant.

ATTACHMENT 1.9-14 (Cont)

Primary containment If the primary containment piping were designed so that pipe stress results indicated that all high-energy systems required a minimum number of breaks to be postulated, approximately 100 breaks would be considered. In light of the separation between the high-energy systems in primary containment, it is reasonable to assume that these high-energy breaks will always govern. Any equipment, systems, or structures must be designed for the extreme environment in primary containment regardless of its particular location. Electrical equipment is routed in conduit or suitable housing so that it is not exposed to the open environment. The combination of separation and redundancy (the preferred method of protection) is also integral to components and piping routed in the primary

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ATTACHMENT 1.9-14 (Cont)

containment. This is verified in the jet impingement evaluation where breaks are postulated at various elevations and azimuths. Additional investigation is only repetitive. It is therefore concluded that this will not degrade the safety of the plant.

Difference 2

Section B.1.c.1.d states that "if intermediate break locations cannot be determined by (b), (B.1.c.1.b) and (c), (B.1.c.1.c) above, two highest stress locations based on equation (10) should be selected." Unit 2 uses a reasonable basis which includes factors such as points of maximum stress intensity and/or cumulative usage factors; however, the points of maximum stress intensity are based on Equation (12) or (13).

Discussion Since all postulated intermediate breaks require evaluation of Equations (12) and (13) and cumulative usage factors, it is reasonable to use these equations to determine points of maximum stress intensity. This approach is conservative. 1

Difference 3

Section B.1.c.4 states that "if a structure separates a high energy line from an essential component, the separating structure should be designed to withstand the consequences of the pipe break in the high energy line which produces the greatest effect at the structure irrespective of the fact that the above criteria might not require such a break location to be postulated." Unit 2 design structures withstand the consequence of pipe breaks postulated at locations in accordance with Sections B.1.c.1, B.1.c.2, and B.1.c.3.

Discussion A systematic logical method must be used to evaluate the effects of pipe breaks in order to address a finite number of potential load cases. By assuming breaks at highly stressed locations and by requiring a minimum number of locations to be selected, a reasonable margin of safety will evolve.

Requiring breaks to be postulated based on structural capability is not prudent and does not enhance the safety of the plant. Several points are:

ATTACHMENT 1.9-14 (Cont'd)

1. Pipe whip loadings are very sensitive to the distance over which unrestrained whip could occur, piping geometry, and break orientation. An infinite number of cases would require consideration particularly if splits are arbitrarily postulated along the length of the pipe. Jet impingement does not have this problem since the load is distributed over a reasonable area. However, pipe whip requires evaluation of local effects, which is much more involved.

2. An excessive number of scab plates would be required on all structures which separate high energy and essential systems, thus causing an unreasonable number of scab plates to be installed.

3. By strengthening the weakest part of a structure, the next weakest part would then be the worst case. This is a perpetual cycle.

4. Additional safety is not really obtained by evaluating the least likely events. Since pipe breaks themselves are extremely unlikely, it is reasonable to postulate them only at the higher stressed locations. Additionally, all walls in the proximity of high energy systems are evaluated for a reasonable number of pipe breaks simply due to the number of breaks which must be postulated using the stress criteria.

Difference 4

Section B.1.c.1.d states that "As a result of piping reanalysis, the highest stress locations may be shifted; however, the initially determined intermediate break locations need not be changed unless... In such conditions, the newly determined highest stress locations should be the intermediate break locations."

Discussion For the feedwater system, no changes to arbitrary intermediate break locations will be postulated as a result of piping reanalysis. This approach is consistent with Branch Technical Position MEB 3-1, Revision 2, which eliminates requirements for arbitrary intermediate pipe breaks. The existing postulated arbitrary intermediate break locations, as determined in accordance with FSAR Section 3.6.2.1.5A, will not be eliminated.

For systems other than the feedwater system, the relaxed requirements in NRC Generic Letter 87-11 (BTP MEB 3-1, Revision 2) are permitted.

ATTACHMENT 1.9-15

STANDARD REVIEW PLAN 3.6.2B, REVISION 1, JULY 1981
DETERMINATION OF RUPTURE LOCATIONS AND DYNAMIC
EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE
OF PIPING

Difference 1

The method of determining pipe rupture locations inside containment differs in that Equation (12) or (13) is not considered, if Equation (10) is less than or equal to 3.0 Sm and the cumulative usage factor is less than 0.1. 27

Discussion This method is consistent with the previous revision of the SRP (November 24, 1975).

Difference 2

Criterion B.1.e identified in NUREG-0800 was not used.

Discussion This method was not part of the November 24, 1975 SRP which was the only guidance available during the period this work was being performed. The method used, cracks and breaks, is consistent with the SRP applicable at the time the work was performed.

ATTACHMENT 1.9-16

STANDARD REVIEW PLAN 3.7.2A, REVISION 1
SEISMIC SYSTEM ANALYSIS

Difference 1

Additional seismicity of ±5 percent of the maximum building dimension at the level under consideration was not assumed.

Discussion Since the three-dimensional seismic models used in the dynamic analyses of Category I structures account for the torsional effects, including the effects of eccentricities between the centers of rigidity and the centers of mass of the structural components, the additional eccentricity of ±5 percent of the maximum building dimension is not considered necessary.

ATTACHMENT 1.9-17

STANDARD REVIEW PLAN 3.7.3B, REVISION 1
SEISMIC SUBSYSTEM ANALYSIS

Difference

For the determination of the number of earthquake cycles for nuclear steam supply system (NSSS) components and equipment other than piping, 10 equivalent peak operating basis earthquake (OBE) cycles are used, as opposed to the 50 OBE cycles specified in the acceptance criteria.

Discussion Fatigue evaluation due to SSE is not necessary 1
since it is a faulted condition and thus not required by ASME Section III (FSAR Section 3.7.3.2B). The criterion requires that 50 OBE cycles be used, and for NSSS piping, 50 cycles are used. For other NSSS components and equipment, 10 equivalent peak OBE cycles are used (FSAR Section 3.7.3.2B). This 10-cycle approach has been approved by the NRC on the basis of equivalent levels of safety (letter from R. Bosnak [NRC] to R. Artigas [GE], Number of OBE Fatigue Cycles in the BWR NSSS Design, dated February 18, 1982).

ATTACHMENT 1.9-18

STANDARD REVIEW PLAN 3.7.4, REVISION 1
SEISMIC INSTRUMENTATION

Difference

Seismic monitoring instrumentation surveillance frequency is not discussed.

Discussion

The seismic monitoring instrument surveillance program has been incorporated in the Technical Specifications. 25

ATTACHMENT 1.9-19

STANDARD REVIEW PLAN 3.8.1, REVISION 1
CONCRETE CONTAINMENT

Difference 1

An analysis was not performed to determine the ultimate capacity of the concrete containment.

Discussion In lieu of performing ultimate capacity analysis of the containment, the structural acceptance test performed prior to the plant operation for 1.15 times the design pressure is considered sufficient assurance for the adequacy of the analysis and design of the concrete containment.

Difference 2

A design report was not prepared in accordance with Appendix C to SRP 3.8.4, Revision 1, for concrete containment.

Discussion In lieu of a design report all the information relevant to the analysis and design of the concrete containment is provided in FSAR Section 3.8.1. 1

Difference 3

Compliance with Regulatory Guide 1.136 and ASME Section III, Division 2, was not used in designing the concrete containment and containment liner.

Discussion As stated in FSAR Table 1.8-1, Regulatory Guide 1.136, since the containment design precedes the issuance of Regulatory Guide 1.136 and ASME Section III, Division 2, it is not feasible to assure full compliance with these documents. While the loads and loading combinations are in accordance with Table CC-3230-1 of ASME III, Division 2, the acceptance criteria for stresses and strains and the procurement of materials for concrete and steel portions of the containment follows PSAR commitments. Consequently, the design, procurement, and construction of concrete and steel portions of the containment are in accordance with ACI 318, ACI 301, Regulatory Guide 1.94, and ASME Section III, Division 1, respectively. At the Construction Permit stage this was accepted by the NRC as an adequate basis for the design, procurement, and construction of the containment.

ATTACHMENT 1.9-20

STANDARD REVIEW PLAN 3.8.3, REVISION 1
CONCRETE AND STEEL INTERNAL STRUCTURES
OF STEEL OR CONCRETE CONTAINMENTS

Difference 1

Compliance with Regulatory Guide 1.142 and ACI 349-76 was not used in designing concrete internal structures.

Discussion As stated in FSAR Table 1.8-1, Regulatory Guide 1.142, since the design of internal structures precedes the issuance of Regulatory Guide 1.142 and ACI 349-76, it is not feasible to assure full compliance to these documents. Several major provisions of ACI 349-76 are identical to those of ACI 318-71 (and ACI 318-77) which was accepted for Unit 2 by the NRC at the Construction Permit Stage as an adequate basis for the design of Category I concrete structures. ACI 318-71 (and ACI 318-77) is used in designing the concrete internal structures. 1

Difference 2

A design report as described in Appendix C to Section 3.8.4 for all internal structures was not prepared.

Discussion Although the information required by the design report is not provided in the format required by Appendix C of NUREG-0800, SRP 3.8.4, FSAR Section 3.8.3 provides the information necessary for evaluation of internal structures of containment.

ATTACHMENT 1.9-21

STANDARD REVIEW PLAN 3.8.4, REVISION 1
OTHER SEISMIC CATEGORY I STRUCTURES

<u>Difference 1</u>

Compliance with Regulatory Guide 1.142 and ACI 349-76 was not used in designing other Category I structures.

<u>Discussion</u> As stated in FSAR Table 1.8-1, Regulatory Guide 1.142, since the design of Category I structures precedes the issuance of Regulatory Guide 1.142 and ACI 349-76, it is not feasible to assure full compliance to these documents. Several major provisions of ACI 349-76 are identical to those of ACI 318-71 (and ACI 318-77) which was accepted by the NRC for the design of Unit 2 Category I concrete structures at the Construction Permit Stage. ACI 318-71 (and ACI 318-77) is used in designing Category I concrete structures.

<u>Difference 2</u>

A design report, as described in Appendix C, was not prepared for all Category I structures.

<u>Discussion</u> Although the information required by the design 1
report is not provided in the format required by Appendix C, FSAR Section 3.8.4 provides the information necessary for evaluation of Category I structures.

<u>Difference 3</u>

Compliance to safety-related masonry wall criteria, as described in Appendix A to this SRP, was not required.

<u>Discussion</u> As described in FSAR Section 3.8.4.4, Unit 2 does not use masonry walls to support any safety-related structure, system, or component. However, removable, solid concrete blocks contained in position by structural steel supports and adjacent concrete structures are used in Category I structures to provide access for equipment removal and/or installation. Since the supports are designed to withstand all the possible loading combinations and remain in place, the solid concrete blocks cannot endanger adjacent structures, systems, or components. Hence, the criteria in Appendix A to this SRP are not applied to the concrete block designs. The concrete blocks are provided to satisfy the shielding requirements for the area.

ATTACHMENT 1.9-22

STANDARD REVIEW PLAN 3.8.5, REVISION 1
FOUNDATIONS

<u>Difference 1</u>

Compliance with Regulatory Guide 1.142 and ACI 349-76 was not used in designing foundations of Category I structures.

<u>Discussion</u> As stated in FSAR Table 1.8-1, since the design of foundations precedes the issuance of Regulatory Guide 1.142 and ACI 349-76, it is not feasible to assure full compliance to these documents. Several major provisions of ACI 349-76 are identical to those of 1
ACI 318-71 (and ACI 318-77) which was accepted by the NRC to provide an adequate basis for the design of Category I concrete structures for Unit 2 at the Construction Permit Stage. ACI 318-71 (and ACI 318-77) is used in designing the foundations of Category I structures.

<u>Difference 2</u>

A design report, as described in Appendix C to Section 3.8.4, was not prepared for all foundations of Category I structures.

<u>Discussion</u> Although the information required by the design report is not provided in the format required by Appendix C of NUREG-0800, SRP 3.8.4, FSAR Section 3.8.5 provides the information necessary for evaluation of foundations of Category I structures.

ATTACHMENT 1.9-23

STANDARD REVIEW PLAN 3.9.2A, REVISION 2 DYNAMIC TESTING AND ANALYSIS OF SYSTEMS, COMPONENTS, AND EQUIPMENT

<u>Difference 1</u>

A list of snubbers on systems which experience sufficient thermal movement to measure snubber travel from the cold to the hot position is not provided.

<u>Discussion</u> This list will be developed prior to the detailed preoperational test program and included in an amendment to the FSAR.

<u>Difference 2</u>

A description of the thermal motion monitoring program, i.e., verification of snubber movement, adequate clearances and gaps, including acceptance criteria and the manner in which motion will be measured, is not provided.

<u>Discussion</u> This will be developed prior to the detailed preoperational test program and included in an amendment to the FSAR.

<u>Difference 3</u> 1

A description of corrective action to assure that a snubber, which did not displace as predicted by analysis, is operable is not provided.

<u>Discussion</u> This will be addressed in the detailed preoperational test program.

<u>Difference 4</u>

The consideration of maximum relative displacements among supports of Category I systems and components is not described in this section.

<u>Discussion</u> Application to piping is discussed in FSAR Section 3.7.3.8.3A.

ATTACHMENT 1.9-24

STANDARD REVIEW PLAN 3.9.2B, REVISION 2
DYNAMIC TESTING AND ANALYSIS OF SYSTEMS, COMPONENTS, AND EQUIPMENT

<u>Difference</u>

For the determination of the number of earthquake cycles for NSSS components other than piping, 10 equivalent peak OBE cycles are used. The SRP acceptance criteria require 50 cycles.

<u>Discussion</u> Fatigue evaluation due to SSE is not necessary 1
since it is a faulted condition and thus not required by ASME Section III (FSAR Section 3.7.3.2B). The criterion requires that 50 OBE cycles be used, and for NSSS piping, 50 cycles are used. For other NSSS components and equipment, 10 equivalent peak OBE cycles are used (FSAR Section 3.7.3.2B). This 10-cycle approach has been approved by the NRC on the basis of equivalent levels of safety (letter from R. Bosnak [NRC] to R. Artigas [GE], Number of OBE Fatigue Cycles in the BWR NSSS Design, dated February 18, 1982).

ATTACHMENT 1.9-25

STANDARD REVIEW PLAN 3.9.3B, REVISION 1
ASME CODE CLASS 1, 2, AND 3 COMPONENTS, COMPONENT SUPPORTS, AND CORE SUPPORT STRUCTURES

Difference

Combination of loadings does not use the criteria in Regulatory Guides 1.124 and 1.130.

Discussion Load combination and acceptance criteria for
Category I component supports are described in FSAR
Sections 3.9.3.1B, 3.9.3.4B, and Table 3.9B-2. Regulatory
Guides 1.124 and 1.130 apply respectively to Class 1 linear
and Class 1 plate and shell component support designs.
Their issue dates of January 1978 (Regulatory Guide 1.124, 1
Revision 1) and July 1977 (Regulatory Guide 1.130) are after
the Unit 2 Construction Permit docketing date requirement.
However, the design utilizes ultimate strength temperature
correlations of regulatory position C2 of these guides in
regions adjacent to the pipe having high temperatures.
Additionally, the critical buckling strength limits of ASME
Section III, Appendix XVII, paragraph 2110(b), are observed
in Regulatory Guide 1.124. Regulatory position C4 with
alternate conservative collapse criteria for plates-shells
is being used in lieu of regulatory position C3 in
Regulatory Guide 1.130. The remaining design analysis
criteria of these regulatory guides are considered to be
adequately addressed by conservatisms present in the
existing ASME Section III code.

ATTACHMENT 1.9-26

STANDARD REVIEW PLAN 3.9.5B, REVISION 2
REACTOR PRESSURE VESSEL INTERNALS

<u>Difference 1</u>

The design and construction of the core support structures do not conform to the requirements of subsection NG, Core Support Structures, of the ASME Code (Reference 5), and SRP Section 3.9.3.

<u>Discussion</u> Unit 2 core support structures were designed and
purchased in 1971 prior to the issue of ASME Section III,
subsection NG, in 1974. However, an earlier draft of ASME
Section III, subsection NB, was used as a guide in
developing the design of these supports. These criteria are 1
presented in Section 3.9.5.3B. Subsequent to the issuance
of ASME Section III, subsection NG, comparisons were made to
assure that the pre-ASME Section III, subsection NG, design
provides the equivalent level of safety as prescribed by
ASME Section III, subsection NG, 1974.

<u>Difference 2</u>

The design criteria, loading conditions, and analyses that provide the basis for the design of reactor internals other than the core support structures do not meet the guidelines of NG-3000.

<u>Discussion</u> Unit 2 reactor internals other than core support structures were designed and purchased prior to the initial issuance of ASME Section III, subsection NG. Design guidelines for these components and later safety comparisons against subsection NG criteria were selected as described for core support structures in criteria II.b under Difference 1.

ATTACHMENT 1.9-27

STANDARD REVIEW PLAN 3.9.6, REVISION 2
INSERVICE TESTING OF PUMPS AND VALVES

Difference 1

The acceptance criteria for NUREG-0800 require that pumps and valves not categorized as Code Class 1, 2, or 3 but which are considered to be safety related be added to the inspection program.

Discussion The Unit 2 inspection and testing program will conform to these criteria by meeting the relevant requirements set forth in General Design Criteria (GDC) 37, 40, 43, 46, 54, and 10CFR50, 50.55a(g).

ATTACHMENT 1.9-28

STANDARD REVIEW PLAN 3.10A, REVISION 2, JULY 1981
SEISMIC AND DYNAMIC QUALIFICATION OF
MECHANICAL AND ELECTRICAL EQUIPMENT 1

<u>Difference</u>

The position for Regulatory Guide 1.148 is not provided in this section.

<u>Discussion</u> The Unit 2 degree of compliance with Regulatory Guide 1.148 is in FSAR Section 1.8.

ATTACHMENT 1.9-29

STANDARD REVIEW PLAN 3.11, REVISION 2
ENVIRONMENTAL QUALIFICATION OF MECHANICAL
AND ELECTRICAL EQUIPMENT

Difference 1

The submittal of the environmental qualification document which demonstrates equipment environmental capability is not included.

Discussion The environmental qualification document is maintained as part of the Nine Mile Point Unit 2 Equipment Qualification Program. This document is maintained separately from the FSAR, and it is not considered a part of the FSAR.

ATTACHMENT 1.9-30

STANDARD REVIEW PLAN 4.2, REVISION 2
FUEL SYSTEM DESIGN

<u>Difference 1</u>

Factors of 2 on stress or 20 on cycles are not used with the current method.

<u>Discussion</u> Design limits for fatigue failure are provided in GESTAR-II, Section 2.5, and have been approved by the NRC.

<u>Difference 2</u>

Allowable fretting wear is not stated in the FSAR. 1

<u>Discussion</u> See GESTAR-II, Appendix A, Section A.4.2.1.1.3, and Section 2.6.3.

<u>Difference 3</u>

Separate design limits for oxidation, hydriding, and corrosion buildup are not stated in the FSAR.

<u>Discussion</u> See GESTAR-II, Appendix A, Section A.4.2.1.1.4.

<u>Difference 4</u>

There is no limit for internal gas pressure stated in the FSAR.

<u>Discussion</u> See GESTAR-II, Appendix A, Section A.4.2.1.1.6. Current methodology has been approved by the NRC.

<u>Difference 5</u>

Allowable fretting wear is not stated in the FSAR.

<u>Discussion</u> See GESTAR-II, Appendix A, Section A.4.2.1.2.3.

<u>Difference 6</u>

There is no centerline melt criterion for abnormal operational events stated in the FSAR.

Discussion See GESTAR-II, Appendix A, Section A.4.2.1.2.5. (Reference Subsections 2.4.2.5 and 2.4.1.1, GESTAR-II)

Difference 7

A description of elastic strain limits is not included in the FSAR. There is no centerline melt criterion for abnormal operational events described in the FSAR.

Discussion The 1 percent plastic strain criterion is applied to all abnormal operating events. No fuel melt criterion is applied. This methodology has been approved by the NRC. See GESTAR-II, Appendix A, Section A.4.2.1.2.7.

Difference 8

1 The fuel system description does not provide all the information discussed in the acceptance criteria.

Discussion The level of descriptive information and detail in the FSAR is consistent with that previously approved and accepted by the NRC. Quantitative information is provided in GESTAR-II, Chapters 2 and S.2, and is referenced in FSAR Section 4.2.

Difference 9

Surveillance of control rods for boron leaching is not provided in the FSAR.

Discussion Periodic reactivity testing of the control rods (beyond the beginning of cycle shutdown margin demonstration) is performed only if there is reason to suspect control absorber loss or other degradation of the control blades.

ATTACHMENT 1.9-31

STANDARD REVIEW PLAN 4.4, REVISION 1
THERMAL AND HYDRAULIC DESIGN 1

Difference

Compliance with TMI Action Plan requirements (NUREG-0737) is not assessed in this section.

Discussion See FSAR Section 1.10, Tasks II.D.1, II.F.1, and II.F.2.

ATTACHMENT 1.9-32

STANDARD REVIEW PLAN 4.5.1, REVISION 2
CONTROL ROD DRIVE STRUCTURAL MATERIALS

Difference 1

Only those parts of the control rod drive (CRD) forming part of the primary pressure boundary are code materials.

Discussion Jurisdiction of ASME Section III does not extend
to the nonpressure parts of the CRD. ASME materials are 1
identified in the materials list of FSAR Section 4.5.1.1.

Difference 2

Some CRD structural materials were not purchased to code requirements, but there is no difference for tempering and aging temperatures.

Discussion Non-code materials are not required to be purchased to code requirements. The materials specified were, however, selected for their compatibility with the reactor coolant. Tempering and aging are done according to standards which are discussed in FSAR Section 4.5.1.

ATTACHMENT 1.9-33

STANDARD REVIEW PLAN 4.5.2, REVISION 2
REACTOR INTERNAL AND CORE
SUPPORT MATERIALS

Difference 1

ASME Section III, NG-2000, specifications were not used.

Discussion For core support and reactor internals, the material specifications given in ASME Section III, NG-2000, were not used. Article NG-2000 was not part of Section III at the time these materials were procured for Unit 2. All core support structures were fabricated from ASME- and ASTM-specified materials and designed using ASME Section III as a guide. The other reactor internals are non-coded and are fabricated from ASME or ASTM specification materials. Material requirements in the ASTM specifications are identical to the requirements given in the corresponding ASME specifications. The material specifications for Unit 2 reactor internal and core support materials are given in FSAR Section 4.5.2.1.

Difference 2

ASME Section III, NG-4000, and NG-5000 were not imposed. 1

Discussion The requirements of Articles NG-4000 and NG-5000 were not part of ASME Section III when fabrication welding was performed for Unit 2. As specified in FSAR Section 4.5.2.2., welding was performed to the requirements of ASME Section IX. Conformance to regulatory guides applicable to welding (i.e., Regulatory Guides 1.31, 1.34, 1.37, 1.44, and 1.71) is presented in FSAR Section 4.5.2.4.

Difference 3

ASME Section III, NG-2500, and NG-5300 were not imposed.

Discussion Articles NG-2500 and NG-5300 were not part of ASME Section III at the time the Unit 2 wrought seamless tubular products and fittings were procured. As contained in FSAR Section 4.5.2.3, wrought seamless tubular products for CRD guide tubes, CRD housings, and peripheral fuel supports were supplied in accordance with applicable ASME material specifications. These specifications require a hydrostatic test on each length of tubing. No other nondestructive testing was specified for the tubes.

ATTACHMENT 1.9-34

STANDARD REVIEW PLAN 5.2.1.1, REVISION 2 COMPLIANCE WITH THE CODES AND STANDARDS RULE, 10CFR50.55a

<u>Difference</u> 1

Differences exist between the Unit 2 design and Regulatory Guide 1.26, Quality Group Classification and Standards.

<u>Discussion</u> Justification for all differences listed in Table 3.2-1 are discussed in the notes to the table.

ATTACHMENT 1.9-35

THE INFORMATION ON THIS PAGE HAS BEEN DELETED.

ATTACHMENT 1.9-36

STANDARD REVIEW PLAN 5.2.2, REVISION 2
OVERPRESSURE PROTECTION

1

Difference

TMI Task are not discussed in this section.

Discussion NUREG-0737, Task II.D.3, is discussed in FSAR Sections 1.10 and 5.4.12.

ATTACHMENT 1.9-37

STANDARD REVIEW PLAN 5.2.3, REVISION 2
REACTOR COOLANT PRESSURE BOUNDARY MATERIALS

<u>Difference 1</u>

Material Specifications - Part of the RCPB materials comply with ASME Section II, Parts A and C, only.

Not all ASME code cases used are listed in Regulatory Guides 1.84 and 1.85.

