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1.0 INTRODUCTION AND SUMMARY				1.0 INTRODUCTION AND SUMMARY	
1.1	PROJECT IDENTIFICATION		Modified section 1.1, Applicant, changed title to Introduction, included information regarding origin and purpose of Defueled Safety Analysis Report, combined select information applicable in the defueled state from sections 1.1.1, Applicant, 1.1.2, Engineer-Construction, 1.1.3, Nuclear Steam Supply System Supplier, and 1.1.4, Turbine Generator Supplier, deleted historical and obsolete information.	1.1	Introduction
	1.1.1	Applicant	Deleted section 1.1.1, Applicant, moved relevant information to Section 1.1, Introduction		
	1.1.2	Engineer - Construction	Deleted section 1.1.2, Engineer - Construction, moved relevant information to Section 1.1, Introduction		
	1.1.3	Nuclear Steam Supply System Supplier	Deleted section 1.1.3, Nuclear Steam Supply System Supplier, moved relevant information to Section 1.1, Introduction		
	1.1.4	Turbine Generator Supplier	Deleted section 1.1.4, Turbine Generator Supplier, moved relevant information to Section 1.1, Introduction		
1.2	DEFINITIONS		Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therrefore, many of the definitions in this section are no longer applicable. The SSC classification process described in UFSAR section 1.4, Classification of BWR Systems, Criteria, and Requirements is not applicable in the defueled state and has been deleted. Many of the definitions provided in this section define terms related to the SSC Classification process and therefore, are no longer applicable Additionally, many of the definitions provided in this section are already defined in more appropriate documentation, including Technical Specifications and the Technical Requiements Manual. Finally, many of the definitions provided in this section are not used in the DSAR. Therefore, the information in this section is historcal and obsolete and may be deleted.		
	Table 1.2.1(A)	Actual Plant Design and Operation, Safety Considerations	Delete Table 1.2.1(A). Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. The classification process described in section 1.4.1 allowed classification of any BWR aspect relative to either personnel hazard or the generation of electrical power. This process is no longer required and is not applicable to a permanently defueled station. Consequently, the information in Table 1.2.1(A) is no longer required and is obsolete.		
	Table 1.2.1(B)	Actual Plant Design and Operation, Power Generation Considerations	Delete Table 1.2.1(B). Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. The classification process described in section 1.4.1 allowed classification of any BWR aspect relative to either personnel hazard or the generation of electrical power. This process is no longer required and is not applicable to a permanently defueled station. Consequently, the information in Table 1.2.1(B) is no longer required and is obsolete.		
	Table 1.2.1(C)	Actual Plant Design and Operation, Safety Considerations	Delete Table 1.2.1(C). Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. The classification process described in section 1.4.1 allowed classification of any BWR aspect relative to either personnel hazard or the generation of electrical power. This process is no longer required and is not applicable to a permanently defueled station. Consequently, the information in Table 1.2.1(C) is no longer required and is obsolete.		
	Table 1.2.1(D)	Actual Plant Design and Operation, Safety Considerations	Delete Table 1.2.1(D). Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. The classification process described in section 1.4.1 allowed classification of any BWR aspect relative to either personnel hazard or the generation of electrical power. This process is no longer required and is not applicable to a permanently defueled station. Consequently, the information in Table 1.2.1(D) is no longer required and is obsolete.		
	Table 1.2.1(E)	Actual Plant Design and Operation, Safety Considerations	Delete Table 1.2.1(E). Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. The classification process described in section 1.4.1 allowed classification of any BWR aspect relative to either personnel hazard or the generation of electrical power. This process is no longer required and is not applicable to a permanently defueled station. Consequently, the information in Table 1.2.1(E) is no longer required and is obsolete.		
	Table 1.2.1(F)	Actual Plant Design and Operation, Safety Considerations	Delete Table 1.2.1(F). Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. The classification process described in section 1.4.1 allowed classification of any BWR aspect relative to either personnel hazard or the generation of electrical power. This process is no longer required and is not applicable to a permanently defueled station. Consequently, the information in Table 1.2.1(F) is no longer required and is obsolete.		
	Figure 1.2.1	Relationship Between Safety Action and Protective Action	Delete Figure1.2.1. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. The classification process described in section 1.4.1 allowed classification of any BWR aspect relative to either personnel hazard or the generation of electrical power. This process is no longer required and is not applicable to a permanently defueled station. Consequently, the information in Figure 1.2.1 is no longer required and is obsolete.		

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	Figure 1.2.2	Relationship Between Protective Functions and Protective Actions	Delete Figure1.2.2. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. The classification process described in section 1.4.1 allowed classification of any BWR aspect relative to either personnel hazard or the generation of electrical power. This process is no longer required and is not applicable to a permanently defueled station. Consequently, the information in Figure 1.2.2 is no longer required and is obsolete.		
	Figure 1.2.3	Relationships Between Different Types of Systems, Actions and Objectives	Delete Figure1.2.3. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. The classification process described in section 1.4.1 allowed classification of any BWR aspect relative to either personnel hazard or the generation of electrical power. This process is no longer required and is not applicable to a permanently defueled station. Consequently, the information in Figure 1.2.3 is no longer required and is obsolete.		
1.3	METHODS OF TECHNICAL PRESENTATION				
	1.3.1	Purpose	1. Modified Paragraph 1.3.1 to reflect defueled state and associated change is purpose of the SAR.	1.1	Introduction
	1.3.2	Radioactive Material Barrier Concept	Deleted Section 1.3.2, Radioactive Material Barrier Concept, provided a methodology for ranking SSCs with depth of information presented regarding each SSC determined based on relationship with the radioactive barrier concept. Although this approach was used for operation, the approach is not applicable following certification of permanent defueling, is obsolete post defueling, and was not used in the development of the DSAR.		
	1.3.3	Organization of Contents	This section was deleted since it outlined the format and presentation for the UFSAR. The DSAR format and presentation and depth of material presented was based on predecessor DSARs and Reg Guide 1.184.		
	1.3.4	Format Organization of Sections	Information in this section was deleted since it outlined the format and presentation for the UFSAR. The DSAR format and presentation and depth of material presented was based on predecessor DSARs and Reg Guide 1.184. Delete section 1.3.3, Organization of Contents, to delete incorporation by reference of the following special references since they are no longer applicable following certification of cessation of operations and permanent defueling. - APED 5286 Design Basis for Critical Heat Flux in Boiling Water Reactors (September 1966). - APED 5450, Design Provisions for In-Service Inspection (April 1967) - APED 5453 Vibration Analysis and Testing of Reactor Internals (April 1967). - APED 5448 Analysis Methods of Hypothetical Super Prompt Critical Reactivity Transients in Large Power Reactors (April 1968). - APED 5458 Effectiveness of Core Standby Cooling Systems for General Electric Boiling Water Reactors (March 1968). - APED 5460 Design and Performance of General Electric Boiling Water Reactor Jet Pumps (September 1968). - APED 5454 Metal Water Reactions Effects on Core Cooling and Containment (March 1968). - APED 5654 Considerations Pertaining to Containment Inerting (August 1968). - APED 5706 In Core Neutron Monitoring System for General Electric Boiling Water Reactors (November 1968). - APED 5698 Summary of Results Obtained from a Typical General Electric Boiling Water Reactor Startup and Power Test Program (February 1969). - APED 5736 Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards (April 1969). - APED 5750 Design and Performance of General Electric BWR Main Steam Isolation Valves (March 1969). - APED 5758 Quality System Summary for BWR Nuclear Systems Projects (July 1969). - APED 5703 Design and Analysis of Control Rod Drive Reactor Vessel Penetrations (November 1968). - APED 5652 Stability and Dynamic Performance of the General Electric Boiling Water Reactor (April 1969). - APED 5446 Control Rod Velocity Limiter (March 1967). - APED 5449 Control Rod Worth Minimizer (March 1967). - APED 5447 Depressurization Performance of HPCI (June 1969). - APED 5455 The Mechanical Effects of Reactivity Transients (January 1968). - APED 5528 Nuclear Excursion Technology (August 1967). - APED 5640 Xenon Considerations in Design of Large Boiling Water Reactors (June 1968). - NEDO 10017 Field Testing Requirements for Fuel, Curtains and Control Rods (June 1969). - NEDO 10029 An Analytical Study of Brittle Fracture of GE BWR Vessel Subject to the Design Basis Accident (July 1969). - NEDO 10045 Consequences of a Steam Line Break in a General Electric Boiling Water Reactor (July 1969). - NEDO 10173 Current State of Knowledge High Performance BWR Zircaloy Clad UO2 Fuel (May 1970). - NEDO 10179 Effects of Cladding Temperature and Material on ECCS Performance (June 1970). - NEDO 10139 Compliance of Protection Systems to BWR Industry Criteria; General Electric Nuclear Steam Supply System (June 1970). - NEDO 10174 Consequences of a Postulated Flow Blockage Incident in a Boiling Water Reactor (May 1970). - NEDO 10208 Effects of Fuel Rod Failure on ECCS Performance (August 1970). - NEDO 10189 An Analysis of Functional Common Mode Failures in GE BWR Protection and Control Instrumentation (July 1970). - Safe Shutdown Capability Analysis (Section 10.11.3)		

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	1.3.5	Power Level Basis for Analysis of Abnormal Operational Transients and Accidents		Deleted section 1.3.5, Power Level Basis for Analysis of Abnormal Operational Transients and Accidents. Information in this section is not applicable following certification of permanent defueling. Since the reactor may no longer be operated and nuclear fuel may no longer be emplaced or retained in the reactor vessel, information regarding the power level basis for analysis o AOTs and accidents is no longer applicable.				
	Figure 1.3-1	Piping and Instrument Symbols, G-191155		Delete, reference no longer required since entire section has been deleted.				
1.4	CLASSIFICATION OF BWR SYSTEMS, CRITERIA, AND REQUIREMENTS			Delete Section 1.4, CLASSIFICATION OF BWR SYSTEMS, CRITERIA, AND REQUIREMENTS				
	1.4.1	Introduction		Delete Section 1.4.1, Introduction. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. The classification process described in this section allowed classification of any BWR aspect relative to either personnel hazard or the generation of electrical power. This process is no longer required and is not applicable to a permanently defueled station.				
	1.4.2	Classification Basis		Delete Section 1.4.2, Classification Basis. Since the classification process described in section 1.4.1 is no longer used, the classification basis is no longer required and is not applicable to a permanently defueled station.				
	1.4.3	Use of the Classification Basis		Delete Section 1.4.3, Use of the Classification Basis. Since the classification process described in section 1.4.1 is no longer used, the section describing use of the classification basis is no longer required and is not applicable to a permanently defueled station.				
	Table 1.4.1	Concept for Classification of BWR Systems, Criteria and Requirements		Delete Table 1.4.1, Concept for Classification of BWR Systems, Criteria and Requirements. Since the classification process described in section 1.4.1 is no longer used, the table listing the safet considerations and power generation considerations for Planned Operation, Abnormal Operating Transients and Accidents is no longer required.				
1.5	PRINCIPAL DESIGN CRITERIA			1. Deleted paragraph in section 1.5 which stated that design criteria are grouped according to the classification plan given in Table 1.2.1, since table 1.2.1 and the classification plan are not applicable to the defueled state. Obsolete information 2. Consolidated design criteria which remain applicable in the defueled state into appropriate general sections .	1.2	DESIGN CRITERIA		
	1.5.1	General Criteria		2. Section 1.5.1, General Criteria, General, deleted criteria which states that the station shall be designed, fabricated and erected so that it can be operated to produce electrical power in a safe, reliable and effecient manner. Obsolete information 3. Section 1.5.1, General Criteria, Nuclear, deleted paragraph 3 pertaining to the criteria for the reactor core and reactivity control systems. Not applicable in the defueled state, obsolete information 4. Consolidated remaining information into one section titled "Design Criteria" and deleted section 1.5.1				
	1.5.2	Power Generation Design Criteria (Planned Operation)		4. Deleted section 1.5.2, Power Generation Design Criteria, (Planned Operations). Not applicable in the defueled state, obsolete information				
	1.5.3	Power Generation Design Criteria (Abnormal Operational Transients)		5. Deleted section 1.5.3, Power Generation Design Criteria, (Abnormal Operational Transients. Not applicable in the defueled state, obsolete information.				
	1.5.4	Nuclear Safety Design Criteria (Planned Operation)		6. Deleted section 1.5.4, Nuclear Safety Design Criteria (Planned Operation). Not applicable in the defueled state, obsolete information				
	1.5.5	Nuclear Safety Design Criteria (Abnormal Operational Transients)		7. Deleted section 1.5.5, Nuclear Safety Design Criteria, (Abnormal Operational Transients). Not applicable in the defueled state, obsolete information				
	1.5.6	Nuclear Safety Design Criteria (Accidents)		8. Deleted Section 1.5.6, Nuclear Safety Design Criteria (Accidents). Not applicable in the defueled state, obsolete information				
	1.5.7	Special Safety Design Criteria		Deleted Section 1.5.7, Special Safety Design Criteria. This section lists the general design control for reactivity control. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since nuclear fuel can no longer be placed in the reactor vessel, reactivity control functions are no longer required. Consequently, the Reactivity Control Design Criteria are no longer applicable.				
1.6	PLANT DESCRIPTION				1.3	FACILITY DESCRIPTION		
	1.6.1	General				1.3.1	General	
		1.6.1.1	SITE AND ENVIRONS	No Change			1.3.1.1	SITE AND ENVIRONS
		1.6.1.2	FACILITY ARRANGEMENT	No Change			1.3.1.2	FACILITY ARRANGEMENT
						1.3.2	Fuel Storage and Handling	
		1.6.1.3	NUCLEAR SYSTEM	Delete section 1.6.1.3, Not applicable in the defueled state, obsolete information EXCEPT section 1.6.1.3.1, Renamed section to Nuclear Fuel and Control Rods. Fuel will be stored in the spent fuel pool until transferred to the site ISFSI, control rods will be stored either in the spent fuel pool or remain installed in the reactor vessel			1.3.2.1	Nuclear Fuel and Control Rods
		1.6.1.4	POWER CONVERSION SYSTEMS	Delete this section, Not applicable in the defueled state, obsolete information				

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	1.6.1.5	Electrical Power Systems		Modify to remove reference to power generation at the station, reference to the main generator and excitation system, the station main transformer, and the 345 kV transmission lines. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, the equipment mentioned above is no longer required and is obsolete.			1.3.5.1	Electrical Power Systems
	1.6.1.6	Radioactive Waste Systems		Modify to remove reference to methods used to control release of radioactive material to within applicable limits. Listing of the methods used is excess detail and is not reqired.		1.3.3	Radioactive Waste Management	
		1.6.1.6.1	Liquid Radwaste System	Modify to remove reference to discharge of radioactive process liquid wasate through the circulating water discharge canal and dilution with condenser effluent circulating water or river water The statement "in accordance with applicable permits" ensures that VY discharges will comply with all applicable requirements. Information removed is considered excessive detail.			1.3.3.2	Liquid Radwaste System
		1.6.1.6.2	Solid Radwaste System	Modify to remove excessive detail regarding methods for preparing and shipping solid radioactive waste. All VY solid radwaste shipped will be in accordance with applicable requirements.			1.3.3.3	Solid Radwaste System
		1.6.1.6.3	Gaseous Radwaste System	Delete section, Not applicable in the defueled state, obsolete information				
1.6.2	Nuclear Safety and Engineered Safeguards							
	1.6.2.1	Reactor Protection System		Delete section, Not applicable in the defueled state, obsolete information				
	1.6.2.2	Neutron Monitoring System		Delete section, Not applicable in the defueled state, obsolete information				
	1.6.2.3	Control Rod Drive System		Delete section, Not applicable in the defueled state, obsolete information				
	1.6.2.4	Nuclear System Pressure Relief System		Delete section, Not applicable in the defueled state, obsolete information				
	1.6.2.5	Reactor Core Isolation Cooling System		Delete section, Not applicable in the defueled state, obsolete information				
	1.6.2.6	Primary Containment		Delete section, Not applicable in the defueled state, obsolete information				
	1.6.2.7	Primary Containment and Reactor Vessel Isolation Control System		Delete section, Not applicable in the defueled state, obsolete information				
	1.6.2.8	Secondary Containment		Delete this section. License Amendment 262, approved by the NRC in February 2015, eliminated operability requirements for secondary containment, the Standby Gas Treatment (SGT) system and the reactor building ventilation isolation and the SGT system initiation instrumentation when handling irradiated fuel or a fuel cask. The changes require the subject systems to be operable only during movement of "recently" irradiated fuel assemblies in secondary containment and during operations with the potential to drain the reactor vessel. The period of sufficient radioactiv decay was determined to be 13 days. Since VYNPS certified permanent defueling in January 2015, all fuel in the fuel pool is considered sufficiently decayed. Consequently, the secondary containment function is no longer required.				
	1.6.2.9	Main Steam Line Isolation Valves		Delete section, Not applicable in the defueled state, obsolete information				
	1.6.2.10	Main Steam Line Flow Restrictors		Delete section, Not applicable in the defueled state, obsolete information				
	1.6.2.11	Core Standby Cooling Systems		Delete this section. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longe possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions provided by the Core Standby Cooling Systems are no longer required. Information regarding the Core Standby Cooling Systems is obsoleted and may be deleted.				
	1.6.2.12	Residual Heat Removal System (Containment Cooling)		Delete this section. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longe possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions provided by the Residual Heat Removal System (Containment Cooling) are no longer required. Information regarding the Residual Heat Removal System (Containment Cooling) is obsoleted and may be deleted.				
	1.6.2.13	Control Rod Velocity Limiter		Delete this section. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longe possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since nuclear fuel can no longer be placed in the reactor vessel, reactivity control functions are no longer required. Consequently, the functions provided by the Control Rod Velocity Limiters are no longer applicable.				
	1.6.2.14	Control Rod Drive Housing Supports		Delete this section. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longe possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since nuclear fuel can no longer be placed in the reactor vessel, reactivity control functions, inxluding the control rods, are no longer required. Consequently, the functions provided by the Control Rod Drive Housing Supports are no longer applicable				
	1.6.2.15	Standby Gas Treatment System		Delete section, Not applicable in the defueled state, obsolete information				

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	1.6.2.16	Station Diesel Generator System			Modify Section1.6.2.16, Station Diesel Generator System to remove reference to operating and accident conditions, and reflect that the diesel-generators now supply back-up power only to ensure the safe storage and handling of irradiated fuel. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the Diesel-Generators are only required to supply backup power to ensure the safe storage of irradiated fuel.			1.3.5.1	Electrical Power Systems
	1.6.2.17	Station Battery System			Modify Section 1.6.2.17, 125 Volt Battery System, to remove reference to operating and accident conditions, and reflect that the 125 V DC system now supplies power only to ensure the safe storage and handling of irradiated fuel. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the safety functions of the 125 V DC system are no longer required.			1.3.5.1	Electrical Power Systems
	1.6.2.18	Station Service Water System			Modify Section 1.6.2.18 Station Service Water System, to remove reference to operating and accident conditions, and reflect that the Station Service Water System now supplies only cooling necessary to ensure the safe storage and handling of irradiated fuel.			1.3.5.2	Service Water System
	1.6.2.19	Reactor Building Closed Cooling Water System			Delete section, Not applicable in the defueled state, obsolete information See Section 10.9				
	1.6.2.20	Main Steam Line Radiation Monitoring System			Delete section 1.6.2.20, Main Steam Line Radiation Monitoring System. The Main Steam Line Radiation Monitoring System monitored for the gross release of fission products from the fuel and, upon indication of such failure, initiated appropriate action to contain the released fission products. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the Main Steam Line Radiation Monitoring System are no longer required. Main Steam Line Radiation Monitoring System information is obsolete.				
	1.6.2.21	Reactor Building Ventilation Radiation Monitoring System			Modify as indicate to reflect defueled state			1.3.4.1	Reactor Building Ventilation Radiation Monitoring System
1.6.3	Special Safety Systems								
	1.6.3.1	Standby Liquid Control System			Deleted, Not applicable in the defueled state, obsolete information				
	1.6.3.2	Station Equipment Outside the Main Control Room			Deleted, Not applicable in the defueled state, obsolete information				
1.6.4	Process Control and Instrumentation								
	1.6.4.1	Nuclear System Process Control and Information			Delete				
		1.6.4.1.1	Reactor Manual Control System		Deleted, Not applicable in the defueled state, obsolete information				
		1.6.4.1.2	Recirculation Flow Control System		Deleted, Not applicable in the defueled state, obsolete information				
		1.6.4.1.3	Neutron Monitoring System		Deleted, Not applicable in the defueled state, obsolete information				
		1.6.4.1.4	Refueling Interlocks		Deleted, Not applicable in the defueled state, obsolete information				
		1.6.4.1.5	Refueling Vessel Instrumentation		Deleted, Not applicable in the defueled state, obsolete information				
		1.6.4.1.6	Process Computer System		Deleted, Not applicable in the defueled state, obsolete information				
	1.6.4.2	Power Conversion Systems Process Control and Instrumentation			Delete Section 1.6.4.2				
		1.6.4.2.1	Pressure Regulator and Turbine Generator Control		Deleted, Not applicable in the defueled state, obsolete information				
		1.6.4.2.2	Feedwater System Control		Deleted, Not applicable in the defueled state, obsolete information				
	1.6.4.3	Radiation Monitoring and Control					1.3.4	Radiation Monitoring and Control	