<u>Discussion</u> Section 5.2.3.1 and Table 5.2-5 list components 27
and their material specifications. These specifications comply with ASME Section II, Parts A and C, and are augmented with ASME Section III, Code Cases 1562 and 1572, and code cases approved by Regulatory Guides 1.84 and 1.85. 27

Code Cases 1141-1, 1332-6, 1361-2, 1557-2, 1620, and N-1588, 27
which are imposed by Regulatory Guides 1.84 and 1.85, were
used as noted in FSAR Table 5.2-1. Other code cases used 27
but not imposed by these regulatory guides are 1562 and 1572. These code cases have been annulled and incorporated into ASME Section III.

<u>Difference 2</u>

The FSAR does not address the qualification of welding procedures at the minimum preheat.

<u>Discussion</u> FSAR Section 5.2.3.3.2 indicates that components were either held for an extended time at preheat temperature to assure removal of hydrogen or were maintained at preheat temperature until post weld treatment. Minimum preheat and maximum interpass temperatures were specified and monitored.

<u>Difference 3</u>

Some ferrite tubular products do not meet all requirements of Regulatory Guide 1.66 and ASME Section III, paragraph NB-2550.

<u>Discussion</u> See FSAR Section 5.2.3.3.3. Nondestructive examination of ferrite tubular products met existing ASME Section III and 10CFR50 criteria at order placement which in some cases predated Regulatory Guide 1.66. CRD housing tubes do meet ASME Section III, paragraph NB-2550.

ATTACHMENT 1.9-37 (Cont)

Difference 4

Regulatory Guide 1.44 was not applied completely by the Unit 2 project design basis for the NSSS.

Discussion For NSSS components alternate criteria for sensitization controls of stainless steel which satisfy NUREG-0313 are discussed in FSAR Section 5.2.3.4.

Difference 5

The NSSS QA program complies with Regulatory Guide 1.37 except for Section 5, paragraph G, which recommends that local rusting on corrosion resistant alloys be removed by mechanical means.

Discussion GE Topical Report NEDO-11209 (accepted by the NRC) describes the NSSS QA program and does not preclude the
1 use of other than mechanical means for local rust removal.

Difference 6

Some austenitic tubular products were procured prior to the creation of ASME Section III, paragraph NB-2550.

Discussion See FSAR Section 5.2.3.3.3 and the assessment of criterion II.3.c of this SRP for positions on ASME Section III, paragraph NB-2550, requirements.

ATTACHMENT 1.9-38

STANDARD REVIEW PLAN 5.3.1, REVISION 1
REACTOR VESSEL MATERIALS

<u>Difference 1</u>

Special Methods for Nondestructive Examination - Ultrasonic examination methods meet ASME Section XI, Appendix I, rather than ASME Section III requirements.

<u>Discussion</u> FSAR Section 5.3.1.2 describes radiographic examination, which is performed on all pressure-containing welds in accordance with requirements of ASME Section III, subsection NB-5320. FSAR Section 5.3.1.3 indicates that materials and welds on the RPV were examined by methods which meet ASME Section III requirements. Special
ultrasonic examination meeting ASME Section XI, Appendix I,
requirements using manual techniques were used. Acceptance 1
standards were equal to or greater than those required by ASME Section XI.

<u>Difference 2</u>

The NSSS QA program complies with the referenced regulatory guides except for Section 5, paragraph 6 of Regulatory Guide 1.37, which recommends that local rusting on corrosion-resistant alloys be removed by mechanical means.

<u>Discussion</u> GE Topical Report NEDO-11209 (accepted by the NRC) describes the NSSS QA program and does not preclude the use of other than mechanical means for local rust removal.

ATTACHMENT 1.9-39

STANDARD REVIEW PLAN 5.4.6, REVISION 2
REACTOR CORE ISOLATION COOLING SYSTEM (BWR)

Difference 1

TMI action items are not discussed in this section.

Discussion See FSAR Section 1.10 for NUREG-0737.

ATTACHMENT 1.9-40

STANDARD REVIEW PLAN 5.4.8, REVISION 2
REACTOR WATER CLEANUP SYSTEM

Difference 1

All RWCU system components are not drained and vented through closed systems.

Discussion Vents and drains associated with the pumps and the regenerative and non-regenerative heat exchangers are routed to the reactor building equipment drain system through open drains which are vented to secondary containment atmosphere. The pumps and heat exchanger vents are used to vent the equipment when filling the system. The drains are used to empty the components prior to maintenance.

The temperature of the water will be low enough during these draining and venting operations that the possibility of airborne contamination will be minimal. Therefore, the routing of these lines to an open drain connection is acceptable.

Difference 2

Evaluation of compliance with the technical specifications for water chemistry parameter limits is not provided.

Discussion Reactor water purity will be maintained by the system to yield effluent water in accordance with the requirements of Regulatory Guide 1.56 (FSAR Section 5.4.8.1.2) and the Technical Specifications for Water Chemistry within limits described in Technical Specifications.

ATTACHMENT 1.9-41

STANDARD REVIEW PLAN 6.1.2, REVISION 2
PROTECTIVE COATING SYSTEMS
(PAINTS) - ORGANIC MATERIALS

Difference

For a small fraction of the exposed surfaces in the drywell, the recommendation of Regulatory Guide 1.54 is not met.

Discussion See FSAR Sections 6.1.2.1 and 6.1.2.2. Protective coatings are generally not used in the suppression pool. The majority of the exposed surfaces within the drywell (i.e., primary containment lines, drywell head, biological shield wall, structural steel, cranes, pipe rupture restraints, pipe supports, piping, and concrete) are coated with materials qualified in accordance with ANSI N101.2 and applied in accordance with Regulatory Guide 1.54. The balance of the exposed surfaces within the drywell (i.e., valve bodies, hand wheels, electrical and control panels, loudspeakers, and emergency light cases), constituting a small fraction of the total exposed surfaces, do not satisfy Regulatory Guide 1.54 conditions. 22

ATTACHMENT 1.9-42

STANDARD REVIEW PLAN 6.2.1.1.c, REVISION 4, JULY 1981
APPENDIX I TO STANDARD REVIEW PLAN 6.2.1.1.c,
REVISION 1, JULY 1981
PRESSURE-SUPPRESSION TYPE BWR CONTAINMENTS

<u>Difference 1</u>

Peak calculated temperature for the wetwell airspace exceeds the design temperature of the suppression pool.

<u>Discussion</u> Peak calculated containment pressure and deck
differential pressure are within design limits. Drywell
calculated environment temperature is below its design
value. However, following the steam bypass transient, the
atmospheric temperature in the suppression chamber is
greater than 212°F (super heated). For a small break LOCA
with steam bypass, the temperature is determined to be
approximately 250°F. Any Category 1 equipment in the
suppression chamber will be qualified to the maximum
envelope value of 270°F, which has been specified in 5
environmental qualification documents. However, the
structure temperature, i.e., steel liner, remains below the
saturation temperature of the suppression chamber atmosphere
for the duration of the transient. Since the liner
temperature is below 212°F, the design temperature of the
suppression chamber structure is not exceeded.

<u>Difference 2</u>

Suppression chamber spray is not autoactuated following a LOCA.

<u>Discussion</u> One of the SRP requirements concerns the
automatic suppression chamber spray limiting containment
pressure to 45 psig considering steam bypass. Analysis for
Unit 2 shows that containment spray is not necessary for the
first 20 min following a LOCA. Considering no operator
action for the initial 10 min and 5- to 10-min operator 5
action to switch on containment spray mode, manual spray is
justified. This will eliminate the potential for
inadvertent spray due to the malfunction of an automatic
control.

<u>Difference 3</u>

A redundant position indicator for each vacuum relief valve and an alarm for vacuum breaker valves are not provided.

ATTACHMENT 1.9-42 (Cont)

Discussion Each vacuum breaker flow path has two relief valves mounted horizontally in series to ensure a leaktight boundary. Three flow paths are required for the vacuum breaker design basis; however, four flow paths (eight valves) are provided. Each vacuum relief valve is provided with three position-sensing devices mounted 120 degrees apart around the circumference of each disc. One of the position-sensing devices is mounted at the bottom of the disc. These devices are designed such that all three positions must be within 0.05 inches of the full closed 26
position before a closed signal can be initiated. Total detectable opening for the vacuum breakers is ≤ 0.044 ft^2 vs 26
the allowable bypass leakage capacity of 0.05 ft^2, thus providing adequate sensitivity. Although redundant position indication does not exist on each vacuum relief valve, redundancy is achieved due to redundant valves in each flow path. Indication in the control room is achieved by red/green lights.

Difference 4

Visual inspection at each refueling outage for vacuum relief valves and piping is not described.

Discussion 4 This is addressed in the technical specification. 26

Difference 5

Vacuum breaker operability test at monthly intervals is not described.

Discussion 5 This is addressed in the technical specification. 26

ATTACHMENT 1.9-42 (Cont)

Discussion Each vacuum breaker flow path has two relief valves mounted horizontally in series to ensure a leaktight boundary. Three flow paths are required for the vacuum breaker design basis; however, four flow paths (eight valves) are provided. Each vacuum relief valve is provided with three position-sensing devices mounted 120 degrees apart around the circumference of each disc. One of the position-sensing devices is mounted at the bottom of the disc. These devices are designed such that all three positions must be within 0.05 inches of the full closed 26
position before a closed signal can be initiated. Total detectable opening for the vacuum breakers is ≤0.044 ft^2 vs 26
the allowable bypass leakage capacity of 0.05 ft^2, thus providing adequate sensitivity. Although redundant position indication does not exist on each vacuum relief valve, redundancy is achieved due to redundant valves in each flow path. Indication in the control room is achieved by red/green lights.

Difference 4

Visual inspection at each refueling outage for vacuum relief valves and piping is not described.

Discussion 4 This is addressed in the technical specification. 26

Difference 5

Vacuum breaker operability test at monthly intervals is not described.

Discussion 5 This is addressed in the technical specification. 26

ATTACHMENT 1.9-43

STANDARD REVIEW PLAN 6.2.1.2, REVISION 2, JULY 1981
SUBCOMPARTMENT ANALYSIS

Difference

The acceptable model for subcompartment initial conditions is to assume air at the maximum allowable temperature, minimum absolute pressure, and zero percent relative humidity. 1

One of the Unit 2 annulus pressurization analyses assumes 20 percent relative humidity instead of zero percent.

Discussion The governing case for the design of the annulus considers zero percent relative humidity; therefore, Unit 2 meets the intent of the acceptance criteria.

ATTACHMENT 1.9-44

STANDARD REVIEW PLAN 6.2.1.3, REVISION 1, JULY 1981
MASS AND ENERGY RELEASE ANALYSIS FOR POSTULATED LOCAs

1

Difference

The ability of the containment and its associated systems, including subcompartments, to withstand calculated pressure and temperature conditions resulting from any LOCA without exceeding design temperature is not discussed in this section.

Discussion See discussion for SRP 6.2.1.1.c (steam bypass temperature of wetwell).

ATTACHMENT 1.9-45

STANDARD REVIEW PLAN 6.2.2, REVISION 3, JULY 1981
CONTAINMENT HEAT REMOVAL SYSTEMS

<u>Difference</u> 1

The spray drop efficiency calculation is not provided.

<u>Discussion</u> An analysis of the spray drop thermal effectiveness was not performed due to the unavailability of drop size test data from the nozzle manufacturer. When the required drop size data become available, the spray thermal effectiveness will be calculated by the method referenced in this SRP.

ATTACHMENT 1.9-46

THIS INFORMATION HAS BEEN DELETED. 5

ATTACHMENT 1.9-47

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ATTACHMENT 1.9-48

STANDARD REVIEW PLAN 6.2.7, REVISION 0
FRACTURE PREVENTION OF CONTAINMENT
PRESSURE BOUNDARY

Difference

Not all of the Class 2 piping and/or components (valves) have had actual impact testing performed.

Ferritic materials of construction for the containment pressure boundary have been toughness tested as follows:

1. All ferritic material of the primary containment liner (e.g., drywell and suppression pool liner plate, equipment and personnel hatches, drywell head, penetration sleeves, etc) requiring notch toughness have been Charpy impact tested and conform to NE-2300 of ASME Section III. This information may be found in FSAR Section 3.8.1, specifically Item 3.8.1.6.2. 1

2. Class 1 ferritic process piping has been impact tested and conforms to NB-2300 of ASME Section III.

Discussion Class 2 ferritic process piping has not been impact tested, except for that portion included in the penetration assembly which penetrates the containment liner. It has been impact tested and conforms to NB-2300 of ASME Section III.

An initial review indicates that similar construction materials have been used on those items which were not subjected to actual impact testing. This indicates that inherent toughness may be substantiated.

ATTACHMENT 1.9-49

STANDARD REVISION PLAN 6.3, REVISION EMERGENCY CORE COOLING SYSTEM

Difference

The requirements of the following Task Action Plans are not addressed in this section:

1. Task Action Plan II.B.8 of NUREG-0718 (Reference 14). 1
2. Task Action Plan III.D.1.1 of NUREG-0694 and NUREG-0718.
3. Task Action Plan II.E.2.1 of NUREG-0737.
4. Task Action Plan II.K.3(10) of NUREG-0737 and NUREG-0718.
5. Task Action Plan II.K.3(15) of NUREG-0737 and NUREG-0718.
6. Task Action Plan II.K.3(18) of NUREG-0737 and NUREG-0718.
7. Task Action Plan II.K.3(21) of NUREG-0737 and NUREG-0718.

Discussion Items 1, 2, 5, 6, and 7 are discussed in FSAR Section 1.10. Items 3 and 4 are not applicable to Unit 2.

ATTACHMENT 1.9-50

STANDARD REVIEW PLAN 6.5.1, REVISION 2, JULY 1981
ENGINEERED SAFETY FEATURES ATMOSPHERE CLEANUP SYSTEMS

1

Difference

Exception is taken with compliance to Regulatory Guide 1.52.

Discussion See FSAR Table 1.8-1.

ATTACHMENT 1.9-51

THE INFORMATION ON THIS PAGE HAS BEEN DELETED.

ATTACHMENT 1.9-52

STANDARD REVIEW PLAN 8.3.1, REVISION 2
AC POWER SYSTEMS (ONSITE)

Difference

The Division III (HPCS) standby diesel generator (GM-EMD) is provided with a standard duty turbocharger mechanical drive gear assembly.

Discussion The Division III standby diesel generator is retrofitted with a heavy duty turbocharger drive gear assembly. 23

ATTACHMENT 1.9-52A

STANDARD REVIEW PLAN 9.1.3, REVISION 1
SPENT FUEL POOL COOLING AND CLEANUP SYSTEM

Difference

The acceptance criteria of Section II.1.d.(4) require computation of decay heat loads based on one refueling load after 150 hr decay plus one refueling load after one year decay. FSAR Section 9.1.3.2 describes the conditions for the spent fuel heat load as one refueling load after 288 hr decay plus additional refuelings decayed in multiples of 18 months after reactor shutdown.

Discussion The decay times used to compute the spent fuel heat loads are consistent with expected operating procedures and refueling cycles for NMP2.

ATTACHMENT 1.9-53

STANDARD REVIEW PLAN 9.1.5, REVISION 0 1
OVERHEAD HEAVY LOAD HANDLING SYSTEM

There is no FSAR Section 9.1.5. All material relating to this subject is in Section 9.1.4.

ATTACHMENT 1.9-54

STANDARD REVIEW PLAN 9.2.2, REVISION 1
REACTOR AUXILIARY COOLING WATER SYSTEM

Difference 1

Task II.K.2.16 of NUREG-0718 and Task II.K.3.25 of NUREG-0737, as they relate to loss of cooling water to reactor coolant pump seals, are not addressed in this section.

Discussion NUREG-0718 is not applicable to Unit 2. It is applicable to applicants for construction permit or manufacturing license only.

Task II.K.3.25 is addressed in FSAR Section 1.10. 18

Difference 3

The ability of the reactor coolant pumps to withstand a complete loss of cooling water for 20 min is not demonstrated by testing.

Discussion An analysis was used to demonstrate that the cooling water systems have been designed such that cooling water will be provided whenever it is needed.

ATTACHMENT 1.9-55

STANDARD REVIEW PLAN 9.2.3, REVISION 2
DEMINERALIZED WATER MAKEUP SYSTEM 1

<u>Difference</u>

Acceptance Criterion II, 1, is not addressed.

<u>Discussion</u> Unit 2 is in compliance with SRP 9.2.3, Acceptance Criterion II, 1, although it is not addressed. All makeup water system piping in the reactor building is seismically analyzed.

ATTACHMENT 1.9-56

STANDARD REVIEW PLAN 9.3.2, REVISION 2
PROCESS AND POST-ACCIDENT SAMPLING SYSTEMS

Difference 1

The post-accident sampling system is not completely addressed in Section 9.3.2.

Discussion Additional information on the post-accident sampling system is provided in Task II.B.3. 18

ATTACHMENT 1.9-57

STANDARD REVIEW PLAN 9.4.1, REVISION 2, JULY 1981
CONTROL ROOM AREA VENTILATION SYSTEM 1

Difference

Unit 2 does not meet the guidance of Regulatory Guides 1.52 and 1.140.

Discussion Unit 2 complies with the intent of Regulatory Guides 1.52 and 1.140 (paragraph c of these guides) through the alternate approaches discussed in FSAR Section 1.8.

ATTACHMENT 1.9-58

STANDARD REVIEW PLAN 9.4.2, REVISION 2, JULY 1981
SPENT FUEL POOL AREA VENTILATION SYSTEM 1

Difference

Unit 2 does not meet the guidance of Regulatory Guides 1.52 and 1.140.

Discussion Unit 2 complies with the intent of Regulatory Guides 1.52 and 1.140 (paragraph c of these guides) through the alternate approaches discussed in FSAR Section 1.8.

ATTACHMENT 1.9-59

STANDARD REVIEW PLAN 9.4.3, REVISION 2, JULY 1981
AUXILIARY AND RADWASTE AREA VENTILATION SYSTEM

Difference 1

Unit 2 does not meet the guidance of Regulatory Guide 1.140.

Discussion Unit 2 complies with the intent of Regulatory Guide 1.140 (paragraph c) through the alternate approach discussed in FSAR Section 1.8.

ATTACHMENT 1.9-60

STANDARD REVIEW PLAN 9.4.5, REVISION 2, JULY 1981
ENGINEERED SAFETY FEATURE VENTILATION SYSTEM

1

Difference

Unit 2 does not meet the guidance of Regulatory Guide 1.52.

Discussion Unit 2 complies with the intent of Regulatory Guide 1.52 (paragraph c) through the alternate approach discussed in FSAR Section 1.8.

ATTACHMENT 1.9-61

STANDARD REVIEW PLAN 9.5.1, REVISION 3, JULY 1981
FIRE PROTECTION PROGRAM (FIRE PROTECTION SYSTEM)

Deviations to BTP CMEB 9.5-1
Attached to Standard Review Plan 9.5.1
Fire Protection Program

Difference 1

Section C.1.c.(3) states that "the fire suppression system should be capable of delivering water to manual hose stations located within hose reach of areas containing equipment required for safe shutdown following the safe shutdown earthquake (SSE)."

Discussion Unit 2 standpipe and hose connection design is in accordance with Appendix A (dated August 1976) to BTP 9.5-1 (dated May 1, 1976) and Appendix R to 10CFR50, and is not seismically qualified.

The design does not contemplate simultaneous earthquake and fire conditions; therefore, this requirement was not incorporated into the design. Further, justification is that Unit 2 is not in an area of high seismic activity.

Difference 2

Section C.5.a(3)(b) of Unit 2 design incorporates fire boot-type penetration seals (approximately 200 of 11,000 fire rated seals) for which temperature levels on the unexposed side reached 393°F during the acceptance test.

Discussion Fixed combustibles potentially within close proximity have ignition temperatures of >500°F. Cables are generally installed in raceways (i.e., conduit or cable trays).

Difference 3

Section C.5.a(5) - Unit 2 fire doors are administratively supervised to verify that they are in the closed position.

Discussion Fire doors are maintained in the closed position.

ATTACHMENT 1.9-61 (Cont)

The doors are administratively verified to be in the closed position on a daily basis. Additionally, fire doors in areas protected by automatic total flooding CO_2-systems are provided with heavy-duty door closures. Halon 1301 suppression systems are used in computer rooms and control rooms. Doors to these areas are inherently supervised by the occupants in the area, in addition to the daily inspection, to verify that the doors are in the proper position.

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ATTACHMENT 1.9-61 (Cont)

Difference 4 | 27

Section C.5.a(14) - Unit 2 floor drains are conservatively sized and are in accordance with the National Plumbing Code. They were not sized based on firefighting water flows.

Discussion Unit 2 fixed water suppression systems incorporate the use of closed-heads and closed-water spray nozzles, which inherently limit the amount of water discharged to the area of involvement during a fire. Refer to Section 9A.3.5.1.12 for the results of an evaluation completed to determine the effects of firefighting water flows on floor drains.

Difference 5 | 27

Section C.5.b.(2) - Credit is taken in the Unit 2 reactor building for separation of cables, equipment, and associated circuits of redundant trains of safe shutdown equipment by a horizontal distance of more than 20 ft. Fire detection and automatic suppression systems are provided in the zone. Nonsafe-shutdown-related cable trays traverse the 20-ft zone.

Discussion Fire detection, automatic area suppression, and automatic cable tray suppression systems are provided for the cables in this zone in accordance with Section 9A.3.5.5.3. The cables are IEEE 383 qualified.

Difference 6 | 27

Section C.5.e.(2) - Unit 2 safety-related cable trays are provided with ionization type detectors in lieu of line type and ionization detectors. Unit 2 safety-related cable trays are provided with closed head pre-action sprinkler systems in lieu of open head deluge or open directional spray nozzle systems.

Discussion Safety-related cable trays are provided with ionization-type smoke detectors which provide an earlier warning system than line-type heat detectors. Safety-related cable trays that are not accessible for manual fire-fighting are protected by zoned automatic closed head reaction sprinkler systems. Water spray systems that

ATTACHMENT 1.9-61 (Cont)

incorporate the use of open directional spray nozzles discharge an excessive amount of water in protected areas, requiring substantially larger drainage and processing capabilities than areas protected by sprinkler systems which minimize the potential for damage to safety-related structures and components.

<u>Difference 7</u> 27

Section C.5.g.(1) - Unit 2 emergency lighting capability is provided by means other than individual 8-hr battery supplies.

<u>Discussion</u> Areas which must be manned during safe shutdown will be supplied with 8-hr battery packs for access and egress lighting.

<u>Difference 8</u> 27

Section C.5.g.(3) - The Unit 2 emergency communications system is not independent of the plant communication system.

<u>Discussion</u>

Fixed emergency communications systems independent of normal plant communications systems are not necessary because:

1. The systems are connectible to uninterruptible power sources, which provide reliability during emergency conditions.

2. In case of total loss of power to all communication systems, the Sound Powered Communication (SPC) system can be utilized.

3. The system is set up as described in Section 9.5.2.

4. The system and important components are supervised.

<u>Difference 9</u> 27

Section C.6.a.(3) - The fire detector spacing criteria for Unit 2 meet the intent of NFPA 72E.

<u>Discussion</u> NFPA 72E recommends one detector per bay for beam depth greater than 8 in and bay width <u>greater</u> than

8 ft. NFPA 72E does not address beam depth greater than 8 in and bay width less than 8 ft. In this situation, the Unit 2 design incorporates one detector for every other bay mounted on the bottom flange of structural steel.

Difference 10

Section C.6.c.(4) - Unit 2 design does not incorporate a cross connection to the service water system for firefighting capability post-SSE.

Discussion Standpipes and hose connections for manual fire fighting are seismically supported in safety-related areas and in areas containing safety-related equipment. The design bases do not contemplate simultaneous earthquake and fire conditions; therefore, this requirement was not incorporated into the design. Further justification is that Unit 2 is not in an area of high seismic activity.

Difference 11

Section C.7.a.(1), part (c) - During normal operation, the Unit 2 design does not incorporate the use of general area fire detection in the primary containment.

Discussion The Unit 2 containment is inerted during normal operation.

Difference 12

In general, Section C endorses the use of the National Fire Protection Association (NFPA) standards. Unit 2 deviates from a number of these NFPA standards.

Discussion Each Unit 2 deviation to the NFPA standards is described and justified in Table 9.5-3.

Difference 13

Section C.7.b - Unit 2 design incorporates the use of carpet in the control room.

Discussion Carpet exceeds NFPA 101, Class I, interior floor finish requirements and is required to satisfy human factors guidelines.

8 ft. NFPA 72E does not address beam depth greater than 8 in and bay width <u>less</u> than 8 ft. In this situation, the Unit 2 design incorporates one detector for every other bay mounted on the bottom flange of structural steel.

<u>Difference 9</u>

Section C.6.c.(4) - Unit 2 design does not incorporate a cross connection to the service water system for firefighting capability post-SSE.

<u>Discussion</u> Standpipes and hose connections for manual fire fighting are seismically supported in safety-related areas and in areas containing safety-related equipment. The design bases do not contemplate simultaneous earthquake and fire conditions; therefore, this requirement was not incorporated into the design. Further justification is that Unit 2 is not in an area of high seismic activity.

<u>Difference 10</u>

Section C.7.a.(1), part (c) - During normal operation, the Unit 2 design does not incorporate the use of general area fire detection in the primary containment.

<u>Discussion</u> The Unit 2 containment is inerted during normal operation.

<u>Difference 11</u>

In general, Section C endorses the use of the National Fire Protection Association (NFPA) standards. Unit 2 deviates from a number of these NFPA standards. 26

<u>Discussion</u> Each Unit 2 deviation to the NFPA standards is described and justified in Table 9.5-3.

ATTACHMENT 1.9-62

THE INFORMATION ON THIS PAGE HAS BEEN DELETED.

ATTACHMENT 1.9-63

STANDARD REVIEW PLAN 10.3, REVISION 2
MAIN STEAM SUPPLY SYSTEM

1

Difference

Acceptance Criterion II, 2, is not addressed in Section 10.3 with respect to internally or externally generated missiles.

Discussion Unit 2 complies with this criterion as discussed in FSAR Section 3.5.1.

ATTACHMENT 1.9-64

STANDARD REVIEW PLAN 10.4.2, REVISION 2
MAIN CONDENSER EVACUATION SYSTEM

Difference

Regulatory Guides 1.33 and 1.123 are not addressed in this section.

Discussion Regulatory Guides 1.33 and 1.123 are discussed in Section 1.8.

ATTACHMENT 1.9-65

STANDARD REVIEW PLAN 10.4.3, REVISION 2
TURBINE GLAND SEALING SYSTEM 1

<u>Difference</u>

Regulatory Guides 1.33 and 1.123 are not addressed in this section.

<u>Discussion</u> Regulatory Guides 1.33 and 1.123 are discussed in Section 1.8.

ATTACHMENT 1.9-66

STANDARD REVIEW PLAN 11.2, REVISION 2
LIQUID WASTE MANAGEMENT

Difference

The Unit 2 position on Regulatory Guide 1.143 is not addressed in this section. 1

Discussion The Unit 2 project complies with Regulatory Guide 1.143 through the alternate approach discussed in Section 1.8.

ATTACHMENT 1.9-67

STANDARD REVIEW PLAN 11.3, REVISION 2
GASEOUS WASTE MANAGEMENT SYSTEMS

1

Difference

The Unit 2 position on Regulatory Guide 1.143 is not addressed in this section.

Discussion The Unit 2 project complies with Regulatory Guide 1.143 through the alternate approach discussed in Section 1.8.

ATTACHMENT 1.9-68

STANDARD REVIEW PLAN 11.3, REVISION 0
BRANCH TECHNICAL POSITION ETSB 11-5
POSTULATED RADIOACTIVE RELEASES DUE TO A WASTE GAS SYSTEM LEAK OR FAILURE

Difference

A comparison of the main parameters of the waste gas system event analysis, as presented in this SRP and those actually used in FSAR Section 15.7.1, is provided below.

Parameter	NUREG-0800 BTP ETSB 11-5	FSAR Section 15.7.1	
Accident/event	Bypass of charcoal delay units, release of undelayed offgas activities	Failure of charcoal delay beds, release of total bed activity	
Source term	7 x normal operation source term 7x50,000 uCi/s = 350,000 uCi/s	100 uCi/s/MWt (100x3,489MWt) = 350,000 uCi/s	1
Source term decay time	30 min	30 min	
Isotopes considered	Xe, Kr, Ar	Xe, Kr	
Holdup time on charcoal beds	Not applicable	Xe - 178 days Kr - 278 hr	
Release point	Ground level	Ground level	
Duration of release	2 hr	2 hr	
Value of X/Q	5% overall site short term	.5% maximum sector short term	
Duration of exposure	2 hr	2 hr	
Dose calculations	Semi-infinite cloud	Semi-infinite cloud	

ATTACHMENT 1.9-68 (Cont)

Exposure limit	<0.5 Rem total body	<5 Rem whole body (calculated .12 Rem); <30 Rem Beta (calculated .11 Rem)

1 Discussion The analysis of the failure of the offgas system, provided in Section 15.7.1, is more conservative than the analysis proposed in this SRP, in terms of duration, X/Q, and transit time. Therefore, the existing analysis envelops that proposed by BTP ETSB 11-5.

ATTACHMENT 1.9-69

THE INFORMATION ON THIS PAGE HAS BEEN DELETED.

ATTACHMENT 1.9-69

STANDARD REVIEW PLAN 11.4
BRANCH TECHNICAL POSITION ETSB 11-3
SOLID WASTE MANAGEMENT SYSTEMS

<u>Difference</u>

The bases for storage time are not provided. 4

<u>Discussion</u> The basis for onsite storage time will be described in a future amendment.

ATTACHMENT 1.9-70

STANDARD REVIEW PLAN 12.2, REVISION 2
RADIATION SOURCES

Difference 1 | 25

Shielding and ventilation design fission product source terms were not developed using these bases:

1. An offgas rate of 100,000 uCi/sec after 30 min delay for BWRs.

2. 0.25 percent fuel cladding defects for PWRs.

Discussion The general basis for the shielding design is stated in Section 12.2.1.1. Sections 12.2.1.2 through 12.2.1.5 provide source data that were used in shielding designs. Sources of airborne radiation to be considered in ventilation design are discussed in Section 12.2.2. Criterion (1) is discussed in Section 11.1, and criterion (2) does not apply.

ATTACHMENT 1.9-71

STANDARD REVIEW PLANS 12.3 AND 12.4, REVISION 2
RADIATION PROTECTION DESIGN FEATURES

Difference 1

The following items required by NUREG-0800, Section II.1, are not presented in the FSAR.

1. Access control to spent fuel transfer canal should be more stringent than that required by 10CFR20.203.

2. All accessible portions of the spent fuel transfer canal that are capable of having radiation levels greater than 100 rads/hr shall be shielded during fuel transfer.

3. Removable shielding may be used (for Item b) but must be explicitly marked. Local audible and visible alarming radiation monitors must be installed to alert personnel if the temporary shielding is removed during fuel transfer operations.

4. All accessible portions of the spent fuel transfer tube shall be clearly marked with a sign stating that potentially lethal radiation fields are possible during fuel transfer. 1

5. Similar precautions to those described in Items a through d shall also apply to any other radiation source having radiation levels higher than 100 Rem/hr.

Discussion

1. Because of the procedures and shield design described below, access control in accordance with 10CFR20 is considered to be adequate.

2. A portable shield or access control will be used to limit dose rates in areas of the drywell accessible during fuel transfer to <20 mRem/hr.

3. Refueling procedures will either mandate the placement of the radiation shield or implement access controls before fuel transfer operations. Portable monitors will be used to alarm audibly and

ATTACHMENT 1.9-71 (Cont)

visibly in the drywell if the portable shield is not installed or is removed during fuel transfer.

4. Not applicable to Unit 2 design.

5. Precautions similar to those described above may also be taken for other radiation sources having radiation levels in excess of 100 Rem/hr.

1

Difference 2

Area radiation monitors are required by NUREG-0800, paragraph II.4.A.3, to remain on-scale when measuring dose rates during accidents and anticipated operational occurrences. A description of vital area monitoring has not been provided in Section 12.3.4.

Discussion Post-accident vital area monitors meet the criterion of NUREG-0800, paragraph II.4.A.3, and will be addressed in an amendment to FSAR Section 12.3.

ATTACHMENT 1.9-72

STANDARD REVIEW PLAN 12.5, REVISION 2
OPERATIONAL RADIATION PROTECTION PROGRAM

Difference 1

No audible alarm dosimeters, personnel air samplers, or count rate meters are provided.

Discussion Audible alarm dosimeters available to the industry have reliability deficiencies. Therefore, no commitment to use audible alarm dosimeters is to be made at this time.

Personnel air samplers are not provided at Unit 2 due to the extensive radiation and contamination survey procedures provided (Section 12.5.3.1.), which are considered sufficient to protect personnel from airborne radioactivity.

The use of count rate meters on protective clothing will provide little, if any, additional radiation protection in view of the extensive personnel monitoring that will be implemented.

ATTACHMENT 1.9-73

STANDARD REVIEW PLAN 13.4, REVISION 2
OPERATIONAL REVIEW

Difference

1

Independent review is not performed by an Independent Safety Engineering Group.

Discussion Independent review is performed by the Safety Review and Audit Board and the Onsite Technical Services Group, as described in Section 1.10 and Chapter 13. The approach given meets the intent of the requirements stated.

ATTACHMENT 1.9-74

STANDARD REVIEW PLAN 14.2, REVISION 2
INITIAL PLANT TEST PROGRAM

Difference 1

The test abstracts contain significant parameters but do not include plant performance characteristics.

Discussion The preoperational test descriptions, which will be available for NRC review at least 60 days before the test is to be run, will include plant performance characteristics.

ATTACHMENT 1.9-75

STANDARD REVIEW PLANS 15.3.3 AND 15.3.4, REVISION 2 REACTOR COOLANT PUMP ROTOR SEIZURE AND REACTOR COOLANT PUMP SHAFT BREAK

<u>Difference</u> 1

Accident analysis of these faulted events does not include the assumption of turbine trip and coincident loss of offsite power and coastdown pumps.

<u>Discussion</u> The consequences of this combination would be less severe than the transient analyzed in FSAR Section 15.2.6. The turbine trip, or indirect loss of offsite power, will initiate scram and cause rapid power reduction. The severity of shaft seizure or shaft break, without loss of offsite power, is evidenced by the fast coastdown of core flow which reduces thermal margin significantly before the L8-initiated scram.

ATTACHMENT 1.9-76

STANDARD REVIEW PLAN 15.4.7, REVISION 1
INADVERTENT LOADING AND OPERATION OF A FUEL ASSEMBLY IN AN IMPROPER POSITION

Difference

Plant operating procedures will not contain provisions to search for fuel loading errors with nuclear instrumentation. 1

Discussion As addressed in FSAR Section 15.4.7.1, the probability of a fuel bundle being misplaced is extremely small.

The Unit 2 approach is to analyze the worst case (misplaced bundle accident) and show compliance with fuel limits. The analysis and results demonstrating compliance with these limits is presented in FSAR Section 15.4.7.3. (See also GESTAR-II, Section S.2.5.4.)

ATTACHMENT 1.9-77

STANDARD REVIEW PLAN 15.6.4, REVISION 2 RADIOLOGICAL CONSEQUENCES OF MAIN STEAM LINE FAILURE OUTSIDE CONTAINMENT (BWR)

Difference

The iodine concentration in the primary coolant is stated in NUREG-0800, paragraph III.2.b, to correspond to the following two cases:

1. The concentration is the maximum value permitted and corresponds to the conditions of an assumed pre-accident spike (meets the 10CFR100 Dose Guidelines).

2. The concentration is the maximum equilibrium value permitted for continued full power operation (meets 10 percent of the 10CFR100 Dose Guidelines). 1

The FSAR presents the results of the main steam line failure analysis performed using only Case 1.

Discussion The main steam line failure analysis performed using the more conservative assumption that the iodine concentration in the primary coolant is the maximum value permitted by the BWR standard technical specifications, results in doses that are less than 10 percent of the limits of 10CFR100.

Therefore, FSAR Section 15.6.4 is considered to meet or exceed the requirements of NUREG-0800, Section 15.6.4, without performing the other analysis.

ATTACHMENT 1.9-78

STANDARD REVIEW PLAN 15.6.5, REVISION 2
LOSS-OF-COOLANT ACCIDENTS RESULTING FROM SPECTRUM OF POSTULATED PIPING BREAKS WITHIN THE REACTOR COOLANT PRESSURE BOUNDARY

Difference

The TMI Task Action Plan requirements for II.E.2.3, II.K.3.25, II.K.3.30, and II.K.3.31 have not been addressed.

Discussion See FSAR Section 1.10 for Tasks II.K.3.25, II.K.3.30, and II.K.3.31. Resolution of Task II.E.2.3 is not addressed in the FSAR but has been generically approved by the NRC. 27

ATTACHMENT 1.9-79

STANDARD REVIEW PLAN 15.8, REVISION 1
ANTICIPATED TRANSIENTS WITHOUT SCRAM

Difference 1

GDC 10, 15, 26, 27, and 29 are not applied for the anticipated transient without scram (ATWS) event.

Discussion The postulated ATWS event is so remote that it is outside the range of design basis accidents to which 1
these GDC apply. The RCPB pressure design has sufficient margin to meet GDC 15.

Difference 2

NUREG-0460, Volume 2, Section IV-4, criterion j, is not applicable to the recirculation pump trip (RPT) design.

Discussion The NRC reviewed RPT design features during 1978 and 1979 and, after the publication of Volume 2 of NUREG-0460, determined a set of design criteria to determine RPT acceptablity. These criteria are essentially the same as criteria a through i of NUREG-0460, Volume 2, Section IV-4. The NRC has deemed the Monticello and Hatch RPT designs as being acceptable since they meet these criteria as noted in SRP 15.8.

ATTACHMENT 1.9-80

SRP DEVIATION WRITEUPS
CHAPTER 16 - TECHNICAL SPECIFICATIONS

The information contained in Chapter 16 was finalized in July 1987 when the full power license was issued. The results of an analysis to determine conformance to the Standard Review Plan will be provided in a future update.

ATTACHMENT 1.9-81

STANDARD REVIEW PLAN 17.1, REVISION 2
QUALITY ASSURANCE DURING THE DESIGN AND CONSTRUCTION PHASES

This SRP is not applicable to Unit 2. A review of Section 17.1 shows that the program is in conformance to this SRP for the operations phase QA program, as defined in FSAR Appendix B (QA Topical Report).

ATTACHMENT 1.9-82

STANDARD REVIEW PLAN 17.2, REVISION 2
QUALITY ASSURANCE DURING THE OPERATIONS PHASE

QA during the operations phase is discussed in FSAR Appendix B (QA Topical Report). There are no differences noted.

ATTACHMENT 1.9-83

STANDARD REVIEW PLAN 18.0, REVISION 0
HUMAN FACTORS ENGINEERING 1

SRP acceptance criteria for this section are still being developed. An analysis will be performed when the acceptance criteria are finalized.

TABLE 1.9-1

STANDARD REVIEW PLAN CONFORMANCE TO ACCEPTANCE CRITERIA

SRP Number	Title	Revision	Conformance	Difference	
CHAPTER 1: INTRODUCTION AND GENERAL DESCRIPTION OF PLANT					
1.8	Interfaces for Standard Design	1	NA	NA	
CHAPTER 2: SITE CHARACTERISTICS					
2.1.1	Site Location and Description	2	X		
2.1.2	Exclusion Area Authority and Control	2	X		
2.1.3	Population Distribution	2	X		7
2.2.1-2.2.2	Identification of Potential Hazards in Site Vicinity	2	X		
2.2.3	Evaluation of Potential Accidents	2	X		
2.3.1	Regional Climatology	2	X		
2.3.2	Local Meteorology	2		Attachment 1.9-2	
2.3.3	Onsite Meteorological Measurements Programs	2		Attachment 1.9-3	
2.3.4	Short-Term Diffusion Estimates for Accidental Atmospheric Releases	1	X		
2.3.5	Long-Term Diffusion Estimates	2	X		
2.4.1	Hydrologic Description	2	X		
	Appendix A	2	X		
2.4.2	Floods	2	X		
2.4.3	Probable Maximum Flood (PMF) on Streams and Rivers	2	NA	NA	
2.4.4	Potential Dam Failures	2	NA	NA	
2.4.5	Probable Maximum Surge and Seiche Flooding	2	X		
2.4.6	Probable Maximum Tsunami Flooding	2	NA	NA	
2.4.7	Ice Effects	2	X		
2.4.8	Cooling Water Canals and Reservoirs	2	NA	NA	
2.4.9	Channel Diversions	2	NA	NA	
2.4.10	Flood Protection Requirements	2	X		
2.4.11	Cooling Water Supply	2	X		
2.4.12	Groundwater	2		Attachment 1.9-4	
	BTP HMB/GSB 1	1	NA	NA	
	BTP HGEB 1	2	NA	NA	
2.4.13	Accidental Releases of Liquid Effluents in Ground and Surface Waters	2		Attachment 1.9-5	
2.4.14	Technical Specifications and Emergency Operation Requirements	2	NA	NA	
2.5.1	Basic Geologic and Seismic Information	2	X		
2.5.2	Vibratory Ground Motion	1	X		
2.5.3	Surface Faulting	2	X		

TABLE 1.9-1 (Cont'd)

SRP Number	Title	Revision	Conformance	Difference
2.5.4	Stability of Subsurface Materials and Foundations	2		Attachment 1.9-6
2.5.5	Stability of Slopes	2		Attachment 1.9-7
CHAPTER 3:	DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS			
3.2.1	Seismic Classification	1		Attachment 1.9-8
3.2.2	System Quality Group Classification	1		Attachment 1.9-9
3.3.1	Wind Loadings	2		Attachment 1.9-10
3.3.2	Tornado Loadings	2	X	
3.4.1	Flood Protection	2	X	
3.4.2	Analysis Procedures	2	X	
3.5.1.1	Internally Generated Missiles (Outside Containment)	2	X	
3.5.1.2	Internally Generated Missiles (Inside Containment)	2	X	
3.5.1.3	Turbine Missiles	2		Attachment 1.9-11
3.5.1.4	Missiles Generated by Natural Phenomena	2	X	
	BTP AAB 3-2	1	NA	NA
	BTP ASB 3-2	2	NA	NA
3.5.1.5	Site Proximity Missiles (Except Aircraft)	1	X	
3.5.1.6	Aircraft Hazards	1	X	
3.5.2	Structures, Systems, and Components to be Protected from Externally Generated Missiles	2	X	
3.5.3	Barrier Design Procedures	1	X	Attachment 1.9-12
3.6.1	Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	1	X	
	BTP ASB 3-1	1		Attachment 1.9-13
3.6.2A	Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping	1		Attachment 1.9-14
	BTP MEB 3-1	2		Attachment 1.9-14
3.6.2B	Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping	1		Attachment 1.9-15
	BTP MEB 3-1	2		Attachment 1.9-15
3.7.1A	Seismic Design Parameters	1	X	
3.7.1B	Seismic Design Parameters	1	X	
3.7.2A	Seismic System Analysis	1		Attachment 1.9-16
3.7.2B	Seismic System Analysis	1	X	
3.7.3A	Seismic Subsystem Analysis	1	X	
3.7.3B	Seismic Subsystem Analysis	1		Attachment 1.9-17
3.7.4	Seismic Instrumentation	1		Attachment 1.9-18

TABLE 1.9-1 (Cont)

SRP Number	Title	Revision	Conformance	Difference	
3.8.1	Concrete Containment			Attachment 1.9-19	
3.8.2	Steel Containment	1	NA	NA	
3.8.3	Concrete and Steel Internal Structures of Steel or Concrete Containments	1		Attachment 1.9-20	
3.8.4	Other Seismic Category I Structures	1		Attachment 1.9-21	
3.8.5	Foundations	1		Attachment 1.9-22	
3.9.1A	Special Topics for Mechanical Components	2	X		
3.9.1B	Special Topics for Mechanical Components	2	X		
3.9.2A	Dynamic Testing and Analysis of Systems, Components, and Equipment	2		Attachment 1.9-23	
3.9.2B	Dynamic Testing and Analysis of Systems, Components, and Equipment	2		Attachment 1.9-24	
3.9.3A	ASME Code Class 1, 2 and 3 Components, Component Supports, and Core Support Structures	1	X		
3.9.3B	ASME Code Class 1, 2 and 3 Components, Component Supports, and Core Support Structures	1		Attachment 1.9-25	1
3.9.4B	Control Rod Drive Systems	1	X		
3.9.5B	Reactor Pressure Vessel Internals	2		Attachment 1.9-26	
3.9.6	Inservice Testing of Pumps and Valves	2		Attachment 1.9-27	
3.10A	Seismic Qualification of Category I Instrumentation and Electrical Equipment	2		Attachment 1.9-28	
3.10B	Seismic Qualification of Category I Instrumentation and Electrical Equipment	2	X		
3.11	Environmental Design of Mechanical and Electrical Equipment	2		Attachment 1.9-29	
CHAPTER 4: REACTOR					
4.2	Fuel System Design	2		Attachment 1.9-30	
4.3	Nuclear Design	2	X		
	BTP CPB 4.3-1	2	NA	NA	
4.4	Thermal and Hydraulic Design	1		Attachment 1.9-31	
	Appendix	1	NA	NA	
4.5.1	Control Rod Drive Structural Materials	2		Attachment 1.9-32	
4.5.2	Reactor Internals and Core Support Materials	2		Attachment 1.9-33	
4.6	Functional Design of Control Rod Drive System	1	X		

TABLE 1.9-1 (Cont)

SRP Number	Title	Revision	Conformance	Difference
CHAPTER 5: REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS				
5.2.1.1	Compliance with the Codes and Standard Rule 10CFR50.55a	2		Attachment 1.9-34
5.2.1.2	Applicable Codes Cases	2	X	
5.2.2	Overpressurization Protection	1		Attachment 1.9-36
	BTP RSB 5-2	0	NA	NA
5.2.3	Reactor Coolant Pressure Boundary Materials	2		Attachment 1.9-37
	BTP MTEB 5-7	2	NA	NA
5.2.4	Reactor Coolant Pressure Boundary Inservice Inspection and Testing	1	X	
5.2.5	Reactor Coolant Pressure Boundary Leakage Detection	1	X	
5.3.1	Reactor Vessel Materials	1		Attachment 1.9-38
5.3.2	Pressure-Temperature Limits	1	X	
	BTP MTEB 5-2	1	X	
5.3.3	Reactor Vessel Integrity	1	X	
5.4	Preface	1	NA	NA
5.4.1.1	Pump Flywheel Integrity (PWR)	1	NA	NA
5.4.2.1	Steam Generator Materials	2	NA	NA
	BTP MTEB 5-3	2	NA	NA
5.4.2.2	Steam Generator Tube Inservice Inspection	1	NA	NA
5.4.6	Reactor Core Isolation Cooling System (BWR)	2		Attachment 1.9-39
5.4.7	Residual Heat Removal (RHR) System	2	X	
	BTP RSB 5-1	2	X	
5.4.8	Reactor Water Cleanup System (BWR)	2		Attachment 1.9-40
5.4.11	Pressurizer Relief Tank	2	NA	NA
5.4.12	Reactor Coolant System High Point Vents	0	X	
CHAPTER 6: ENGINEERED SAFETY FEATURES				
6.1.1	Engineered Safety Features Materials	2	X	
	BTP MTEB 6-1	2	NA	NA
6.1.2	Protective Coating Systems (Paints) - Organic Materials	2		Attachment 1.9-41
6.2.1	Containment Functional Design	2	NA	NA
6.2.1.1A	PWR Dry Containments, Including Subatmospheric Containments	2	NA	NA
6.2.1.1B	Ice Condenser Containments	2	NA	NA
6.2.1.1c	Pressure Suppression Type BWR Containments	4		Attachment 1.9-42
	Appendix I	1		Attachment 1.9-42
6.2.1.2	Subcompartment Analysis	2		Attachment 1.9-43
6.2.1.3	Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents	1		Attachment 1.9-44

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TABLE 1.9-1 (Cont)

SRP Number	Title	Revision	Conformance	Difference
6.2.1.4	Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures	1	NA	NA
6.2.1.5	Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies	2	NA	NA
6.2.2	Containment Heat Removal Systems	3		Attachment 1.9-45
6.2.3	Secondary Containment Functional Design	2	X	
6.2.4	Containment Isolation System	2	X	
6.2.5	Combustible Gas Control in Containment	2	X	
	Appendix A	2	NA	NA
	BTP CSB 6-2	2	NA	NA
6.2.6	Containment Leakage Testing	2	X	
6.2.7	Fracture Prevention of Containment Pressure Boundary	0		Attachment 1.9-48
6.3	Emergency Core Cooling System			Attachment 1.9-49
	BTP RSB 6-1	1	NA	NA
6.4	Control Room Habitability System	2	X	
	Appendix A	2	X	
6.5.1	Engineered Safety Feature Atmosphere Cleanup System	2		Attachment 1.9-50
6.5.2	Containment Spray as a Fission Product Cleanup System	1	NA	NA
6.5.3	Fission Product Control Systems and Structures	2	X	
6.5.4	Ice Condenser as a Fission Product Cleanup System	2	NA	NA
6.6	Inservice Inspection of Class 2 and 3 Components	1	X	
6.7	Main Steam Isolation Valve Leakage Control System (BWR)	2	NA	NA
CHAPTER 7: INSTRUMENTATION AND CONTROLS				
7.1	Instrumentation and Controls - Introduction	2	X	
	Table 7-1 - Acceptance Criteria and Guidelines for Instrumentation and Controls Systems Important to Safety	2	X	
7.2	Reactor Trip System	2	X	
	Appendix A	2	NA	NA
7.3	Engineered Safety Features System	2	X	
	Appendix A	2	NA	NA
7.4	Safe Shutdown Systems	2	X	
7.5	Information Systems Important to Safety	2	X	