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1.6.4.3.1	Process Radiation Monitoring	Deleted reference to AOG System Off Gas, Primary Containment System Purge Ventilation System Discharge, Standby Gas Treatment system discharge, Circulating Water Discharge, Cooling Tower. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the above Radiation Monitoring System are no longer required. Information provided regarding those Radiation Monitoring Systems is obsolete.	1.3.4.2	Process Radiation Monitoring
1.6.4.3.2	Area Radiation Monitors	Section 1.6.4.3.2, Area Radiation Monitors modified to remove excess information regarding locations of area radiation monitors in the Reactor Building.	1.3.4.3	Area Radiation Monitors
1.6.4.3.3	Liquid Radwaste System Control	Section 1.6.4.3.3,Liquid Radwaste System control editorial changes only.	1.3.3.2	Liquid Radwaste System
1.6.4.3.4	Solid Radwaste Control	No changes to Solid Radwaste Control, information consolidated into Solid Radwaste System.	1.3.3.3	Solid Radwaste System
1.6.4.3.5	Gaseous Radwaste System Control	DeleteSection 1.6.4.3.5, Gaseous Radwaste System Control. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the Gaseous Radwaste System are no longer required. Information provided regarding the Gaseous Radwaste System is obsolete.		
1.6.5	Auxiliary Systems		1.3.5	Auxiliary Systems
1.6.5.1	Normal Auxiliary AC Power	Modified to reflect that facility power is supplied from 115 kV swyds via SU transformers	1.3.5.1	Electrical Power Systems
1.6.5.2	Turbine Building Closed Cooling Water System	Deleted, Not applicable in the defueled state, obsolete information. See Section 10.10		
1.6.5.3	Alternate Cooling System	Delete Section 1.6.5.3, Alternate Cooling System. The Alternate Cooling System (ACS) provided an alternate means of heat removal in the event that the service water pumps become inoperable. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, decay and sensible heat removal from the primary system is no longer required. A backup system for heat removal from the Fuel Pool is no longer required, other strategies are in place to maintain cooling and inventory in the spent fuel pool. Consequently, the information provided regarding the Alternate Cooling System is no longer applicable. Alternate Cooling System information is obsolete.		
1.6.5.4	Fire Protection System	Modified as indicated to remove excess specificity	1.3.5.3	Fire Protection System
1.6.5.5	Heating, Ventilating, and Air Conditioning Systems	Editorial Changes only	1.3.5.4	Heating, Ventilating, and Air Conditioning Systems
1.6.5.6	New and Spent Fuel Storage	Modified to accommodate transfer of irradiated fuel to the ISFSI. Removed statement that irradiated fuel is transferred underwater since transfer to the ISFSI does not occur underwater. See Section 10.3 Removed the reference to spent fuel, since not all irradiated fuel is "spent".	1.3.2.2	Irradiated Fuel Storage
1.6.5.7	Fuel Pool Cooling and Clean-up System	Editorial changes to reflect changes to section 10.5	1.3.2.3	Standby Fuel Pool Cooling and Demineralizer System
1.6.5.8	Service and Instrument Air Systems	Deleted reference to the Primary Containment Atmospeheric Control System since that system is no longer required	1.3.5.5	Service and Instrument Air Systems
			1.3.7	Station Water Purification, Treatment and Storage
1.6.5.9	Makeup Water Treatment System	Editorial Changes only	1.3.7.1	Makeup Water Treatment System
1.6.5.10	Potable and Sanitary Water System	Editorial Changes only	1.3.7.2	Potable and Sanitary Water System
1.6.5.11	Equipment and Floor Drainage Systems	Editorial Changes only	1.3.3.1	Equipment and Floor Drainage Systems
1.6.5.12	Process Sampling System	Editorial Changes only	1.3.5.6	Process Sampling System
1.6.5.13	Post-Accident Sampling System	Deleted, Not applicable in the defueled state, obsolete information See Section 10.20		
1.6.5.14	Station Communications System	Editorial Changes only	1.3.6	Station Communications System
1.6.6	Station Shielding, Access Control and RP Procedures		1.3.8	Shielding, Access Control and Radiation Protection Procedures

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UFSAR Section				FSAR Conversion to DSAR Change Summary	DSAR Section			
		1.6.6.1	General	Deleted reference to table 1.6.1, shielding in containment required to make radiation levels on the station consistent with operational requirements			1.3.8.1	General
		1.6.6.2	Shielding	Section modified to present summary to reflect conditions in permanently defueled state and to eliminate duplication between 1.6.6 and section 12.3, Radiation Shielding and 13.4 Radiation Protection. Consolidated section with section 1.6.6.1.				
		1.6.6.3	Access Control Procedures	Section modified to present summary to reflect conditions in permanently defueled state and to eliminate duplication between 1.6.6 and section 12.3, Radiation Shielding and 13.4 Radiation Protection. Consolidated section with section 1.6.6.1.				
		1.6.6.4	RP Procedures	Section modified to present summary to reflect conditions in permanently defueled state and to eliminate duplication between 1.6.6 and section 12.3, Radiation Shielding and 13.4 Radiation Protection. Consolidated section with section 1.6.6.1.				
	1.6.7	Implementation of Structural Loading Criteria		No Change		1.3.9	Structural Loading Criteria	
	1.6.8	Plant Design Changes		Delete Section 1.6.8. This section is redundant to the regulatory requirement to submit summaries of 50.59 evaluations.				
	1.6.9	References		Delete Section 1.6.9, References. References 1 and 2 were deleted in earlier UFSAR revisions, Reference 3 , VYC-2423, Updated Vital Area Dose for EPU, documented a review of plant shielding for vital equipment and post accident procedural controls in response to NUREG 0737, Item II.B.2.2. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the information provided in reference 3 is no longer applicable and is obsolete. Since no additional references exist, this section may be deleted.				
	Table 1.6.1	Shielding Design Basis Limitations		Delete Table 1.6.1, Shielding Design Basis Limitations. Reference to this table has been deleted from section 1.6.6.1, no other references to this table exist in this section. Therefore, this table may be deleted.				
	Figure 1.6-1	Reactor Heat Balance, Rated Power		Delete Figure 1.6-1, Reactor Heat Balance, Rated Power. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. The information provided by figure 1.6-1, Reactor Heat Balance, Rated Power, is no longer applicable. This information is obsolete.				
	Figure 1.6-2	Turbine Generator Heat Balance, Rated Power (at 0.96 Power Factor)		Delete Figure 1.6-2, Turbine Generator Heat Balance, Rated Power. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. The information provided by figure 1.6-1, Reactor Heat Balance, Rated Power, is no longer applicable. This information is obsolete.				
1.7	COMPARISON OF PRINCIPAL DESIGN CHARACTERISTICS			Delete Section 1.7, Comparison of Principal Design Characteristics. This section primarily lists the design characteristics of the VY station at initial licensing and compares them to similar plants in operation at the time. This information is historical in nature and is no longer applicable following permanent defueling. Where applicable, information provided was verified to also exist in the appropriate SSC description in other locations in the DSAR. Additionally, the information presented in Table 1.7.7, AEC 1967 Draft General Design Criteria Indexed to applicable UFSAR Section(s) is historical information and is no longer applicable.				
	1.7.1	Nuclear System Design Characteristics		Delete, not applicable to the defueled state, obsolete information				
	1.7.2	Power Conversion System Design Characteristics		Delete, not applicable to the defueled state, obsolete information				
	1.7.3	Electrical Power Systems Design Characteristics		Delete, not applicable to the defueled state, obsolete information				
	1.7.4	Containment Design Characteristics		Delete, not applicable to the defueled state, obsolete information				
	1.7.5	Structural Design Characteristics		Delete, not applicable to the defueled state, obsolete information				
	Table 1.7.1	Comparison Nuclear System Design Characteristics		Delete, not applicable to the defueled state, obsolete information				
	Table 1.7.2	Comparison Power Conversion System Design Characteristics		Delete, not applicable to the defueled state, obsolete information				
	Table 1.7.3	Comparison Electrical System Design Characteristics		Delete, not applicable to the defueled state, obsolete information				
	Table 1.7.4	Comparison of Containment Design Characteristics		Delete, not applicable to the defueled state, obsolete information				

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UFSAR Section			FSAR Conversion to DSAR Change Summary	DSAR Section		
	Table 1.7.5	Comparison of Structural Design Characteristics	Delete, not applicable to the defueled state, obsolete information			
	Table 1.7.6	Comparison of System Design Characteristics	Delete, not applicable to the defueled state, obsolete information			
	Table 1.7.7	AEC 1967 Draft General Design Criteria Indexed to Applicable UFSAR Section(s)	Delete, not applicable to the defueled state, obsolete information			
1.8	SUMMARY OF RADIATION EFFECTS			1.4	SUMMARY OF RADIATION EFFECTS	
	1.8.1	Normal Operations	1. Changed section 1.8.1 Title to "Fuel Storage and Handling and Waste Management" 2. Changed "The station will be operated " to " Spent fuel storage and handling and waste management operations will be conducted".... 3. Deleted "based on operating experience", inserted "During fuel storage and handling and waste management operations"		1.4.1	Fuel Storage and Handling and Waste Management
	1.8.2	Abnormal Operational Transients	Deleted section 1.8.2, Abnormal Operating Transients. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, fuel damage due Abnormal Operational Transients are no longer possible. Consequently, the information provided in this section is obsolete and may be deleted.			
	1.8.3	Accidents	6. Reworded FHA section to reflect only 2 accidents, a FHA in the pool and a radwaste transfer cask drop accident.		1.4.2	Accidents and Events
	Table 1.8.1	Summary of the Maximum Offsite Effects of Design Basis Accidents	7. Deleted table 1.8.1, Summary of Maximum Offsite Effects of Design Basis Accidents, not applicable in defueled state, obsolete information			
1.9	QA PROGRAM			5.4	QA Program	
	1.9.1	Scope	No Changes		5.4.1	Scope
	1.9.2	Responsibilities	1. Deleted reference to ANSI N18.7-1976, inserted "as committed to within the QAPM" 2. Deleted reference to "Section A.2", inserted "The Organization section of the QAPM".		5.4.2	Responsibilities
	1.9.3	Implementation	3. Section 1.9.3, Implementation, deleted "in Table 1 of". Removing information which is too specific		5.4.3	Implementation
	1.9.4	Management Evaluation	4. Section 1.9.4, Management Evaluation, substituted DSAR for UFSAR, deleted "Entergy's, inserted "applicable". Removing information which is too specific		5.4.4	Management Evaluation
1.10	STATION RESEARCH DEVELOPMENT AND FURTHER INFORMATION;REQUIREMENTS AND RESOLUTION SUMMARY		1. Delete entire section, not applicable post-defueling, Obsolete information			
1.11	GENERAL CONCLUSIONS		Section reworded to state:" Based on the design of the facility, and the analysis of credible events, there is reasonable assurance that the facility can safely manage irradiated fuel and radioactive waste without endangering the health and safety of the public."	1.5	General Conclusions	
2.0 STATION SITE AND ENVIRONS				2.0 STATION SITE AND ENVIRONS		
2.1	SUMMARY DESCRIPTION		No Changes, retained as supporting historical information	2.1	Summary Description	
2.2	STATION SITE AND ENVIRONS			2.2	Site Description	
	2.2.1	Location and Area	No Changes		2.2.1	Location and Area
	2.2.2	Population	No Changes		2.2.2	Population
	2.2.3	Land Use	Deleted reference to Hinsdale Raceway, raceway has been demolished.		2.2.3	Land Use
	2.2.4	Site Area Boundaries, Exclusion Area, and Low Population Zone	Section 3.0, Exclusion Areas, deleted "during normal plant operation"		2.2.4	Site Area Boundaries, Exclusion Area, and Low Population Zone
	2.2.5	Conclusions	No Changes		2.2.5	Conclusions
	Table 2.2.1	Population Density Comparision	No Changes		Table 2.2.1	Population Density Comparision
	Table 2.2.2	2000 Population Distribution	No Changes		Table 2.2.2	2000 Population Distribution

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UFSAR Section			FSAR Conversion to DSAR Change Summary	DSAR Section		
	Table 2.2.3	Projected Population Distribution for Year 2010	No Changes		Table 2.2.3	Projected Population Distribution for Year 2010
	Table 2.2.4	Urban Centers Within 30 Miles of Site	No Changes		Table 2.2.4	Urban Centers Within 30 Miles of Site
	Table 2.2.5	Table of Land Use - Square Miles	No Changes		Table 2.2.5	Table of Land Use - Square Miles
	Table 2.2.6	Agricultural Statistics for Counties Within 50 Miles	No Changes		Table 2.2.6	Agricultural Statistics for Counties Within 50 Miles
	Table 2.2.7	Agricultural Statistics for Counties Within 50 Miles	No Changes		Table 2.2.7	Agricultural Statistics for Counties Within 50 Miles
	Figure 2.2-1	Location Map - 2 Mile Radius	No Changes		Figure 2.2-1	Location Map - 2 Mile Radius
	Figure 2.2-2	Location Map - 10 Mile Radius	No Changes		Figure 2.2-2	Location Map - 10 Mile Radius
	Figure 2.2-3	Location Map - 25 Mile Radius	No Changes		Figure 2.2-3	Location Map - 25 Mile Radius
	Figure 2.2-4	G-191142 Station Plan	No Changes		Figure 2.2-4	G-191142 Station Plan
	Figure 2.2-5	5920-6245 Plan Showing Exclusion Area and Restricted Area Boundaries	No Changes		Figure 2.2-5	5920-6245 Plan Showing Exclusion Area and Restricted Area Boundaries
	Figure 2.2-6	Station Site - Area Population Distribution 5 Mile Radius - Year 2000	No Changes		Figure 2.2-6	Station Site - Area Population Distribution 5 Mile Radius - Year 2000
	Figure 2.2-7	Station Site - Area Population Distribution 10 Mile Radius - Year 2000	No Changes		Figure 2.2-7	Station Site - Area Population Distribution 10 Mile Radius - Year 2000
	Figure 2.2-8	Station Site - Area Population Distribution 50 Mile Radius - Year 2000	No Changes		Figure 2.2-8	Station Site - Area Population Distribution 50 Mile Radius - Year 2000
2.3	METEOROLOGY			2.3	Meteorology	
	2.3.1	General	No Changes		2.3.1	General
	2.3.2	On-site Meteorological Programs	No Changes		2.3.2	On-site Meteorological Programs
	2.3.3	Diffusion Climatology	No Changes		2.3.3	Diffusion Climatology
	2.3.4	Winds and Wind Loading	No Changes		2.3.4	Winds and Wind Loading
	2.3.5	Temperature and Precipitation	No Changes		2.3.5	Temperature and Precipitation
	2.3.6	Storms	No Changes		2.3.6	Storms
	2.3.7	Conclusions	No Changes		2.3.7	Conclusions
	2.3.8	References	No Changes		2.3.8	References
	Table 2.3.1	Meteorology Record	No Changes		Table 2.3.1	Meteorology Record
	Table 2.3.2	TEMPERATURE DATA FOR THE VERNON AREA	No Changes		Table 2.3.2	TEMPERATURE DATA FOR THE VERNON AREA
	Table 2.3.3	PRECIPITATION DATA FOR THE VERNON AREA	No Changes		Table 2.3.3	PRECIPITATION DATA FOR THE VERNON AREA
	Table 2.3.4	WINDS DURING THUNDERSTORMS	No Changes		Table 2.3.4	WINDS DURING THUNDERSTORMS
	Table 2.3.5	RAINFALL DATA FROM HURRICANE CONNIE	No Changes		Table 2.3.5	RAINFALL DATA FROM HURRICANE CONNIE

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UFSAR Section			FSAR Conversion to DSAR Change Summary	DSAR Section		
	Figure 2.3-1	Station Site - Westover AFB, Massachusetts Area- Annual Surface Windrose	No Changes		Figure 2.3-1	Station Site - Westover AFB, Massachusetts Area- Annual Surface Windrose
	Figure 2.3-2	Station Site - Westover AFB, Massachusetts Area- Seasonal Surface Windroses	No Changes		Figure 2.3-2	Station Site - Westover AFB, Massachusetts Area- Seasonal Surface Windroses
	Figure 2.3-3	Station Site - Concord New Hampshire Area, Periods of Rainfall - (For Extremely - Short Intervals)	No Changes		Figure 2.3-3	Station Site - Concord New Hampshire Area, Periods of Rainfall - (For Extremely - Short Intervals)
2.4	HYDROLOGY AND BIOLOGY		No Changes	2.4	Hydrology and Biology	
	2.4.1	General	No Changes		2.4.1	General
	2.4.2	Land Area Ground Hydrology	No Changes		2.4.2	Land Area Ground Hydrology
	2.4.3	Hydrology	Section 2.4.3.4, Hydrology, Floods. Delete information regarding equipment required to operate during a maximum probable flood since that equipment was necessary to ensure adequate decay heat removal from a recently shutdown reactor. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since power operations cannot occur and there is no fuel in the reactor vessel, core decay heat removal during a PMF from a recently shutdown reactor is no longer a concern. Therefore, this information is no longer applicable and may be deleted. Section 2.4.3.4, Hydrology, Floods. Insert the statement regarding fuel pool cooling during a PMF. Fuel pool cooling will be maintained until either service water is lost or normal electrical power is lost. DG backups are available for power. Alternate cooling strategies exist in the event service water or electrical power is lost. Replaced discussion of flooding and wave runup with discussion generated during Fukushima Flooding Evaluation. Information in previous discussion was superseded by the information in the Fukushima Evaluation. Fukushima Evaluation information was located in the conclusion section and was moved into the text body.		2.4.3	Hydrology
	2.4.4	Uses Of River	No Changes		2.4.4	Uses Of River
	2.4.5	Biology	No Changes		2.4.5	Biology
	2.4.6	Chemical and Bacteriological Quality of Water	No Changes		2.4.6	Chemical and Bacteriological Quality of Water
	2.4.7	River Field Program	No Changes		2.4.7	River Field Program
	2.4.8	Conclusions	1. Section 2.4.8, Conclusions, deleted statement "Since the entrances to all structures cntaining equipment necessary for reactor shutdown and cooling are at elevation 252.5 feet MSL, they are protected against the PMF. " No equipment is required for reactor shutdown or cooling post defueling.		2.4.8	Conclusions
	2.4.9	References	No Changes		2.4.9	References
	Table 2.4.1	Average and Extreme Values of Stream Flow Connecticut River at Vernon, Vermont Water Years 1945-1965	No Changes		Table 2.4.1	Average and Extreme Values of Stream Flow Connecticut River at Vernon, Vermont Water Years 1945-1965
	Table 2.4.2	Vermont Yankee Nuclear Power Station, Daily Stream Flow for October 1964 to September 1965, Connecticut River at Vernon, Vermont	No Changes		Table 2.4.2	Vermont Yankee Nuclear Power Station, Daily Stream Flow for October 1964 to September 1965, Connecticut River at Vernon, Vermont
	Table 2.4.3	Municipal and Industrial Groundwater Usage Within a 10-Mile Radius of the Vernon Site	No Changes		Table 2.4.3	Municipal and Industrial Groundwater Usage Within a 10-Mile Radius of the Vernon Site
	Table 2.4.4	Public Water Supplies Within a 10-Mile Radius of the Vernon Site	No Changes		Table 2.4.4	Public Water Supplies Within a 10-Mile Radius of the Vernon Site
	Table 2.4.5	Water Supplies Within a l-Mile Radius of the Site	No Changes		Table 2.4.5	Water Supplies Within a l-Mile Radius of the Site
	Table 2.4.6	Six-Hour PMP and Runoff Increments - Connecticut River Basin above Vernon, Vermont	No Changes		Table 2.4.6	Six-Hour PMP and Runoff Increments - Connecticut River Basin above Vernon, Vermont

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UFSAR Section			FSAR Conversion to DSAR Change Summary	DSAR Section		
	Table 2.4.7	Maximum Annual Floods on Connecticut River at Vernon, Vermont - Arranged in Descending Order (1927, 1936, 1938, 1945-1973)	No Changes		Table 2.4.7	Maximum Annual Floods on Connecticut River at Vernon, Vermont - Arranged in Descending Order (1927, 1936, 1938, 1945 1973)
	Table 2.4.8	Time - Varying PMF Stage - Discharge Table Vermont Yankee Nuclear Plant Site	No Changes		Table 2.4.8	Time - Varying PMF Stage - Discharge Table Vermont Yankee Nuclear Plant Site
	Table 2.4.9	Time - Varying Modified PMF Stage - Discharge Table Vermont Yankee Nuclear Plant Site	No Changes		Table 2.4.9	Time - Varying Modified PMF Stage - Discharge Table Vermont Yankee Nuclear Plant Site
	Table 2.4.10	Checklist of Connecticut River Fishes Found Near Vernon, Vermont	No Changes		Table 2.4.10	Checklist of Connecticut River Fishes Found Near Vernon, Vermont
	Table 2.4.11	Fishes of the Connecticut River in the Vicinity of Vernon, Vermont - All Collections, 1980	No Changes		Table 2.4.11	Fishes of the Connecticut River in the Vicinity of Vernon, Vermont All Collections, 1980
	Figure 2.4-1	Station Site - Area Public Water Supplies - 10-Mile Radius	No Changes		Figure 2.4-1	Station Site - Area Public Water Supplies - 10-Mile Radius
	Figure 2.4-2	Station Site - Area Private Water Supplies - 1-Mile Radius	No Changes		Figure 2.4-2	Station Site - Area Private Water Supplies - 1-Mile Radius
	Figure 2.4-3	Enveloping Depth-Duration-Area Values of PMP for Susquehannna River Basin	No Changes		Figure 2.4-3	Enveloping Depth-Duration-Area Values of PMP for Susquehannna River Basin
	Figure 2.4-4	6-Hour Unit Hydrograph	No Changes		Figure 2.4-4	6-Hour Unit Hydrograph
	Figure 2.4-5	Total SPF Hydrograph	No Changes		Figure 2.4-5	Total SPF Hydrograph
	Figure 2.4-6	Total PMF Hydrograph (Natural and Modified)	No Changes		Figure 2.4-6	Total PMF Hydrograph (Natural and Modified)
	Figure 2.4-7	Connecticut River Basin - Federal Power Commission Water Resource Appraisals for Hydroelectric Licensing - Summary of Planning Status	No Changes		Figure 2.4-7	Connecticut River Basin - Federal Power Commission Water Resource Appraisals for Hydroelectric Licensing - Summary of Planning Status
	Figure 2.4-8	Vermont Yankee Nuclear Plant - Location of River Cross-Sections	No Changes		Figure 2.4-8	Vermont Yankee Nuclear Plant - Location of River Cross-Sections
	Figure 2.4-9	Stage-Discharge Curve at the Vermont Yankee Nuclear Plant	No Changes		Figure 2.4-9	Stage-Discharge Curve at the Vermont Yankee Nuclear Plant
	Figure 2.4-10	Cross Section of the Critical Fetch	No Changes		Figure 2.4-10	Cross Section of the Critical Fetch
	Figure 2.4-11	Vermont Yankee Sample Stations on Connecticut River	No Changes		Figure 2.4-11	Vermont Yankee Sample Stations on Connecticut River
2.5	GEOLOGY AND SEISMOLOGY		No Changes	2.5	Geology and Seismology	
	2.5.1	General	No Changes		2.5.1	General
	2.5.2	Geology	No Changes		2.5.2	Geology
	2.5.3	Seismology	No Changes		2.5.3	Seismology
	2.5.4	Conclusions	No Changes		2.5.4	Conclusions
	Table 2.5.1	Available Information Concerning Geology and Seismic Activity Related to the Vermont Yankee	No Changes		Table 2.5.1	Available Information Concerning Geology and Seismic Activity Related to the Vermont Yankee Nuclear Power Station Site