TABLE 1.9-1 (Cont)

SRP Number	Title	Revision	Conformance	Difference
7.6	Interlock Systems Important to Safety	2	X	
7.7	Control Systems	2	X	
Appendix 7-A	Branch Technical Positions (ICSB)	2		
	BTP ICSB 1 (DOR)	2	NA	NA
	BTP ICSB 3	2	X	
	BTP ICSB 4 (PSB)	2	NA	NA
	BTP ICSB 5	2	NA	NA
	BTP ICSB 9	2	NA	NA
	BTP ICSB 12	2	NA	
	BTP ICSB 13	2	NA	NA
	BTP ICSB 14	2	NA	NA
	BTP ICSB 16	2	NA	NA
	BTP ICSB 19	2	NA	NA
	BTP ICSB 20	2	X	
	BTP ICSB 21	2	X	
	BTP ICSB 22	2	X	
	BTP ICSB 25	2	NA	NA
	BTP ICSB 26	2	X	
Appendix 7-B	General Agenda, Station Site Visits	1	NA	NA

CHAPTER 8: ELECTRIC POWER

SRP Number	Title	Revision	Conformance	Difference	
8.1	Electric Power - Introduction	2	NA	NA	
	Table 8-1 - Acceptance Criteria and Guidelines for Electric Power Systems	2	X		23
8.2	Offsite Power System	2	X		
8.3.1	AC Power Systems (Onsite)	2		Attachment 1.9-52	
8.3.2	DC Power Systems (Onsite)	2	X		
Appendix 8-A	Branch Technical Positions (PSB)	2			
	BTP ICSB 2 (PSB)	2	NA	NA	
	BTP ICSB 4 (PSB)	2	NA	NA	
	BTP ICSB 8 (PSB)	2	X		
	BTP ICSB 11 (PSB)	2	X		
	BTP ICSB 15 (PSB)	2	NA	NA	
	BTP ICSB 17 (PSB)	2	NA	NA	
	BTP ICSB 18 (PSB)	2	X		
	BTP ICSB 21 (PSB)	2	X		
	BTP PSB 1	0	X		
	BTP PSB 2	0	X		
Appendix 8-B	General Agenda, Station Site Visits	0	NA	NA	

CHAPTER 9: AUXILIARY SYSTEMS

SRP Number	Title	Revision	Conformance	Difference
9.1.1	New Fuel Storage	2	X	

TABLE 1.9-1 (Cont)

SRP Number	Title	Revision	Conformance	Difference
9.1.2	Spent Fuel Storage	3	X	
9.1.3	Spent Fuel Pool Cooling and Cleanup System	1		Attachment 1.9-52A
9.1.4	Light Load Handling System (Related to Refueling)	2	X	
	BTP ASB 9-1	2	NA	NA
9.1.5	Overhead Heavy Load Handling Systems	0		Attachment 1.9-53
9.2.1	Station Service Water System	2	X	
9.2.2	Reactor Auxiliary Cooling Water Systems	1		Attachment 1.9-54
9.2.3	Demineralized Water Makeup Systems	2		Attachment 1.9-55
9.2.4	Potable and Sanitary Water System	2	X	
9.2.5	Ultimate Heat Sink	2	X	
	BTP ASB 9-2	2	X	
9.2.6	Condensate Storage Facilities	2	NA	NA
9.3.1	Compressed Air System	1	NA	NA
9.3.2	Process and Post-Accident Sampling System	2		Attachment 1.9-56
9.3.3	Equipment and Floor Drainage System	2	X	
9.3.4	Chemical and Volume Control System (PWR) (Including Boron Recovery System)	2	NA	NA
9.3.5	Standby Liquid Control System (BWR)	2	X	
9.4.1	Control Room Area Ventilation System	2		Attachment 1.9-57
9.4.2	Spent Fuel Pool Area Ventilation System	2		Attachment 1.9-58
9.4.3	Auxiliary and Radwaste Area Ventilation System	2		Attachment 1.9-59
9.4.4	Turbine Area Ventilation System	2	X	
9.4.5	Engineered Safety Feature Ventilation System	2		Attachment 1.9-60
9.5.1	Fire Protection Program	3		Attachment 1.9-61
	BTP CMEB 9.5.1	2		Attachment 1.9-61
9.5.2	Communications System	2	X	
9.5.3	Lighting Systems	2	X	
9.5.4	Emergency Diesel Engine Fuel Oil Storage and Transfer System	2	X	
9.5.5	Emergency Diesel Engine Cooling Water System	2	X	
9.5.6	Emergency Diesel Engine Starting System	2		
9.5.7	Emergency Diesel Engine Lubrication System	2	X	
9.5.8	Emergency Diesel Engine Combustion Air Intake and Exhaust System	2	X	
CHAPTER 10: STEAM AND POWER CONVERSION SYSTEM				
10.2	Turbine Generator	2	X	
10.2.3	Turbine Disk Integrity	1	X	
10.3	Main Steam Supply System	2		Attachment 1.9-63
10.3.6	Steam and Feedwater System Materials	2	X	
10.4.1	Main Condensers	2	X	
10.4.2	Main Condenser Evacuation System	2		Attachment 1.9-64

TABLE 1.9-1 (Cont)

SRP Number	Title	Revision	Conformance	Difference
10.4.3	Turbine Gland Sealing System	2		Attachment 1.9-65
10.4.4	Turbine Bypass System	2	X	
10.4.5	Circulating Water System	2	X	
10.4.6	Condensate Cleanup System	2	X	
10.4.7	Condensate and Feedwater System	2	X	
	BTP ASB 10-2	2	NA	NA
10.4.8	Steam Generator Blowdown System (PWR)	2	NA	NA
10.4.9	Auxiliary Feedwater System (PWR)	2	NA	NA
	BTP ASB 10-1	2	NA	NA
CHAPTER 11: RADIOACTIVE WASTE MANAGEMENT				
11.1	Source Terms	2	X	
11.2	Liquid Waste Management Systems	2		Attachment 1.9-66
11.3	Gaseous Waste Management Systems	2		Attachment 1.9-67
	BTP ETSB 11-5	0		Attachment 1.9-68
11.4	Solid Waste Management Systems	2	X	
	BTP ETSB 11-3	2	X	
	Appendix 11.4-A	0	NA	NA
11.5	Process and Effluent Radiological Monitoring and Sampling Systems	3	X	
	Appendix 11.5-A	1	X	
CHAPTER 12: RADIATION PROTECTION				
12.1	Assuring that Occupational Radiation Exposures are as Low as Is Reasonably Achievable	2	X	
12.2	Radiation Sources	2		Attachment 1.9-70
12.3-12.4	Radiation Protection Design Features	2		Attachment 1.9-71
12.5	Operational Radiation Protection Program	2		Attachment 1.9-72
CHAPTER 13: CONDUCT OF OPERATIONS				
13.1.1	Management and Technical Support Organization	2	X	
13.1.2-13.1.3	Operating Organization	2	X	
13.2.1	Reactor Operating Training	0	X	
13.2.2	Training for Non-Licensed Plant Staff	0	X	
13.3	Emergency Planning	2	X	
13.4	Operational Review	2		Attachment 1.9-73
13.5.1	Administration Procedures	0	X	
13.5.2	Operating and Maintenance Procedures	0	X	
13.6	Physical Security	2	X	

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TABLE 1.9-1 (Cont)

SRP Number	Title	Revision	Conformance	Difference
CHAPTER 14: INITIAL TEST PROGRAM				
14.1	Initial Plant Test Programs - PSAR	2	NA	NA
14.2	Initial Plant Test Programs - FSAR	2		Attachment 1.9-74
14.3	Standard Plant Designs Initial Test Program Final Design Approval (FDA)	1	NA	NA
CHAPTER 15: ACCIDENT ANALYSIS				
15.0	Introduction	2	NA	NA
15.1 Increase in Heat Removal by the Secondary System				
15.1.1-15.1.4	Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve	1	X	
15.1.5	Steam System Piping Failures Inside and Outside of Containment (PWR)	2	NA	NA
	Appendix A	2	NA	NA
15.2 Decrease in Heat Removal by the Secondary System				
15.2.1-15.2.5	Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, Closure of Main Steam Isolation Valve (BWR), and Steam Pressure Regulator Failure (Closed)	1	X	
15.2.6	Loss of Non-Emergency AC Power to the Station Auxiliaries	1	X	
15.2.7	Loss of Normal Feedwater Flow	1	X	
15.2.8	Feedwater System Pipe Breaks Inside and Outside Containment (PWR)	1	NA	NA
15.3 Decrease in Reactor Coolant System Flow Rate				
15.3.1-15.3.2	Loss of Forced Reactor Coolant Flow Including Trip of Pump and Flow Controller Malfunctions	1	X	
15.3.3-15.3.4	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break	2		Attachment 1.9-75

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TABLE 1.9-1 (Cont)

SRP Number	Title	Revision	Conformance	Difference
15.4 Reactivity and Power Distribution Anomalies				
15.4.1	Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition	2	X	
15.4.2	Uncontrolled Control Rod Assembly Withdrawal at Power	2	X	
15.4.3	Control Rod Misoperation (System Malfunction or Operator Error)	2	NA	NA
15.4.4-15.4.5	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate	1	X	
15.4.6	Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant (PWR)	1	NA	NA
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	1		Attachment 1.9-76
15.4.8	Spectrum of Rod Ejection Accidents (PWR)	2	NA	NA
	Appendix A	1	NA	NA
15.4.9	Spectrum of Rod Drop Accidents (BWR)	2	X	
	Appendix A	2	X	
15.5 Increase in Reactor Coolant Inventory				
15.5.1-15.5.2	Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory	1	X	
15.6 Decrease in Reactor Coolant Inventory				
15.6.1.	Inadvertent Opening of a PWR Pressurizer Relief Valve or a BWR Relief Valve	1	X	
15.6.2	Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment	2	X	
15.6.3	Radiological Consequences of Steam Generator Tube Failure (PWR)	2	NA	NA
15.6.4	Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)	2		Attachment 1.9-77

TABLE 1.9-1 (Cont)

SRP Number	Title	Revision	Conformance	Difference
15.6.5	Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	2		Attachment 1.9-78
15.7 Radioactive Release from a Subsystem or Component				
15.7.1	Waste Gas System Failure	1	NA	NA
15.7.2	Radioactive Liquid Waste System Leak or Failure (Release to Atmosphere)	1	NA	NA
15.7.3	Postulated Radioactive Releases Due to Liquid-Containing Tank Failures	2	X	
15.7.4	Radiological Consequences of Fuel Handling Accidents	1	X	
15.7.5	Spent Fuel Cask Drop Accidents	2	X	
15.8 Anticipated Transients Without Scram				1
15.8	Anticipated Transients Without Scram	1		Attachment 1.9-79
CHAPTER 16: TECHNICAL SPECIFICATIONS				
16.0	Technical Specifications	1		Attachment 1.9-80
CHAPTER 17: QUALITY ASSURANCE				
17.1	Quality Assurance During the Design and Construction Phases	2		Attachment 1.9-81
17.2	Quality Assurance During the Operations Phase	2		Attachment 1.9-82
CHAPTER 18: HUMAN FACTCRS ENGINEERING				
18.0	Human Factors Engineering/Standard Review Plan Development	0		Attachment 1.9-83

KEY: NA = Not applicable

(1) SRP section has been combined with SRP Section 12.3.
(2) SRP section has been combined with SRP Section 13.1.2.

1.10 UNIT 2 RESPONSE TO REGULATORY ISSUES RESULTING FROM TMI

The information contained in this section provides the Unit 2 project position in response to the requirements for Operating License Applicants identified in NUREG-0737. Table 1.10-1 provides a listing of the tasks from NUREG-0737 for which a project position has been developed.

TABLE 1.10-1

NUREG-0737 TMI-2 ITEMS

	Section Number	Title	Page Number
	I.A.1.1	Shift Technical Advisor	1.10-5
	I.A.1.2	Shift Supervisor Responsibilities	1.10-7
	I.A.1.3	Shift Manning	1.10-11
	I.A.2.1	Immediate Upgrade of Reactor Operator and Senior Reactor Operator Training and Qualifications	1.10-14
	I.A.2.3	Administration of Training Programs	1.10-16
	I.A.3.1	Revise Scope and Criteria for Licensing Examinations - Simulator Exams	1.10-17
9	I.B.1.2	Independent Safety Engineering Group	1.10-18
	I.C.1	Short-Term Accident and Procedure Review	1.10-20
	I.C.2	Shift and Relief Turnover Procedures	1.10-25
9	I.C.3	Shift Supervisor Responsibility	1.10-27
	I.C.4	Control Room Access	1.10-30
9	I.C.5	Procedures for Feedback of Operating Experience to Plant Staff	1.10-31
	I.C.6	Guidance on Procedures for Verifying Correct Performance of Operating Procedures	1.10-34
	I.C.7	NSSS Vendor Review of Procedures	1.10-36

TABLE 1.10-1 (Cont)

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TABLE 1.10-1 (Cont)

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TABLE 1.10-1 (Cont)

NINE MILE POINT UNIT 2 RESPONSE TO TMI REQUIREMENTS

I.A.1.1 SHIFT TECHNICAL ADVISOR

FSAR Cross Reference

Sections 13.1, 13.2.2

NUREG-0737 Position

Each licensee shall provide an on-shift Technical Advisor to the Shift Supervisor. The Shift Technical Advisor (STA) may serve more than one unit at a multiunit site if qualified to perform the advisor function for the various units.

The STA shall have a bachelor's degree or equivalent in a scientific or engineering discipline and have received specific training in the response and analysis of the plant for transients and accidents. The STA shall also receive training in plant design and layout, including the capabilities of instrumentation and controls in the control room. The licensee shall assign normal duties to the STAs that pertain to the engineering aspects of assuring safe operations of the plant, including the review and evaluation of operating experience.

The need for the STA position may be eliminated if the qualifications of the Shift Supervisors and Senior Operators have been upgraded and the man-machine interface in the control room has been acceptably upgraded. However, until long-term improvements are attained, the need for an STA program will continue.

The NRC staff will establish the detailed elements of the academic and training requirements of the STA at a later date. The level of upgrading required for licensed operating personnel and the man-machine interface in the control room acceptable for eliminating the need of an STA will also be determined at a later date. Until these requirements for eliminating the STA position have been established, the staff continues to require that an STA be available for duty on each operating shift when a plant is being operated in Modes 1-3 for a BWR. At other times, an STA is not required to be on duty.

Since the accident at TMI several efforts have been made to establish, for the long term, the minimum level of experience, education, and training for STAs. These efforts include work on the revision to ANS-3.1, work by the

Institute of Nuclear Power Operations (INPO), and internal staff efforts.

INPO has made available a document entitled Nuclear Power Plant Shift Technical Advisor--Recommendations for Position Description, Qualifications, Education and Training. A copy of Revision 0 of this document, dated April 30, 1980, is attached as a supplement to this task. Sections 5 and 6 of the INPO document describe the education, training, and experience requirements for STAs. The NRC staff finds that the descriptions as set forth in Sections 5 and 6 of Revision 0 to the INPO document are an acceptable approach for the selection and training of personnel to staff the STA positions. The INPO document provides interim guidance for a utility in planning its STA program over the long term.

Applicants for operating licenses shall provide a description of their STA training program and their plans for requalification training on a schedule consistent with the NRC licensing review schedule. This description shall indicate the level of training attained by STAs and demonstrate conformance with the qualification and training requirements.

Applicants for operating licenses shall provide a description of the long-term STA program, including qualification, selection criteria, training plans, and plans, if any, for the eventual phase-out of the STA program on a schedule consistent with the NRC licensing review schedule.

<u>Nine Mile Point Unit 2 Position</u>

The person fulfilling the STA position will meet the qualification requirements of Option 2 (i.e., dedicated STA) of the Policy Statement on Engineering Expertise on Shift described in 50FR43621 and Generic Letter 86-04. However, if a dedicated STA cannot be provided on a shift, then the Assistant Station Shift Supervisor (ASSS) will function in a dual role (ASSS/STA) and assume the duties of the STA when the Emergency Plan is activated during normal operation, startup, and hot shutdown conditions. Training requirements (Section 13.2) include specific training in the response and analysis of the unit for transients and accidents and in unit design and layout, including capabilities of instrumentation and controls in the control room. The responsibilities of the STA are described in Section 13.1.2.

conditions. During this time he shall perform no duties unrelated to assessment or diagnosis.

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I.A.1.2 SHIFT SUPERVISOR RESPONSIBILITIES

FSAR Cross Reference

Sections 13.1, 13.5.1

NUREG-0737 Position[1] 23

The highest level of corporate management of each licensee shall issue and periodically reissue a management directive that emphasizes the primary management responsibility of the Shift Supervisor for safe operation of the plant under all conditions on his shift and that clearly establishes his command duties.

Plant procedures shall be reviewed to assure that the duties, responsibilities, and authority of the Shift Supervisor and Control Room Operators are properly defined to effect the establishment of a definite line of command and clear delineation of the command decision authority of the Shift Supervisor in the control room relative to other plant management personnel. Particular emphasis shall be placed on the following:

1. The responsibility and authority of the Shift Supervisor shall be to maintain the broadest perspective of operational conditions affecting the safety of the plant as a matter of highest priority at all times when on duty in the control room. The idea shall be reinforced that the Shift Supervisor should not become totally involved in any single operation in times of emergency when multiple operations are required in the control room.

2. The Shift Supervisor, until properly relieved, shall remain in the control room at all times during accident situations to direct the activities of Control Room Operators. Persons authorized to relieve the Shift Supervisor shall be specified.

3. If the Shift Supervisor is temporarily absent from the control room during routine operations, a lead control room operator shall be designated to assume the control room command function. These temporary duties, responsibilities, and authorities shall be clearly specified.

Training programs for Shift Supervisors shall emphasize and reinforce the responsibility for safe operation and the

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[1] Text of NUREG position can be found in NUREG-0578. 23

management function of the Shift Supervisor is to provide for assuring safety.

The administrative duties of the Shift Supervisor shall be reviewed by the senior officer of each utility responsible for plant operations. Administrative functions that detract from or are subordinate to the management responsibility for assuring the safe operation of the plant shall be delegated to other operations personnel not on duty in the control room.

The following table provides clarification to the above position.

SHIFT SUPERVISOR RESPONSIBILITY (2.2.1.a)

NUREG-0578 Position (Position No.)	Clarification
Highest level of corporate management (1)	Executive Vice President Nuclear
Periodically reissue (1)	Annual reinforcement of company policy
Management direction (1)	Formal documentation of shift personnel, all plant management, copy to IE region
Properly defined (2.0)	Defined in writing in a plant procedure
Until properly relieved (2.B)	Formal transfer of authority, valid SRO license, recorded in plant log
Temporarily absent (2.C)	Any absence
Control room defined (2.C)	Includes shift supervisor office adjacent to control room
Designated (2.C)	In administrative procedures
Clearly specified (2.C)	Defined in administrative procedures

NUREG-0578 Position (Position No.)	Clarification
SRO training (3)	Specified in ANS 3.1 (Draft) Section 5.2.1.8
Administrative duties (4)	Not affecting plant safety
Administrative duties reviewed (4)*	On same interval as reinforcement; i.e., annual by Executive Vice President Nuclear

Nine Mile Point Unit 2 Position

Prior to fuel loading and annually thereafter, the Vice President Nuclear Generation shall issue a management directive that emphasizes the primary management responsibility of the Station Shift Supervisor (SSS) for safe operation of the plant under all conditions on his shift and clearly establishes his command duties.

Plant procedures are written to ensure that the duties, responsibilities, and authority of the SSS and other licensed control room operators are properly defined to implement the chain of command.

Administrative duties of the SSS have been reviewed, and many administrative functions have been assigned to other personnel not involved with actual operation of the reactor. Administrative duties of the SSS are reviewed annually by the Vice President Nuclear Generation to ensure that such functions do not detract from safe plant operation.

The responsibilities of the SSS are described in Section 13.1.2.

* This requirement shall be met before fuel loading. See NUREG-0578, Section 22.1a, Item 4 and NRC letters of September 27, and November 9, 1979.

Training programs for the SROs reinforce the responsibility for safe operation and the management function of the control room supervisor to ensure safety.

I.A.1.3 SHIFT MANNING

FSAR Cross Reference

Sections 13.1, 13.5.1, 16.6.2 23

NUREG-0737 Position

Licensees of operating plants and applicants for operating licenses shall include in their administrative procedures (required by license conditions) provisions governing required shift staffing and movement of key individuals about the plant. These provisions are required to assure that plant personnel qualified to man the operational shifts are readily available in the event of an abnormal or emergency situation.

These administrative procedures shall also set forth a policy, the objective of which is to operate the plant with the required staff and develop working schedules such that use of overtime is avoided, to the extent practicable, for the plant staff who perform safety-related functions (e.g., Senior Reactor Operators, Reactor Operators, Health Physicists, Auxiliary Operators, I&C technicians, and key maintenance personnel).

IE Circular No. 80-02, Nuclear Power Plant Staff Work Hours, dated February 1, 1980, discusses the concern of overtime work for members of the plant staff who perform safety-related functions (see WNP-2 position).

The staff recognizes that there are diverse opinions on the amount of overtime that would be considered permissible and that there is a lack of hard data on the effects of overtime beyond the generally recognized normal 8-hr working day, the effects of shift rotation, and other factors. The NRC has initiated studies in this area. Until a firmer basis is developed on working hours, the administrative procedures shall include as an interim measure the following guidance, which generally follows that of IE Circular No. 80-02.

In the event that overtime must be used (excluding extended periods of shutdown for refueling, major maintenance, or major plant modifications), the following overtime restrictions should be followed:

1. An individual should not be permitted to work more than 12 hr straight (not including shift turnover time).

2. There should be a break of at least 12 hr (which can include shift turnover time) between all work periods.

3. An individual should not work more than 72 hr in any 7-day period.

4. An individual should not be required to work more than 14 consecutive days without having 2 consecutive days off.

However, recognizing that circumstances may arise requiring deviation from the above restrictions, such deviation shall be authorized by the Plant Manager or his deputy, or higher levels of management in accordance with published procedures and with appropriate documentation of the cause.

If a Reactor Operator or Senior Reactor Operator has been working more than 12 hr during periods of extended shutdown (e.g., at duties away from the control board), such individuals shall not be assigned shift duty in the control room without at least a 12-hr break preceding such an assignment.

The NRC encourages the development of a staffing policy that would permit the Licensed Reactor Operators and Senior Reactor Operators to be periodically assigned to other duties away from the control board during their normal tours of duty.

If a Reactor Operator is required to work in excess of 8 continuous hours, he shall be periodically relieved of primary duties at the control board, so that periods of duty at the board do not exceed about 4 hr at a time. The guidelines on overtime do not apply to the Shift Technical Advisor provided he or she is provided sleeping accommodations and a 10-min availability is assured.

Operating license applicants shall complete these administrative procedures before fuel loading. Development and implementation of the administrative procedures at operating plants will be reviewed by the Office of Inspection and Enforcement beginning 90 days after July 31, 1980.

See Task III.A.1.2 for minimum staffing and augmentation capabilities for emergencies.

Nine Mile Point Unit 2 Position

Shift manning for Unit 2 is consistent with the requirements of the Technical Specifications and Site Administrative Procedures, and meets all minimum requirements for this task.

I.A.2.1 IMMEDIATE UPGRADE OF REACTOR OPERATOR AND SENIOR REACTOR OPERATOR TRAINING AND QUALIFICATIONS

FSAR Cross Reference

Section 13.2.1

NUREG-0737 Position

Effective December 1, 1980, an applicant for a Senior Reactor Operator (SRO) license will be required to have been a licensed operator for 1 yr. Applicants for SRO either come through the operations chain (C operator to B operator to A operator, etc) or are degree-holding Staff Engineers who obtain licenses for backup purposes.

In the past, many individuals who came through the operator ranks were administered SRO examinations without first being operators. This was clearly a poor practice and the letter of March 28, 1980 requires reactor operator experience for SRO applicants.

However, the NRC does not wish to discourage Staff Engineers from becoming licensed SROs. This effort is encouraged because it forces engineers to broaden their knowledge about the plant and its operation.

In addition, in order to attract degree-holding engineers to consider the Shift Supervisor's job as part of their career development, the NRC should provide an alternate path to holding an operator's license for one year.

The track followed by a high school graduate (a nondegreed individual) to become an SRO would be 4 yr as a Control Room Operator, at least 1 of which would be as a Licensed Operator, and participation in an SRO training program that includes 3 months on shift as an extra person.

The track followed by a degree-holding engineer would be, at a minimum, 2 yr of responsible nuclear power plant experience as a Staff Engineer, participation in an SRO training program equivalent to a cold applicant training program, and 3 months on shift as an extra person in training for an SRO position.

Holding these positions assures that individuals who will direct the licensed activities of Licensed Operators have had the necessary combination of education, training, and actual operating experience prior to assuming a supervisory role at the facility.

The staff realizes that the necessary knowledge and experience can be gained in a variety of ways. Consequently, credit for equivalent experience should be given to applicants for SRO licenses.

Applicants for SRO licenses at a facility may obtain their 1 yr operating experience in a licensed capacity (Operator or Senior Operator) at another nuclear power plant. In addition, actual operating experience in a position that is equivalent to a Licensed Operator or Senior Operator at military propulsion reactors will be acceptable on a one-for-one basis. Individual applicants must document this experience in their individual applications in sufficient detail so that the staff can make a finding regarding equivalency.

Applicants for SRO licenses who possess a degree in engineering or applicable sciences are deemed to meet the above requirement, provided they meet the requirements set forth in Sections A.1.a and A.2 in enclosure 1 in the letter from H. R. Denton and all power reactor applicants and licensees, dated March 28, 1980, and have participated in a training program equivalent to that of a cold Senior Operator Applicant.

The NRC has not imposed the 1-yr experience requirement on cold applicants for SRO licenses. Cold applicants are to work on a facility not yet in operation; their training programs are designed to supply the equivalent of the experience not available to them.

<u>Nine Mile Point Unit 2 Position</u>

The Upgrading of Operator Training and Senior Operator Training for Unit 2 is being performed as described in Section 13.2 of the FSAR. This is also in accordance with the Site Administrative Procedures.

I.A.2.3 ADMINISTRATION OF TRAINING PROGRAMS

FSAR Cross Reference

Section 13.2.1

NUREG-0737 Position

Pending accreditation of training institutions, licensees and applicants for operating licenses will assure that training center and facility instructors who teach systems, integrated responses, transient, and simulator courses demonstrate SRO qualifications and are enrolled in appropriate requalification programs.

The above position is a short-term position. In the future, accreditation of training institutions will include review of the procedure for certification of instructors. The certification of instructors may or may not include successful completion of a Senior Operator examination.

The purpose of the examination is to provide the NRC with reasonable assurance during the interim period that instructors are technically competent. The requirement is directed to permanent members of the training staff who teach the subjects enumerated above, including members of other organizations who routinely conduct training at the facility. There is no intention to require guest lecturers who are experts in particular subjects (reactor theory, instrumentation, thermodynamics, health physics, chemistry, etc.) to successfully complete a Senior Operator examination. Nor do we intend to require a system expert, such as the Supervisor-Instrument and Control Maintenance teaching the rod control drive system to sit for a Senior Operator examination. The use of guest lecturers should be limited.

Nine Mile Point Unit 2 Position

Instructors who teach systems, integrated responses, transient, and simulator courses to operators hold SRO licenses or certifications and participate in operator requalification programs, as outlined in Nuclear Training Procedures. Vendor instructors hold SRO certifications from their respective companies and participate in the operator requalification program.

The qualification of the training instructors meets the requirements of this task, as described in Section 13.2 of the FSAR.

I.A.3.1 REVISE SCOPE AND CRITERIA FOR LICENSING EXAMINATIONS - SIMULATOR EXAMS

FSAR Cross Reference

Section 13.2.1

NUREG-0737 Position

Simulator examinations will be included as part of the licensing examinations. The administration of simulator examinations will be deferred for applicants whose facilities do not have simulators onsite as of October 1, 1980. These deferred simulator examinations will be initiated by October 1, 1981.

The clarification provides additional preparation time for utility companies and the NRC to meet examination requirements as stated. A study is under way to consider how similar a nonidentical simulator should be for a valid examination. In addition, present simulators are fully booked months in advance.

Application of this requirement was stated on June 1, 1980 to applicants where a simulator is located at the facility. Starting October 1, 1981, simulator examinations will be conducted for applicants of facilities that do not have simulators at the site.

NRC simulator examinations normally require 2 to 3 hr. Normally, two applicants are examined during this time period by two examiners.

Utility companies should make the necessary arrangements with an appropriate simulator training center to provide time for these examinations. Preferably these examinations should be scheduled consecutively with the balance of the examination. However, they may be scheduled no sooner than 2 weeks prior to and not later than 2 weeks after the balance of the examination.

Nine Mile Point Unit 2 Position

All new licensing examinations will utilize a control room simulator. The simulator for Unit 2 has been ordered, and it is expected to be operational in January 1985.

I.B.1.2 INDEPENDENT SAFETY ENGINEERING GROUP

FSAR Cross Reference

Sections 13.4 and the Technical Specifications

NUREG-0737 Position

Each applicant for an operating license shall establish an onsite independent safety engineering group (ISEG) to perform independent reviews of plant operations.

The principal function of the ISEG is to examine plant operating characteristics, NRC issuances, Licensing Information Service advisories, and other appropriate sources of plant design and operating experience information that may indicate areas for improving plant safety. The ISEG is to perform independent review and audits of plant activities, including maintenance, modifications, operational problems, and operational analysis and to aid in the establishment of programmatic requirements for plant activities. Where useful improvements can be achieved, it is expected that this group will develop and present detailed recommendations to corporate management for such things as revised procedures or equipment modifications.

Another function of the ISEG is to maintain surveillance of plant operations and maintenance activities to provide independent verification that these activities are performed correctly and that human errors are reduced as far as practicable. ISEG will then be in a position to advise utility management on the overall quality and safety of operations. ISEG need not perform detailed audits of plant operations and shall not be responsible for signoff functions such that it becomes involved in the operating organization.

The new ISEG shall not replace the plant operations review committee (PORC) and the utility's independent review and audit group as specified by current staff quidelines (Standard Review Plan, Regulatory Guide 1.33, Standard Technical Specifications). Rather, it is an additional independent group of a minimum of five dedicated, full-time engineers, located onsite but reporting offsite to a corporate official who holds a high level, technically-oriented position that is not in the management chain for power production. The ISEG will increase the available technical expertise located onsite and will provide continuing, systematic, and independent assessment of plant activities. Integrating the Shift Technical

Advisors (STAs) into the ISEG in some way would be desirable in that it could enhance the group's contact with the knowledge of day-to-day plant operations to provide additional expertise. However, the STA on shift is necessarily a member of the operating staff and cannot be independent of it.

It is expected that the ISEG may interface with the quality assurance (QA) organization, but preferably should not be an integral part of the QA organization.

The functions of the ISEG require daily contact with the operating personnel and continued access to plant facilities and records. The ISEG review functions can therefore best be carried out by a group physically located onsite. However, for utilities with multiple sites, it may be possible to perform portions of the independent safety assessment function in a centralized location for all the utility's plants. In such cases, an onsite group still is required, but it may be slightly smaller than would be the case if it were performing the entire independent safety assessment function. Such cases will be reviewed on a case-by-case basis.

At this time, the requirement for establishing an ISEG is being applied only to applicants for operating licenses in accordance with Task I.B.1.2. The staff intends to review this activity in about a year to determine its effectiveness and to see whether changes are required. Applicability to operating plants will be considered in implementing long-term improvements in organization and management for operating plants (Task I.B.1.1).

Nine Mile Point Unit 2 Position

An onsite independent safety engineering group (ISEG) will be established to perform independent reviews of plant operation. The principal function of the ISEG is to examine plant operating characteristics and the various NRC and industry licensing and service advisories, and to recommend areas for improving plant operations or safety. The ISEG also performs investigative functions as requested and performs independent review of plant activities, including maintenance, modifications, operational concerns and analysis and make recommendations to the Vice President Nuclear Engineering.

The Vice President Nuclear Engineering (or his designee) is the Chairman of the Safety Review and Audit Board (SRAB). The Unit 1 and 2 Plant Managers are members of the SRAB. The Plant Managers in turn chair the Station Operations Review Committee (SORC). This ensures that the flow of internal and external operating experience is communicated to SRAB and SORC while maintaining the independent status of ISEG. ISEG chairs the SRAB subcommittee which evaluates the effectiveness of dispositioned operating experiences identified by the DER process.

The ISEG will observe plant operations and maintenance activities to determine that these activities are being performed properly and provide written recommendations (when useful improvements can be achieved). The ISEG does not perform detailed (QA-type) audits and is not responsible for signoff functions associated with daily operational activities. The ISEG is independent of the SORC and SRAB, but may make recommendations to these groups.

The ISEG shall be composed of at least five dedicated, full-time engineers located onsite, assigned to Unit 2, who report to the Vice President Nuclear Engineering. Each shall have a bachelor's degree in engineering or related science and at least 2 years professional level experience in his field, at least 1 year of which experience shall be in the nuclear field. The Vice President Nuclear Engineering is responsible for technical support and is independent from plant operations.

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I.C.1 SHORT-TERM ACCIDENT AND PROCEDURE REVIEW

FSAR Cross Reference

Sections 13.5.2, 14.2

NUREG-0737 Position

In letters of September 13 and 27, October 10 and 30, and November 9, 1979, the Office of Nuclear Reactor Regulation required licensees of operating plants, applicants for operating licenses, and licensees of plants under construction to perform analyses of transients and accidents, prepare emergency procedure guidelines, upgrade emergency procedures, including procedures for operating with natural circulation conditions, and to conduct operator retraining (see also Task I.A.2.1). Emergency procedures are required to be consistent with the actions necessary to cope with the transients and accidents analyzed. Analyses of transients and accidents were to be completed in early 1980 and implementation of procedures and retraining were to be completed 3 months after emergency procedure guidelines were established; however, some difficulty in completing these requirements has been experienced. Clarification of the scope of the task and appropriate schedule revisions are being developed. In the course of review of these matters on Babcock and Wilcox (B&W)-designed plants, the staff will follow up on the bulletin and orders matters relating to analysis methods and results, as listed in NUREG-0660, Appendix C.

Changes to Previous Requirements and Guidance

1. Modification to Clarification:

   a. Addresses Owners Group and vendor submittals.

   b. References to Tasks I.C.8 and I.C.9.

   c. Scope of procedures review is explained.

   d. Establishes configuration control of guidelines for emergency procedures.

2. Modification to Implementation: Deleted reference to NUREG-0578, Recommendation 2.1.9 for Task I.C.1(a)2, Inadequate Core Cooling.

The letters of September 13 and 27, October 10 and 30, and November 9, 1979, required that procedures and operator

training be developed for transients and accidents. The initiating events to be considered were to include the events presented in the FSAR: loss of instrumentation buses, and natural phenomena such as earthquakes, floods, and tornadoes. The purpose of this paper is to clarify the requirements and add additional requirements for the reanalysis of transients and accidents and inadequate core cooling.

Based on staff reviews to date, there appear to be some recurring deficiencies in the guidelines being developed. Specifically, the staff has found a lack of justification for the approach used (i.e., symptom-, event-, or function-oriented) in developing diagnostic guidance for the operator and in procedural development. It has also been found that although the guidelines take implicit credit for operation of many systems or components, they do not address the availability of these systems under expected plant conditions nor do they address corrective or alternative actions that should be performed to mitigate the event should these systems or components fail.

The analyses conducted to date for guideline and procedure development contain insufficient information to assess the extent to which multiple failures are considered. NUREG-0578 concludes that the single-failure criterion was not considered appropriate for guideline development and calls for the consideration of multiple failures and operator errors. Therefore, the analyses that support guideline and procedure development should consider the occurrences of multiple and consequential failures. In general, the sequence of events for the transients and accidents and inadequate core cooling analyzed should postulate multiple failures such that, if the failures were unmitigated, conditions of inadequate core cooling would result.

Examples of multiple failure events include:

1. Multiple tube ruptures in a single steam generator and tube rupture in more than one steam generator.

2. Failure of main and auxiliary feedwater.

3. Failure of high pressure reactor coolant makeup system.

4. Anticipated transient without scram (ATWS) event following a loss of offsite power, stuck-open

relief valve or safety/relief valve, or loss of main feedwater.

5. Operator errors of omission or commission.

The analyses should be carried out far enough into the event to assure that all relevant thermal/hydraulic/neutronic phenomena are identified (e.g., upper head voiding due to rapid cooldown, steam generator stratification). Failures and operator errors during the long-term cooldown period should also be addressed.

The analyses should support development of guidelines that define a logical transition from the emergency procedures into the inadequate core cooling procedure including the use of instrumentation to identify inadequate core cooling conditions. Rationale for this transition should be discussed. Additional information that should be submitted includes:

1. A detailed description of the methodology used to develop the guidelines.

2. Associated control function diagrams, sequence-of-event diagrams, or others, if used.

3. The bases for multiple and consequential failure considerations.

4. Supporting analysis, including a description of any computer codes used.

5. A description of the applicability of any generic results to plant-specific applications.

Owners Group or vendor submittals may be referenced as appropriate to support this reanalysis. If Owners Group or vendor submittals have already been forwarded to the staff for review, a brief description of the submittals and justification of their adequacy to support guideline development is all that is required.

Pending staff approval of the revised analysis and guidelines, the staff will continue the pilot monitoring of emergency procedures described in Task I.C.8 (NUREG-0660). Since the analysis and guidelines submitted by the GE Owners' Group that comply with the requirements stated above have been reviewed and approved for trial implementation on six plants with applications for operating licenses pending,

the interim program for BWRs will consist of trial implementation on these six plants.

Following approval of analysis and guidelines and the pilot monitoring of emergency procedures, the staff will advise all licensees of the adequacy of the guidelines for application to their plants. Consideration will be given to human factors engineering and system operational characteristics, such as information transfer under stress, compatibility with operator training and control room design, the time required for component and system response, clarity of procedural actions, and control room-personnel interactions. When this determination has been made by the staff, a long-term plan for emergency procedure review, as described in Task I.C.9, will be made available. At that time, the reviews currently being conducted on NTOLs under Task I.C.8 will be discontinued, and the review required for applicants for operating licenses will be as described in the long-term plan.

Depending on the information submitted to support development of emergency procedures for each reactor type or vendor, this transition may take place at different times. For example, if the GE guidelines are shown to be effective on the six plants chosen for pilot monitoring, the long-term plan for BWRs may be complete in early 1981. Operating plants and applicants will then have the option of implementing the long-term plan in a manner consistent with their operating schedule, provided they meet the final date required for implementation. This may require a plant that was reviewed for an operating license under Task I.C.8 to revise its emergency procedures again prior to the final implementation date for Task I.C.9.

The extent to which the long-term program will include review and approval of plant-specific procedures for operating plants has not been established. Our objective, however, is to minimize the amount of plant-specific procedure review and approval required. The staff believes this objective can be acceptably accomplished by concentrating the staff review and approval on generic guidelines. A key element in meeting this objective is the use of staff-approved generic guidelines and guideline revisions by licensees to develop procedures. For this approach to be effective, it is imperative that, once the staff has issued approval of a guideline, subsequent revisions of the guideline should not be implemented by licensees until reviewed and approved by the staff. Any changes in plant-specific procedures based on unapproved guidelines could constitute an unreviewed safety issue under

10CFR50.59. Deviations from this approach on a plant-specific basis would be acceptable provided the basis is submitted by the licensee for staff review and approval. In this case, deviations from generic guidelines should not be implemented until staff approval is formally received in writing. Interim implementation of analysis and procedures for small break loss-of-coolant accident and inadequate core cooling should remain on the schedule contained in NUREG-0578, Recommendation 2.1.9.

Nine Mile Point Unit 2 Position

The Unit 2 project has monitored the development of the BWR Owners Group emergency procedures.

NMP2 EOPs are developed using the BWROG EPG (Revision 4). Deviations (additions, deletions or changes) from the generic document are documented as an appendix to the plant-specific technical guideline (PSTG). Based upon discussions between the BWROG and the NRC, NRC approval of deviations from the NRC-approved EPGs (Revision 4) is not required unless deviations are major strategy changes (BWROG letter OG90-791-62). All deviations are reviewed for technical content and approved by NMPC operations and engineering departments.

I.C.2 SHIFT AND RELIEF TURNOVER PROCEDURES

FSAR Cross Reference

Section 13.5.1

NUREG-0737 Position[1] | 23

The licensees shall review and revise as necessary the plant procedure for shift and relief turnover to assure the following:

1. A checklist shall be provided for the oncoming and offgoing Control Room Operators and the oncoming Shift Supervisor to complete and sign. The following items, as a minimum, shall be included in the checklist:

   a. Assurance that critical plant parameters are within allowable limits (parameters and allowable limits shall be listed on the checklist).

   b. Assurance of the availability and proper alignment of all systems essential to the prevention and mitigation of operational transients and accidents by a check of the control console (what to check and criteria for acceptable status shall be included in the checklist).

   c. Identification of systems and components that are in a degraded mode of operation permitted by the Technical Specifications. For such systems and components, the length of time in the degraded mode shall be compared with the Technical Specifications action statement (this shall be recorded as a separate entry on the checklist).

2. Checklists or logs shall be provided for completion by the offgoing and ongoing auxiliary operators and technicians. Such checklists or logs shall include any equipment under maintenance or test that by itself could degrade a system critical to the prevention and mitigation of operational transients and accidents or initiate an operational transient (what to check and criteria for acceptable status shall be included on the checklist).

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[1] Text of NUREG position can be found in NUREG-0578. | 23

3. A system shall be established to evaluate the effectiveness of the shift and relief turnover procedure.

Nine Mile Point Unit 2 Position

The Unit 2 shift relief turnover procedures will include:

1. A checklist for incoming and outgoing control room operators and incoming Shift Supervisor to complete and sign. The following items, as a minimum, shall be included in the checklist:

   a. Assurance that critical plant parameters are within allowable limits. (Parameters and allowable limits shall be listed on the checklist.)

   b. Assurance of the availability and proper alignment of all systems essential to the prevention and mitigation of operational transients and accidents by a check of the control console. (What to check and criteria for acceptable status shall be included in the checklist.)

   c. Identification of systems and components that are in an off-normal or out-of-service mode of operation permitted by the technical specifications. For such systems and components, the length of time in the off-
17 normal or out-of-service mode shall be compared with the technical specifications action statement, if any. (This shall be recorded as a separate entry on the checklist.)

2. Checklists or logs shall be provided for completion by outgoing and incoming operators. Such checklists or logs shall include any equipment under maintenance or test that by itself could degrade a system critical to the prevention and mitigation of operational transient and accidents,
17 initiate operational transients and accidents, or initiate an operational transient. (What to check and criteria for acceptable status shall be included on the checklist.)

3. A system shall be established to evaluate the effectiveness of the shift and relief turnover procedure.

I.C.3 SHIFT SUPERVISOR RESPONSIBILITY

FSAR Cross Reference

Sections 13.1, 13.5.1

NUREG-0737 Position[1] 23

The highest level of corporate management of each licensee shall issue and periodically reissue a management directive that emphasizes the primary management responsibility of the Shift Supervisor for safe operation of the plant under all conditions on his shift and that clearly establishes his command duties.

Plant procedures shall be reviewed to assure that the duties, responsibilities, and authority of the Shift Supervisor and Control Room Operators are properly defined to effect the establishment of a definite line of command and clear delineation of the command decision authority of the Shift Supervisor in the control room relative to other plant management personnel. . Particular emphasis shall be placed on the following:

1. The responsibility and authority of the Shift Supervisor shall be to maintain the broadest perspective of operational conditions affecting the safety of the plant as a matter of highest priority at all times when on duty in the control room. The idea shall be reinforced that the Shift Supervisor should not become totally involved in any single operation in times of emergency when multiple operations are required in the control room.

2. The Shift Supervisor, until properly relieved of duty, shall remain in the control room at all times during accident situations to direct the activities of Control Room Operators. Persons authorized to relieve the Shift Supervisor shall be specified.

3. If the Shift Supervisor is temporarily absent from the control room during routine operations, a lead Control Room Operator shall be designated to assume the control room command function. These temporary duties, responsibilities, and the authority shall be clearly specified.

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[1] Text of NUREG position can be found in NUREG-0578. 23

Training programs for Shift Supervisors shall emphasize and reinforce the responsibility for safe operation and the

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management function the Shift Supervisor is to provide for assuring safety.

The administrative duties of the Shift Supervisor shall be reviewed by the senior officer of each utility responsible for plant operations. Administrative functions that detract from or are subordinate to the management responsibility for assuring the safe operation of the plant shall be delegated to other operations personnel not on duty in the control room.

The following table clarifies this position:

SHIFT SUPERVISOR RESPONSIBILITY (2.2.1.a)

NUREG-0578 Position (Position No.)	Clarification
Highest level of corporate management (1)	Executive Vice President Nuclear
Periodically reissue (1)	Annual reinforcement of company policy
Management direction (1)	Formal documentation of shift personnel, all plant management, copy to IE region
Properly defined (2.0)	Defined in writing in a plant procedure
Until properly relieved (2.8)	Formal transfer of authority, valid SRO license, recorded in plant log
Temporarily absent (2.C)	Any absence
Control room defined (2.C)	Includes shift supervisor office adjacent to the control room
Designated (2.C)	In administrative procedures
Clearly specified	Defined in administrative procedures

NUREG-0578 Position (Position No.)	Clarification
SRO training	Specified in ANS 3.1 (Draft) Section 5.2.1.8
Administrative duties (4)	Not affecting plant safety
Administrative duties reviewed (4)	On same interval as reinforcement: i.e., annual by Executive Vice President Nuclear

Nine Mile Point Unit 2 Position

The response to this task is contained in I.A.1.2.

Refer to Task I.A.1.2 position statement for response to I.C.3.

I.C.4 CONTROL ROOM ACCESS

FSAR Cross Reference

Section 13.5

23 NUREG-0737 Position[1]

The licensee shall make provisions for limiting access to the control room to those individuals responsible for the direct operation of the nuclear power plant (e.g., Operations Supervisor, Shift Supervisor, and Control Room Operators), to technical advisors who may be requested or required to support the operation, and to predesignated NRC personnel. Provisions shall include the following:

1. Develop and implement an administrative procedure that establishes the authority and responsibility of the person in charge of the control room to limit access, and

2. Develop and implement procedures that establish a clear line of authority and responsibility in the control room in the event of an emergency. The line of succession for the person in charge of the control room shall be established and limited to persons possessing a current senior reactor operator's license. The plan shall clearly define the lines of communication and authority for plant management personnel not in direct command of operations, including those who report to stations outside of the control room.

Nine Mile Point Unit 2 Position

Unit 2 will utilize procedures that limit control room access to those individuals responsible for direct operation of the plant, technical advisors requested or required to support operation, and NRC personnel as described below:

1. Procedures establish the authority and responsibility of the person in charge of the control room to limit access.

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23 [1]Text of NUREG position can be found in NUREG-0578.

2. Procedures establish a clear line of authority and responsibility in the control room in the event of an emergency. The line of succession for the person in charge of the control room includes those holding a senior reactor operator's license. The Emergency Plan clearly defines the lines of communication and authority for plant management personnel not in direct command of operations, including those who report to stations outside the control room.

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I.C.5 PROCEDURES FOR FEEDBACK OF OPERATING EXPERIENCE TO PLANT STAFF

FSAR Cross Reference

Section 13.5.1

NUREG-0737 Position

In accordance with Task I.C.5, Procedures for Feedback of Operating Experience to Plant Staff (NUREG-0660), each applicant for an operating license shall prepare procedures to assure that operating information pertinent to plant safety originating both within and outside the utility organization is continually supplied to operators and other personnel and is incorporated into training and retraining programs. These procedures shall:

1. Clearly identify organizational responsibilities for review of operating experience, the feedback of pertinent information to operators and other personnel, and the incorporation of such information into training and retraining programs;

2. Identify the administrative and technical review steps necessary in translating recommendations by the operating experience assessment group into plant actions (e.g., changes to procedures, operating orders);

3. Identify the recipients of various categories of information from operating experience (i.e., supervisory personnel, shift technical advisors, operators, maintenance personnel, health physics technicians) or otherwise provide means through which such information can be readily related to the job functions of the recipients;

4. Provide means to assure that affected personnel become aware of and understand information of sufficient importance that it should not wait for emphasis through routine training and retraining programs;

5. Assure that plant personnel do not routinely receive extraneous and unimportant information on operating experience in such volume that it would obscure priority information or otherwise detract from overall job performance and proficiency;

6. Provide suitable checks to assure that conflicting or contradictory information is not conveyed to operators and other personnel until resolution is reached; and,

7. Provide periodic internal audit to assure that the feedback program functions effectively at all levels.

Each utility shall carry out an operating experience assessment function that will involve utility personnel having collective competence in all areas important to plant safety. In connection with this assessment function, it is important that procedures exist to assure that important information on operating experience originating both within and outside the organization is continually provided to operators and other personnel and that it is incorporated into plant operating procedures and training and retraining programs.