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UFSAR Section			FSAR Conversion to DSAR Change Summary	DSAR Section		
	Table 2.5.2	Vernon Pluton: Estimated Mode of the Oliverian Magma Series	No Changes		Table 2.5.2	Vernon Pluton: Estimated Mode of the Oliverian Magma Series
	Figure 2.5-1	Deleted	Previously Deleted			
	Figure 2.5-2	Station Site - Geological Survey - General Plan - Location of Test Borings	No Changes		Figure 2.5-2	Station Site - Geological Survey - General Plan - Location of Test Borings
	Figure 2.5-3	Station Site - Geological Survey - Subsurface Profile - Log of Test Borings (1A, 2A, 3A, 4, 5, 8)	No Changes		Figure 2.5-3	Station Site - Geological Survey - Subsurface Profile - Log of Test Borings (1A, 2A, 3A, 4, 5, 8)
	Figure 2.5-4	Station Site - Tectonic Map - State of Vermont	No Changes		Figure 2.5-4	Station Site - Tectonic Map - State of Vermont
	Figure 2.5-5	Station Site - Tectonic Map - State of New Hampshire	No Changes		Figure 2.5-5	Station Site - Tectonic Map - State of New Hampshire
	Figure 2.5-6	Station Site - Geological Survey - Area Bedrock Geology	No Changes		Figure 2.5-6	Station Site - Geological Survey - Area Bedrock Geology
	Figure 2.5-7	Station Site - Geological Survey - Area Geological Section	No Changes		Figure 2.5-7	Station Site - Geological Survey - Area Geological Section
	Figure 2.5-8	Station Site - Geological Survey - Subsurface Profile (Section AA) - Log of Test Borings (5, 8, S9, 11, and 21)	No Changes		Figure 2.5-8	Station Site - Geological Survey - Subsurface Profile (Section AA) - Log of Test Borings (5, 8, S9, 11, and 21)
	Figure 2.5-9	Station Site - Geological Survey - Subsurface Profile (Section BB) - Log of Test Borings (2A, 3A, ST6-1/2, and S9)	No Changes		Figure 2.5-9	Station Site - Geological Survey - Subsurface Profile (Section BB) - Log of Test Borings (2A, 3A, ST6-1/2, and S9)
	Figure 2.5-10	Station Site - Geological Survey - Subsurface Profile (Section CC) - Log of Test Borings (2, 2A, 5, 7, 7A, 13, and 15)	No Changes		Figure 2.5-10	Station Site - Geological Survey - Subsurface Profile (Section CC) - Log of Test Borings (2, 2A, 5, 7, 7A, 13, and 15)
	Figure 2.5-11	Station Site - Geological Survey - Subsurface Profile (Section DD) - Log of Test Borings (3, 3A, 4, 8, 8A, 12, and 16)	No Changes		Figure 2.5-11	Station Site - Geological Survey - Subsurface Profile (Section DD) - Log of Test Borings (3, 3A, 4, 8, 8A, 12, and 16)
	Figure 2.5-12	Station Site - Tectonic Map - New England Area	No Changes		Figure 2.5-12	Station Site - Tectonic Map - New England Area
	Figure 2.5-13	Station Site - Compilation of Earthquakes - New England Area	No Changes		Figure 2.5-13	Station Site - Compilation of Earthquakes - New England Area
	Figure 2.5-14	Station Site - Earthquake Intensity - Modified Mercalli and Rossi - Forel Scales	No Changes		Figure 2.5-14	Station Site - Earthquake Intensity - Modified Mercalli and Rossi - Forel Scales
	Figure 2.5-15	Station Site - Compilation of Earthquakes - Central New England Area	No Changes		Figure 2.5-15	Station Site - Compilation of Earthquakes - Central New England Area
2.6	Radiological Environmental Monitoring Program		Section 2.6.1, deleted the word safety from the section title. Since the classification process described in the UFSAR section 1.4.1 is no longer used, the term "Safety" objective is no longer applicable.	2.6	Radiological Environmental Monitoring Program	
	2.6.1	Safety Objectives	No Changes		2.6.1	Objectives
	2.6.2	Monitoring Network	No Changes		2.6.2	Monitoring Network
	2.6.3	Land Use Census	No Changes		2.6.3	Land Use Census
	2.6.4	Emergency Surveillance	No Changes		2.6.4	Emergency Surveillance
	2.6.5	Reports	No Changes		2.6.5	Reports
3.0	Reactor		Rename section "Facility Design and Operation" to provide a central location for information pertinent to the design and operation of Vermont Yankee in the permanently defueled state.	3.0	Facility Design and Operation	

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UFSAR Section			FSAR Conversion to DSAR Change Summary	DSAR Section	
3.1	SUMMARY DESCRIPTION		Delete the information in Section 3.1, Summary Description, rename the section 3.1, Design Criteria to provide a central location for SSC design information pertinent in the permanently defueled state. Information in the old summary description section summmarized aspects of the design and operation of the reactor which are no longer applicable in the permanently defueled state. Section 3.1, Design Criteria, establishes a central location for design information which remains applicable in the permanently defueled state.	3.1	Design Criteria
3.2	Fuel Mech Design			3.3.1.1	Nuclear Fuel
	3.2.1	Power Generation Objectives	Modify Section 3.2.1, Power Generation Objectives, to reflect the fact that fuel may no longer be emplaced or retained in the reactor vessel. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Therefore, power operations cannot occur. All fuel is stored either in the spent fuel pool or at the ISFSI. This objective has been rewritten and applies to the fuel stored in the Spent Fuel Pool.	3.3.1.1.1	Objective
	3.2.2	Safety Design Basis	Delete Section 3.2.2, Fuel Mech Design Safety Design Bases. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vesse IAW 10CFR50.82(a)(2). Therefore, power operations cannot occur. The Safety Design Basis is not applicable. Portions of this objective remaining applicable have been consolidated with the Power Generation Objective to form the Objective section.		
	3.2.3	Description	Modify Section 3.2.3, Fuel Mechanical Design Description to remove all discussion soecific to fuel emplacement in the reactor vessel. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Therefore, power operations cannot occur. Information provided in the Fuel Mechanical Design Description section regarding nuclear fuel operation or emplacement in the reactor vessel is no longer applicable;all nuclear fuel has been removed from the reactor vessel and is stored in the spent fuel pool or at the ISFSI. Information regarding fuel design applicable to storage in the spent fuel pool has been retained.	3.3.1.1.2	Description
	3.2.4	Safety Evaluation	Delete Section 3.2.4, Fuel Mech Design Safety Evaluation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Therefore, power operations cannot occur. The information provided in the Fuel Mech Design Safety Evaluation section is no longer applicable.		
	3.2.5	Inspection and Testing	Delete Section 3.2.5, Fuel Mech Design Inspection and testing. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Therefore, power operations cannot occur. The information provided in the Fuel Mech Design Inspection and Testing section is no longer applicable.		
	3.2.6	References	Delete Section 3.2.6, Fuel Mech Design References. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Therefore, power operations cannot occur. The information provided in the Fuel Mech Design References section is no longer applicable.		
3.3	REACTOR VESSEL INTERNALS Mech Design		Delete entire section, Sections 3.3.1 through 3.3.7 are no longer applicable, no information remains in this section		
	3.3.1	Power Generation Objectives	Delete section 3.3.1, Reactor Vessel Internals Mech Design, Power Generation Objectives. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. The functions performed by the reactor vessel internals are no longer required. Consequently, the Reactor Vessel Internals Mech Design information is no longer applicable, the safety objective is not required. Reactor Vessel Internals Mech Design information is obsolete.		
	3.3.2	Power Generation Design Bases	Delete section 3.3.2, Reactor Vessel Internals Mech Design, Power Generation Design Bases. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. The functions performed by the reactor vessel internals are no longer required. Consequently, the Reactor Vessel Internals Mech Design information is no longer applicable, the power generation design bases are not required. Reactor Vessel Internals Mech Design information is obsolete.		
	3.3.3	Safety Design Basis	Delete section 3.3.3, Reactor Vessel Internals Mech Design, Safety Design Bases. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. The functions performed by the reactor vessel internals are no longer required. Consequently, the Reactor Vessel Internals Mech Design information is no longer applicable, the safety design basis is not required. Reactor Vessel Internals Mech Design information is obsolete.		
	3.3.4	Description	Delete section 3.3.4, Reactor Vessel Internals Mech Design, Description. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. The functions performed by the reactor vessel internals are no longer required. Consequently, the Reactor Vessel Internals Mech Design information is no longer applicable, the information provided in the description section is not required. Reactor Vessel Internals Mech Design information is obsolete.		
	3.3.5	Safety Evaluation	Delete section 3.3.5, Reactor Vessel Internals Mech Design, Safety Evaluation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. The functions performed by the reactor vessel internals are no longer required. Consequently, the Reactor Vessel Internals Mech Design information is no longer applicable, the information provided in the safety evaluation section is not required. Reactor Vessel Internals Mech Design information is obsolete.		

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	3.3.6	Inspection and Testing	Delete section 3.3.6, Reactor Vessel Internals Mech Design, Inspection and Testing. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. The functions performed by the reactor vessel internals are no longer required. Consequently, the Reactor Vessel Internals Mech Design information is no longer applicable, the information provided in the Inspection and Testing section is not required. Reactor Vessel Internals Mech Design information is obsolete.	
	3.3.7	References	Delete section 3.3.7, Reactor Vessel Internals Mech Design, References. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. The functions performed by the reactor vessel internals are no longer required. Consequently, the Reactor Vessel Internals Mech Design information is no longer applicable, the references provided no longer apply. Reactor Vessel Internals Mech Design information is obsolete.	
	Table 3.3.1	Reactor Vessel Internals Design Data	Delete Table 3.3.1, Reactor Vessel Internals Design Data. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. The functions performed by the reactor vessel internals are no longer required. Consequently, the Reactor Vessel Internals Mech Design information is no longer applicable, the information provided no longer applies. Reactor Vessel Internals Mech Design information is obsolete.	
	Figure 3.3.1	Reactor Vessel Internals	Delete Figure 3.3.1, Reactor Vessel Internals. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. The functions performed by the reactor vessel internals are no longer required. Consequently, the Reactor Vessel Internals information is no longer applicable, the information provided no longer applies. Reactor Vessel Internals information is obsolete.	
	Figure 3.3.2	Reactor Vessel Internal Flow Schematic	Delete Figure 3.3.2, Reactor Vessel Internals Flow Schematic. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. The functions performed by the reactor vessel internals are no longer required. Consequently, the Reactor Vessel Internals information is no longer applicable, the information provided no longer applies. Reactor Vessel Internals information is obsolete.	
	Figure 3.3.3	Steam Separator	Delete Figure 3.3.3, Steam Separator. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. The functions performed by the reactor vessel internals are no longer required. Consequently, the Reactor Vessel Internals information is no longer applicable, the information provided no longer applies. Reactor Vessel Internals information is obsolete.	
	Figure 3.3.4	Fuel Support Pieces	Delete Figure 3.3.4, Fuel Support Pieces. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. The functions performed by the reactor vessel internals are no longer required. Consequently, the Reactor Vessel Internals information is no longer applicable, the information provided no longer applies. Reactor Vessel Internals information is obsolete.	
	Figure 3.3.5	Jet Pump Isometric	Delete Figure 3.3.5, Jet Pump Isometric. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. The functions performed by the reactor vessel internals are no longer required. Consequently, the Reactor Vessel Internals information is no longer applicable, the information provided no longer applies. Reactor Vessel Internals information is obsolete.	
	Figure 3.3.6	Steam Dryer	Delete Figure 3.3.6, Steam Dryer. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. The functions performed by the reactor vessel internals are no longer required. Consequently, the Reactor Vessel Internals information is no longer applicable, the information provided no longer applies. Reactor Vessel Internals information is obsolete.	
	Figure 3.3.7	Locations of Calculated Pressure Differentials	Delete Figure 3.3.7, Locations of Calculated Pressure Differentials. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. The functions performed by the reactor vessel internals are no longer required. Consequently, information regarding the Locations of Calculated Pressure Differentials is no longer applicable, the information provided no longer applies. This information is obsolete.	
	Figure 3.3.8A	Pressure Differentials Following Steam Line Break Inside the Drywell 102% Power/107% Core Flow	Delete Figure 3.3.8A, Pressure Differentials Following Steam Line Break Inside the Drywell 102% Power/107% Core Flow. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. The functions performed by the reactor vessel internals are no longer required. Consequently, information regarding the Pressure Differentials Following Steam Line Break Inside the Drywell 102% Power/107% Core Flow is no longer applicable, the information provided no longer applies. This information is obsolete.	
	Figure 3.3.8B	Pressure Differentials Following Steam Line Break Inside the Drywell 102% Power/107% Core Flow	Delete Figure 3.3.8B, Pressure Differentials Following Steam Line Break Inside the Drywell 102% Power/107% Core Flow. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. The functions performed by the reactor vessel internals are no longer required. Consequently, information regarding the Pressure Differentials Following Steam Line Break Inside the Drywell 102% Power/107% Core Flow is no longer applicable, the information provided no longer applies. This information is obsolete.	
	Figure 3.3.8C	3.3-8C Pressure Differentials Following Steam Line Break Inside the Drywell 16.7% Power/110% Core Flow	Delete Figure 3.3.8C, Pressure Differentials Following Steam Line Break Inside the Drywell 16.7% Power/110% Core Flow. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. The functions performed by the reactor vessel internals are no longer required. Consequently, information regarding the Pressure Differentials Following Steam Line Break Inside the Drywell 16.7% Power/110% Core Flow is no longer applicable, the information provided no longer applies. This information is obsolete.	

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	Figure 3.3.8D	Pressure Differentials Following Steam Line Break Inside the Drywell 16.7% Power/110% Core Flow	Delete Figure 3.3.8D, Pressure Differentials Following Steam Line Break Inside the Drywell 16.7% Power/110% Core Flow. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. The functions performed by the reactor vessel internals are no longer required. Consequently, information regarding the Pressure Differentials Following Steam Line Break Inside the Drywell 16.7% Power/110% Core Flow is no longer applicable, the information provided no longer applies. This information is obsolete.	
	Figure 3.3.8E	Steam Dryer Pressure Differential Following Steam Line Break Inside the Drywell 16.7% Power/110% Core Flow	Delete Figure 3.3.8E, Steam Dryer Pressure Differential Following Steam Line Break Inside the Drywell 16.7% Power/110% Core Flow. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. The functions performed by the reactor vessel internals are no longer required. Consequently, information regarding the Steam Dryer Pressure Differential Following Steam Line Break Inside the Drywell 16.7% Power/110% Core Flow is no longer applicable, the information provided no longer applies. This information is obsolete.	
	Figure 3.3.9	Thermal Shock Transient Analysis Zones	Delete Figure 3.3.9, Thermal Shock Transient Analysis Zones. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. The functions performed by the reactor vessel internals are no longer required. Consequently, information regarding the Thermal Shock Transient Analysis Zones is no longer applicable, the information provided no longer applies. This information is obsolete.	
	Figure 3.3.10	Material Behavior Graph Cycles Versus Stress for Stainless Steel	Delete Figure 3.3.10,Material Behavior Graph Cycles Versus Stress for Stainless Steel. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. The functions performed by the reactor vessel internals are no longer required. Consequently, information regarding the Material Behavior Graph Cycles Versus Stress for Stainless Steel is no longer applicable, the information provided no longer applies. This information is obsolete.	
	Figure 3.3.11	Deleted	Previously Deleted	
	Figure 3.3.12A	Vessel Dome Pressure Following Steam Line Break Inside the Drywell 102% Power/107% Core Flow	Delete Figure 3.3.12A, Vessel Dome Pressure Following Steam Line Break Inside the Drywell 102% Power/107% Core Flow. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. The functions performed by the reactor vessel internals are no longer required. Consequently, information regarding the Vessel Dome Pressure Following Steam Line Break Inside the Drywell 102% Power/107% Core Flow is no longer applicable, the information provided no longer applies. This information is obsolete.	
	Figure 3.3.12B	Vessel Dome Pressure Following Steam line Break Inside the Drywell 16.7% Power/110% Core Flow	Delete Figure 3.3.12B, Vessel Dome Pressure Following Steam Line Break Inside the Drywell 16.7% Power/110% Core Flow. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. The functions performed by the reactor vessel internals are no longer required. Consequently, information regarding the Vessel Dome Pressure Following Steam Line Break Inside the Drywell 16.7% Power/110% Core Flow is no longer applicable, the information provided no longer applies. This information is obsolete.	
	Figure 3.3.13A	MSLB Bundle Lift Margin at 102% Power/107% Core Flow	Delete Figure 3.3.13A, MSLB Bundle Lift Margin at 102% Power/107% Core Flow. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel i the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. The functions performed by the reactor vessel internals are no longer required. Consequently, information regarding the MSLB Bundle Lift Margin at 102% Power/107% Core Flow is no longer applicable, the information provided no longer applies. This information is obsolete.	
	Figure 3.3.13B	MSLB Bundle Lift Margin at 16.7% Power/110% Core Flow	Delete Figure 3.3.13B, MSLB Bundle Lift Margin at 16.7% Power/110% Core Flow. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. The functions performed by the reactor vessel internals are no longer required. Consequently, information regarding the MSLB Bundle Lift Margin at 16.7% Power/110% Core Flow is no longer applicable, the information provided no longer applies. This information is obsolete.	
3.4	REACTIVITY CONTROL Mech Design		Delete entire section, Sections 3.4.1 through 3.4.8 are no longer applicable, no information remains in this section	
	3.4.1	Power Generation Objective	Delete section 3.4.1, Reactivity Control Mech Design, Power Generation Objective. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since nuclear fuel can no longer be placed in the reactor vessel, reactivity control functions are no longer required. Consequently, the Reactivity Control Mech Design information is no longer applicable, the Power Generation Objective provided no longer applies. Reactivity Control Mech Design information is obsolete.	
	3.4.2	Safety Objectives	Delete section 3.4.2, Reactivity Control Mech Design, Safety Objective. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since nuclear fuel can no longer be placed in the reactor vessel, reactivity control functions are no longer required. Consequently, the Reactivity Control Mech Design information is no longer applicable, the Safety Objective provided no longer applies. Reactivity Control Mech Design information is obsolete.	

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	3.4.3	Power Generation Design Bases	Delete section 3.4.3, Reactivity Control Mech Design, Power Generation Design Basis. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since nuclear fuel can no longer be placed in the reactor vessel, reactivity control functions are no longer required. Consequently, the Reactivity Control Mech Design information is no longer applicable, the Power Generation Design Basis provided no longer applies. Reactivity Control Mech Design information is obsolete.	
	3.4.4	Safety Design Basis	Delete section 3.4.4, Reactivity Control Mech Design, Safety Design Basis. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since nuclear fuel can no longer be placed in the reactor vessel, reactivity control functions are no longer required. Consequently, the Reactivity Control Mech Design information is no longer applicable, the Safety Design Basis provided no longer applies. Reactivity Control Mech Design information is obsolete.	
	3.4.5	Description	Delete section 3.4.5, Reactivity Control Mech Design, Description. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since nuclear fuel can no longer be placed in the reactor vessel, reactivity control functions are no longer required. Consequently, the Reactivity Control Mech Design information is no longer applicable, the information provided in the description section no longer applies. Reactivity Control Mech Design information is obsolete.	
	3.4.6	Safety Evaluation	Delete section 3.4.6, Reactivity Control Mech Design, Safety Evaluation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since nuclear fuel can no longer be placed in the reactor vessel, reactivity control functions are no longer required. Consequently, the Reactivity Control Mech Design information is no longer applicable, the information provided in the safety evaluation section no longer applies. Reactivity Control Mech Design information is obsolete.	
	3.4.7	Inspection and Testing	Delete section 3.4.7, Reactivity Control Mech Design, Inspection and Testing. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since nuclear fuel can no longer be placed in the reactor vessel, reactivity control functions are no longer required. Consequently, the Reactivity Control Mech Design information is no longer applicable, the information provided in the Inspection and Testing section no longer applies. Reactivity Control Mech Design information is obsolete.	
	3.4.8	References	Delete section 3.4.8, Reactivity Control Mech Design, References. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since nuclear fuel can no longer be placed in the reactor vessel, reactivity control functions are no longer required. Consequently, the Reactivity Control Mech Design information is no longer applicable, the information provided in the References section no longer applies. Reactivity Control Mech Design information is obsolete.	
	Figure 3.4-1	Original Equipment Control Rod Isometric	Delete Figure 3.4-1, Original Equipment Control Rod Isometric. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since nuclear fuel can no longer be placed in the reactor vessel, reactivity control functions are no longer required. Consequently, Original Equipment Control Rod information is no longer applicable, the information provided in the Original Equipment Control Rod Isometric no longer applies. Reactivity Control Mech Design information is obsolete.	
	Figure 3.4-2	Typical Control Rod to Control Rod Drive Coupling - Isometric	Delete Figure 3.4-2, Typical Control Rod to Control Rod Drive Coupling - Isometric. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since nuclear fuel can no longer be placed in the reactor vessel, reactivity control functions are no longer required. Consequently, Typical Control Rod to Control Rod Drive Coupling information is no longer applicable, the information provided in the Typical Control Rod to Control Rod Drive Coupling - Isometric no longer applies. Reactivity Control Mech Design information is obsolete.	
	Figure 3.4-3	Typical Control Rod Velocity Limiter - Isometric	Delete Figure 3.4-3, Typical Control Rod Velocity Limiter - Isometric. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since nuclear fuel can no longer be placed in the reactor vessel, reactivity control functions are no longer required. Consequently, Typical Control Rod Velocity Limiter information is no longer applicable, the information provided in the Typical Control Rod Velocity Limiter - Isometric no longer applies. Reactivity Control Mech Design information is obsolete.	
	Figure 3.4-4	Control Rod Drive, Simplified Component Illustration	Delete Figure 3.4-4, Control Rod Drive, Simplified Component Illustration. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since nuclear fuel can no longer be placed in the reactor vessel, reactivity control functions are no longer required. Consequently, Control Rod Drive, Simplified Component information is no longer applicable, the information provided in the Control Rod Drive, Simplified Component Illustration no longer applies. Reactivity Control Mech Design information is obsolete.	
	Figure 3.4-5	Control Rod Drive, Schematic Diagram	Delete Figure 3.4-5, Control Rod Drive, Schematic Diagram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since nuclear fuel can no longer be placed in the reactor vessel, reactivity control functions are no longer required. Consequently, Control Rod Drive, information is no longer applicable, the information provided in the Control Rod Drive, Schematic Diagram no longer applies. Reactivity Control Mech Design information is obsolete.	