Those involved in the assessment of operating experience will review information from a variety of sources. These include operating information from the licensee's own plant(s), publications such as IE Bulletins, Circulars, and Notices, and pertinent NRC or industrial assessments of operating experience. In some cases, information may be of sufficient importance that it must be dealt with promptly (through instructions, changes to operating and emergency procedures, issuance of special precautions, etc) and must be handled in such a manner to assure that operations management personnel would be directly involved in the process. In many other cases, however, important information will become available which should be brought to the attention of operators and other personnel for their general information to assure continued safe plant operation.

Since the total volume of information handled by the assessment group may be large, it is important that assurance be provided that high priority matters are dealt with promptly and that discrimination is used in the feedback of other information so that personnel are not deluged with unimportant and extraneous information to the detriment of their overall proficiency. It is important, also, that technical reviews be conducted to preclude premature dissemination of conflicting or contradictory information.

Nine Mile Point Unit 2 Position

Unit 2 will utilize administrative and training procedures to implement operating experience feedback to the plant staff. These procedures will:

1. Clearly identify organizational responsibilities for review of operating experience, the feedback of pertinent information to operators and other personnel, and the incorporation of such information in training and requalification training programs (Section 13.2.4.1.1, Item 9).

2. Identify the administrative and technical review steps necessary to translate recommendations that are the result of an operating experience assessment function, which is performed by the processing of deviation event reports (DERs) into plant actions (e.g., changes to procedures and operating orders). Sections 13.4 and 1.10 provide information concerning the operating experience assessment function.

3. Identify the recipients of various categories of operating experience information (i.e., shift or supervisor, personnel) or otherwise provide means through which such information can be readily related to the job functions of the recipients (Section 13.2.4.1.3).

4. Provide means to ensure that affected personnel become aware of and understand information of sufficient importance so that this information should not wait for emphasis through routine training and retraining, standing orders or night orders. (For example, required reading assignments are made on an ongoing basis to address this concern.)

5. Ensure that plant personnel do not routinely receive extraneous information on operating experience in such volume that it could obscure priority information.

6. Provide suitable checks to ensure that correct information is conveyed to operators and other personnel.

7. Provide periodic audits to ensure that the feedback program functions effectively (e.g., training audits).

Operating experience assessment is performed on an ongoing basis by the processing of DERs by the responsible organization which is considered most cognizant over the subject matter of the DER. The individuals involved review information from a variety of sources such as IE Bulletins, IE Information Notices, INPO reports, LERs, and vendor information letters, such as SILs.

The feedback system provides for early notification of significant information to operating personnel and management. The DER evaluation process provides assurance that the information is correct and that unimportant and extraneous information does not impact overall proficiency.

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I.C.6 GUIDANCE ON PROCEDURES FOR VERIFYING CORRECT PERFORMANCE OF OPERATING PROCEDURES

FSAR Cross Reference

23 Section 13.5

NUREG-0737 Position

It is required that licensees' procedures be reviewed and revised, as necessary, to assure that an effective system of verifying the correct performance of operating activities is provided as a means of reducing human errors and improving the quality of normal operations. This will reduce the frequency of occurrence of situations that could result in or contribute to accidents. Such a verification system may include automatic system status monitoring, human verification of operations and maintenance activities independent of the people performing the activity, or both.

Implementation of automatic status monitoring if required will reduce the extent of human verification of operations and maintenance activities but will not eliminate the need for such verification in all instances. The procedures adopted by the licensees may consist of two phases - one before and one after installation of automatic status monitoring equipment, if required, in accordance with Task I.D.3.

Task I.C.6 of the NRC Task Action Plan (NUREG-0660) and Recommendation 5 of NUREG-0585 propose requiring that licensees' procedures be reviewed and revised, as necessary, to assure that an effective system of verifying the correct performance of operating activities is provided. An acceptable program for verification of operating activities is described below.

The American Nuclear Society has prepared a draft revision to ANSI Standard N18.7-1972 (ANS 3.2), Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants. A second proposed revision to Regulatory Guide 1.33, Quality Assurance Program Requirements (Operation), which is to be issued for public comment in the near future, will endorse the latest draft revision to ANS 3.2 subject to the following supplemental provisions:

1. Applicability of the guidance of Section 5.2.6 should be extended to cover surveillance testing in addition to maintenance.

2. In lieu of any designated Senior Reactor Operator (SRO), the authority to release systems and equipment for maintenance or surveillance testing or return-to-service may be delegated to an onshift SRO, provided provisions are made to ensure that the Shift Supervisor is kept fully informed of system status.

3. Except in cases of significant radiation exposure, a second qualified person should verify correct implementation of equipment control measures such as tagging of equipment.

4. Equipment control procedures should include assurance that control room operators are informed of changes in equipment status and the effects of such changes.

5. For the return-to-service of equipment important to safety, a second qualified operator should verify proper systems alignment unless functional testing can be performed without compromising plant safety, and can prove that all equipment, valves, and switches involved in the activity are correctly aligned.

NOTE: A licensed operator possessing knowledge of the systems involved and the relationship of the systems to plant safety would be a "qualified" person. The staff is investigating the level of qualification necessary for other operators to perform these functions.

For plants that have or will have automatic system status monitoring as disussed in Task I.D.3, NUREG-0660, the extent of human verification of operations and maintenance activities will be reduced. However, the need for such verification will not be eliminated in all instances.

Nine Mile Point Unit 2 Position

Unit 2 will utilize procedures and equipment to ensure an effective system of verifying correct performance of operating activities. As part of the overall program, the Unit 2 design incorporates an automatic status system and bypass inoperability system. Unit 2 is committed to Regulatory Guide 1.33 (Section 1.8) and the following: 14

1. ANSI N18.7-1976, Section 5.26, is applied to both maintenance and surveillance testing.

2. The authority to release systems and equipment for maintenance or surveillance testing or return to service may be delegated to either the SSS (SRO) or Chief Shift Operator (SRO), provided the SSS is kept informed.

3. Except in cases of significant radiation exposure, a second qualified person shall verify correct implementation of equipment control measures such as tagging of equipment.

14 4. Equipment control procedures shall include assurance that control room operators are informed of changes in equipment status and the effects of such changes.

5. For the return to service of safety-related equipment, a second qualified operator shall verify proper systems alignment, unless functional testing can be performed without compromising plant safety and can prove that equipment valves and switches involved in the activity are correctly aligned.

Equipment control procedures described in Section 13.5.1.3.3 provide assurance that this guidance is implemented.

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I.C.7 NSSS VENDOR REVIEW OF PROCEDURES

FSAR Cross Reference

23 Sections 13.5.2, 14.2

NUREG-0737 Position[1]

Obtain nuclear steam supply system (NSSS) vendor review of low power testing procedures to further verify their adequacy. This requirement must be met before fuel loading (NUREG-0694).

Nine Mile Point Unit 2 Position

On April 14, 1983, Niagara Mohawk committed in a letter from C. V. Mangan to Darrell G. Eisenhut, to provide a procedures generation package based upon NRC-approved BWR Owners Group Emergency Procedure Guidelines. Additionally the NSSS vendor, GE, and the architect engineer, SWEC, will be reviewing procedures by their participation on the Joint Test Group. This commitment fulfills this requirement.

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23 [1] Text of NUREG position can be found in NUREG-0694.

I.C.8 PILOT MONITORING OF SELECTED EMERGENCY PROCEDURES FOR NTOL APPLICANTS

FSAR Cross Reference

Section 13.5.2

NUREG-0737 Position[1] 23

Correct emergency procedures, as necessary, based on the NRC audit of selected plant emergency operating procedures (e.g., small break LOCA, loss of feedwater, restart of engineered safety features following a loss of ac power, steam line break, or steam generator tube rupture). This action will be completed prior to issuance of a full power license (NUREG-0694).

Nine Mile Point Unit 2 Position

On April 14, 1983, in a letter from C. V. Mangan to Darrel G. Eisenhut, Niagara Mohawk committed to provide a procedure generation package based upon NRC-approved BWR Owners Group Emergency Procedure Guidelines. This commitment fulfills this requirement.

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[1] Text of NUREG position can be found in NUREG-0694. 23

I.D.1 CONTROL ROOM DESIGN REVIEWS

FSAR Cross Reference

Section - 18.1

NUREG-0737 Position

All licensees and applicants for operating licenses will be required to conduct a detailed control room design review to identify and correct design deficiencies. This detailed control room design review is expected to take about a year. Therefore, the Office of Nuclear Reactor Regulation requires that applicants for operating licenses who are unable to complete this review prior to issuance of a license make preliminary assessments of their control rooms to identify significant human factors and instrumentation problems and establish a schedule approved by the NRC for correcting deficiencies. These applicants will be required to complete the more detailed control room reviews on the same schedule as licensees with operating plants (NUREG-0737).

The Office of Nuclear Reactor Regulation is presently developing human engineering guidelines to assist each licensee and applicant in performing a detailed control room review. A draft of the guidelines has been published for public comment as NUREG/CR-1580, Human Engineering Guide to Control Room Evaluation. The Office of Nuclear Reactor Regulation issued the final version of the guidelines as NUREG-0700 in September 1981, after receiving, reviewing, and incorporating substantive public comments from operating reactor licensees, applicants for operating licenses, human factors engineering experts, and other interested parties. The Office of Nuclear Reactor Regulation will evaluate the applicants' preliminary assessments including the performance by the Office of Nuclear Reactor Regulation of onsite review/audit. The Office of Nuclear Reactor Regulation onsite review/audit will be on a schedule consistent with licensing needs and will emphasize the following aspects of the control room:

1. Adequacy of information presented to the operator to reflect plant status for normal operation, anticipated operational occurrences, and accident conditions.

2. Groupings of displays and the layout of panels.

3. Improvements in the safety monitoring and human factors enhancement of controls and control displays.

4. Communications from the control room to points outside the control room, such as the onsite technical support center, remote shutdown panel, offsite telephone lines, and to other areas within the plant for normal and emergency operation.

5. Use of direct rather than derived signals for the presentation of process and safety information to the operator.

6. Operability of the plant from the control room with multiple failures of nonsafety-grade and nonseismic systems.

7. Adequacy of operating procedures and operator training with respect to limitations of instrumentation displays in the control room.

8. Categorization of alarms, with unique definition of safety alarms.

9. Physical location of the shift supervisor's office either adjacent to or within the control room complex.

Prior to the onsite review/audit, the Office of Nuclear Reactor Regulation will require a copy of the applicant's preliminary assessment and additional information which will be used in formulating the details of the onsite review/audit.

Nine Mile Point Unit 2 Position

The Unit 2 project will utilize the guidance provided by the NRC Committee to Review Generic Requirements (CRGR) as stated in SECY 82-111.

NMPC has provided a commitment to follow the guidance provided by Supplement 1 to NUREG-0737 in NMPC Letter No. 6438 dated April 14, 1983, to D. G. Eisenhut, Division of Licensing of the NRC.

NMPC has performed a preliminary control room design review based on the BWR Owners Group program. The survey was structured with a team consisting of representatives from NMPC, other utilities, the NSSS supplier, and a human factors consultant. This group included licensed Senior Reactor Operators.

The review included panel layout and design, instrumentation, hardware, and annunciators. The

preliminary review was set up to identify areas where potential changes could be made in the PGCC shop prior to shipment to the site in early 1983. The final control room design review will be conducted during 1983 or 1984 based on the guidance of NUREG-0700. The following paragraphs provide a description of this review.

Purpose and Scope The purpose of the control room design review described is to 1) review and evaluate the control room workspace, instrumentation, controls, and other equipment from a human factors engineering point of view that takes into account both system demands and operator capabilities; and 2) to identify, assess, and implement control room design modifications that improve control room man-machine interfaces. The scope of the Unit 2 control room design review described covers the human factors engineering aspects of the completed control room.

Objectives The control room design review will accomplish the following objectives:

1. To determine whether the control room provides the system status information, control capabilities, feedback, and analytic aids necessary for control room operators to accomplish their functions effectively.

2. To identify characteristics of existing control room instrumentation, controls, other equipment, and physical arrangements that may detract from operator performance.

3. To analyze and evaluate the problems that could arise from discrepancies of Items 1 and 2, and to analyze means of correcting those discrepancies.

4. To define and put into effect a plan of action that applies human factors principles to improve control room design and enhance operator effectiveness. Particular emphasis will be placed on improvements affecting control room design and operator performance under abnormal or emergency conditions.

5. To integrate the control room design review with other areas of human factors inquiry identified as a result of TMI-related requirements.

Relationship to Other Human Factors Programs The control room design review will be integrated with other TMI issues that are also concerned with or may affect control room human factors. Most of these programs are described in NRC Task Action Plan NUREG-0660 and NUREG-0737. Aspects of the TMI-Related Requirements for New Operating Licenses (NUREG-0694) and Functional Criteria for Emergency Response Facilities (NUREG-0696) are also pertinent.

Control Room Design Review Process The control room design review process addresses four major phases of activity, as illustrated in Exhibit 1A: planning, review, assessment and implementation, and reporting. Methods and procedures, as suggested by NUREG-0700, are used as guidance for accomplishing that portion of the review. Alternatives to that guidance may be used and will be clearly documented.

Planning A formal planning phase will be performed which will take into account the data and information needs of related control room human factors efforts, so that a data base can be developed to meet common needs. Other features of the planning phase will involve management in the overall control room design review process, to ensure that all objectives and tasks are fully understood, to develop a well defined work plan and schedule that takes operational constraints into account, and to ensure that the resources needed to complete the review on schedule will be available. A preliminary report summarizing the formal planning phase will be prepared.

EXHIBIT I.D.1-1

OVERVIEW OF CONTROL ROOM (CR) DESIGN REVIEW PHASES AND PRODUCTS

Analyze CR design review objectives, resource requirements, constraints, related action plan concerns

Planning Phase

Design review work plan and schedule

Develop planning report

Perform review

Review Phase

Human engineering discrepancies

Analyze safety implications and ways of correcting discrepancies

Assessment and Implementation Phase

Work plan and schedule for improving CR human engineering

Initiate CR improvement program

Reporting Phase

Develop report on design review and CR improvement plan

Review The review phase is directed toward identifying and documenting human engineering discrepancies. A system point of view is used so that the assessment of control room characteristics will be tied to their functional applications and operational interrelationships. The term system, as used here, includes personnel as well as hardware; the design review addresses the man-machine system configuration. The processes are defined for the review phase:

1. Review and analysis of system functions and control room operator tasks, to establish the instrumentation and equipment requirements and the performance criteria for the tasks operators are expected to accomplish:

2. Inventory of the control room to identify and describe the performance features of the existing instrumentation and equipment.

3. Survey of the control room in which the instrumentation, controls, other equipment, ambient conditions, and other features are checked against human engineering guidelines.

4. Verification of task performance capabilities, in which the instrument and equipment requirements derived from task analysis are compared to the items presently in the control room inventory.

5. Validation of the control room functions, in which the relationships and dependencies in operating crew activities and between the operators and plant processes are examined in the context of operational sequences.

Assessment and Implementation The processes of the review phase will identify and describe control room design features that may adversely affect operator performance. In the assessment and implementation phase, human engineering discrepancies will be assessed and the process of correcting them (implementation) initiated. Assessment involves determining the safety significance of discrepancies and analyzing them to select design improvements. Discrepancies that have no particular safety significance will also be assessed and analyzed for correction, but on a lower priority basis. Cost-benefit or cost-effectiveness analyses will be a part of the assessment process. Assessment also involves establishing priorities and schedules for corrective action,

determining the extent of corrections, and addressing any recommendations or decisions not to implement modifications.

The assessment process will ensure that the control room design review has been appropriately integrated with those other control room-related projects that are concerned with or may affect human factors. Corrective actions identified in this phase will be reviewed to ensure their consistency with the goals of related projects.

Design improvements that can be executed without interfering with normal control room operation (e.g., changes in surface features such as labeling and location aids) will be implemented in a timely fashion consistent with the project schedule. Other improvements that involve changes to control room equipment or design or that require operator retraining will be scheduled for introduction on a schedule consistent with their significance to plant safety and with operational considerations.

<u>Reporting</u> Reports will be prepared to summarize the review process:

1. <u>Program Planning</u> A preliminary report will be prepared at the conclusion of the planning phase. This report will summarize the planned review process, including methods, staff qualifications, scope of the review effort, and plans for integrating the review.

2. <u>Design Review</u> A final report will be prepared that summarizes the overall review process. Three principal topics will be addressed:

    a. <u>Methodology</u> This section will reference the program planning report and update that material with any changes made during the course of the review.

    b. <u>Review Findings</u> This section will summarize the documentation of the status of the control room with respect to the guidelines for system function review and task analysis, and human engineering guidelines. The section will identify equipment, controls, or displays that may be in the control room and missing features that are needed for improved performance of control room evaluation. Problems related to operational dynamics in crew performance and procedures will be discussed.

Discrepancies with safety consequences will be described, as well as the design and design review processes (including any reallocation of functions) selected for their resolution. Decisions not to take corrective action will be addressed.

c. <u>Implementation</u> This section will provide a summary of control room design improvements already completed and will identify improvements to be completed at a later date. Improvements scheduled for long-term implementation will be described. A suggested schedule will be developed based upon priority and safety significance.

I.D.2 DATA ACQUISITION SYSTEM-PLANT SAFETY PARAMETER DISPLAY CONSOLE

FSAR Cross Reference

Sections 7.5, 7.6, 18.2

NUREG-0737 Position

Plant Safety Parameter Display Console (I.D.2) In accordance with Task I.D.2, Plant Safety Parameter Display Console (NUREG-0660), each applicant and licensee shall install a safety parameter display system (SPDS) that will display to operating personnel a minimum set of parameters that define the safety status of the plant. This can be attained through continuous indication of direct and derived variables as necessary to assess plant safety status (NUREG-0737).

The requirements for the SPDS have been developed in NUREG-0696.

Nine Mile Point Unit 2 Position

Refer to Section 18.2 for the Unit 2 NUREG-0737 position.

NMPC has provided a commitment to follow the guidance provided by Supplement 1 to NUREG-0737 in NMPC Letter No. 6438, dated April 14, 1983, to D. G. Eisenhut, Director, Division of Licensing of the NRC.

I.G.1 TRAINING DURING LOW-POWER TESTING

FSAR Cross Reference

Sections 13.2, 14.2

NUREG-0737 Position[1] 23

The objective is to increase the capability of the shift crews to operate facilities in a safe and competent manner by assuring that training for plant changes and off-normal events is conducted. Near-term operating license facilities will be required to develop and implement intensified training exercises during the low-power testing programs. This may involve the repetition of startup tests on different shifts for training purposes. Based on experiences from the near-term operating license facilities, requirements may be applied to other new facilities or incorporated into the plant drill requirement (Task I.A.2.5). Review comprehensiveness of test programs.

The NRR will require new operating licensees to conduct a set of low-power tests to accomplish the requirement. The set of tests will be determined on a case-by-case basis for the first few plants. Then NRR will develop acceptance criteria for low-power test programs to provide hands-on training for plant evaluation and off-normal events for each operating shift. It is not expected that all tests will be required to be conducted by each operating shift. Observation by one shift of training of another shift may be acceptable.

The NRR will develop criteria in conjunction with initial near-term operating license reviews.

Licensees will 1) define training plan prior to loading fuel, and 2) conduct training prior to full-power operation.

Nine Mile Point Unit 2 Position

The conduct of the initial startup and test program including preoperational testing and startup testing is described in Chapter 14. Personnel training is described in Section 13.2. The training described includes the use of a plant-specific simulator and meets the intent of this TMI action item.

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[1] Text of NUREG position can be found in NUREG-0660. 23

Operator training by exposure to operational and testing evolutions is inherent in the integrated testing program being performed at Unit 2.

Integrated ECCS testing, typical cold and hot functional tests, and startup tests, including loss of power with or without simulated LOCAs, will be possible on the Unit 2 plant-specific simulator.

The simulator training program will provide all shift personnel with experience in evolutions scheduled, not just those in which they participate during the plant testing phases. These evolutions can be repeated to allow various aspects and responses to be emphasized and test procedures to be critiqued. Documentation of operations personnel participation in these training evolutions is required as part of the operator training program.

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II.B.1 REACTOR COOLANT SYSTEM VENTS

FSAR Cross Reference

23 Sections 5.1, 5.2

NUREG-0737 Position

Each applicant and licensee shall install reactor coolant system (RCS) and reactor vessel head high point vents remotely operated from the control room. Although the purpose of the system is to vent noncondensable gases from the RCS which may inhibit core cooling during natural circulation, the vents must not lead to an unacceptable increase in the probability of a loss-of-coolant accident (LOCA) or a challenge to containment integrity. Since these vents form a part of the reactor coolant pressure boundary (RCPB), the design of the vents shall conform to the requirements of Appendix A to 10CFR50, General Design Criteria. The vent system shall be designed with sufficient redundancy to assure a low probability of inadvertent or irreversible actuation.

Each licensee shall provide the following information concerning the design and operation of the high point vent system:

1. Submit a description of the design, location, size, and power supply for the vent system along with results of analyses for LOCAs initiated by a break in the vent pipe. The results of the analyses should demonstrate compliance with the acceptance criteria of 10CFR50.46.

2. Submit procedures and supporting analysis for operator use of the vents that also include the information available to the operator for initiating or terminating vent usage.

The important safety function enhanced by this venting capability is core cooling. For events beyond the present design basis, this venting capability will substantially increase the plant's ability to deal with large quantities of noncondensable gas which could interfere with core cooling.

Procedures addressing the use of the RCS vents should define the conditions under which the vents should be used as well as the conditions under which the vents should not be used. The procedures should be directed toward achieving a substantial increase in the plant's capability to maintain core

cooling without loss of containment integrity for events beyond the design basis. The use of vents for accidents within the normal design basis must not result in a violation of the requirements of 10CFR50.44 or 10CFR50.46.

The size of the reactor coolant vents is not a critical issue. The desired venting capability can be achieved with vents in a fairly broad spectrum of sizes. The criteria for sizing a vent can be developed in several ways. One approach that may be considered is to specify a volume of noncondensable gas to be vented and in a specific venting time. For containments particularly vulnerable to failure from large hydrogen releases over a short period of time, the necessity and desirability for contained venting outside the containment must be considered (e.g., into a decay gas collection and storage system).

Where practical, the RCS vents should be kept smaller than the size corresponding to the definition of LOCA (10CFR50, Appendix A). This will minimize the challenges to the emergency core cooling system (ECCS) since the inadvertent opening of a vent smaller than the LOCA definition would not require ECCS actuation, although it may result in leakage beyond technical specification limits. On PWRs, the use of new or existing lines whose smallest orifice is larger than the LOCA definition will require a valve-in-series valve that can be closed from the control room to terminate the LOCA that would result if an open vent valve could not be reclosed.

A positive indication of valve position should be provided in the control room.

The reactor coolant vent system shall be operable from the control room.

Since the RCS vent will be part of the RCPB, all requirements for the RCPB must be met, and, in addition, sufficient redundancy should be incorporated into the design to minimize the probability of an inadvertent actuation of the system. Administrative procedures may be a viable option to meet the single-failure criterion. For vents larger than the LOCA definition, an analysis is required to demonstrate compliance with 10CFR50.46.

The probability of a vent path failing to close, once opened, should be minimized; this is a new requirement. Each vent must have its power supplied from an emergency bus. A single failure within the power and control aspects of the reactor coolant vent system should not prevent

isolation of the entire vent system when required. On BWRs, block valves are not required in lines with safety valves that are used for venting.

Vent paths from the primary system to within containment should go to those areas that provide good mixing with containment air.

The reactor coolant vent system (i.e., vent valves, block valves, position indication devices, cable terminations, and piping) shall be seismically and environmentally qualified in accordance with IEEE-344-1975 as supplemented by Regulatory Guides 1.100 and 1.92 and SRP 3.9.2, 3.9.3, and 3.10. Environmental qualifications are to be in accordance with the May 23, 1980 Commission Order and memorandum (CLI-80-21).

Provisions to test for operability of the reactor coolant vent system should be part of the design. Testing should be performed in accordance with subsection IWV of Section XI of the ASME Code for Category B valves.

It is important that the displays and controls added to the control room as a result of this requirement not increase the potential for operator error. A human-factor analysis should be performed taking into consideration:

1. The use of this information by an operator during both normal and abnormal plant conditions.

2. Integration into emergency procedures.

3. Integration into operator training.

4. Other alarms during emergency and need for prioritization of alarms.

Specific BWR Design Considerations

Since the BWR Owners Group has suggested that the present BWR designs have an inherent capability to vent, a question relating to the capability of existing systems arises. The ability of these systems to vent the RCS of noncondensable gas generated during an accident must be demonstrated. Because of differences among the head vent systems for BWRs, each licensee or applicant should address the specific design features of this plant and compare them with the generic venting capability proposed by the BWR Owners Group. In addition, the ability of these systems to meet the same requirements as the PWR vent system must be documented.

In addition to RCS venting, each BWR licensee should address the ability to vent other systems, such as the isolation condenser which may be required to maintain adequate core cooling. If the production of a large amount of noncondensable gas would cause the loss of function of such a system, remote venting of that system is required. The qualifications of such a venting system should be the same as that required for PWR venting systems.

Nine Mile Point Unit 2 Position

The Unit 2 design philosophy is in agreement with the BWR Owners Group position on this subject, which is described in detail in NEDO-24782[2].

The Unit 2 design includes 18 main steam safety/relief valves (SRVs), of which 7 are used for automatic depressurization (ADS). Redundant divisional power is supplied to the ADS valves. The discharge lines from the ADS valves (as well as the discharge lines from the 11 non-ADS SRVs) run individually to the suppression pool. The ADS valves and discharge lines satisfy the NUREG-0737 requirements for RCS venting.

In addition to the ADS valves, RCS venting can also take place through the reactor core isolation cooling (RCIC) system which directs steam from one of the main steam lines to a turbine-driven pump; the steam then exhausts from the turbine to the suppression pool. The RCIC system can serve as a vent path during hot standby or during reactor isolation.

The reactor vessel top head vent line can also be used to direct steam and noncondensable gases from the reactor upper dome to the suppression pool. Two Class 1E divisionally powered motor-operated valves (MOVs) are located in series on this line. These MOVs are operated remote manually from the main control room. The reactor vessel top head vent line can be operated over the range of plant operations from cold shutdown through full power operation. This line is used principally to vent the reactor during the final stages of normal shutdown from power operation.

Each RHR heat exchanger is equipped with a shellside (RHR side) vent line, which is routed back to the suppression pool. These lines are each provided with two Class 1E divisionally powered MOVs, located in series, which are 5
operated remote manually from the main control room. These lines may be used at any time, including post LOCA operation, although they are used principally to vent

5 noncondensable gases from the heat exchangers during the steam-condensing mode of operation.

The BWR Owners Group developed the following position with regard to a potential break in a vent line: "The result of a break in the safety-relief valve discharge line, or any of the other systems enumerated above, would be the same as a small steam line break. A complete steam line break is part of the plant's design basis, and smaller-sized breaks have been shown to be of lesser severity...Thus, no new analyses to show conformance with 10CFR50.46 are required." The BWR

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Owners Group also developed the following position regarding operating procedures for ADS and RCIC for vent purposes:

"Under most circumstances, there would be no choice as to where to vent to or when to vent, since the relief valves (as part of the Automatic Depressurization System)...and RCIC [ICS] will function automatically in their designed modes to ensure adequate core cooling, and these will provide continuous venting to the suppression pool. The current assessment is that it would not be desirable to interfere with emergency core cooling functions in order to prevent venting."

II.B.2 PLANT SHEILDING/POST-ACCIDENT ACCESS TO VITAL AREAS

FSAR Cross Reference

Sections 3.11, 12.1, 12.2, 12.3

NUREG-0737 Position

With assumption of a post-accident release of radioactivity equivalent to that described in Regulatory Guides 1.3 and 1.4 (i.e., the equivalent of 50 percent of the core radioiodine, 100 percent of the core noble gas inventory, and 1 percent of the core solids are contained in the primary coolant), each licensee shall perform a radiation and shielding design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review should identify the location of vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during post-accident operations of these systems.

Each licensee shall provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or post-accident procedural controls. The design review shall determine which types of corrective actions are needed for vital areas throughout the facility.

The purpose of this item is to ensure that licensees examine their plants to determine what actions can be taken over the short term to reduce radiation levels and increase the capability of operators to control and mitigate the consequences of an accident. These actions should be taken pending conclusions resulting in the long-term degraded core rulemaking, which may result in a need to consider additional sources.

Any area that will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident is designated as a vital area. For the purposes of this evaluation, vital areas and equipment are not necessarily the same vital areas or equipment defined in 10CFR73.2 for security purposes. The security center is listed as an area to be considered potentially vital, since access to this area may be necessary to gain access to other areas in the plant.

The control room, technical support center (TSC), sampling station, and sample analysis area must be included among those areas where access is considered vital after an accident. (See Task III.A.1.2 for discussion of the TSC and emergency operations facility.) The evaluation to determine the necessary vital areas should also include, but not be limited to, consideration of the post-LOCA hydrogen control system, containment isolation reset control area, manual ECCS alignment area (if any), motor control centers, instrument panels, emergency power supplies, security center, and radwaste control panels. Dose rate determinations need not be noted for these areas if they are not considered vital.

As a minimum, necessary modifications must be sufficient to provide for vital system operation and for occupancy of the control room, TSC, sampling station, and sample analysis area.

To ensure that personnel can perform necessary post-accident operations in the vital areas, the following guidance is to be used by licensees to evaluate the adequacy of radiation protection to the operators:

1. <u>Source Term</u> The minimum radioactive source term should be equivalent to the source terms recommended in Regulatory Guides 1.3, 1.4, and 1.7 and Standard Review Plan 15.6.5 with appropriate decay times based on plant design (i.e., the radioactive decay that occurs before fission products can be transported to various systems can be assumed).

   a. <u>Liquid-Containing Systems</u> 100 percent of the core equilibrium noble gas inventory, 50 percent of the core equilibrium halogen inventory, and 1 percent of all others are assumed to be mixed in the reactor coolant and liquids recirculated by residual heat removal (RHR), high pressure coolant injection (HPCI), and low pressure coolant injection (LPCI), or the equivalent of these systems. In determining the source term for recirculated, depressurized cooling water, it can be assumed that the water contains no noble gases.

   b. <u>Gas-Containing Systems</u> 100 percent of the core equilibrium noble gas inventory and 25 percent of the core equilibrium halogen activity are assumed to be mixed in the containment atmosphere. For vapor-containing lines

connected to the primary system (e.g., BWR steam lines), the concentration of radioactivity shall be determined, assuming the activity is contained in the vapor space in the primary coolant system.

2. Systems Containing the Source Systems assumed in the analysis to contain high levels of radioactivity in a post-accident situation should include, but not be limited to, containment, RHR system, safety injection systems, chemical and volume control system (CVCS), containment spray recirculation system, sample lines, gaseous radwaste systems, and standby gas treatment systems (or equivalent of these systems). If any of these systems or others that could contain high levels of radioactivity were excluded, explain why such systems were excluded. Radiation from leakage of systems located outside of containment need not be considered for this analysis. Leakage measurement and reduction is treated under Task III.D.1.1. Liquid waste systems need not be included in this analysis. Modifications to liquid waste systems will be considered after completion of Task III.D.1.4.

3. Dose Rate Criteria The design dose rate for personnel in a vital area should be such that the guidelines of General Design Criterion 19 will not be exceeded during the course of the accident. General Design Criterion 19 requires that adequate radiation protection be provided so that the dose to personnel would not be in excess of 5 Rem whole body, or its equivalent to any part of the body for the duration of the accident. When determining the dose to an operator, care must be taken to determine the necessary occupancy times in a specific area. For example, areas requiring continuous occupancy will require much lower dose rates than areas where minimal occupancy is required. Therefore, allowable dose rates will be based upon expected occupancy, as well as the radioactive source terms and shielding. However, to provide a general design objective, we are providing the following dose rate criteria with alternatives to be documented on a case-by-case basis. The recommended dose rates are average rates in the area. Local hot spots may exceed the dose rate guidelines. These doses are design objectives and

are not to be used to limit access in the event of an accident.

23 a. Areas Requiring Continuous Occupancy Less than fifteen mRem/hr (averaged over 30 days). These areas will require full-time occupancy during the course of the accident. The control room and onsite TSC are areas where continuous occupancy will be required. The dose rate for these areas is based on the control room occupancy factors contained in Standard Review Plan 6.4.

b. Areas Requiring Infrequent Access These areas may require access on an irregular basis, not continuous occupancy. Shielding should be provided to allow access at a frequency and duration estimated by the licensee. The plant radiochemical/chemical analysis laboratory, radwaste panel, motor control center, instrumentation locations, and reactor coolant and containment gas sample stations are examples of sites where occupancy may be needed often, but not continuously.

4. Radiation Qualification of Safety-Related Equipment The review of safety-related equipment that may be unduly degraded by radiation during post-accident operation of this equipment relates to equipment inside and outside the primary containment. Radiation source terms calculated to determine environmental qualification of safety-related equipment consider the following:

a. LOCA events that completely depressurize the primary system should consider releases of the source term (100 percent noble gases, 50 percent iodines, and 1 percent particulates) to the containment atmosphere.

b. LOCA events in which the primary system may not depressurize should consider the source term (100 percent nobles gases, 50 percent iodines, and 1 percent particulates) to remain in the primary coolant. This method is used to determine qualification doses for equipment in close proximity to recirculating fluid systems inside and outside of containment. Non-LOCA events both inside and outside of containment should use 10 percent noble gases,

10 percent iodines, and 0 percent particulates as a source term.

The following table summarizes these considerations:

Containment	LOCA Source Term (Noble Gas/Iodine/ Particulate)	Non-LOCA High Energy Line Break Source Term (Noble Gas/Iodine/ Particulate)
Outside	% (100/50/1) in RCS	% (10/10/0) in RCS
Inside	Larger of (100/50/1) in containment or (100/50/1) in RCS	(10/10/0) in RCS

Nine Mile Point Unit 2 Position

Analyses have been performed to quantify the post-accident radiation levels throughout the Unit 2 plant, based upon the source terms presented below.

These radiation conditions are being used in conjunction with other environmental conditions (pressure, temperature, and humidity) for the equipment qualification program. Safety-related equipment is being qualified in accordance with NUREG-0588.

A description of the Unit 2 post-accident shield design review is given in Section 12.3.1.3. Areas where access is vital after an accident are analyzed for personnel occupancy 13
to ensure that doses to personnel performing vital post-accident functions are less than GDC-19 limits. This information is provided in Table 12.3-3. A dose rate map for potentially occupied areas is provided on Figure 12.3-69 and corresponding Table 12.3-4.

Source Term

Radioactive source release and distribution assumptions for Unit 2 are as follows.

Radioactive Source Release

1. The percentages of core inventory radioactive fission products assumed to be released from the fuel rods are:

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Noble gases (Kr, Xe)	100%
Halogens	50%
Others	1%
Cesium	50%

2. This entire release is assumed to occur instantaneously at the start of the accident.

<u>Radioactive Source Distribution</u>

To envelop the full spectrum of break sizes and depressurization rates, two bounding events and source distributions were considered.

1. <u>LOCA</u> The following fission products are considered to be uniformly mixed in the following volumes:

   a. <u>Suppression Pool and Reactor Coolant System</u>

Noble gases	0%
Halogens	50%
Others	1%
Cesium	50%

   This distribution assumes short-term reactor depressurization and is consistent with a scenario which leads to gross fission product release from the fuel.

   b. <u>Combined Drywell/Wetwell Air Space</u>

Noble gases	100%
13 Halogens	50%*

Using this distribution, time history radiation
13 zones are established throughout the Unit 2 plant as follows:

a. The above sources will be distributed in the following system piping to establish time history radiation zones for the primary containment and reactor building. These systems were conservatively assumed to operate concurrently.

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13 *The fraction of airborne halogens available for release to the environment is 25 percent of the core inventory in accordance with Regulatory Guide 1.3.

(1) Main steam system (primary only).
(2) RCIC.
(3) RHR. 13
(4) LPCS and HPCS.
(5) SGTS.

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(6) RCS/RWCU (primary only).
(7) Containment atmosphere monitoring.
(8) Hydrogen recombiner system.

b. In addition to radiation shine from system piping and components, the primary containment is assumed to leak at technical specification limits resulting in an airborne source term that is included in the radiation zoning.

c. For areas outside the reactor building, the following sources are considered:

(1) Airborne releases, including bypass leakage, post-accident SGTS effluent, ESF equipment leakage, and secondary containment overpressurization. Pertinent data for these release scenarios are provided in Table 15.6-13.

(2) Direct radiation from secondary containment.

(3) Air-scattered radiation from secondary containment (sky shine).

2. <u>Pipe Break in Reactor Building</u> The temperature and pressure transients resulting as a consequence of this accident are insufficient to cause fuel damage (see Section 6.3). Therefore, the activity available for release from the break is the reactor coolant and steam that are released prior to system isolation. In addition, the iodine concentrations are normalized to the maximum technical specification limit of 4 uCi/gm I-131 dose equivalent to account for spiking. 25

II.B.3 POST-ACCIDENT SAMPLING

FSAR Cross Reference

Section 9.3.2.

NUREG-0737 Position

The Post-Accident Sampling System (PASS) is evaluated for compliance with NUREG-0737, Section II.B.3. These 11 criteria and clarifications have been copied verbatim from NUREG-0737. A general description of the system has been included in Position 1.

Criterion 1

The licensee shall have the capability to promptly obtain reactor coolant samples and containment atmosphere samples. The combined time allotted for sampling and analysis should be 3 hours or less from the time a decision is made to take a sample.

Clarification 1

Provide information on sampling(s) and analytical
laboratories locations including a discussion of relative
elevations, distances and methods for sample transport.
Responses to this item should also include a discussion of
7 sample recirculation, sample handling and analytical times
to demonstrate that the three-hour time limit will be met
(see (6) below relative to radiation exposure). Also
describe provisions for sampling during loss of offsite
power (i.e. designate an alternative backup power source,
not necessarily the vital (Class IE) bus, that can be
energized in sufficient time to meet the 3-hr sampling and
analysis time limit).

Position 1

PASS is designed to obtain representative liquid and gas samples from within the primary containment for radiological analysis in association with the possible consequences of a loss-of-coolant accident (LOCA). The system consists of: 1) a piping station located in the reactor building, el 250 ft; 2) a sampling station located in the radwaste sample room, el 261 ft; 3) two control panels situated approximately 10 ft from the sample station in the sample room; 4) assorted transport equipment; 5) a ventilation system; and 6) assorted interconnecting tubing.

The underlying design philosophy is to minimize exposure by minimizing the required sample sizes, to optimize the weight of shielded sample containers to facilitate movement through potentially high level radiation areas, and to provide adequate shielding at the sample station and in the laboratory. The system is also designed to provide useful samples under all conditions ranging from normal shutdown and power operation to a full LOCA with fission product releases consistent with Regulatory Guide 1.3. A local area radiation monitor is provided to inform the operator of the ambient radiation level. The PASS flow diagram is shown on Figure II.B.3-1a.

Provision has been made to obtain gas samples from both the drywell and wetwell atmospheres and from the secondary containment atmosphere. The sample system is designed to operate over the range of potential pressures starting at 1 hr after a LOCA. Heat-traced sample lines are used to prevent precipitation of moisture and resultant loss of iodine in the sample lines. The gas samples may be passed through a particulate filter and silver zeolite cartridge for determination of particulate activity and total iodine activity by subsequent counting of the samples on a gamma spectroscopy system.

Alternately, the sample flow can bypass the iodine sampler and be chilled to remove moisture. A 15-ml grab sample can then be taken for determination of gaseous activity and for gas composition by gas chromatography. This size sample has been adopted to be consistent with present off-gas sample vial counting factors. Provision will be made in the laboratory to aliquot fractions of the initial vial contents to other vials if the activity is too high to count directly.

A sample line is provided to obtain reactor coolant samples from two points in the jet pump pressure instrument system when the reactor is at pressure. This sample location was chosen over the normal reactor sample points as the reactor clean-up system is expected to be isolated under accident conditions, and it is possible that the recirculation line containing the normal reactor water sample lines may not be representative. The jet pump pressure instrumentation system has been determined to be an optimum sample point for accident conditions. The pressure taps are well protected from potential damage and debris. There is normally excellent natural circulation of the bulk of the coolant past these taps, and the pressure taps are located sufficiently low to permit sampling at a reactor water level below the lower core support plate.

A single sample line is also connected to both loops in the RHR system. This provides a means of obtaining a reactor coolant sample when the reactor is depressurized and at least one of the RHR loops is operated in the shutdown cooling mode. Similarly, a suppression pool liquid sample can be obtained from the RHR loop lined up in the suppression pool cooling mode.

23

All liquid samples are taken into septum bottles mounted on sampling needles. The sample panel is basically a bypass loop on the sample purge line. In the normal lineup, the sample flows through a conductivity cell (readable range 0.1 to 1000 micromhos/cm), and then through a ball valve bored out to 0.10-ml volume. Flow through the sample panel is established, the valve is rotated 90°, and a syringe is used to flush the sample plus a measured volume of diluent (generally 10 ml) through the valve and into the sample bottle. This provides an initial dilution of 100:1 and supplies a sample for further dilution and subsequent counting on a gamma spectrometer. Alternately, the sample flow can be diverted through a cylinder to obtain a large pressurized volume of coolant. This volume can be circulated and depressurized into a gas sampling chamber. The pressure of the collected gas can be related to total dissolved gas concentration. A sample of the gas can be obtained for H_2O and O_2 analysis. Ten-ml aliquots of the degassed liquid can also be taken for off-site chemical analyses requiring a relatively large sample. A radiation monitor in the liquid sample enclosure monitors liquid flow from the sample station to provide immediate assessment of the sample activity level. This monitor also provides information as to the effectiveness of the demineralized water flushing of the sample system following sample operation.

The piping station includes sample coolers and control valves which select liquid sample points. The sample station consists of a wall-mounted frame and enclosures. Included within the sample station are equipment trays which contain modularized liquid and gas samplers. Each of these modules is approximately 18 in x 14 in x 20 in high. The lower liquid sample portion of the sample station is shielded with 6 in of lead brick, whereas the upper gas

sampler requires 2 in of lead. The total weight of the wall-mounted portion of the system is approximately 7,000 lb. The dimensions of the sample station including shielding are approximately 29 in wide x 27 in deep x 72 in high. The frame is mounted so that the bottom of the frame is approximately 20 in off the floor. The control instrumentation is installed in a 2 ft x 4 ft x 6 ft high standard control cabinet panel. The panel contains the conductivity, radiation level readouts, and the flow, pressure, and temperature indicators, and various control valves and switches. The general front panel arrangement is shown on Figure II.B.3-2.

Appropriate sample handling tools are provided at the sample station. A gas sampler vial positioner and gas vial cask are included. The gas vial is installed and removed by use of the vial positioner through the front of the gas sampler. The vial is then manually dropped into the cask with the positioner which allows the vial to be maintained about 3 ft from the individual performing the operation.

The small volume liquid sample is remotely obtained through the bottom of the sample station by use of the small volume cask and cask positioner. The cask positioner holds the cask and positions the cask directly under the liquid sampler. 7

The sample vial is manually raised within the cask to engage the hypodermic needles. When the sample vial has been filled, the bottle is manually withdrawn into the cask. The sample vial is always contained within lead shielding during this operation. The cask is then lowered and sealed prior to transport to the laboratory.

A large volume cask and cask positioner are used to remotely obtain the large volume liquid sample. The positioner contains the cask and vial.

The cask is transported to the required position under the sample station by a four-wheel dolly cask positioner. When in position, this cask is hydraulically elevated approximately 1.5 in by a small hand pump for contact with the sample station shielding under the liquid sample enclosure. The sample bottle is raised, held, and lowered by a simple push/pull cable. The cask is sealed by a threaded top plug inserted above the sample bottle. The weight of this large volume cask is approximately 700 lb.

The cask may be used for offsite shipment of the large volume sample; however, it will require additional packaging.

A 15-ml bottle is contained within the lead-shielded cask. This sample bottle is raised from its location in the cask to the sample station needles for bottle filling. The sample station will only deliver 10 ml to this sample bottle. When filled, the bottle is withdrawn into the cask. The sample bottle is always shielded by 5 to 6 in of lead when in position under the sample station and during the fill and withdraw cycles; thus, operator exposure is controlled.

The particulate filters and iodine cartridges are removed via a drawer arrangement. The quantity of activity which is accumulated on the cartridges is controlled by a combination of flow orificing and time sequence control of the flow valve opening; in addition, the deposition of iodine is monitored during sampling using a radiation detector installed adjacent to the cartridge. These samples are limited to activity levels which will not require shielded sample carriers to transport the samples to the laboratory.

The chemistry laboratory and counting room to be used for post-accident analysis are located in Nine Mile Point Unit 1, on el 261 ft. The sample transport route to the Unit 1 chemistry room will be from the radwaste sample room, el 261 ft, west to the southeast corner of the screenhouse, then south into the turbine building, el 250 ft, via the truck ramp. The transport route includes the truck aisle that lies to the north and west of the condensers, and the access passage to Unit 1. Once in the Unit 1 shop room, an elevator provides transportation to el 261 ft, where the chemistry laboratory is located. The total distance is approximately 1,000 ft. Table II.B.3-1 shows the times required for sampling, transport, and analysis. The total time is within the 3-hr limit, with the exception of chloride analysis of reactor coolant, which is completed within the 96-hr time limit.

26 The PASS isolation valve control panel (2CES-PNL555) and the PASS sample station control panel (2SSP-IPNL102) are operated from the plant normal, uninterruptible power supply
26 (UPS, 2VBS-PNLA102 and B107), which is backed up by the plant 125-V dc normal battery system. Both the UPS system and the battery chargers are connectable to the standby diesel generators, except under LOCA conditions.

 10

Criterion 5

The time for a chloride analysis to be performed depends upon two factors: (a) if the plant's coolant water is seawater or brackish water, and (b) if there is only a single barrier between primary containment systems and the cooling water. Under both of the above conditions the licensee shall provide for a chloride analysis within 24 hr of the sample being taken. For all other cases, the licensee shall provide for the analysis to be completed within 4 days. The chloride analysis does not have to be done onsite.

Clarification 5

BWRs on sea or brackish water sites, and plants which use sea or brackish water in essential heat exchangers (e.g., shutdown cooling) that have only single barrier protection between the reactor coolant are required to analyze chloride within 24 hr. All other plants have 96 hr to perform a chloride analysis. Samples diluted by up to a factor of 1,000 are acceptable as initial scoping analysis for chloride, provided (1) the results are reported as ±2 ppm Cl (the licensee should establish this value; the number in the blank should be no greater than 10.0 ppm Cl) in the reactor coolant system, and (2) that dissolved oxygen can be verified at <0.1 ppm, consistent with the guidelines above in clarification No. 4. Additionally, if chloride analysis is performed on a diluted sample, an undiluted sample need also be taken and retained for analysis within 30 days, consistent with ALARA.

Position 5

Chloride in the reactor coolant can be determined within 96 hr by using a specific ion electrode. Unit 2 does not use brackish water for plant coolant. An undiluted reactor 10
coolant sample is treated with a sodium bromate-nitric acid solution to eliminate halogen interferences.

The sample is handled in a fume hood with long tongs and lead brick shielding to reduce radiation exposure.

Additionally, NMPC participates in the Pooled Inventory Management Program and should have a post-accident sampling cask from Nuclear Packaging, Inc. available for sample transport to an offsite facility for further analysis.

Criterion 6

The design basis for plant equipment for reactor coolant and containment atmosphere sampling and analysis must assume that it is possible to obtain and analyze a sample without radiation exposures to any individual exceeding the criteria of GDC 19 (Appendix A, 10CFR50) i.e., 5 rem whole body, 75 rem extremities. (Note that the design and operational review criterion was changed from the operational limits of 10CFR20 (NUREG-0578) to the GDC 19 criterion (October 30, 1979, letter from H. R. Denton to all licensees).

Clarification 6

Consistent with Regulatory Guide 1.3 or 1.4 source terms, provide information on the predicted personnel exposures based on person-motion for sampling, transport and analysis of all required parameters.

Position 6

As shown in Table II.B.3-1, whole body exposure and extremity exposure would be less than 0.98 R and 16 R, respectively. Individual exposure would be at even lower levels because more than one person would be performing the required tasks.

Criterion 7

The analysis of primary coolant samples for boron is required for PWRs. (Note that Regulatory Guide 1.97, Rev. 2, specifies the need for primary coolant boron analysis capability at BWR plants).