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	Figure 3.4-6	G-191170 Control Rod Drive Hydraulic Control System Simplified Component Illustration	Delete Figure 3.4-6, Control Rod Drive Hydraulic Control System Simplified Component Illustration. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since nuclear fuel can no longer be placed in the reactor vessel, reactivity control functions are no longer required. Consequently, Control Rod Drive information is no longer applicable, the information provided in the Control Rod Drive Hydraulic Control System Simplified Component Illustration no longer applies. Reactivity Control Mech Design information is obsolete.	
	Figure 3.4-7A	G-191170 Control Rod Drive Hydraulic System Piping and Instrumentation Diagram	Delete Figure 3.4-7A, Control Rod Drive Hydraulic System Piping and Instrumentation Diagram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since nuclear fuel can no longer be placed in the reactor vessel, reactivity control functions are no longer required. Consequently, Control Rod Drive information is no longer applicable, the information provided in the Control Rod Drive Hydraulic System Piping and Instrumentation Diagram no longer applies. Reactivity Control Mech Design information is obsolete.	
	Figure 3.4-7B	G-191170 Hydraulic Control Unit, Piping and Instrumentation Diagram	Delete Figure 3.4-7B, Control Rod Drive Hydraulic System Piping and Instrumentation Diagram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since nuclear fuel can no longer be placed in the reactor vessel, reactivity control functions are no longer required. Consequently, Control Rod Drive information is no longer applicable, the information provided in the Control Rod Drive Hydraulic System Piping and Instrumentation Diagram no longer applies. Reactivity Control Mech Design information is obsolete.	
	Figure 3.4-8	Control Rod Unit Drive--Cutaway Illustration	Delete Figure 3.4-8, Control Rod Unit Drive--Cutaway Illustration. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since nuclear fuel can no longer be placed in the reactor vessel, reactivity control functions are no longer required. Consequently, Control Rod Drive information is no longer applicable, the information provided in the Control Rod Unit Drive--Cutaway Illustration no longer applies. Reactivity Control Mech Design information is obsolete.	
	Figure 3.4-9	5920-164 Control Rod Drive Hydraulic System, Process Diagram	Delete Figure 3.4-9, Control Rod Drive Hydraulic System, Process Diagram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since nuclear fuel can no longer be placed in the reactor vessel, reactivity control functions are no longer required. Consequently, Control Rod Drive information is no longer applicable, the information provided in the Control Rod Drive Hydraulic System, Process Diagram no longer applies. Reactivity Control Mech Design information is obsolete.	
	Figure 3.4-10	Control Rod Hydraulic Control Unit--Isometric	Delete Figure 3.4-10, Control Rod Hydraulic Control Unit--Isometric. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since nuclear fuel can no longer be placed in the reactor vessel, reactivity control functions are no longer required. Consequently, Control Rod Drive information is no longer applicable, the information provided in the Control Rod Hydraulic Control Unit--Isometric no longer applies. Reactivity Control Mech Design information is obsolete.	
3.5	CONTROL ROD DRIVE HOUSING SUPPORTS		Delete entire section, Sections 3.5.1 through 3.5.5 are no longer applicable, no information remains in this section	
	3.5.1	Safety Objective	Delete section 3.5.1, Safety Objective. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Control Rod Drive Housing Supports are no longer required, the safety objective is no longer applicable.Control Rod Drive Housing Support System information is obsolete.	
	3.5.2	Safety Design Basis	Delete section 3.5.2, Safety Design Bases. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Control Rod Drive Housing Supports are no longer required, the safety design bases are no longer applicable.Control Rod Drive Housing Support System information is obsolete.	
	3.5.3	Description	Delete section 3.5.3, CRD Housing Support Description. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Control Rod Drive Housing Supports are no longer required, the information provided in the description section is no longer applicableControl Rod Drive Housing Support System information is obsolete.	
	3.5.4	Safety Evaluation	Delete section 3.5.4, CRD Housing Support Safety Evaluation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Control Rod Drive Housing Supports are no longer required, the information provided in the safety evaluation section is no longer applicableControl Rod Drive Housing Support System information is obsolete.	
	3.5.5	Inspection and Testing	Delete section 3.5.5, CRD Housing Support Inspection and Testing. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Control Rod Drive Housing Supports are no longer required, the information provided in the Inspection and Testing section is no longer applicableControl Rod Drive Housing Support System information is obsolete.	
	Figure 3.5-1	Control Rod Drive Housing Support Isometric	Delete Figure 3.5-1, Control Rod Drive Housing Support Isometric. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Control Rod Drive Housing Supports are no longer required, the information provided in the Control Rod Drive Housing Support Isometric is no longer applicable. Control Rod Drive Housing Support System information is obsolete.	

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UFSAR Section			FSAR Conversion to DSAR Change Summary	DSAR Section
3.6	NUCLEAR DESIGN			
	3.6.1	Power Generation Objectives	Section 3.6.1, Power Generation Objectives, Delete this section, post defueling the licensee will no longer be authorized to operate the reactor or emplace or retain fuel in the reactor vessel. Therefore, power operations cannot occur and these objectives no longer apply	
	3.6.2	Power Generation Design Bases	Section 3.6.2, Power Generation Design Bases, Delete this section, post defueling the licensee will no longer be authorized to operate the reactor or emplace or retain fuel in the reactor vessel. Therefore, power operations cannot occur and these design bases no longer apply	
	3.6.3	Safety Design Basis	Section 3.6.3, Safety Design Bases, Delete this section, post defueling the licensee will no longer be authorized to operate the reactor or emplace or retain fuel in the reactor vessel. Therefore power operations cannot occur and these Safety Design Bases no longer apply	
	3.6.4	Nuclear Requirements	Section 3.6.4, Nuclear Requirements, Delete this section, post defueling the licensee will no longer be authorized to operate the reactor or emplace or retain fuel in the reactor vessel Therefore, power operations cannot occur and these nuclear requirements no longer apply: - Control rods will be removed from core. - Reactor Manual control system will no longer be required - Standby liquid control system will no longer be required since there will be no core	
	3.6.5	Fuel Nuclear Characteristics	Section 3.6.5, Fuel Nuclear Characteristics, Delete this section, post defueling the licensee will no longer be authorized to operate the reactor or emplace or retain fuel in the reactor vessel. Therefore, the reactor will not be taken critical or operated. Fuel Nuclear characteristics no longer apply. Reactivity control will be provided by fuel placement in the Spent Fuel Pool and pool design characteristics, control rod worth is no longer applicable since control rods will be removed from the RPV, reactivity coefficients are no longer applicable and xenon transients cannot occur.	
	3.6.6	Nuclear Evaluations	Section 3.6.6, Nuclear Evaluations, Delete this section, post defueling the licensee will no longer be authorized to operate the reactor or emplace or retain fuel in the reactor vessel. Therefore power operations cannot occur and these nuclear evaluations are no longer required	
	3.6.7	Testing and Verification	Delete this section, post defueling the licensee will no longer be authorized to operate the reactor or emplace or retain fuel in the reactor vessel. Therefore, power operations cannot occur and testing and verification of nuclear design criteria are no longer required.	
	3.6.8	References	Delete Section 4.2.7, References. Post certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Power operation can no longer occur nor will it be possible to load a nuclear core. All control rods have been removed. All nuclear instrumentation has been removed. Consequently, the functions of a reactor vessel are no longer required. Inspection and testing of the Reactor Pressure Vessel is no longer required. Therefore, the listed references no longer apply.	
	Figure 3.6-1	Relative Xenon Stability With No Flux Flattening	Delete Figure 3.6-1 , Relative Xenon Stability With No Flux Flattening, Post defueling the licensee will no longer be authorized to operate the reactor or emplace or retain fuel in the reactor vessel. Therefore, the reactor will not be taken critical or operated. Fuel Nuclear characteristics no longer apply. Reactivity control will be provided by fuel placement in the Spent Fuel Pool and pool design characteristics, control rod worth is no longer applicable since control rods will be removed from the RPV, reactivity coefficients are no longer applicable and xenon transients cannot occur. Therefore, the information provided by Figure 3.6-1 is no longer applicable and is obsolete.	
	Figure 3.6-2	Effect of Power Density on Axial Xenon Stability Including Void Transport	Delete Figure 3.6-2, Effect of Power Density on Axial Xenon Stability Including Void Transport. Post defueling the licensee will no longer be authorized to operate the reactor or emplace or retain fuel in the reactor vessel. Therefore, the reactor will not be taken critical or operated. Fuel Nuclear characteristics no longer apply. Reactivity control will be provided by fuel placement in the Spent Fuel Pool and pool design characteristics, control rod worth is no longer applicable since control rods will be removed from the RPV, reactivity coefficients are no longer applicable and xenon transients cannot occur. Therefore, the information provided by Figure 3.6-2 is no longer applicable and is obsolete.	
	Figure 3.6-3	Azimuthal Xenon Stability	Delete Figure 3.6-3, Azimuthal Xenon Stability. Post defueling the licensee will no longer be authorized to operate the reactor or emplace or retain fuel in the reactor vessel. Therefore, the reactor will not be taken critical or operated. Fuel Nuclear characteristics no longer apply. Reactivity control will be provided by fuel placement in the Spent Fuel Pool and pool design characteristics, control rod worth is no longer applicable since control rods will be removed from the RPV, reactivity coefficients are no longer applicable and xenon transients cannot occur. Therefore, the information provided by Figure 3.6-3 is no longer applicable and is obsolete.	
3.7	THERMAL AND HYDRAULIC DESIGN			
	3.7.1	Power Generation Objective	Delete Section 3.7.1, Power Generation Objective. Post certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Power generation can no longer occur nor will it be possible to load a nuclear core. Consequently, core thermal and hydraulic design is no longer required and power generation objectives are obsolete information.	
	3.7.2	Power Generation Design Bases	Delete Section 3.7.2, Power Generation Design Bases. Post certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Power generation can no longer occur nor will it be possible to load a nuclear core. Consequently, core thermal and hydraulic design is no longer required and power generation design bases are obsolete information.	
	3.7.3	Safety Design Basis	Delete Section 3.7.3, Safety Design Bases. Post certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2) Power generation can no longer occur nor will it be possible to load a nuclear core. Consequently, core thermal and hydraulic design is no longer required and safety design bases are obsolete information.	
	3.7.4	Thermal and Hydraulic Limits	Delete Section 3.7.4, Thermal and Hydraulic Limits. Post certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Power generation can no longer occur nor will it be possible to load a nuclear core. Consequently, core thermal and hydraulic limits are no longer required and are obsolete information.	
	3.7.5	Thermal and Hydraulic Characteristics	Delete Section 3.7.5, Thermal and Hydraulic Characteristics. Post certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Power generation can no longer occur nor will it be possible to load a nuclear core. Consequently, core thermal and hydraulic characteristics are no longer required and are obsolete information	

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	3.7.6	Thermal and Hydraulic Evaluation	Delete Section 3.7.6, Thermal and Hydraulic Evaluation. Post certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Power generation can no longer occur nor will it be possible to load a nuclear core. Consequently, the core thermal and hydraulic evaluation is no longer required and is obsolete information. The design MCPR limit no longer applies, fuel damage or waterlogging cannot occur due to exceeding limits, and flow blockages can no longer occur.	
	3.7.7	Testing and Verification	Delete Section 3.7.7, Testing and Verification. Post certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Power generation can no longer occur nor will it be possible to load a nuclear core. Consequently, testing and verification that operation is occurring within MCPR and LHGR limits is no longer required and is obsolete information.	
	3.7.8	References	Delete Section 3.7.8, References. Since this section is being deleted completely, these references are no longer required and are obsolete information.	
3.8	STANDBY LIQUID CONTROL SYSTEM			
	3.8.1	Safety Objective	Delete Section 3.8.1, Safety Objective. Post certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Power generation can no longer occur nor will it be possible to load a nuclear core. Consequently, a method of shutting down the reactor independent of control rods is no longer required, nor is torus pH control required following a LOCA. These objectives are obsolete information.	
	3.8.2	Safety Design Bases	Delete Section 3.8.2, Safety Design Bases. Post certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Power generation can no longer occur nor will it be possible to load a nuclear core. Consequently, a method of shutting down the reactor independent of control rods is no longer required, nor is torus pH control required following a LOCA, since a LOCA can no longer occur. These design bases are obsolete information.	
	3.8.3	Description	Delete Section 3.8.3, Description. Post certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Power generation can no longer occur nor will it be possible to load a nuclear core. Consequently, a method of shutting down the reactor independent of control rods is no longer required, nor is torus pH control required following a LOCA. The functions of the SBLC system are no longer required.	
	3.8.4	Safety Evaluation	Delete Section 3.8.4, Safety Evaluation. Post certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Power generation can no longer occur nor will it be possible to load a nuclear core. Consequently, a method of shutting down the reactor independent of control rods is no longer required, nor is torus pH control required following a LOCA. The functions of the SBLC system are no longer required. Therefore, a safety evaluation of the SBLC system is no longer required and is obsolete information.	
	3.8.5	Inspection and Testing	Delete Section 3.8.5, Inspection and Testing. Post certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Power generation can no longer occur nor will it be possible to load a nuclear core. Consequently, a method of shutting down the reactor independent of control rods is no longer required, nor is torus pH control required following a LOCA. The functions of the SBLC system are no longer required. Therefore, inspection and testing of the SBLC system is no longer required and is obsolete information. Technical Specification surveillance requirements were deleted from Technical Specifications in permanently defueled technical specifications.	
	Figure 3.8-1	Standby Liquid Control System, Piping and Instrumentation Diagram G191171	Delete Figure 3.8-1, Standby Liquid Control System, Piping and Instrumentation Diagram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Power generation can no longer occur nor will it be possible to load a nuclear core. Consequently, a method of shutting down the reactor independent of control rods is no longer required, nor is torus pH control required following a LOCA. The functions of the SBLC system are no longer required. Consequently, the information provided on Figure 3.8-1 is no longer applicable. Information regarding the Standby Liquid Control System is obsolete.	
	Figure 3.8-2	Standby Liquid Control System, Process Diagram 5920-717	Delete Figure 3.8-2, Standby Liquid Control System, Process Diagram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Power generation can no longer occur nor will it be possible to load a nuclear core. Consequently, a method of shutting down the reactor independent of control rods is no longer required, nor is torus pH control required following a LOCA. The functions of the SBLC system are no longer required. Consequently, the information provided on Figure 3.8-2 is no longer applicable. Information regarding the Standby Liquid Control System is obsolete.	
	Figure 3.8-3	Saturation Temperature of Sodium Pentaborate Solution	Delete Figure 3.8-3, Saturation Temperature of Sodium Pentaborate Solution. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Power generation can no longer occur nor will it be possible to load a nuclear core. Consequently, a method of shutting down the reactor independent of control rods is no longer required, nor is torus pH control required following a LOCA. The functions of the SBLC system are no longer required. Consequently, the information provided on Figure 3.8-3 is no longer applicable. Information regarding the Standby Liquid Control System is obsolete.	
	Figure 3.8-4	Standby Liquid Control System, Functional Control Diagram 5920-40	Delete Figure 3.8-4, Standby Liquid Control System, Functional Control Diagram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Power generation can no longer occur nor will it be possible to load a nuclear core. Consequently, a method of shutting down the reactor independent of control rods is no longer required, nor is torus pH control required following a LOCA. The functions of the SBLC system are no longer required. Consequently, the information provided on Figure 3.8-4 is no longer applicable. Information regarding the Standby Liquid Control System is obsolete.	
	Figure 3.8-5	Sodium Pentaborate Solution Volume-- Concentration Requirements	Delete Figure 3.8-5, Sodium Pentaborate Solution Volume-- Concentration Requirements. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Power generation can no longer occur nor will it be possible to load a nuclear core. Consequently, a method of shutting down the reactor independent of control rods is no longer required, nor is torus pH control required following a LOCA. The functions of the SBLC system are no longer required. Consequently, the information provided on Figure 3.8-5 is no longer applicable. Information regarding the Standby Liquid Control System is obsolete.	
4.0	**Reactor Coolant System**			

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UFSAR Section				FSAR Conversion to DSAR Change Summary	DSAR Section
4.1	SUMMARY DESCRIPTION			Delete Section 4.1, Reactor Coolant System Summary Description. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Power generation can no longer occur nor will it be possible to load a nuclear core. Consequently the structures, systems and components discussed in section 4.1 are no longer required and the information provided is no longer applicable.	
	Table 4.1.1	Equipment Within Reactor Coolant Pressure Boundary Comparison With 10CFR50.55A		Delete Table 4.1.1, Equipment Within Reactor Coolant Pressure Boundary Comparison With 10CFR50.55A. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Power generation can no longer occur nor will it be possible to load a nuclear core. Consequently the structures, systems and components discussed in section 4.1 are no longer required and the information provided in Table 4.1.1 is no longer applicable.	
4.2	REACTOR VESSEL AND APPURTENANCES				
	4.2.1	Power Generation Objectives		Section 4.2.1, Power Generation Objectives, Delete this section, post defueling the licensee will no longer be authorized to operate the reactor or emplace or retain fuel in the reactor vessel. Therefore, power operations cannot occur and these objectives no longer apply.	
	4.2.2	Power Generation Design Bases		Section 4.2.2, Power Generation Design Bases, Delete this section, post defueling the licensee will no longer be authorized to operate the reactor or emplace or retain fuel in the reactor vessel. Therefore, power operations cannot occur and these reactor vessel design bases no longer apply.	
	4.2.3	Safety Design Bases		Section 4.2.3, Safety Design Bases, Delete this section, post defueling the licensee will no longer be authorized to operate the reactor or emplace or retain fuel in the reactor vessel. Therefore, power operations cannot occur and these Safety Design Bases no longer apply.	
	4.2.4	Description			
		4.2.4.1	Reactor Vessel	Delete Section 4.2.4.1, Reactor Vessel. Post certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Power generation can no longer occur nor will it be possible to load a nuclear core. Consequently, the functions of a reactor vessel are no longer required.	
		4.2.4.2	Shroud Support	Delete Section 4.2.4.2, Shroud Support. Post certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Power generation can no longer occur nor will it be possible to load a nuclear core. Consequently, the functions of a reactor vessel, including the shroud and shroud support are no longer required.	
		4.2.4.3	Reactor Vesseil Support Assembly	Delete Section 4.2.4.3, Reactor Vessel Support Assembly. Post certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Power generation can no longer occur nor will it be possible to load a nuclear core. Consequently, the functions of a reactor vessel, including the Reactor Vessel Support Assembly are no longer required. .	
		4.2.4.4	Vessel Stabilizers	Delete Section 4.2.4.4, Vessel Stabilizers. Post certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Power generation can no longer occur nor will it be possible to load a nuclear core. Consequently, the functions of a reactor vessel, including the Reactor Vessel Stabilizers are no longer required.	
		4.2.4.5	Refueling Bellows	Delete Section 4.2.4.5, Refueling Bellows. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Power generation can no longer occur nor will it be permissible to load a nuclear core. Consequently, the functions of a reactor vessel, including the Refueling Bellows are no longer required. Therefore, the information provided regarding the refueling bellows is no longer required and is considered obsolete.	
		4.2.4.6	Control Rod Drive Housings	Delete Section 4.2.4.6, Control Rod Drive Housings. Post certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Power operation can no longer occur nor will it be possible to load a nuclear core. All control rods have been removed. Consequently, the functions of a reactor vessel, including the Control Rod Drive Housings are no longer required.	
		4.2.4.7	Control Rod Drive Housing Supports	Delete Section 4.2.4.7, Control Rod Drive Housing Supports. Post certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Power operation can no longer occur nor will it be possible to load a nuclear core. All control rods have been removed. Consequently, the functions of a reactor vessel, including the Control Rod Drive Housing supports are no longer required.	
		4.2.4.8	Incore Neutron Flux Monitor Housings	Delete Section 4.2.4.8, Incore Neutron Flux Monitor Housings. Post certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Power operation can no longer occur nor will it be possible to load a nuclear core. All control rods have been removed. All nuclear instrumentation has been removed. Consequently, the functions of a reactor vessel, including the Incore Neutron Flux Monitor Housings are no longer required.	
		4.2.4.9	Reactor Vessel Insulation	Delete Section 4.2.4.9, Reactor Vessel Insulation. Post certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Power operation can no longer occur nor will it be possible to load a nuclear core. All control rods have been removed. All nuclear instrumentation has been removed. Consequently, the functions of a reactor vessel, including the Reactor Vessel Insulation are no longer required.	
	4.2.5	Safety Evaluation		Delete Section 4.2.5, Safety Evaluation. Post certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Power operation can no longer occur nor will it be possible to load a nuclear core. All control rods have been removed. All nuclear instrumentation has been removed. Consequently, the functions of a reactor vessel are no longer required. Therefore, a safety evaluation of the reactor pressure vessel is no longer required and is obsolete information.	
	4.2.6	Inspection and Testing		Delete Section 4.2.6, Inspection and Testing. Post certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Power operation can no longer occur nor will it be possible to load a nuclear core. All control rods have been removed. All nuclear instrumentation has been removed. Consequently, the functions of a reactor vessel are no longer required.Therefore, inspection and testing of the Reactor Pressure Vessel is no longer required.	

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	4.2.7	References	Delete Section 4.2.7, References. Post certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Power operation can no longer occur nor will it be possible to load a nuclear core. All control rods have been removed. All nuclear instrumentation has been removed. Consequently, the functions of a reactor vessel are no longer required. Inspection and testing of the Reactor Pressure Vessel is no longer required. Therefore, the listed references no longer apply.	
	Table 4.2.1	Reactor Pressure Vessel Materials	Delete Table 4.2.1, Reactor Pressure Vessel Materials. Post certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Power operation can no longer occur nor will it be possible to load a nuclear core. All control rods have been removed. All nuclear instrumentation has been removed. Consequently, the functions of a reactor vessel are no longer required. Inspection and testing of the Reactor Pressure Vessel is no longer required. Therefore, the list of Reactor Pressure Vessel Materials is obsolete information.	
	Table 4.2.2	Reactor Vessel Data	Delete Table 4.2.2, Reactor Vessel Data. Post certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Power operation can no longer occur nor will it be possible to load a nuclear core. All control rods have been removed. All nuclear instrumentation has been removed. Consequently, the functions of a reactor vessel are no longer required. Inspection and testing of the Reactor Pressure Vessel is no longer required. Therefore, the list of Reactor Vessel Data is obsolete information	
	Table 4.2.3	Reactor Vessel Attachments	Delete Table 4.2.3, Reactor Vessel Attachments. Post certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Power operation can no longer occur nor will it be possible to load a nuclear core. All control rods have been removed. All nuclear instrumentation has been removed. Consequently, the functions of a reactor vessel are no longer required. Inspection and testing of the Reactor Pressure Vessel is no longer required. Therefore, the list of Reactor Vessel Attachments is obsolete information	
	Table 4.2.4	Surveillance Capsule Removal Schedule	Delete Table 4.2.4, Surveillance Capsule Removal Schedule. Post certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Power operation can no longer occur nor will it be possible to load a nuclear core. All control rods have been removed. All nuclear instrumentation has been removed. Consequently, the functions of a reactor vessel are no longer required. Inspection and testing of the Reactor Pressure Vessel is no longer required. Therefore, the Surveillance Capsule Removal Schedule is obsolete information.	
	Figure 4.2-1	Reactor Vessel	Delete Figure 4.2.1, Reactor Vessel. Post certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Power generation can no longer occur nor will it be possible to load a nuclear core. Consequently, the functions of a reactor vessel are no longer required. Therefore, the information provided on Figure 4.2.1, Reactor Vessel is no longer applicable. Information regarding the reactor vessel is obsolete.	
	Figure 4.2-2	Reactor Vessel Nozzles and Penetrations	Delete Figure 4.2.2, Reactor Vessel Nozzles and Penetrations. Post certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Power generation can no longer occur nor will it be possible to load a nuclear core. Consequently, the functions of a reactor vessel are no longer required. Therefore, the information provided on Figure 4.2.2, Reactor Vessel nozzles and penetrations is no longer applicable. Information regarding the reactor vessel is obsolete.	
	Figure 4.2-3	Change in NDTT Versus Neutron Exposure	Delete Figure 4.2.3, Change in NDTT Versus Neutron Exposure. Post certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Power generation can no longer occur nor will it be possible to load a nuclear core. Consequently, the functions of a reactor vessel are no longer required. Therefore, the information provided on Figure 4.2.3, Change in NDTT Versus Neutron Exposure is no longer applicable. Information regarding the reactor vessel is obsolete.	
	Figure 4.2-4	Deleted	Previously Deleted	
4.3	REACTOR RECIRCULATION SYSTEM		The Reactor Recirculation System provided a variable moderator (coolant) flow to the reactor core to adjust reactor power level. The system was designed to assure adequate fuel barrier thermal margin remained following Recirculation Pump System malfunctions and that failure of piping integrity did not compromise the ability of the reactor vessel internals to maintain a refloodable volume.	
	4.3.1	Power Generation Objectives	Delete Section 4.3.1, Power Generation Objectives. The Reactor Recirculation System provided a variable moderator (coolant) flow to the reactor core to adjust reactor power level. The system was designed to assure adequate fuel barrier thermal margin remained following Recirculation Pump System malfunctions and that failure of piping integrity did not compromise the ability of the reactor vessel internals to maintain a refloodable volume. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Reactor Recirculation System are no longer required. Reactor Recirculation System information is obsolete.	
	4.3.2	Power Generation Design Basis	Delete Section 4.3.2, Power Generation Design Basis. The Reactor Recirculation System provided a variable moderator (coolant) flow to the reactor core to adjust reactor power level. The system was designed to assure adequate fuel barrier thermal margin remained following Recirculation Pump System malfunctions and that failure of piping integrity did not compromise the ability of the reactor vessel internals to maintain a refloodable volume. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Reactor Recirculation System are no longer required. Reactor Recirculation System information is obsolete.	
	4.3.3	Safety Design Basis	Delete Section 4.3.3, Safety Design Basis. The Reactor Recirculation System provided a variable moderator (coolant) flow to the reactor core to adjust reactor power level. The system was designed to assure adequate fuel barrier thermal margin remained following Recirculation Pump System malfunctions and that failure of piping integrity did not compromise the ability of the reactor vessel internals to maintain a refloodable volume. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Reactor Recirculation System are no longer required. Reactor Recirculation System information is obsolete.	