Clarification 7

PWRs need to perform boron analysis. The guidelines for BWRs are to have the capability to perform boron analysis, but they do have to do so unless boron was injected.

pH can be measured by use of a flat surface combination pH electrode on 0.1 to 0.3 ml of reactor coolant. The sample can be collected by operating the diluted reactor coolant sampler of the PASS several times without adding dilution water.

Chloride in the reactor coolant can be determined, within 96 hr, by using a specific ion electrode. An undiluted reactor coolant sample is treated with a sodium bromate-nitric acid solution to eliminate halogen interferences. The sample is handled in a fume hood with long tongs and lead brick shielding to reduce radiation exposure.

Criterion 3

Reactor coolant and containment atmosphere sampling during post-accident conditions shall not require an isolated auxiliary system (e.g., the letdown system, reactor water cleanup system [RWCU] to be placed in operation in order to use the sampling system. 23

Clarification 3

System schematics and discussions should clearly demonstrate that post-accident sampling, including recirculation, from each sample source is possible without use of an isolated auxiliary system. It should be verified that valves which are not accessible after an accident are environmentally qualified for the conditions in which they must operate.

Position 3

Isolated auxiliary systems are not required for post-accident sampling of reactor coolant or containment atmospheres. The sample lines for containment atmosphere samples connect into the containment monitoring system (CMS) outside of containment isolation valves on both hydrogen analyzer loops. These hydrogen analyzers are operational post-accident, with the containment isolation valves open. The sample lines for reactor coolant connect to two systems. There are no isolation valves between the reactor vessel and the sample taps into the jet pump instrumentation lines; therefore, a sample is available upon opening the PASS isolation valves. The residual heat removal system (RHS) is 23
also operational post-accident which allows sampling upon

23 opening the RHS sampling valves (operated from the main control room) and the PASS isolation valve.

The atmosphere samples are obtained by operating a gas pump inside the PASS while coolant samples are obtained by system pressure.

The PASS isolation valves are environmentally qualified to IEEE 323-1974 and IEEE 382-1972. All the components located in the PASS piping station in the secondary containment have been selected to assure that materials in these components will withstand the thermal and radiation environment expected during PASS operation.

Criterion 4

Pressurized reactor coolant samples are not required if the licensee can quantify the amount of dissolved gases with unpressurized reactor coolant samples. The measurement of either total dissolved gases or H_2 gas in reactor coolant samples is considered adequate. Measuring the O_2 concentration is recommended, but is not mandatory.

Clarification 4

Discuss the method whereby total dissolved gas or hydrogen and oxygen can be measured and related to reactor coolant system concentrations. Additionally, if chlorides exceed 0.15 ppm, verification that dissolved oxygen is less than 0.1 ppm is necessary. Verification that dissolved oxygen is <0.1 ppm by measurement of a dissolved hydrogen residual of ≥10 cc/kg is acceptable for up to 30 days after the accident. Within 30 days, consistent with minimizing personnel radiation exposures (ALARA), direct monitoring for dissolved oxygen is recommended.

Position 4

Total dissolved gas levels in the reactor coolant can be determined by measuring the pressure of the gas collected from a degassed sample of coolant. The sample flow in the PASS is diverted through a cylindrical volume. The volume is then circulated and depressurized into a gas chamber. The total dissolved gas level is determined from the pressure developed in the chamber. A sample of the gas can also be obtained for H_2 and O_2 analysis.

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Criterion 5

The time for a chloride analysis to be performed depends upon two factors: (a) if the plant's coolant water is seawater or brackish water, and (b) if there is only a single barrier between primary containment systems and the cooling water. Under both of the above conditions the licensee shall provide for a chloride analysis within 24 hr of the sample being taken. For all other cases, the licensee shall provide for the analysis to be completed within 4 days. The chloride analysis does not have to be done onsite.

Clarification 5

BWRs on sea or brackish water sites, and plants which use sea or brackish water in essential heat exchangers (e.g., shutdown cooling) that have only single barrier protection between the reactor coolant are required to analyze chloride within 24 hr. All other plants have 96 hr to perform a chloride analysis. Samples diluted by up to a factor of 1,000 are acceptable as initial scoping analysis for chloride, provided (1) the results are reported as ±2 ppm Cl (the licensee should establish this value; the number in the blank should be no greater than 10.0 ppm Cl) in the reactor coolant system, and (2) that dissolved oxygen can be verified at <0.1 ppm, consistent with the guidelines above in clarification No. 4. Additionally, if chloride analysis is performed on a diluted sample, an undiluted sample need also be taken and retained for analysis within 30 days, consistent with ALARA.

Position 5

Chloride in the reactor coolant can be determined within 96 hr by using a specific ion electrode. Unit 2 does not use brackish water for plant coolant. An undiluted reactor coolant sample is treated with a sodium bromate-nitric acid solution to eliminate halogen interferences. 10

The sample is handled in a fume hood with long tongs and lead brick shielding to reduce radiation exposure.

Additionally, NMPC participates in the Pooled Inventory
10 Management Program and should have a post-accident sampling
cask from Nuclear Packaging, Inc. available for sample
transport to an offsite facility for further analysis.

Criterion 6

The design basis for plant equipment for reactor coolant and containment atmosphere sampling and analysis must assume that it is possible to obtain and analyze a sample without radiation exposures to any individual exceeding the criteria of GDC 19 (Appendix A, 10CFR50) i.e., 5 rem whole body, 75 rem extremities. (Note that the design and operational review criterion was changed from the operational limits of 10CFR20 (NUREG-0578) to the GDC 19 criterion (October 30, 1979, letter from H. R. Denton to all licensees).

Clarification 6

Consistent with Regulatory Guide 1.3 or 1.4 source terms, provide information on the predicted personnel exposures based on person-motion for sampling, transport and analysis of all required parameters.

Position 6

As shown in Table II.B.3-1, whole body exposure and
10 extremity exposure would be less than (later), respectively.
Individual exposure would be at even lower levels because
more than one person would be performing the required tasks.

Criterion 7

The analysis of primary coolant samples for boron is required for PWRs. (Note that Regulatory Guide 1.97, Rev. 2, specifies the need for primary coolant boron analysis capability at BWR plants).

Clarification 7

PWRs need to perform boron analysis. The guidelines for BWRs are to have the capability to perform boron analysis, but they do have to do so unless boron was injected.

Position 7

Boron concentration in the primary coolant can be determined by the carminic acid method of analysis on the diluted reactor coolant samples. Reactor coolant flows into the PASS through a ball valve bored out to 0.10 ml. The valve is rotated 90 degrees, and a syringe is used to flush the

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sample plus 10 ml of demineralized water into a sample bottle. The bottle is transported to the laboratory in a lead-shielded cask. The sample is handled in the laboratory with tongs and lead brick shielding to reduce radiation exposure.

Criterion 8

If inline monitoring is used for any sampling and analytical capability specified herein, the licensee shall provide backup sampling through grab samples, and shall demonstrate the capability of analyzing the samples. Established planning for analysis at offsite facilities is acceptable. Equipment provided for backup sampling shall be capable of providing at least one sample per day for 7 days following onset of the accident, and at least one sample per week until the accident condition no longer exists.

Clarification 8

A capability to obtain both diluted and undiluted backup samples is required. Provisions to flush inline monitors to facilitate access for repair is desirable. If an offsite laboratory is to be relied on for the backup analysis, an explanation of the capability to ship and obtain analysis for one sample per week thereafter until accident condition no longer exists should be provided.

Position 8

The PASS can obtain both diluted and undiluted reactor coolant, dissolved gas, and containment atmosphere samples. Inline monitoring is used to determine hydrogen levels in the containment atmosphere; however, a grab sample can be obtained and analyzed for hydrogen on a gas chromatograph as a backup. No offsite laboratories are relied upon for backup analysis.

However, NMPC participates in the Pooled Inventory Management Program and should have a post-accident sampling cask from Nuclear Packaging, Inc. available for sample transport to an offsite facility. 10

Criterion 9

The licensee's radiological and chemical sample analysis capability shall include provisions to:

1. Identify and quantify the isotopes of the nuclide categories discussed above to levels corresponding

to the source terms given in Regulatory Guide 1.3 or 1.4 and 1.7. Where necessary and practicable, the ability to dilute samples to provide capability for measurement and reduction of personnel exposure should be provided. Sensitivity of onsite liquid

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sample analysis capability should be such as to permit measurement of nuclide concentration in the range from approximately 1 uCi/g to 10 Ci/g.

2. Restrict background levels of radiation in the radiological and chemical analysis facility from sources such that the sample analysis will provide results with an acceptably small error (approximately a factor of 2). This can be accomplished through the use of sufficient shielding around samples and outside sources, and by the use of a ventilation system design which will control the presence of airborne radioactivity.

Clarification 9

1. Provide a discussion of the predicted activity in the samples to be taken and the methods of handling/dilution that will be used to reduce the activity sufficiently to perform the required analysis. Discuss the range of radionuclide concentration which can be analyzed for, including an assessment of, the amount of overlap between post accident and normal sampling capabilities.

2. State the predicted background radiation levels in the counting room, including the contribution from samples which are present. Also provide data demonstrating what the background radiation levels and radiation effect will be on a sample being counted to assure an accuracy within a factor of 2.

Position 9

27 | Under severe accident conditions, the design basis activity level of the reactor coolant sample is 6.9 E-3 Ci/gm. The 0.1 ml of primary coolant diluted to 10.0 ml at the PASS would be used for the initial sample. The sample would be placed in a lead container for transport to the laboratory. A calibrated syringe would be used to obtain an aliquot for further dilution. A dilution factor of 10^6 can be obtained for isotopically analyzing samples up to 10 Ci/gm.

Direct counting of the initial 100:1 dilution sample would permit analysis down to 10 uCi/gm. In addition, the degassed 10-ml sample available from the PASS provides a method to obtain samples in the 10^{-2} to 10^{-3} uCi/gm levels which are encountered during normal operation. The PASS can provide useful samples at coolant activities ranging from

10 Ci/gm to well below the maximum level that can be tolerated at the normal reactor sample station.

The counting room used for post-accident sampling analysis is located in Unit 1. It is surrounded by concrete walls approximately 3 ft thick. The emergency ventilation system inlet duct for this room is 1500 ft from the Unit 2 stack. It has particulate filters. Assuming containment isolation, background radiation levels are predicted to be ≤ .3 mrem/hr. 28

To demonstrate the effect of background radiation, a Cs-137 source was counted with a 5 mR/hr background level from a Eu-152 source. The Cs-137 was counted with an accuracy of 10 percent, which is well within the factor of 2 requirement.

Criterion 10

Accuracy, range, and sensitivity shall be adequate to provide pertinent data to the operator in order to describe radiological and chemical status of the reactor coolant systems.

Clarification 10

The recommended ranges for the required accident sample analyses are given in Regulatory Guide 1.97, Rev. 2. The necessary accuracy within the recommended ranges are as follows:

1. Gross activity, gamma spectrum: Measured to estimate core damage these analyses should be accurate within a factor of 2 across the entire range.

2. Boron: Measured to verify shutdown margin.

   In general, this analysis should be accurate within ±5% of the measured value (i.e., at 6,000 ppm B the tolerance is ±300 ppm while at 1,000 ppm B the tolerance is ±50 ppm.) For concentrations below 1,000 ppm, the tolerance band should remain at ±50 ppm.

3. Chloride: Measured to determine coolant corrosion potential.

For concentrations between 0.5 and 20.0 ppm chloride, the analysis should be accurate within ±10% of the measured value. At concentrations below 0.5 ppm, the tolerance band remains at ±0.05 ppm.

10 Ci/gm to well below the maximum level that can be tolerated at the normal reactor sample station.

The counting room used for post-accident sampling analysis
is located in Unit 1. It is surrounded by concrete walls
approximately 3 ft thick. The emergency ventilation system 10
inlet duct for this room is 1500 ft from the Unit 2 stack.
It has particulate filters. Assuming containment isolation,
background radiation levels are predicted to be (later).

To demonstrate the effect of background radiation, a Cs-137 source was counted with a 5 mR/hr background level from a Eu-152 source. The Cs-137 was counted with an accuracy of 10 percent, which is well within the factor of 2 requirement.

<u>Criterion 10</u>

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   In general, this analysis should be accurate within ±5% of the measured value (i.e., at 6,000 ppm B the tolerance is ±300 ppm while at 1,000 ppm B the tolerance is ±50 ppm.) For concentrations below 1,000 ppm, the tolerance band should remain at ±50 ppm.

3. Chloride: Measured to determine coolant corrosion potential.

For concentrations between 0.5 and 20.0 ppm chloride, the analysis should be accurate within ±10% of the measured value. At concentrations below 0.5 ppm, the tolerance band remains at ±0.05 ppm.

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4. Hydrogen or Total Gas: Monitored to estimate core degradation and corrosion potential of the coolant.

   An accuracy of ±10% is desirable between 50 and 2000 cc/kg but ±20% can be acceptable. For concentration below 50 cc/kg, the tolerance remains at ±5.0 cc/kg.

5. Oxygen: Monitored to assess coolant corrosion potential.

   For concentrations between 0.5 and 20.0 ppm oxygen, the analysis should be accurate within ±10% of the measured value. At concentrations below 0.5 ppm, the tolerance band remains at ±0.05 ppm.

6. pH: Measured to assess coolant corrosion potential.

   Between a pH of 5 to 9, the reading should be accurate within ±0.3 pH units. For all other
7 ranges, ±0.5 pH units is acceptable.

   To demonstrate that the selected procedures and instrumentation will achieve the above listed accuracies, it is necessary to provide information demonstrating their applicability in the post accident water chemistry and radiation environment. This can be accomplished by performing tests utilizing the standard test matrix provided below or by providing evidence that the selected procedure or instrument has been used successfully in a similar environment.

STANDARD TEST MATRIX FOR UNDILUTED REACTOR COOLANT SAMPLES IN A POST-ACCIDENT ENVIRONMENT

Constituent	Nominal Concentrations (ppm)	Added as (chemical salt)
I-	40	Potassium iodide
Cs+	250	Cesium nitrate
Ra+2	10	Barium nitrate
La+3	5	Lanthanum chloride
Ce+4	5	Ammonium cerium nitrate
Cl-	10	
B	2000	Boric acid
Li±	2	Lithium hydroxide
NO_3	150	

Constituent	Nominal Concentrations (ppm)	Added as (Chemical Salt)	
NH_4	5		
K+	20		
Gamma radiation (induced field)	Adsorbed dose (Later) rad/gm of reactor coolant		10

NOTES:

1. Instrumentation and procedures which are applicable to diluted samples only, should be tested with an equally diluted chemical test matrix. The induced radiation environment should be adjusted commensurate with the weight of actual reactor coolant in the sample being tested.

2. For PWRs, procedures which may be affected by spray additive chemicals must be tested in both the standard test matrix plus appropriate spray additives. Both procedures (with and without spray additives) are required to be available.

3. For BWRs, if procedures are verified with boron in the test matrix, they do not have to be tested without boron.

4. In lieu of conducting tests utilizing the standard test matrix for instruments and procedures, provide evidence that the selected instrument or procedure has been used successfully in a similar environment.

All equipment and procedures which are used for post accident sampling and analyses should be calibrated or tested at a frequency which will ensure, to a high degree of reliability, that it will be available if required. Operators should receive initial and refresher training in post-accident sampling, analysis and transport. A minimum frequency for the above efforts is considered to be every 6 months if indicated by testing. These provisions should be submitted in revised Technical Specifications in accordance with Enclosure 1 of NUREG-0737. The staff will provide model Technical Specifications at a later date.

Position 10

The suitability of each method used for analysis was evaluated. A summary is presented in Table II.B.3-2. In order to ensure the stated accuracies are achievable, technicians are regularly trained in the various tasks. Additionally, the various analytical and sampling instrumentation is calibrated.

Technician training consists of an initial 2-day program.
10 Retraining of technicians occurs during plant emergency drills during which various samples are obtained and analyzed. Process instrumentation is calibrated annually, while a calibration verification of the MCA used is performed three times a week. The spectrometer, pH meter, and gas chromatograph are calibrated before use.

Criterion 11

In the design of the post accident sampling and analysis capability, consideration should be given to the following items:

1. Provisions for purging sample lines, for reducing plateout in sample lines, for minimizing sample loss or distortion, for preventing blockage of sample lines by loose material in the RCS or containment, for appropriate disposal of the samples, and for flow restrictions to limit reactor coolant loss from a rupture of the sample line. The post-accident reactor coolant and containment atmosphere samples should be representative of the reactor coolant in the core area and the containment atmosphere following a transient or accident. The sample lines should be as short as possible to minimize the volume of fluid to be taken from containment. The residues of sample collection should be returned to containment or to a closed system.

2. The ventilation exhaust from the sampling station should be filtered with charcoal absorbers and high-efficiency particulate air (HEPA) filters.

10

Clarification 11

1. A description of the provisions which address each of these items should be provided. Such items, as heat tracing and purge velocities, should be addressed. To demonstrate that samples are representative of core conditions a discussion of mixing, both short and long term, is needed. If a given sample location can be rendered inaccurate due to the accident, i.e., sampling from a hot or cold leg loop which may have a steam or gas pocket, describe the backup sampling capabilities or address the maximum time that this condition can exist.

   BWRs should specifically address samples which are taken from the core shroud area and demonstrate how they are representative of core conditions.

   Passive flow restrictors in the sample lines may be replaced by redundant, environmentally qualified, remotely operated isolation valves to limit potential leakage from sampling lines. The automatic containment isolation valves should close on containment isolation or safety injection signals.

2. A dedicated sample station filtration system is not required, provided a positive exhaust exists which is subsequently routed through charcoal absorbers and HEPA filters.

Position 11

Purging capability is provided by two methods:

1. Purging with either demineralized water or nitrogen is available from near the source of the sample in secondary containment, to purge the entire PAS system from source to discharge.

2. Purging with either demineralized water or nitrogen is available from the radwaste sample room to flush the sample station, piping station, and discharge sample lines. Purging the lines before and after sampling will alleviate cross-contamination of the samples when switching from one sample to another.

Prior to taking a sample, the line volume will be replaced several times, thus ensuring a representative sample. 23

23 The sample residue and flush water are returned to the suppression pool. Restriction devices are not used because they are potential crud traps. The small size of the sample lines serves to limit loss of reactor coolant in the case of sample line rupture. The gas sample lines are heat traced to prevent precipitation of moisture and resultant loss of iodine.

Reactor coolant samples obtained from a tap of the jet pump pressure instrument system will provide representative core coolant samples for accident conditions.

In order to ensure that this sample location provides a representative sample, sufficient core flow is needed to circulate water from the core to the jet pump intake.

After a small break or nonbreak accident, the reactor water level is maintained at or near normal water level by the operator using emergency procedures. For decay power above 1 percent of rated power the core flow is estimated to be greater than 10 percent rated flow because of natural circulation. The entire reactor water inventory would be circulated through the jet pumps in about 3 to 4 min, thus ensuring that representative samples of core coolant will be available at the jet pumps.

At power levels of less than 1 percent rated, a sample that is representative of core conditions would be obtained by increasing the reactor water level by 18 in. This will fully flood the standpipes of the moisture separators and will provide a thermally induced recirculation flow path for mixing.

Makeup water does not significantly dilute the sample. Makeup water flow amounts to approximately 2 percent of the core flow for small steam line breaks or nonbreak accidents. For small liquid line breaks, the makeup water flow rate is estimated to be less than 18 percent of the core flow. Thus, no significant dilution occurs and the water circulating through the jet pump is representative of reactor coolant inventory for small break or nonbreak accidents.

Further, sample lines in the RHR system provide for a reactor coolant sample when the reactor is depressurized and at least one of the RHR loops is operating in the shutdown cooling mode.

Finally, for larger line breaks where reactor water level cannot be maintained, reverse flow through the core to the

suppression pool is provided. Suppression pool samples are obtained from the RHR pump discharge.

Water is injected into the reactor pressure vessel by the ECCS. The injected water is from the condensate storage tank and/or from the suppression pool. The injected water floods the reactor vessel and flows through the break into the drywell. Water flowing from the reactor vessel pipe break returns to the suppression pool in the drywell downcomer. The RHR system is manually initiated in the pool cooling mode and maintained in this mode unless containment spray is temporarily needed (1 or 2 loops available) to control containment pressure.

The RHR pool cooling system (i.e., suction and return line arrangement in the suppression pool, type of discharge device, etc), is designed to assure adequate mixing of the suppression pool.

Based on the RHR pool cooling system design and the communication established between the primary coolant in the reactor water vessel and the suppression pool, the proposed post-accident sampling of water from the RHR suppression pool suction line provides a representative water sample. 7

Sample capability has been provided for two points in each of the jet pump instrumentation system, the residual heat removal system and the containment monitoring system. If anything should go wrong in one loop, the other loop is available for sampling.

The lines are as short as possible, thereby maintaining accessability to the sample station.

A dedicated charcoal and HEPA filtration system is provided. The small fan is operated on UPS, backed up by the 125-V dc normal battery system. The effluent from the filtration system discharges to the decontamination area ventilation system.

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TABLE II.B.3-1

TIME AND DOSE PROJECTIONS FOR PASS SAMPLING, TRANSPORT, AND ANALYSIS

Task	Time (min) Start	Time (min) Stop	Persons[1]	Exposure[2] (mR) Whole Body	Exposure[2] (mR) Extremities	Notes
Decision to take sample	0	0	N/A	N/A	N/A	Assumes TSC and OSC activated and sample room habitated
Read containment atmosphere H_2 levels in control room	0	5	1	NEG	N/A	
Operate control panel for dilue reactor coolant	0	20	4	9.5	9.5	6" lead shielding
Transport dilute reactor coolant to laborary	20	42	2	3.6+1	2.5+2	6" lead shielding (Max) 3" lead shielding (Min)
Prepare coolant for isotopic	42	44.5	1	5.0-1	6.3+1	4" lead glass for W.B. (Max) 1/2" lead shielding (Min)
Perform isotopic analysis of coolant	44.5	49.5	1	2.2-4	2.0-1	
Analyze coolant for Boron	49.5	54.2	1	2.5	8.6+1	4" lead glass + 2" lead for W.B. 1/2" lead shielding
Prepare sample panel for containment atmosphere	20	20	2	0	0	6" lead shielding
Operate control panel for containment atmosphere	20	35	2	4.8+0	4.8+0	2" lead shielding
Transfer containment atmosphere to small cask	35	39.8	1	1.8+1	2.4+2	2" lead shielding
Transport containment atmosphere to laboratory	39.8	58.5	2	5.8+2	2.4+3	3" lead shielding
Prepare containment atmosphere for isotopic	58.5	63.9	1	3.3	5.2+2	4" lead glass & 2" lead for W.B. (Max) 1/2" lead shielding (Min)
Perform isotopic analysis of containment atmosphere	63.9	68.9	1	2.7-3	2.0-0	
Operate control panel for dissolved gas	39.8	109.8	3	2.5+1	2.5+1	6" lead shielding
Operate control panel for 10-ml reactor coolant	109.8	119.8	3	3.6+0	3.6+0	6" lead shielding

TABLE II.B.3-1 (Cont)

Task	Time (min) Start	Time (min) Stop	Persons[1]	Exposure[2] (mR) Whole Body	Exposure[2] (mR) Extremities	Notes
Transport 10-ml reactor coolant to laboratory	119.8	179.1	3	6.0+1	3.8+3	6" lead shielding (Max) 2" lead shielding (Min)
Analyze 10-ml reactor coolant for chloride	179.1	183.6	1	2.4+2	8.1+3	4" glass lead & 2" lead for W.B. (Max) 1/2" lead shielding (Min)

(1) Number of persons performing particular task.
(2) Doses are based on the assumption that the decision to take a sample is made 1 hr afer reactor scram.

TABLE II.B.3-2

POST-ACCIDENT SAMPLING ANALYTICAL METHODS

Analysis	Method	Suitability	Range	Accuracy
Boron	Carminic acid	GE NEDC-30088 In-house testing	50-1,000 ppm	±50 ppm
Chloride	Specific ion electrode	ASTM D512D In-house testing	1-10 ppm >10 ppm	±1 ppm ±10%
pH	Combination pH electrode	GE NEDC-30088	2-12 pH	±0.2 pH
Isotopic	Gamma spectral analysis	In-house testing	1μCi/gm-10 Ci/gm	±200%
Total Dissolved Gas[1]	Gas sample pressure measurements	GE testing In-house testing	25-50 cc/kg 50-400 cc/kg	±50% ±30%
Dissolved H_2 or O_2	Gas chromatograph and pressure measurements	GE testing	25-50 cc/kg 50-400 cc/kg	±50% ±30%
Hydrogen[2]	Gas chromatograph	In-house testing	0.1-100 %	±0.1%
Oxygen[2]	Gas chromatograph	In-house testing	0.5-100 %	±0.5%

[1] Verification is inconclusive.
[2] Backup analysis for on-line H_2/O_2 monitoring system