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	4.3.4	Description	Delete Section 4.3.4, Description. The Reactor Recirculation System provided a variable moderator (coolant) flow to the reactor core to adjust reactor power level. The system was designed to assure adequate fuel barrier thermal margin remained following Recirculation Pump System malfunctions and that failure of piping integrity did not compromise the ability of the reactor vessel internals to maintain a refloodable volume. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Reactor Recirculation System are no longer required. Reactor Recirculation System information is obsolete.	
	4.3.5	Safety Evaluation	Delete Section 4.3.5, Safety Evaluation. The Reactor Recirculation System provided a variable moderator (coolant) flow to the reactor core to adjust reactor power level. The system was designed to assure adequate fuel barrier thermal margin remained following Recirculation Pump System malfunctions and that failure of piping integrity did not compromise the ability of the reactor vessel internals to maintain a refloodable volume. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Reactor Recirculation System are no longer required. Therefore, a Reactor Recirculation Safety Evaluation is no longer required. Reactor Recirculation System information is obsolete.	
	4.3.6	Inspection and Testing	Delete Section 4.3.6, Inspection and Testing. The Reactor Recirculation System provided a variable moderator (coolant) flow to the reactor core to adjust reactor power level. The system was designed to assure adequate fuel barrier thermal margin remained following Recirculation Pump System malfunctions and that failure of piping integrity did not compromise the ability of the reactor vessel internals to maintain a refloodable volume. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Reactor Recirculation System are no longer required. Therefore, inspection and testing of the Reactor Recirculation System is no longer required. Reactor Recirculation System inspection and testing information is obsolete.	
	4.3.7	Safety Evaluation	Delete Section 4.3.7, Safety Evaluation. The Reactor Recirculation System provided a variable moderator (coolant) flow to the reactor core to adjust reactor power level. The system was designed to assure adequate fuel barrier thermal margin remained following Recirculation Pump System malfunctions and that failure of piping integrity did not compromise the ability of the reactor vessel internals to maintain a refloodable volume. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Reactor Recirculation System are no longer required. Therefore, Reactor Recirculation System references are no longer required. Reactor Recirculation System references information is obsolete.	
	Table 4.3.1	Reactor Recirculation System Design Characteristics	Delete Table 4.3.1, Reactor Recirculation System Design Characteristics. The Reactor Recirculation System provided a variable moderator (coolant) flow to the reactor core to adjust reactor power level. The system was designed to assure adequate fuel barrier thermal margin remained following Recirculation Pump System malfunctions and that failure of piping integrity did not compromise the ability of the reactor vessel internals to maintain a refloodable volume. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Reactor Recirculation System are no longer required. Therefore, Reactor Recirculation System Design Characteristics information is obsolete.	
	Figure 4.3-1	Recirculation System Elevation, Isometric	Delete Figure 4.3-1, Recirculation System Elevation, Isometric. The Reactor Recirculation System provided a variable moderator (coolant) flow to the reactor core to adjust reactor power level. The system was designed to assure adequate fuel barrier thermal margin remained following Recirculation Pump System malfunctions and that failure of piping integrity did not compromise the ability of the reactor vessel internals to maintain a refloodable volume. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Reactor Recirculation System are no longer required. Therefore, the information provided on Figure 4.3-1 is no longer applicable. Reactor Recirculation System information is obsolete.	
	Figure 4.3-2	Nuclear Boiler, Piping and Instrumentation Diagram G-191167	Delete Figure 4.3-2, Nuclear Boiler, Piping and Instrumentation Diagram. The Reactor Recirculation System provided a variable moderator (coolant) flow to the reactor core to adjust reactor power level. The system was designed to assure adequate fuel barrier thermal margin remained following Recirculation Pump System malfunctions and that failure of piping integrity did not compromise the ability of the reactor vessel internals to maintain a refloodable volume. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Reactor Recirculation System are no longer required. Therefore, the information provided on Figure 4.3-2 is no longer applicable. Reactor Recirculation System information is obsolete.	

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	Figure 4.3-3	Recirculation Pump Piping and Instrumentation Diagram G-191159, Sh5	Delete Figure 4.3-3, Recirculation Pump Piping and Instrumentation Diagram. The Reactor Recirculation System provided a variable moderator (coolant) flow to the reactor core to adjust reactor power level. The system was designed to assure adequate fuel barrier thermal margin remained following Recirculation Pump System malfunctions and that failure of piping integrity did not compromise the ability of the reactor vessel internals to maintain a refloodable volume. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Reactor Recirculation System are no longer required. Therefore, the information provided on Figure 4.3-3 is no longer applicable. Reactor Recirculation System information is obsolete.	
	Figure 4.3-4	Jet Pump Operating Principle	Delete Figure 4.3-4, Jet Pump Operating Principle. The Reactor Recirculation System provided a variable moderator (coolant) flow to the reactor core to adjust reactor power level. The system was designed to assure adequate fuel barrier thermal margin remained following Recirculation Pump System malfunctions and that failure of piping integrity did not compromise the ability of the reactor vessel internals to maintain a refloodable volume. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Reactor Recirculation System are no longer required. Therefore, the information provided on Figure 4.3-4 is no longer applicable. Reactor Recirculation System information is obsolete.	
	Figure 4.3-5	Recirculation System Core Flooding Capability	Delete Figure 4.3-5, Recirculation System Core Flooding Capability. The Reactor Recirculation System provided a variable moderator (coolant) flow to the reactor core to adjust reactor power level. The system was designed to assure adequate fuel barrier thermal margin remained following Recirculation Pump System malfunctions and that failure of piping integrity did not compromise the ability of the reactor vessel internals to maintain a refloodable volume. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Reactor Recirculation System are no longer required. Therefore, the information provided on Figure 4.3-5 is no longer applicable. Reactor Recirculation System information is obsolete.	
4.4	Nuclear System Pressure Relief System			
	4.4.1	Power Generation Objectives	Delete Section 4.4.1, Power Generation Objectives. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of a Nuclear System Pressure Relief System are no longer required. Nuclear System Pressure Relief System information is obsolete.	
	4.4.2	Safety Objectives	Delete Section 4.4.2, Safety Objectives. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of a Nuclear System Pressure Relief System are no longer required. Nuclear System Pressure Relief System information is obsolete.	
	4.4.3	Power Generation Design Bases	Delete Section 4.4.3, Power Generation Design Bases. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of a Nuclear System Pressure Relief System are no longer required. Nuclear System Pressure Relief System information is obsolete	
	4.4.4	Safety Design Bases	Delete Section 4.4.4, Safety Design Bases. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and the RPV cannot be pressurized. Consequently, depressurization is not required Therefore, the functions of a Nuclear System Pressure Relief System are no longer required. Nuclear System Pressure Relief System information is obsolete.	
	4.4.5	Description	Delete Section 4.4.5, Description. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and the RPV cannot be pressurized. Depressurization is no longer required. Consequently, the functions of Nuclear System Pressure Relief System are no longer required. Nuclear System Pressure Relief System information is obsolete.	
	4.4.6	Safety Evaluation	Delete Section 4.4.6, Safety Evaluation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and the RPV cannot be pressurized. Depressurization is no longer required. Consequently, the functions of a Nuclear System Pressure Relief System are no longer required. A Nuclear System Pressure Relief System safety evaluation is not required, the Nuclear System Pressure Relief system information is obsolete.	
	4.4.7	Power Generation Evaluation	Delete Section 4.4.7, Power Gneration Evaluation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and the RPV cannot be pressurized. Depressurization is no longer required. Consequently, the functions of a Nuclear System Pressure Relief System are no longer required. A Nuclear System Pressure Relief System power generation evaluation is not required, the Nuclear System Pressure Relief system information is obsolete.	
	4.4.8	Inspection and Testing	Delete Section 4.4.8, Inspection and Testing. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and the RPV cannot be pressurized. Depressurization is no longer required. Consequently, the functions of a Nuclear System Pressure Relief System are no longer required. Nuclear System Pressure Relief System inspection and testing is not required, the Nuclear System Pressure Relief system information is obsolete.	

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	4.4.9	References	Delete Section 4.4.9, References. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and the RPV cannot be pressurized. Depressurization is no longer required. Consequently, the functions of Nuclear System Pressure Relief System are no longer required. A Nuclear System Pressure Relief System references no longer apply; the Nuclear System Pressure Relief system information is obsolete.	
	Table 4.4.1	Nuclear System Safety and Relief Valves	Delete Table 4.4.1, Nuclear System Safety and Relief Valves. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and the RPV cannot be pressurized. Depressurization is no longer required. Consequently, the functions of a Nuclear System Pressure Relief System are no longer required. The Nuclear System Pressure Relief system information is obsolete.	
	Figure 4.4-1	Nuclear System Pressure Relief System, Piping and Instrumentation Diagram G-191167	Delete Figure 4.4-1, Nuclear System Pressure Relief System, Piping and Instrumentation Diagram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and the RPV cannot be pressurized. Depressurization is no longer required. Consequently, the functions of a Nuclear System Pressure Relief System are no longer required, the information provided on figure 4.4-1 is no longer applicable. Nuclear System Pressure Relief System information is obsolete.	
	Figure 4.4-2	Nuclear System Relief Valve Closed Position	Delete Figure 4.4-2, Nuclear System Relief Valve Closed Position. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and the RPV cannot be pressurized. Depressurization is no longer required. Consequently, the functions of a Nuclear System Pressure Relief System are no longer required, the information provided on figure 4.4-2 is no longer applicable. Nuclear System Pressure Relief System information is obsolete.	
	Figure 4.4-3	Nuclear System Relief Valve Open Position	Delete Figure 4.4-3, Nuclear System Relief Valve Open Position. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and the RPV cannot be pressurized. Depressurization is no longer required. Consequently, the functions of a Nuclear System Pressure Relief System are no longer required, the information provided on figure 4.4-3 is no longer applicable. Nuclear System Pressure Relief System information is obsolete.	
	Figure 4.4-4	Previously Deleted	Previously Deleted	
4.5	MAIN STEAM LINE FLOW RESTRICTOR			
	4.5.1	Safety Objective	1. Delete section 4.5.1, Safety Objective. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur, steam cannot be generated, a steam line break cannot occur. Consequently, the functions of Main Steam Line Flow Restrictors are no longer required. Main Steam Line Flow Restrictor information is obsolete.	
	4.5.2	Safety Design Bases	2. Delete section 4.5.2, Safety Design Basis. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur, steam cannot be generated, a steam line break cannot occur. Consequently, the functions of Main Steam Line Flow Restrictors are no longer required. Main Steam Line Flow Restrictor information is obsolete.	
	4.5.3	Description	3. Delete section 4.5.3, Description. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur, steam cannot be generated, a steam line break cannot occur. Consequently, the functions of Main Steam Line Flow Restrictors are no longer required. Main Steam Line Flow Restrictor information is obsolete.	
	4.5.4	Safety Design Bases	4. Delete section 4.5.4, Safety Design Basis. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur, steam cannot be generated, a steam line break cannot occur. Consequently, the functions of Main Steam Line Flow Restrictors are no longer required. A Main Steam Line Flow Restrictor Safety Evaluation is not required. Main Steam Line Flow Restrictor information is obsolete.	
	4.5.5	Inspection and Testing	5. Delete section 4.5.5, Inspection and Testing. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur, steam cannot be generated, a steam line break cannot occur. Consequently, the functions of Main Steam Line Flow Restrictors are no longer required. Main Steam Line Flow Restrictor Inspection and Testing is not required. Main Steam Line Flow Restrictor information is obsolete.	
	Figure 4.5-1	Main Steamline Flow Restrictor Location	Delete Figure 4.5-1, Main Steamline Flow Restrictor Location. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur, steam cannot be generated, a steam line break cannot occur. Consequently, the functions of Main Steam Line Flow Restrictors are no longer required. Therefore, the information provided on figure 4.5-1 is no longer applicable. Main Steam Line Flow Restrictor information is obsolete.	
4.6	MAIN STEAM LINE ISOLATION VALVES			
	4.6.1	Safety Objectives	1. Delete Section 4.6.1, Safety Objectives. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations are no longer possible. Consequently, the safety objectives of the Main Steam Isolation Valves are not required since the conditions requiring Main Steam Isolation Valve actuation can no longer occur. Information regarding the Main Steam Isolation Valves is obsolete.	
	4.6.2	Safety Design Bases	2. Delete Section 4.6.2, Safety Design Bases. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations are no longer possible. Consequently, the safety Design Bases of the Main Steam Isolation Valves are not required since the conditions requiring Main Steam Isolation Valve actuation can no longer occur. Information regarding the Main Steam Isolation Valves is obsolete.	

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	4.6.3	Description	3. Delete Section 4.6.3, Description. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations are no longer possible. Consequently, the Main Steam Isolation Valves are not required since the conditions requiring Main Steam Isolation Valve actuation can no longer occur. Information regarding the Main Steam Isolation Valves is obsolete	
	4.6.4	Safety Evaluation	4. Delete Section 4.6.4, Safety Evaluation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations are no longer possible. Consequently, the Main Steam Isolation Valves are not required since the conditions requiring Main Steam Isolation Valve actuation can no longer occur. An MSIV safety evaluation is not required. Information regarding the Main Steam Isolation Valves is obsolete.	
	4.6.5	Inspection and Testing	5. Delete Section 4.6.5, Inspection and Testing. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations are no longer possible. Consequently, the Main Steam Isolation Valves are not required since the conditions requiring Main Steam Isolation Valve actuation can no longer occur. MSIV inspection and testing is not required. Information regarding the Main Steam Isolation Valves is obsolete	
	4.6.6	References	6. Delete Section 4.6.6, References. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations are no longer possible. Consequently, the Main Steam Isolation Valves are not required since the conditions requiring Main Steam Isolation Valve actuation can no longer occur; references no longer apply. Information regarding the Main Steam Isolation Valves is obsolete.	
	Figure 4.6-1	Rockwell Main Steam Isolation Valve--Cross Section 5920-2038, sh. 5	Delete Figure 4.6-1, Rockwell Main Steam Isolation Valve--Cross Section. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations are no longer possible. Consequently, the Main Steam Isolation Valves are not required since the conditions requiring Main Steam Isolation Valve actuation can no longer occur. Therefore, the information provided on Figure 4.6-1 is no longer applicable. Information regarding the Main Steam Isolation Valves is obsolete	
	Figure 4.6-2	Main Steam Line Isolation Valve, Schematic Control Diagram 5920-6766	Delete Figure 4.6-2, Main Steam Line Isolation Valve, Schematic Control Diagram . Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations are no longer possible. Consequently, the Main Steam Isolation Valves are no required since the conditions requiring Main Steam Isolation Valve actuation can no longer occur. Therefore, the information provided on Figure 4.6-2 is no longer applicable. Information regarding the Main Steam Isolation Valves is obsolete	
4.7	REACTOR CORE ISOLATION COOLING SYSTEM			
	4.7.1	Power Generation Objectives	Delete section 4.7.1, Power Generation Objective. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Reactor Core Isolation Cooling System are no longer required. Reactor Core Isolation Cooling System information is obsolete.	
	4.7.2	Safety Objective	Delete section 4.7.2, Safety Objective. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Reactor Core Isolation Cooling System are no longer required. Reactor Core Isolation Cooling System information is obsolete.	
	4.7.3	Power Generation Design Basis	Delete section 4.7.3, Power Generation Design Basis. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Reactor Core Isolation Cooling System are no longer required. Reactor Core Isolation Cooling System information is obsolete.	
	4.7.4	Safety Design Bases	Delete section 4.7.4, Safety Design Basis. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Reactor Core Isolation Cooling System are no longer required. Reactor Core Isolation Cooling System information is obsolete.	
	4.7.5	Description	Delete section 4.7.5, Safety Evaluation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2) Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Reactor Core Isolation Cooling System are no longer required; a RCIC safety evaluation is not required. Reactor Core Isolation Cooling System information is obsolete.	
	4.7.6	Safety Evaluation	Delete section 4.7.6, Safety Evaluation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2) Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Reactor Core Isolation Cooling System are no longer required; a RCIC safety evaluation is not required. Reactor Core Isolation Cooling System information is obsolete.	
	4.7.7	Inspection and Testing	Delete section 4.7.7, Inspection and Testing. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Reactor Core Isolation Cooling System are no longer required; RCIC inspection and testing is not required. Reactor Core Isolation Cooling System information is obsolete.	

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	Table 4.7.1	Reactor Core Isolation Cooling System Turbine Pump Design Data	Delete Table 4.7.1, Reactor Core Isolation Cooling System Turbine Pump Design Data. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Reactor Core Isolation Cooling System are no longer required. RCIC system Turbine Pump Design Data is not required. Reactor Core Isolation Cooling System information is obsolete.	
	Figure 4.7-1a	Reactor Core Isolation Cooling System, Piping and Instrumentation Diagram G-191174, Sh 1	Delete Figure 4.7-1a, Reactor Core Isolation Cooling System, Piping and Instrumentation Diagram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Reactor Core Isolation Cooling System are no longer required. The information provided on figure 4.7-1a is no longer applicable. Reactor Core Isolation Cooling System information is obsolete.	
	Figure 4.7-1b	Reactor Core Isolation Cooling System Piping and Instrumentation Diagram, Sheet 2 G-191174, Sh 2	Delete Figure 4.7-1b, Reactor Core Isolation Cooling System, Piping and Instrumentation Diagram, sheet 2. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Reactor Core Isolation Cooling System are no longer required. The information provided on figure 4.7-1b is no longer applicable. Reactor Core Isolation Cooling System information is obsolete.	
	Figure 4.7-2a	Reactor Core Isolation Cooling System, Functional Control Diagram 5920-25	Delete Figure 4.7-2a, Reactor Core Isolation Cooling System, Functional Control Diagram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Reactor Core Isolation Cooling System are no longer required. The information provided on figure 4.7-2a is no longer applicable. Reactor Core Isolation Cooling System information is obsolete.	
	Figure 4.7-2b	Reactor Core Isolation Cooling System, Functional Control Diagram 5920-26	Delete Figure 4.7-2b, Reactor Core Isolation Cooling System, Functional Control Diagram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Reactor Core Isolation Cooling System are no longer required. The information provided on figure 4.7-2b is no longer applicable. Reactor Core Isolation Cooling System information is obsolete.	
	Figure 4.7-2c	Reactor Core Isolation Cooling System, Functional Control Diagram 5920-490	Delete Figure 4.7-2c, Reactor Core Isolation Cooling System, Functional Control Diagram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Reactor Core Isolation Cooling System are no longer required. The information provided on figure 4.7-2b is no longer applicable. Reactor Core Isolation Cooling System information is obsolete.	
	Figure 4.7-3	Reactor Core Isolation Cooling System, Process Diagram 5920-605	Delete Figure 4.7-3, Reactor Core Isolation Cooling System, Process Diagram . Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Reactor Core Isolation Cooling System are no longer required. The information provided on figure 4.7-2b is no longer applicable. Reactor Core Isolation Cooling System information is obsolete.	
4.8	RESIDUAL HEAT REMOVAL SYSTEM		The RHR system removed decay heat and residual heat from the nuclear system so that refueling and nuclear system servicing can be performed, and supplemented the Fuel Pool Cooling System when necessary to provide additional cooling capacity.	
	4.8.1	Power Generation Objectives	Delete section 4.8.1, Residual Heat Removal System, Power Generation Objectives. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, removal of decay and residual heat is no longer required. Since a LOCA is no longer possible, post the post LOCA functions of RHR are also no longer required. Additionally, the irradiated fuel decay heat load in the fuel pool is within the capacity of fuel pool cooling, heat removal supplementation by the RHR system is no longer required. Therefore, the power generation objectives are no longer applicable. Residual Heat Removal System information is obsolete.	
	4.8.2	Safety Objectives	Delete section 4.8.2, Residual Heat Removal System, Safety Objectives. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, restoration and maintenance of RPV water level following a LOCA, Suppression Pool Cooling following a LOCA, and Containment Cooling following a LOCA are no longer required. Therefore, the safety objectives are no longer applicable. Residual Heat Removal System information is obsolete.	
	4.8.3	Power Generation Design Basis	Delete section 4.8.3, Residual Heat Removal System, Power Generation Design Basis. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, removal of decay and residual heat is no longer required. Additionally, the irradiated fuel decay heat load in the fuel pool is within the capacity of fuel pool cooling, heat removal supplementation by the RHR system is no longer required. Therefore, an additional source of water for containment flooding is not required, the RHR power generation design basis is no longer applicable. Residual Heat Removal System information is obsolete.	
	4.8.4	Safety Design Bases	Delete section 4.8.2, Residual Heat Removal System, Safety Objectives. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, restoration and maintenance of RPV water level following a LOCA, Suppression Pool Cooling following a LOCA, and Containment Cooling following a LOCA are no longer required. Therefore, the safety design bases are no longer applicable. Residual Heat Removal System information is obsolete.	
	4.8.5	Description	Delete section 4.8.5, Residual Heat Removal System, Description. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the RHR system are no longer required. Therefore, the information provided in the description section is no longer applicable. Residual Heat Removal System information is obsolete.	

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	4.8.5.1	Shutdown Cooling Subsystem	Delete section 4.8.5.1, Description of the Shutdown Cooling mode of operation of RHR. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently a reactor shutdown and cooldown will no longer be conducted. Therefore, the Shutdown Cooling functions of the RHR system are no longer required. Residual Heat Removal System information is obsolete.	
	4.8.5.2	Containment Cooling Subsystem	Delete section 4.8.5.2, Description of the Containment Cooling mode of operation of RHR. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, heat removal from the suppression pool following a LOCA is no longer required. Therefore, the Containment Cooling functions of the RHR system are no longer required. Residual Heat Removal System information is obsolete.	
	4.8.5.3	Low Pressure Coolant Injection Subsystem	Delete section 4.8.5.3, Description of the Low Pressure Coolant Injection mode of operation of RHR. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, restoration and, maintenance of the coolant inventory in the reactor vessel after a loss of coolant accident is no longer required.Therefore, the Low Pressure Coolant Injection functions of the RHR system are no longer required. Residual Heat Removal System information is obsolete.	
	4.8.6	Safety Evaluation	Delete section 4.8.6, Residual Heat Removal System, Safety Evaluation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since the functions of the RHR system are no longer required a safety evaluation is also no longer required. Residual Heat Removal System information is obsolete.	
	4.8.7	Inspection and Testing	Delete section 4.8.7, Residual Heat Removal System, Inspection and Testing. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions of the RHR system are no longer required, inspection and testing of the RHR system is also no longer required. Residual Heat Removal System information is obsolete.	
	Table 4.8.1	Residual Heat Removal System Equipment Design Data	DeleteTable 4.8.1, Residual Heat Removal System Equipment Design Data. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions of the RHR system are no longer required. The information provided in Table 4.8.1 is no longer required. Residual Heat Removal System information is obsolete.	
	Figure 4.8-1	Residual Heat Removal System Process Diagram 5920-725	Delete Figure 4.8-1, Residual Heat Removal System Process Diagram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the RHR system are no longer required. Therefore, the information provided on Figure 4.8-1 is no longer applicable. Residual Heat Removal System information is obsolete.	
	Figure 4.8-2	Residual Heat Removal System, Piping and Instrumentation Diagram G191172	Delete Figure 4.8-2, Residual Heat Removal System, Piping and Instrumentation Diagram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the RHR system are no longer required. Therefore, the information provided on Figure 4.8-2 is no longer applicable. Residual Heat Removal System information is obsolete.	
4.9	REACTOR WATER CLEANUP SYSTEM			
	4.9.1	Power Generation Objectives	Delete section 4.9.1, Power Generation Objectives. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Reactor Water Cleanup System are no longer required. Reactor Water Cleanup System information is obsolete.	
	4.9.2	Power Generation Design Bases	Delete section 4.9.2, Power Generation Design Bases. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Reactor Water Cleanup System are no longer required. Reactor Water Cleanup System information is obsolete.	
	4.9.3	Description	Delete section 4.9.3, Description. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Reactor Water Cleanup System are no longer required. Since noble metal durability and electrochemical corrosion potential are not a concern, the Mitigation Monitoring System does not require flow from the cleanup system. Reactor Water Cleanup System information is obsolete.	
	4.9.4	Inspection and Testing	Delete section 4.9.4, Inspection and Testing. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Reactor Water Cleanup System are no longer required; inspection and testing of the Reactor Water Cleanup System is not required. Reactor Water Cleanup System information is obsolete.	
	Table 4.9.1	Reactor Water Cleanup System Equipment Design	5. Delete Table 4.9.1, Reactor Water Cleanup System Equipment Design Data. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Reactor Water Cleanup System are no longer required. Reactor Water Cleanup System information is obsolete.	

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	Figure 4.9-1	Reactor Water Cleanup System Piping and Instrumentation Diagram G-191178, Sh1		Delete Figure 4.9-1, Reactor Water Cleanup System Piping and Instrumentation Diagram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Reactor Water Cleanup System are no longer required. Since noble metal durability and electrochemical corrosion potential are not a concern, the Mitigation Monitoring System does not require flow from the cleanup system. Therefore, the information provided on Figure 4.9-1 is no longer applicable. Reactor Water Cleanup System information is obsolete.			
	Figure 4.9-2	Reactor Water Cleanup System Process Diagram 5920-429		Delete Figure 4.9-2, Reactor Water Cleanup System Process Diagram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Reactor Water Cleanup System are no longer required. Since noble metal durability and electrochemical corrosion potential are not a concern, the Mitigation Monitoring System does not require flow from the cleanup system. Therefore, the information provided on Figure 4.9-2 is no longer applicable. Reactor Water Cleanup System information is obsolete.			
	Figure 4.9-3	Reactor Water Cleanup Filter Demineralizer System, Piping and Instrument Diagram G-191178, Sh2		Delete Figure 4.9-3, Reactor Water Cleanup Filter Demineralizer System, Piping and Instrument Diagram . Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Reactor Water Cleanup System are no longer required. Since noble metal durability and electrochemical corrosion potential are not a concern, the Mitigation Monitoring System does not require flow from the cleanup system. Therefore, the information provided on Figure 4.9-3 is no longer applicable. Reactor Water Cleanup System information is obsolete.			
	Figure 4.9-4	Reactor Water Cleanup System Functional Control Diagram 5920-3867		Delete Figure 4.9-4, Reactor Water Cleanup System Functional Control Diagram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Reactor Water Cleanup System are no longer required. Since noble metal durability and electrochemical corrosion potential are not a concern, the Mitigation Monitoring System does not require flow from the cleanup system. Therefore, the information provided on Figure 4.9-4 is no longer applicable. Reactor Water Cleanup System information is obsolete.			
4.10	NUCLEAR SYSTEM LEAKAGE RATE LIMITS AND LEAKAGE DETECTION SYSTEMS						
	4.10.1	Safety Objectiver		Delete section 4.10.1, Safety Objective. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Nuclear System Leakage Detection Systems are no longer required. Nuclear System Leakage Detection Systems information is obsolete.			
	4.10.2	Safety Design Bases		Delete section 4.10.2, Safety Design Basis. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Nuclear System Leakage Detection Systems are no longer required. Nuclear System Leakage Detection Systems information is obsolete.			
	4.10.3	Description		Delete section 4.10.3, Description. Sections 4.10.3.1, 4.10.3.2, and 4.10.3.3 are no longer applicable. Since there is no information in Section 4.10.3, it has been deleted.			
		4.10.3.1	Identified Leakage Rate	Delete section 4.10.3.1, Identified Leakage Rate. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Determination of Identified Leakage rates is no longer required. Consequently, the functions of the Nuclear System Leakage Detection Systems are no longer required. Nuclear System Leakage Detection Systems information is obsolete.			
		4.10.3.2	Unidentified Leakage Rate	Delete section 4.10.3.2, Unidentified Leakage Rate. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Determination of Identified Leakage rates is no longer required. Consequently, the functions of the Nuclear System Leakage Detection Systems are no longer required. Nuclear System Leakage Detection Systems information is obsolete.			
		4.10.3.3	Total Leakage Rate	Delete section 4.10.3.3, Total Leakage Rate. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Determination of Total Leakage rates is no longer required. Consequently, the functions of the Nuclear System Leakage Detection Systems are no longer required. Nuclear System Leakage Detection Systems information is obsolete.			
	4.10.4	Actions Upon Discovery of a Leak		Delete section 4.10.4, Actions Upon Discovery of a Leak. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Nuclear System Leakage Detection Systems are no longer required. Therefore, no actions are required since no leakage will occur. All leakage limits have been removed from Permanently defueled technical specifications. Nuclear System Leakage Detection Systems information is obsolete.			
	Table 4.10.1	Leakage and Drainage Inputs to Drywell sumps		Delete Table 4.10.1, Leakage and Drainage Inputs to Drywell sumps. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Nuclear System Leakage Detection Systems are no longer required. Nuclear System Leakage Detection Systems information is obsolete.			
	Figure 4.10-1	Drywell Sumps Diagram G-191177, Sh1		No changes		Figure 4.5.2-4	Drywell Sumps Diagram G-191177, Sh1

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5.0	**Containment**				
5.1	SUMMARY DESCRIPTION				
	5.1.1	General		Delete section 5.1.1, Containment System, Summary, General. Containment Systems are not required following certification of permanent defueling and analysis of fuel handling accident. See sections 5.2, Primary Containment and Section 5.3, Secondary Containment.	
	5.1.2	Primary Containment		Delete section 5.1.2, Primary Containment description has been deleted from the SAR. See section 5.2, Primary Containment	
	5.1.3	Secondary Containment		Delete section 5.1.3, Secondary Containment description has been deleted from the SAR. See section 5.3, Secondary Containment	
5.2	PRIMARY CONTAINMENT SYSTEM				
	5.2.1	Safety Objective		Delete section 5.2.1, Safety Objective. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Primary Containment System are no longer required. Primary Containment System information is obsolete.	
	5.2.2	Safety Design Bases		Delete section 5.2.2, Safety Design Bases. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Primary Containment System are no longer required and the safety design bases are no longer applicable. Primary Containment System information is obsolete.	
	5.2.3	Description		Delete section 5.2.3, Description. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Primary Containment System are no longer required. The information provided in the Description section is no longer applicable. Primary Containment System information is obsolete.	
	5.2.4	Safety Evaluation		Delete section 5.2.4, Safety Evaluation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Primary Containment System are no longer required. A primary containmen safety evaluaion is no longer applicable. Primary Containment System information is obsolete.	
	5.2.5	Inspection and Testing		Delete section 5.2.5, Inspection and Testing. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Primary Containment System are no longer required. Primary Containment Inspection and Testing is not required. Primary Containment System information is obsolete.	
	5.2.6	Primary Containment Inerting		Delete Section 5.2.6, Containment Inerting. Subsections 5.2.6.1, Safety Design Basis, 5.2.6.2, System Description, 5.2.6.3, Safety Evaluation and 5.2.6.4, Nitrogen Supply System Inspection and Testing have been deleted.	
		5.2.6.1	Safety Design Basis	Delete section 5.2.6.1, Primary Containment Inerting Safety Design Bases. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Primary Containment System are no longer required, therefore, the Primary Containment Inerting system is also no longer required. Primary Containment Inerting System information is obsolete.	
		5.2.6.2	System Description	Delete section 5.2.6.2, Primary Containment Inerting System Description. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Primary Containment System are no longer required, therefore, the Primary Containment Inerting system is also no longer required.The information provided in the Primary Containment Inerting System Description section is no longer applicable. Primary Containment Inerting System information is obsolete.	
		5.2.6.3	Safety Evaluation	Delete section 5.2.6.3, Primary Containment Inerting System Safety Evaluation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Primary Containment System are no longer required, therefore, the Primary Containment Inerting system is also no longer required. The information provided in the Primary Containment Inerting System Safety Evaluation section is no longer applicable. Primary Containment Inerting System information is obsolete.	
		5.2.6.4	NSS Inspection and Testing	Delete section 5.2.6.4, Nitrogen Supply System Inspection and Testing. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Primary Containment System are no longer required, therefore, the Primary Containment Inerting system is also no longer required. The information provided in the Nitrogen Supply System Inspection and Testing section is no longer applicable. Primary Containment Inerting System information is obsolete.	
	5.2.7	Containment Atmosphere Dilution System		Delete Section 5.2.7, Containment Atmospheric Dilution (CAD) System. Subsections 5.2.7.1, Safety Design Basis, 5.2.7.2, System Description, 5.2.7.3, System Operation, 5.2.7.4, Safety Evaluation and 5.2.7.5, Containment Atmospheric Dilution (CAD) System Inspection and Testing have been deleted.	
		5.2.7.1	Safety Design Basis	Delete Section 5.2.7.1, Containment Atmospheric Dilution (CAD) System Safety Design Basis. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Primary Containment System are no longer required, therefore, the Primary Containment Atmospheric Dilution system is also no longer required. The information provided in the Containment Atmospheric Dilution Safety Design Basis section is no longer applicable. Containment Atmospheric Dilution System information is obsolete.	

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		5.2.7.2	System Description	Delete Section 5.2.7.2, Containment Atmospheric Dilution (CAD) System Description. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Primary Containment System are no longer required, therefore, the Primary Containment Atmospheric Dilution system is also no longer required. The information provided in the Containment Atmospheric Dilution System Description section is no longer applicable. Containment Atmospheric Dilution System information is obsolete.	
		5.2.7.3	System Operation	Delete Section 5.2.7.3, Containment Atmospheric Dilution (CAD) System Operation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Primary Containment System are no longer required, therefore, the Primary Containment Atmospheric Dilution system is also no longer required. The information provided in the Containment Atmospheric Dilution System Operation section is no longer applicable. Containment Atmospheric Dilution System information is obsolete.	
		5.2.7.4	Safety Evaluation	Delete Section 5.2.7.4, Containment Atmospheric Dilution (CAD) System Operation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Primary Containment System are no longer required, therefore, the Primary Containment Atmospheric Dilution system is also no longer required. The information provided in the Containment Atmospheric Dilution System Safety Evaluation section is no longer applicable. Containment Atmospheric Dilution System information is obsolete	
		5.2.7.5	System Inspection and Testing	Delete Section 5.2.7.5, Containment Atmospheric Dilution (CAD) System Inspection and Testing. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Primary Containment System are no longer required, therefore, the Primary Containment Atmospheric Dilution system is also no longer required. The information provided in the Containment Atmospheric Dilution System Inspection and Testing section is no longer applicable.	
	5.2.8	References		Delete Section 5.2.8, References. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations are no longer possible. Consequently, the functions of the Primary Containment System are no longer required, therefore, the Primary Containment Atmospheric Dilution system is also no longer required. The information provided in the Containment Atmospheric Dilution System References section is no longer applicable.	
	Table 5.2.1	Principal Design Parameters of Primary Containment		Delete Table 5.2.1, Principal Design Parameters of Primary Containment. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations are no longer possible and core related design basis accidents are no longer possible Consequently, the functions of the Primary Containment System are no longer required; the information provided in Table 5.2.1 is no longer applicable.	
	Table 5.2.2	Primary Containment System Penetrations and Associated Containment Isolation Valves		Delete Table 5.2.2, Primary Containment System Penetrations and Associated Containment Isolation Valves. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations are no longer possible and core related design basis accidents are no longer possible. Consequently, the functions of the Primary Containment System are no longer required; the information provided in Table 5.2.2 is no longer applicable.	
	Figure 5.2-1	Primary Containment Vessels 5920-45		Delete Figure 5.2-1, Primary Containment Vessels. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Primary Containment System are no longer required. The information provided on figure 5.2-1 is no longer applicable. Primary Containment System information is obsolete.	
	Figure 5.2-2	Primary Containment System Piping and Instrumentation Diagram G-191175, Sh1		Delete Figure 5.2-2, Primary Containment System Piping and Instrumentation Diagram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Primary Containment System are no longer required. The information regarding primary containment system piping and instrumentation provided on figure 5.2-2 is no longer applicable. Primary Containment System information is obsolete.	
	Figure 5.2-3	Types of Penetrations for Process Lines		Delete Figure 5.2-3, Types of Penetrations for Process Lines. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Primary Containment System are no longer required. The information regarding types of primary containment penetrations for process lines provided on figure 5.2-3 is no longer applicable. Primary Containment System information is obsolete.	
	Figure 5.2-4	Penetration Assembly for Process Lines		Delete Figure 5.2-4, Penetration Assembly for Process Lines. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Primary Containment System are no longer required. The information regarding penetration assemblies for process lines provided on figure 5.2-4 is no longer applicable. Primary Containment System information is obsolete.	
	Figure 5.2-5	Electrical and Control Penetrations		Delete Figure 5.2-5, Electrical and Control Penetrations. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Primary Containment System are no longer required. The information regarding electrical and control containment penetrations provided on figure 5.2-5 is no longer applicable. Primary Containment System information is obsolete.	

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	Figure 5.2-6	Energy Absorber Detail Installed Against Drywell		Delete Figure 5.2-6, Energy Absorber Detail Installed Against Drywell. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Primary Containment System are no longer required. The information regarding energy absorber installed against the drywell, provided on figure 5.2-6, is no longer applicable. Primary Containment System information is obsolete.	
	Figure 5.2-7	Nitrogen Supply Subsystem Primary Containment Atmosphere Control System G-191175, Sh2		Delete Figure 5.2-7, Nitrogen Supply Subsystem Primary Containment Atmosphere Control System. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Primary Containment System are no longer required. The information regarding the containment inerting nitrogen supply subsystem, provided on figure 5.2-7, is no longer applicable. Primary Containment System information is obsolete.	
	Figure 5.2-8	Engineering Flow Diagram Containment Atmosphere Dilution System (CAD) VY-E-75-002		Delete Figure 5.2-8, Engineering Flow Diagram Containment Atmosphere Dilution System. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Primary Containment System are no longer required. The information regarding the containment containment atmosphere dilution subsystem, provided on figure 5.2-8, is no longer applicable. Primary Containment System information is obsolete.	
	Figure 5.2-9	Drywell Temperature Response to Small Steam Line Breaks		Delete Figure 5.2-9, Drywell Temperature Response to Small Steam Line Breaks. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Primary Containment System are no longer required. The information regarding the temperature response to small steam line breaks in the drywell provided on figure 5.2-9, is no longer applicable. Primary Containment System information is obsolete.	
5.3	SECONDARY CONTAINMENT SYSTEM				
	5.3.1	Safety Objective		Delete Section 5.3.1, Secondary Containment System Safety Objective. Secondary Containment minimized the ground level release of airborne radioactive materials and provided a means for a controlled release of the building atmosphere should a design basis accident occur. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations are no longer possible and core related design basis accidents are no longer possible. License Amendment 262, approved by the NRC in February 2015, eliminated operability requirements for secondary containment, the Standby Gas Treatment (SGT) system, and the reactor building ventilation isolation and the SGT system initiation instrumentation when handling irradiated fuel or a fuel cask. The changes require the subject systems to be operable only during movement of "recently" irradiated fuel assemblies in secondary containment and during operations with the potential to drain the reactor vessel. The period of sufficient radioactive decay was determined to be 13 days. Consequently, the Secondary Containment safety objective is not applicable. Secondary Containment information is obsolete.	
	5.3.2	General		Delete Section 5.3.2, Secondary Containment System General. Secondary Containment minimized the ground level release of airborne radioactive materials and provided a means for a controlled release of the building atmosphere should a design basis accident occur. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations are no longer possible and core related design basis accidents are no longer possible. Analysis has demonstrated that secondary containment is no longer required to mitigate the consequences of the remaining design basis accident, a fuel handling accident. Consequently, the information provided in the Secondary Containment General section is not applicable. Secondary Containment information is obsolete.	
	5.3.3	Reactor Building			
		5.3.3.1	Safety Objective	Delete Section 5.3.3.1, Reactor Building Secondary Containment Safety Objective. Secondary Containment minimized the ground level release of airborne radioactive materials and provided a means for a controlled release of the building atmosphere should a design basis accident occur. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.License Amendment 262, approved by the NRC in February 2015, eliminated operability requirements for secondary containment, the Standby Gas Treatment (SGT) system, and the reactor building ventilation isolation and the SGT system initiation instrumentation when handling irradiated fuel or a fuel cask. The changes require the subject systems to be operable only during movement of "recently" irradiated fuel assemblies in secondary containment and during operations with the potential to drain the reactor vessel. The period of sufficient radioactive decay was determined to be 13 days. Since VYNPS certified permanent dufueling in January 2015, all fuel in the fuel pool is considered sufficiently decayed. Consequently, the Reactor Building Safety Objective is not applicable. Secondary Containment information is obsolete.	

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		5.3.3.2	Safety Design Basis	Delete Section 5.3.3.2, Reactor Building Secondary Containment Safety Design Basis. Secondary Containment minimized the ground level release of airborne radioactive materials and provided a means for a controlled release of the building atmosphere should a design basis accident occur. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. License Amendment 262, approved by the NRC in February 2015, eliminated operability requirements for secondary containment, the Standby Gas Treatment (SGT) system, and the reactor building ventilation isolation and the SGT system initiation instrumentation when handling irradiated fuel or a fuel cask. The changes require the subject systems to be operable only during movement of "recently" irradiated fuel assemblies in secondary containment and during operations with the potential to drain the reactor vessel. The period of sufficient radioactive decay was determined to be 13 days. Since VYNPS certified permanent dufueling in January 2015, all fuel in the fuel pool is considered sufficiently decayed. Consequently, the Reactor Building Safety Design Basis is not applicable. Secondary Containment information is obsolete.	
		5.3.3.3	Description	Delete Section 5.3.3.3, Reactor Building Secondary Containment Description. Secondary Containment minimized the ground level release of airborne radioactive materials and provided a means for a controlled release of the building atmosphere should a design basis accident occur. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. License Amendment 262, approved by the NRC in February 2015, eliminated operability requirements for secondary containment, the Standby Gas Treatment (SGT) system, and the reactor building ventilation isolation and the SGT system initiation instrumentation when handling irradiated fuel or a fuel cask. The changes require the subject systems to be operable only during movement of "recently" irradiated fuel assemblies in secondary containment and during operations with the potential to drain the reactor vessel. The period of sufficient radioactive decay was determined to be 13 days. Consequently, the information provided in the Reactor Building Description Section is not applicable. Secondary Containment information is obsolete.	
		5.3.3.4	Safety Evaluation	Delete Section 5.3.3.4, Reactor Building Secondary Containment Safety Evaluation. Secondary Containment minimized the ground level release of airborne radioactive materials and provided a means for a controlled release of the building atmosphere should a design basis accident occur. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Analysis has demonstrated that the secondary containment function of the Reactor Building is no longer required to mitigate the consequences of the remaining design basis accident, a fuel handling accident. Consequently, a safety evaluation of the Reactor Building Secondary Containment function is no longer required. Secondary Containment information is obsolete.	
		5.3.3.5	Inspection and Testing	Delete Section 5.3.3.5, Reactor Building Secondary Containment Inspection and Testing. Secondary Containment minimized the ground level release of airborne radioactive materials and provided a means for a controlled release of the building atmosphere should a design basis accident occur. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Analysis has demonstrated that the secondary containment function of the Reactor Building is no longer required to mitigate the consequences of the remaining design basis accident, a fuel handling accident. Consequently, a inspection and testing of the Reactor Building Secondary Containment function is no longer required. Secondary Containment information is obsolete.	
	5.3.4	Standby Gas Treatment System		The Standby Gas Treatment System limited airborne fission product release to the environment by the use of high efficiency filters and by the maintenance of a negative pressure in the Reactor Building.	
		5.3.4.1	Safety Objective	Delete Section 5.3.4.1, Standby Gas Treatment System Safety Objective. The Standby Gas Treatment System limited airborne fission product release to the environment by the use of high efficiency filters and by the maintenance of a negative pressure in the Reactor Building. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. License Amendment 262, approved by the NRC in February 2015, eliminated operability requirements for secondary containment, the Standby Gas Treatment (SGT) system, and the reactor building ventilation isolation and the SGT system initiation instrumentation when handling irradiated fuel or a fuel cask. The changes require the subject systems to be operable only during movement of "recently" irradiated fuel assemblies in secondary containment and during operations with the potential to drain the reactor vessel. The period of sufficient radioactive decay was determined to be 13 days. Since VYNPS certified permanent defueling in January 2015, all fuel in the fuel pool is considered sufficiently decayed. Consequently, the Standby Gas Treatment System Safety Objective is not applicable. Secondary Containment information is obsolete.	

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	5.3.4.2	Safety Design Basis	Delete Section 5.3.4.2, Standby Gas Treatment System Safety Design Basis. The Standby Gas Treatment System limited airborne fission product release to the environment by the use of high efficiency filters and by the maintenance of a negative pressure in the Reactor Building. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. License Amendment 262, approved by the NRC in February 2015, eliminated operability requirements for secondary containment, the Standby Gas Treatment (SGT) system, and the reactor building ventilation isolation and the SGT system initiation instrumentation when handling irradiated fuel or a fuel cask. The changes require the subject systems to be operable only during movement of "recently" irradiated fuel assemblies in secondary containment and during operations with the potential to drain the reactor vessel. The period of sufficient radioactive decay was determined to be 13 days. Since VYNPS certified permanent defueling in January 2015, all fuel in the fuel pool is considered sufficiently decayed. Consequently, the Standby Gas Treatment System Safety Design Basis is not applicable. Secondary Containment information is obsolete.	
	5.3.4.3	Description	Delete Section 5.3.4.3, Standby Gas Treatment System Description. The Standby Gas Treatment System limited airborne fission product release to the environment by the use of high efficiency filters and by the maintenance of a negative pressure in the Reactor Building. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. License Amendment 262, approved by the NRC in February 2015, eliminated operability requirements for secondary containment, the Standby Gas Treatment (SGT) system, and the reactor building ventilation isolation and the SGT system initiation instrumentation when handling irradiated fuel or a fuel cask. The changes require the subject systems to be operable only during movement of "recently" irradiated fuel assemblies in secondary containment and during operations with the potential to drain the reactor vessel. The period of sufficient radioactive decay was determined to be 13 days. Since VYNPS certified permanent defueling in January 2015, all fuel in the fuel pool is considered sufficiently decayed. Consequently, the information provided in the Standby Gas Treatment Description Section is no longer applicable. Secondary Containment information is obsolete.	
	5.3.4.4	Safety Evaluation	Delete Section 5.3.4.4, Standby Gas Treatment System Safety Evaluation. The Standby Gas Treatment System limited airborne fission product release to the environment by the use of high efficiency filters and by the maintenance of a negative pressure in the Reactor Building. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Analysis has demonstrated that the Standby Gas Treatment System is no longer required to mitigate the consequences of the remaining design basis accident, a fuel handling accident. Consequently, a safety evaluation of the Standby Gas Treatment System is no longer required. Secondary Containment information is obsolete.	
	5.3.4.5	Inspection and Testing	Delete Section 5.3.4.5, Standby Gas Treatment System Inspection and Testing. The Standby Gas Treatment System limited airborne fission product release to the environment by the use of high efficiency filters and by the maintenance of a negative pressure in the Reactor Building. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Analysis has demonstrated that the Standby Gas Treatment System is no longer required to mitigate the consequences of the remaining design basis accident, a fuel handling accident. Consequently, inspection and testing of the Standby Gas Treatment System is no longer required. Secondary Containment information is obsolete.	
5.3.5	Reactor Building Normal HVAC			
	5.3.5.1	Power Generation Objective	Modify Section 5.3.5.1, RB HVAC Power Generation Objective to delete "Power Generation", since following certification of permanent defueling power operations are no longer permitted. Information moved to section 10.12.3.1, RB Normal HVAC	
	5.3.5.2	Power Generation Design Basis	Modify Section 5.3.5.2, RB HVAC Power Generation Design Basis to delete "Power Generation", since following certification of permanent defueling power operations are no longer permitted, remove references to station operations, refueling, shutdown and hot standby. Information moved to section 10.12.3.1, RB Normal HVAC	
	5.3.5.3	Safety Design Basis	Delete Section 5.3.5.3, RB HVAC Safety Design Basis. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. License Amendment 262, approved by the NRC in February 2015, eliminated operability requirements for secondary containment, the Standby Gas Treatment (SGT) system, and the reactor building ventilation isolation and the SGT system initiation instrumentation when handling irradiated fuel or a fuel cask. The changes require the subject systems to be operable only during movement of "recently" irradiated fuel assemblies in secondary containment and during operations with the potential to drain the reactor vessel. The period of sufficient radioactive decay was determined to be 13 days. Since VYNPS certified permanent defueling in January 2015, all fuel in the fuel pool is considered sufficiently decayed. Consequently, the RB HVAC safety design basis is no longer required.	
	5.3.5.4	Description	Modify Section 5.3.5.4, RB HVAC Description to remove reference to plant operation, accidents and other conditions which are no longer applicable. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the RB HVAC system to address those conditions are no longer required. Information moved to section 10.12.3.1, RB Normal HVAC	

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	5.3.5.5	Safety Evaluation		Delete Section 5.3.5.5, RB HVAC Safety Evaluation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. License Amendment 262, approved by the NRC in February 2015, eliminated operability requirements for secondary containment, the Standby Gas Treatment (SGT) system, and the reactor building ventilation isolation and the SGT system initiation instrumentation when handling irradiated fuel or a fuel cask. The changes require the subject systems to be operable only during movement of "recently" irradiated fuel assemblies in secondary containment and during operations with the potential to drain the reactor vessel. The period of sufficient radioactive decay was determined to be 13 days. Since VYNPS certified permanent defueling in January 2015, all fuel in the fuel pool is considered sufficiently decayed. Consequently, a safety evaluation of the RB HVAC system is no longer required.	
	5.3.5.6	Inspection and Testing		Modify Section 5.3.5.4, RB HVAC Description to remove reference to plant operation, accidents and other conditions which are no longer applicable. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the RB HVAC system to address those conditions are no longer required. Inspection and testing of the RB HVAC system will be conducted as appropriate for the defueled state. Information moved to section 10.12.3.1 RB Normal HVAC	
5.3.6	Reactor Building Penetrations			Reactor Building penetrations provided a relatively tight secondary containment such that ground level release of radioactive materials was minimized in the event of a design basis accident.	
	5.3.6.1	Safety Objective		Delete Section 5.3.6.1, Reactor Building Penetrations Safety Objective. The Reactor Building penetrations provided a relatively tight secondary containment such that ground level release of radioactive materials was minimized in the event of a design basis accident. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. License Amendment 262, approved by the NRC in February 2015, eliminated operability requirements for secondary containment, the Standby Gas Treatment (SGT) system, and the reactor building ventilation isolation and the SGT system initiation instrumentation when handling irradiated fuel or a fuel cask. The changes require the subject systems to be operable only during movement of "recently" irradiated fuel assemblies in secondary containment and during operations with the potential to drain the reactor vessel. The period of sufficient radioactive decay was determined to be 13 days. Since VYNPS certified permanent defueling in January 2015, all fuel in the fuel pool is considered sufficiently decayed. Consequently, the safety objective of the RB Penetration system is no longer applicable. RB Penetration System information is obsolete.	
	5.3.6.2	Safety Design Basis		Delete Section 5.3.6.2, Reactor Building Penetrations Safety Design Basis The Reactor Building penetrations provided a relatively tight secondary containment such that ground level release of radioactive materials was minimized in the event of a design basis accident. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Analysis has demonstrated that the RB Penetrations are no longer required to mitigate the consequences of the remaining design basis accident, a fuel handling accident. Consequently, the safety design basis of the RB Penetration system is no longer applicable. RB Penetration System information is obsolete.	
	5.3.6.3	Description		Delete Section 5.3.6.3, Reactor Building Penetrations Description. The Reactor Building penetrations provided a relatively tight secondary containment such that ground level release of radioactive materials was minimized in the event of a design basis accident. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Analysis has demonstrated that the RB Penetrations are no longer required to mitigate the consequences of the remaining design basis accident, a fuel handling accident. Consequently, the information provided in the RB Penetration Description section is longer applicable. RB Penetration System information is obsolete.	
	5.3.6.4	Safety Evaluation		Delete Section 5.3.6.4, Reactor Building Penetrations Safety Evaluation. The Reactor Building penetrations provided a relatively tight secondary containment such that ground level release of radioactive materials was minimized in the event of a design basis accident. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Analysis has demonstrated that the RB Penetrations are no longer required to mitigate the consequences of the remaining design basis accident, a fuel handling accident. Consequently, a safety evaluation of the RB Penetration system is longer required. RB Penetration System information is obsolete.	
	5.3.6.5	Inspection and Testing		Delete Section 5.3.6.5, Reactor Building Penetrations Inspection and Testing. The Reactor Building penetrations provided a relatively tight secondary containment such that ground level release of radioactive materials was minimized in the event of a design basis accident. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Analysis has demonstrated that the RB Penetrations are no longer required to mitigate the consequences of the remaining design basis accident, a fuel handling accident. Consequently, inspection and testing of the RB Penetration system is longer required. RB Penetration System information is obsolete.	
5.3.7	MSIV Alternate Leakage Treatment Path			The MSIV ALT path reduced Post LOCA doses to within the guidelines of 10CFR 50.67 and the acceptance criteria of GDC-19 with a leakage rate of 62 scfh per MSIV with a total leakage rate of 124 scfh through the main steam pathways.	

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	5.3.7.1	Background	Delete Section 5.3.7.1, MSIV ALT Path, Background. The MSIV ALT path reduced Post LOCA doses to within the guidelines of 10CFR 50.67 and the acceptance criteria of GDC-19 with a leakage rate of 62 scfh per MSIV with a total leakage rate of 124 scfh through the main steam pathways. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.The additional MSIV leakage allowed by the ALT path is no longer applicable since the conditions addressed by ALT are no longer possible. Therefore, ALT related information is obsolete.	
	5.3.7.2	Safety Objective	Delete Section 5.3.7.2, MSIV ALT Path, Safety Objective. The MSIV ALT path reduced Post LOCA doses to within the guidelines of 10CFR 50.67 and the acceptance criteria of GDC-19 with a leakage rate of 62 scfh per MSIV with a total leakage rate of 124 scfh through the main steam pathways. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.The additional MSIV leakage allowed by the ALT path is no longer applicable since the conditions addressed by ALT are no longer possible. The ALT safety objective is no longer applicable. ALT related information is obsolete.	
	5.3.7.3	Safety Design Bases	Delete Section 5.3.7.3, MSIV ALT Path, Safety Design Bases The MSIV ALT path reduced Post LOCA doses to within the guidelines of 10CFR 50.67 and the acceptance criteria of GDC-19 with a leakage rate of 62 scfh per MSIV with a total leakage rate of 124 scfh through the main steam pathways. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.The additional MSIV leakage allowed by the ALT path is no longer applicable since the conditions addressed by ALT are no longer possible. The ALT safety design bases are no longer applicable. ALT related information is obsolete.	
	5.3.7.4	Description	Delete Section 5.3.7.4, MSIV ALT Path, Description. The MSIV ALT path reduced Post LOCA doses to within the guidelines of 10CFR 50.67 and the acceptance criteria of GDC-19 with a leakage rate of 62 scfh per MSIV with a total leakage rate of 124 scfh through the main steam pathways. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.The additional MSIV leakage allowed by the ALT path is no longer applicable since the conditions addressed by ALT are no longer possible. The information provided in the ALT description section is no longer applicable. ALT related information is obsolete.	
	5.3.7.5	Safety Evaluation	Delete Section 5.3.7.5, MSIV ALT Path, Safety Evaluation. The MSIV ALT path reduced Post LOCA doses to within the guidelines of 10CFR 50.67 and the acceptance criteria of GDC-19 with a leakage rate of 62 scfh per MSIV with a total leakage rate of 124 scfh through the main steam pathways. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.The additional MSIV leakage allowed by the ALT path is no longer applicable since the conditions addressed by ALT are no longer possible. Therefore, a safety evaluation of the ALT path is no longer required. ALT path related information is obsolete.	
	5.3.7.6	Seismic Verification	Delete Section 5.3.7.6, MSIV ALT Path, Seismic Verification. The MSIV ALT path reduced Post LOCA doses to within the guidelines of 10CFR 50.67 and the acceptance criteria of GDC-19 with a leakage rate of 62 scfh per MSIV with a total leakage rate of 124 scfh through the main steam pathways. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.The additional MSIV leakage allowed by the ALT path is no longer applicable since the conditions addressed by ALT are no longer possible. Therefore, the systems in the ALT pathway are no longer required to be seismic to support the ALT function. ALT path related information is obsolete.	
	5.3.7.7	Radiological Analysis	Delete Section 5.3.7.7, MSIV ALT Path, Radiological Analysis. The MSIV ALT path reduced Post LOCA doses to within the guidelines of 10CFR 50.67 and the acceptance criteria of GDC-19 with a leakage rate of 62 scfh per MSIV with a total leakage rate of 124 scfh through the main steam pathways. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.The additional MSIV leakage allowed by the ALT path is no longer applicable since the conditions addressed by ALT are no longer possible. Therefore, a Radiological Analysis of the effectiveness of the ALT path during a DBA is no longer required. ALT path related information is obsolete.	
	5.3.7.8	Inspection and Testing	Delete Section 5.3.7.8, MSIV ALT Path, Inspection and Testing. The MSIV ALT path reduced Post LOCA doses to within the guidelines of 10CFR 50.67 and the acceptance criteria of GDC-19 with a leakage rate of 62 scfh per MSIV with a total leakage rate of 124 scfh through the main steam pathways. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.The additional MSIV leakage allowed by the ALT path is no longer applicable since the conditions addressed by ALT are no longer possible. Therefore, inspection and testing of the components in the ALT path is no longer required. ALT path related information is obsolete.	

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	5.3.8	References	Delete Section 5.3.8, Secondary Containment System References. Secondary Containment minimized the ground level release of airborne radioactive materials and provided a means for a controlled release of the building atmosphere should a design basis accident occur. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations are no longer possible and core related design basis accidents are no longer possible. Analysis has demonstrated that secondary containment is no longer required to mitigate the consequences of the remaining design basis accident, a fuel handling accident. consequently, the Secondary Containment references are no longer applicable. Secondary Containment information ia obsolete.	
	Ta ble 5.3.1	Reactor Building Vacuum Data With One SGT Train Operating	Delete Table 5.3.1, Standby Gas Treatment System, Reactor Building Vacuum Data With One SGT Train Operating. The Standby Gas Treatment System limited airborne fission product release to the environment by the use of high efficiency filters and by the maintenance of a negative pressure in the Reactor Building. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Analysis has demonstrated that the Standby Gas Treatment System is no longer required to mitigate the consequences of the remaining design basis accident, a fuel handling accident. consequently, the information provided in table 5.3.1 is not applicable since SBGT is no longer relied on to mitigate the consequences of a fuel handling accident and. therefore, will no longer be operated. Secondary Containment information is obsolete.	
	Figure 5.3-1	Secondary Containment System – Reactor Building– Calculated Exfiltration Rate Versus Wind Velocity	Delete Figure 5.3-1, Secondary Containment System – Reactor Building– Calculated Exfiltration Rate Versus Wind Velocity. Secondary Containment minimized the ground level release of airborne radioactive materials and provided a means for a controlled release of the building atmosphere should a design basis accident occur. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Analysis has demonstrated that the secondary containment function of the Reactor Building is no longer required to mitigate the consequences of the remaining design basis accident, a fuel handling accident. Consequently, the information provided on Figure 5.3-1 is not applicable. Secondary Containment information is obsolete.	
	Figure 5.3-2	Reactor Building Heating Ventilation and Air-Conditioning G-191238	Moved to section 10.12	
	Figure 5.3-3	Main Steam Isolation Valve Alternate Leakage Treatment Path 5920-13407	Delete Figure 5.3-3, Main Steam Isolation Valve Alternate Leakage Treatment Path. The MSIV ALT path reduced Post LOCA doses to within the guidelines of 10CFR 50.67 and the acceptance criteria of GDC-19 with a leakage rate of 62 scfh per MSIV with a total leakage rate of 124 scfh through the main steam pathways. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.The additional MSIV leakage allowed by the ALT path is no longer applicable since the conditions addressed by ALT are no longer possible. The information provided on Figure 5.3-3 is no longer applicable. ALT related information is obsolete.	
6.0	**Core Standby Cooling Systems**			
6.1	Core Standby Cooling Systems			
	6.1.1	Safety Objective	Delete Section 6.1.1, Core Standby Cooling Systems Safety Objective. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since a LOCA can no longer occur, the Core Standby Cooling Systems are no longer required to cool or reflood the core to limit the release of radioactive material to the environs. Therefore, the Safety Objective is no longer applicable. Information regarding the Core Standby Cooling Systems is obsolete.	
6.2	Safety Design Bases		Delete Section 6.2, Core Standby Cooling Systems Safety Design Bases. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since a LOCA can no longer occur, the Core Standby Cooling Systems are no longer required to cool or reflood the core to limit the release of radioactive material to the environs. Therefore, the Safety Design Bases are no longer applicable. Information regarding the Core Standby Cooling Systems is obsolete.	
6.3	Summary Description		Delete Section 6.3,Summary Description Core Standby Cooling Systems. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since a LOCA can no longer occur, the Core Standby Cooling Systems are no longer required to cool or reflood the core to limit the release of radioactive material to the environs. Therefore, the information provided in the CSCS summary description section is no longer applicable. Information regarding the CSCSs is obsolete.	
	Table 6.3.1	Core Standby Cooling Systems Equipment Design Data Summary	Delete Table 6.3.1, Core Standby Cooling Systems Equipment Design Data Summary. Since the CSCSs are no longer required and have been deleted, the information provided in this table is obsolete.	
6.4	DESCRIPTION			
	6.4.1	High Pressure Coolant Injection System	Delete Section 6.4.1, High Pressure Coolant Injection System Description. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since a LOCA can no longer occur, the High Pressure Coolant Injection System is no longer required to cool or reflood the core to limit the release of radioactive material to the environs. Therefore, the information provided in the HPCI description section is no longer applicable. Information regarding the HPCI system is obsolete.	

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	6.4.2	Automatic Depressurization System		Delete Section 6.4.2, Automatic Depressurization System Description. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since a LOCA can no longer occur and all fuel is stored in the fuel pool or on the ISFSI, the Automatic Depressurization System is no longer required to function to allow low pressure systems to inject t cool or reflood the core to limit the release of radioactive material to the environs. Therefore, the information provided in the ADS description section is no longer applicable. Information regarding the ADS is obsolete.	
	6.4.3	Core Spray System		Delete Section 6.4.3, Core Spray System Description. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since a LOCA can no longer occur and all fuel is stored in the fuel pool or on the ISFSI, theCore Spray System is no longer required to function to cool the core following a LOCA to limit the release of radioactive material to the environs. Therefore, the information provided in the Core Spray description section is no longer applicable. Information regarding the Core Spray System is obsolete.	
	6.4.4	Low Pressure Coolant Injection		Delete Section 6.4.4,description of the Low Pressure Coolant Injection function of the Residual Heat Removal System . Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since a LOCA can no longer occur and all fuel is stored in the fuel pool or on the ISFSI, the LPCI function of RHR to cool the core following a LOCA to limit the release of radioactive material to the environs is no longer required. Therefore, the information provided in the LPCI description section is no longer applicable. Information regarding theLPCI function of the RHR System is obsolete.	
	6.4.5	References		Delete Section 6.4.5, References. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations are no longer possible. Consequently, the functions of the Core Standby Cooling Systems are no longer required. The information provided in the Core Standby Cooling Systems References section is no longer applicable.	
	Figure 6.4-1	High Pressure Coolant Injection System, Process Diagram 5920-784		Delete Figure 6.4-1, High Pressure Coolant Injection System, Process Diagram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in th reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since a LOCA can no longer occur, the High Pressure Coolant Injection System is no longer required to cool or reflood the core to limit the release of radioactive material to the environs. Therefore, the information provided on figure 6.4-1is no longer applicable. Information regarding the HPCI system is obsolete.	
	Figure 6.4-2	Core Spray System, Process Diagram 5920-718		Delete Figure 6.4-2, Core Spray System, Process Diagram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since a LOCA can no longer occur and all fuel is stored in the fuel pool or on the ISFSI, the Core Spray System is no longer required to function to cool the core following a LOCA to limit the release of radioactive material to the environs. Therefore, the information provided in the Core Spray description section is no longer applicable. Information regarding the Core Spray System is obsolete.	
	Figure 6.4-3	Residual Heat Removal System, Process Diagram 5920-725		Delete Figure 6.4-3, Residual Heat Removal System, Process Diagram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since a LOCA can no longer occur and all fuel is stored in the fuel pool or on the ISFSI, the LPCI function of RHR to cool the core following a LOCA to limit the release of radioactive material to the environs is no longer required. Therefore, the information provided on Figure 6.4-3 is no longer applicable. Information regarding theLPCI function of the RHR System is obsolete.	
6.5	SAFETY EVALUATION				
	6.5.1	Summary		Delete Section 6.5.1, Core Standby Cooling Systems Safety Evaluation, Summary. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since a LOCA can no longer occur, the Core Standby Cooling Systems are no longer required to cool or reflood the core to limit the release of radioactive material to the environs. Therefore, a safety evaluation of the Core Standby Cooling Sysems is not required. The information provided in the Safety Evaluation, Summary section is no longer applicable. Information regarding the Core Standby Cooling Systems is obsolete.	
	6.5.2	Performance Analysis		Delete Section 6.5.2, Safety Evaluation Performance Analysis. All subsections of this summary section have been evaluated and deleted. The information in this section is not required and is obsolete.	
		6.5.2.1	LOCA Analysis Methods and Results	Delete Section 6.5.1, Core Standby Cooling Systems Safety Evaluation, Performance Analysis, LOCA Analysis Methods and Results. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since a LOCA can no longer occur, the Core Standby Cooling Systems are no longer required to cool or reflood the core to limit the release of radioactive material to the environs. Therefore, a LOCA Analysis is not required. The information provided in the LOCA Analysis Methods and Results section is no longer applicable. Information regarding the Core Standby Cooling Systems is obsolete.	
		6.5.2.2	High Pressure Coolant Injection (HPCI) System	Delete Section 6.5.2.2, High Pressure Coolant Injection System Performance Analysis. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since a LOCA can no longer occur, the High Pressure Coolant Injection System is no longer required to cool or reflood the core to limit the release of radioactive material to the environs. Therefore, a safety evaluation and performance analysis of the HPCI System is not required. Information regarding the HPCI system is obsolete.	

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		6.5.2.3	Automatic Depressurization System (ADS)	Delete Section 6.5.2.3, High Pressure Coolant Injection System Performance Analysis. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since a LOCA can no longer occur, the Automatic Depressurization System is no longer required to rapidly depressurize the pressure vessel to allow low pressure systems to cool or reflood the core to limit the release of radioactive material to the environs. Therefore, a safety evaluation and performance analysis of the Automatic Depressurization System is not required. Information regarding ADS is obsolete.	
		6.5.2.4	Core Spray System (CSS)	Delete Section 6.5.2.4, Core Spray System Performance Analysis. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since a LOCA can no longer occur, the Core Spray System is no longer required to cool or reflood the core to limit the release of radioactive material to the environs. Therefore, a safety evaluation and performance analysis of the Core Spray System is not required. Information regarding the Core Spray system is obsolete.	
		6.5.2.5	Low Pressure Coolant Injection (LPCI)	Delete Section 6.5.2.5, Performance Analysis of the Low Pressure Coolant Injection function of the Residual Heat Removal System . Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since a LOCA can no longer occur and all fuel is stored in the fuel pool or on the ISFSI, the LPCI function of RHR to cool the core following a LOCA to limit the release of radioactive material to the environs is not required. Therefore, a safety evaluation and performance analysis of the Low Pressure Coolant Injection function of RHR is not required. Information regarding theLPCI function of the RHR System is obsolete.	
	6.5.3	Compliance With 10CFR50.46 Requirements		Delete Section 6.5.3, Core Standby Cooling Systems Safety Evaluation, Compliance with 10CFR50.46 Requirements. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since a LOCA can no longer occur, demonstration of compliance with 10CFR50.46 requirements is not required. The information provided in the Compliance with 10CFR50.46 Requirements section is no longer applicable. Informationprovided in this section is obsolete.	
	6.5.4	Core Standby Cooling Systems Redundancy		Delete Section 6.5.4, Core Standby Cooling Systems Safety Evaluation, Core Standby Cooling Systems Redundancy. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since a LOCA can no longer occur, demonstration of compliance with 10CFR50.46 requirements, including redundancy, is not required. The information provided in the Core Standby Cooling Systems Redundancy section is no longer applicable. Informationprovided in this section is obsolete.	
	6.5.5	References		Delete Section 6.5.4, References. Since all parts of Section 6.5 have been deleted, References are no longer applicable.	
6.6	INSPECTION AND TESTING				
	6.6.1	Summary		Delete Section 6.6.1, Core Standby Cooling Systems Inspection and Testing. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since a LOCA can no longer occur,the functions of the Core Standby Cooling Systems are not required. Consequently, demonstration of the ability of the CSCSs to meet their function through inspection and testing is not required. The information provided in the Core Standby Cooling Systems Inspection and testing section is no longer applicable. Information provided in this section is obsolete.	
	6.6.2	References		Delete Section 6.5.4, References. Since section 6.6.1 has been deleted, References are no longer applicable.	
7.0	**Control and Instrumentation**			Delete this heading, remaining section information moved to other sections	
7.1	SUMMARY DESCRIPTION			Delete this heading, remaining section information moved to other sections	
	7.1.1	Protection Systems		Delete Section 7.1.1, Control and Instrumentation, Protection Systems. The BWR SSC classification process described in UFSAR section 1.4 is not applicable to a permanently defueled station and the functions of Protection Systems are no longer applicable in the permanently defueled state.	
	7.1.2	Power Generation Systems		Delete Section 7.1.2, Control and Instrumentation, Power Generation Systems. The BWR SSC classification process described in UFSAR section 1.4 is not applicable to a permanently defuele station, and the functions of Power Generation Systems are no longer applicable in the permanently defueled state.	
	7.1.3	Safety Functions		Delete Section 7.1.3, Control and Instrumentation, Safety Functions. The BWR SSC classification process described in UFSAR section 1.4 is not applicable to a permanently defueled station, an the functions of Safety Systems are no longer applicable in the permanently defueled state.	
	7.1.4	Station Operational Control		Delete Section 7.1.4, Control and Instrumentation, Safety Functions. The BWR SSC classification process described in UFSAR section 1.4 is not applicable to a permanently defueled station, an the functions of Station Operational Control Systems are no longer applicable in the permanently defueled state.	

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	7.1.5	Definitions	Delete Section 7.1.5, Definitions. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, many of the definitions in this section are no longer applicable. The SSC classification process described in UFSAR section 1.4, Classification of BWR Systems, Criteria, and Requirements is not applicable in the defueled state and has been deleted. Many of the definitions provided in this section define terms related to the SSC Classification process and therefore, are no longer applicable Additionally, many of the definitions provided in this section are already defined in more appropriate documentation, including Technical Specifications and the Technical Requiements Manual. Finally, many of the definitions provided in this section are no longer used in the DSAR. Therefore, the information in this section is historcal and obsolete and may be deleted.	
	7.1.6	Identification of Safety Systems	Delete Section 7.1.6, Control and Instrumentation, Identification of Safety Systems. Since the functions of safety systems are no longer required following permanent defueling, specific identification of those SSCs is also no longer required.	
	7.1.7	Detailed Control Room Design Review	Delete Section 7.1.7, Control and Instrumentation, Detailed Design Review. This section contains historic information which is not applicable following certification of permanent defueling.	
	7.1.8	Environmental Qualification	Delete section 7.1.8, Environmental Qualification. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAV 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the assurances provided by the Environmental Qualification Program are no longer required.	
	7.1.9	References	Delete section 7.1.9, References. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the referenes cited in this section are no longer applicable. All text citing these references has been deleted from this section.	
	Figure 7.1-1	Use of Protection System Control and Instrumentation Definitions	Delete Figure 7.1-1, Use of Protection System Control and Instrumentation Definitions. Figure is no longer required since the entire section has been deleted.	
7.2	REACTOR PROTECTION SYSTEM			
	7.2.1	Safety Objective	Delete Section 7.2.1, Reactor Protection System Safety Objective. The Reactor Protection System functioned to prevent the uncontrolled release of radioactive material from the fuel and nuclear system process barrier by terminating excessive temperature and pressure increases through the initiation of an automatic scram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Reactor Protection System are no longer required; the RPS Safety Objective is not applicable since there is no core to protect. Information regarding the Reactor Protection System is obsolete.	
	7.2.2	Safety Design Bases	Delete Section 7.2.2, Reactor Protection System Safety Design Bases. The Reactor Protection System functioned to prevent the uncontrolled release of radioactive material from the fuel and nuclear system process barrier by terminating excessive temperature and pressure increases through the initiation of an automatic scram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Reactor Protection System are no longer required; the RPS Safety Objective is not applicable since there is no core to protect. Information regarding the Reactor Protection System is obsolete.	
	7.2.3	Description	Delete Section 7.2.3, Reactor Protection System Description. The Reactor Protection System functioned to prevent the uncontrolled release of radioactive material from the fuel and nuclear system process barrier by terminating excessive temperature and pressure increases through the initiation of an automatic scram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Reactor Protection System are no longer required; the Reactor Protection System Description provided is not applicable since there is no core to protect. Information regarding the Reactor Protection System is obsolete.	
	7.2.4	Safety Evaluation	Delete Section 7.2.4, Reactor Protection System Safety Evaluation. The Reactor Protection System functioned to prevent the uncontrolled release of radioactive material from the fuel and nuclear system process barrier by terminating excessive temperature and pressure increases through the initiation of an automatic scram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Reactor Protection System are no longer required; a safety evaluation of the Reactor Protection System is not required since the functions of the Reactor Protection System are not required. Information regarding the Reactor Protection System is obsolete.	

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	7.2.5	Inspection and Testing	Delete Section 7.2.5, Reactor Protection System Inspection and Testing. The Reactor Protection System functioned to prevent the uncontrolled release of radioactive material from the fuel and nuclear system process barrier by terminating excessive temperature and pressure increases through the initiation of an automatic scram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Reactor Protection System are no longer required; inspection and testing of the Reactor Protection System is not required since the functions of the Reactor Protection System are not required. The Reactor Protection System has been categorized as abandoned in accordance with EC 48440. Information regarding the Reactor Protection System is obsolete.	
	Table 7.2.1	Reactor Protection System Instrumentation	Delete Table 7.2.1, Reactor Protection System Instrumentation. The Reactor Protection System functioned to prevent the uncontrolled release of radioactive material from the fuel and nuclear system process barrier by terminating excessive temperature and pressure increases through the initiation of an automatic scram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Reactor Protection System are no longer required.The Reactor Protection System has been categorized as abandoned in accordance with EC 48440. Information regarding the Reactor Protection System is obsolete.	
	Table 7.2.2	Reactor Protection System Instrumentation Environmental Conditions	Delete Table 7.2.2, Reactor Protection System Instrumentation Environmental Conditions. The Reactor Protection System functioned to prevent the uncontrolled release of radioactive material from the fuel and nuclear system process barrier by terminating excessive temperature and pressure increases through the initiation of an automatic scram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Reactor Protection System are no longer required. Information regarding the Reactor Protection System is obsolete.	
	Figure 7.2-1	Reactor Protection System Instrument Electrical Diagram 5920-272	Delete Figure 7.2-1, Reactor Protection System Instrument Electrical Diagram. The Reactor Protection System functioned to prevent the uncontrolled release of radioactive material from the fuel and nuclear system process barrier by terminating excessive temperature and pressure increases through the initiation of an automatic scram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Reactor Protection System are no longer required; the Reactor Protection System information provided on figure 7.2-1 is not applicable since there is no core to protect. Information regarding the Reactor Protection System is obsolete.	
	Figure 7.2-2	Reactor Protection System, Control Room Panel for One Trip System	Delete Figure 7.2-2, Reactor Protection System, Control Room Panel for One Trip System. The Reactor Protection System functioned to prevent the uncontrolled release of radioactive material from the fuel and nuclear system process barrier by terminating excessive temperature and pressure increases through the initiation of an automatic scram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Reactor Protection System are no longer required; the Reactor Protection System information provided on figure 7.2-2 is not applicable since there is no core to protect. Information regarding the Reactor Protection System is obsolete.	
	Figure 7.2-3	Reactor Protection System, Function Control Diagram for One Trip System 5920-273	Delete Figure 7.2-3, Reactor Protection System, Function Control Diagram for One Trip System. The Reactor Protection System functioned to prevent the uncontrolled release of radioactive material from the fuel and nuclear system process barrier by terminating excessive temperature and pressure increases through the initiation of an automatic scram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Reactor Protection System are no longer required; the Reactor Protection System information provided on figure 7.2-3 is not applicable since there is no core to protect. Information regarding the Reactor Protection System is obsolete.	
	Figure 7.2-4	Schematic Diagram of Logics in One Trip System	Delete Figure 7.2-4, Schematic Diagram of Logics in One Trip System. The Reactor Protection System functioned to prevent the uncontrolled release of radioactive material from the fuel and nuclear system process barrier by terminating excessive temperature and pressure increases through the initiation of an automatic scram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Reactor Protection System are no longer required; the Reactor Protection System information provided on figure 7.2-4 is not applicable since there is no core to protect. Information regarding the Reactor Protection System is obsolete.	
	Figure 7.2-5	Schematic Diagram of Actuator and Actuator Logics	Delete Figure 7.2-5, Schematic Diagram of Actuator and Actuator Logics. The Reactor Protection System functioned to prevent the uncontrolled release of radioactive material from the fuel and nuclear system process barrier by terminating excessive temperature and pressure increases through the initiation of an automatic scram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Reactor Protection System are no longer required; the Reactor Protection System information provided on figure 7.2-5 is not applicable since there is no core to protect. Information regarding the Reactor Protection System is obsolete.	

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	Figure 7.2-6	Reactor Protection System, Scram Functions	Delete Figure 7.2-6, Reactor Protection System, Scram Functions. The Reactor Protection System functioned to prevent the uncontrolled release of radioactive material from the fuel and nuclear system process barrier by terminating excessive temperature and pressure increases through the initiation of an automatic scram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Reactor Protection System are no longer required; the Reactor Protection System information provided on figure 7.2-6 is not applicable since there is no core to protect. Information regarding the Reactor Protection System is obsolete.	
	Figure 7.2-7	Reactor Protection System Instrumentation 5920 274	Delete Figure 7.2-7, Reactor Protection System Instrumentation. The Reactor Protection System functioned to prevent the uncontrolled release of radioactive material from the fuel and nuclea system process barrier by terminating excessive temperature and pressure increases through the initiation of an automatic scram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Reactor Protection System are no longer required; the Reactor Protection System information provided on figure 7.2-7 is not applicable since there is no core to protect. Information regarding the Reactor Protection System is obsolete.	
	Figure 7.2-8	Relationship Between Neutron Monitoring System and Reactor Protection System	Delete Figure 7.2-8, Relationship Between Neutron Monitoring System and Reactor Protection System. The Reactor Protection System functioned to prevent the uncontrolled release of radioactive material from the fuel and nuclear system process barrier by terminating excessive temperature and pressure increases through the initiation of an automatic scram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Reactor Protection System are no longer required; the Reactor Protection System information provided on figure 7.2-8 is not applicable since there is no core to protect. Information regarding the Reactor Protection System is obsolete.	
	Figure 7.2-9	Functional Control Diagram for Neutron Monitoring System Logics 5920-1807	Delete Figure 7.2-9, Functional Control Diagram for Neutron Monitoring System Logics. The Reactor Protection System functioned to prevent the uncontrolled release of radioactive material from the fuel and nuclear system process barrier by terminating excessive temperature and pressure increases through the initiation of an automatic scram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Reactor Protection System are no longer required; the Reactor Protection System information provided on figure 7.2-9 is not applicable since there is no core to protect. Information regarding the Reactor Protection System is obsolete.	
	Figure 7.2-10	Typical Arrangement of Channels and Logics	Delete Figure 7.2-10, Typical Arrangement of Channels and Logics. The Reactor Protection System functioned to prevent the uncontrolled release of radioactive material from the fuel and nuclear system process barrier by terminating excessive temperature and pressure increases through the initiation of an automatic scram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Reactor Protection System are no longer required; the Reactor Protection System information provided on figure 7.2-10 is not applicable since there is no core to protect. Information regarding the Reactor Protection System is obsolete.	
	Figure 7.2-11	Typical Configuration for Turbine Stop Valve Closure Scram	Delete Figure 7.2-11, Typical Configuration for Turbine Stop Valve Closure Scram. The Reactor Protection System functioned to prevent the uncontrolled release of radioactive material from th fuel and nuclear system process barrier by terminating excessive temperature and pressure increases through the initiation of an automatic scram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can nc longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Reactor Protection System are no longer required; the Reactor Protection System information provided on figure 7.2-11 is not applicable since there is no core to protect. Information regarding the Reactor Protection System is obsolete.	
	Figure 7.2-12	Typical Configuration for Main Steamline Isolation Scram	Delete Figure 7.2-12, Typical Configuration for Main Steamline Isolation Scram. The Reactor Protection System functioned to prevent the uncontrolled release of radioactive material from the fuel and nuclear system process barrier by terminating excessive temperature and pressure increases through the initiation of an automatic scram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can nc longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Reactor Protection System are no longer required; the Reactor Protection System information provided on figure 7.2-12 is not applicable since there is no core to protect. Information regarding the Reactor Protection System is obsolete.	
	Figure 7.2-13	Deleted	Previously Deleted	
	Figure 7.2-14	Deleted	Previously Deleted	
	Figure 7.2-15	Deleted	Previously Deleted	
7.3	PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM		The Primary Containment and Reactor Vessel Isolation Control System initiated automatic isolation of appropriate pipelines which penetrate the primary containment whenever monitored variables exceed preselected operational limits to provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and nuclear system process barrier	

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	7.3.1	Safety Objective	Delete Section 7.3.1, Primary Containment and Reactor Vessel Isolation Control System Safety Objective. The Primary Containment and Reactor Vessel Isolation Control System initiated automatic isolation of appropriate pipelines which penetrate the primary containment whenever monitored variables exceeded preselected operational limits to provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and nuclear system process barrier, Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Primary Containment and Reactor Vessel Isolation Control System are no longer required. Information regarding the Primary Containment and Reactor Vessel Isolation Control System is obsolete.	
	7.3.2	Definitions	Delete Section 7.3.2, Primary Containment and Reactor Vessel Isolation Control System Definitions. The Primary Containment and Reactor Vessel Isolation Control System initiated automatic isolation of appropriate pipelines which penetrate the primary containment whenever monitored variables exceeded preselected operational limits to provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and nuclear system process barrier, Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Primary Containment and Reactor Vessel Isolation Control System are no longer required. Information regarding the Primary Containment and Reactor Vessel Isolation Control System is obsolete.	
	7.3.3	Safety Design Bases	Delete Section 7.3.3, Primary Containment and Reactor Vessel Isolation Control System Safety Design Bases The Primary Containment and Reactor Vessel Isolation Control System initiated automatic isolation of appropriate pipelines which penetrate the primary containment whenever monitored variables exceed preselected operational limits to provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and nuclear system process barrier, Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Primary Containment and Reactor Vessel Isolation Control System are no longer required.The Primary Containment and Reactor Vessel Isolation Control System has been categorized as abandoned in accordance with EC 48440 Information regarding the Primary Containment and Reactor Vessel Isolation Control System is obsolete.	
	7.3.4	Description	Delete Section 7.3.4, Primary Containment and Reactor Vessel Isolation Control System Description. The Primary Containment and Reactor Vessel Isolation Control System initiated automatic isolation of appropriate pipelines which penetrate the primary containment whenever monitored variables exceeded preselected operational limits to provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and nuclear system process barrier, Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Primary Containment and Reactor Vessel Isolation Control System are no longer required. Information provided in the Description section is no longer applicable. Information regarding the Primary Containment and Reactor Vessel Isolation Control System is obsolete.	
	7.3.5	Safety Evaluation	Delete Section 7.3.5, Primary Containment and Reactor Vessel Isolation Control System Safety Evaluation. The Primary Containment and Reactor Vessel Isolation Control System initiated automatic isolation of appropriate pipelines which penetrate the primary containment whenever monitored variables exceed preselected operational limits to provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and nuclear system process barrier, Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Primary Containment and Reactor Vessel Isolation Control System are no longer required. A safety evaluation of the Primary Containment and Reactor Vessel Isolation Control System is no longer required. The Primary Containment and Reactor Vessel Isolation Control System has been categorized as abandoned in accordance with EC 48440. Information regarding the Primary Containment and Reactor Vessel Isolation Control System is obsolete.	
	7.3.6	Inspection and Testing	Delete Section 7.3.6, Primary Containment and Reactor Vessel Isolation Control System Inspection and Testing. The Primary Containment and Reactor Vessel Isolation Control System initiated automatic isolation of appropriate pipelines which penetrate the primary containment whenever monitored variables exceed preselected operational limits to provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and nuclear system process barrier, Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Primary Containment and Reactor Vessel Isolation Control System are no longer required. Inspection and Testing of the Primary Containment and Reactor Vessel Isolation Control System is no longer required . The Primary Containment and Reactor Vessel Isolation Control System has been categorized as abandoned in accordance with EC 48440. Information regarding the Primary Containment and Reactor Vessel Isolation Control System is obsolete.	

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	7.3.7	References	Delete Section 7.3.7, Primary Containment and Reactor Vessel Isolation Control System References. The Primary Containment and Reactor Vessel Isolation Control System initiated automatic isolation of appropriate pipelines which penetrate the primary containment whenever monitored variables exceeded preselected operational limits to provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and nuclear system process barrier, Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Primary Containment and Reactor Vessel Isolation Control System are no longer required. References are no longer applicable. Information regarding the Primary Containment and Reactor Vessel Isolation Control System is obsolete.	
	Table 7.3.1	Primary Containment and Reactor Vessel Isolation Actuation	Delete Table 7.3.1, Primary Containment and Reactor Vessel Isolation Actuation. The Primary Containment and Reactor Vessel Isolation Control System initiated automatic isolation of appropriate pipelines which penetrate the primary containment whenever monitored variables exceed preselected operational limits to provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and nuclear system process barrier, Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Primary Containment and Reactor Vessel Isolation Control System are no longer required. The information presented in Table 7.3.1 is no longer applicable. Information regarding the Primary Containment and Reactor Vessel Isolation Control System is obsolete.	
	Table 7.3.2	Primary Containment and Reactor Vessel Isolation Control System Instrumentation	Delete Table 7.3.2, Primary Containment and Reactor Vessel Isolation Control System Instrumentation. The Primary Containment and Reactor Vessel Isolation Control System initiated automatic isolation of appropriate pipelines which penetrate the primary containment whenever monitored variables exceed preselected operational limits to provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and nuclear system process barrier, Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Primary Containment and Reactor Vessel Isolation Control System are no longer required. The information presented in Table 7.3.2 is no longer applicable. Information regarding the Primary Containment and Reactor Vessel Isolation Control System is obsolete.	
	Figure 7.3-1	Nuclear System, Piping and Instrumentation Diagram G-191167	Delete Figure 7.3-1, Nuclear System, Piping and Instrumentation Diagram. The Primary Containment and Reactor Vessel Isolation Control System initiated automatic isolation of appropriate pipelines which penetrate the primary containment whenever monitored variables exceeded preselected operational limits to provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and nuclear system process barrier, Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Primary Containment and Reactor Vessel Isolation Control System are no longer required. Information provided on figure 7.3-1 is no longer applicable. Information regarding the Primary Containment and Reactor Vessel Isolation Control System is obsolete.	
	Figure 7.3-2	Typical Isolation Control System Using Motor Operated Valves	Delete Figure 7.3-2, Typical Isolation Control System Using Motor Operated Valves. The Primary Containment and Reactor Vessel Isolation Control System initiated automatic isolation of appropriate pipelines which penetrate the primary containment whenever monitored variables exceeded preselected operational limits to provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and nuclear system process barrier, Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Primary Containment and Reactor Vessel Isolation Control System are no longer required. Information provided o n figure 7.3-2 is no longer applicable. Information regarding the Primary Containment and Reactor Vessel Isolation Control System is obsolete.	
	Figure 7.3-3	Typical Isolation Control System for Main Steamline Isolation Valves	Delete Figure 7.3-3, Typical Isolation Control System for Main Steamline Isolation Valves. The Primary Containment and Reactor Vessel Isolation Control System initiated automatic isolation of appropriate pipelines which penetrate the primary containment whenever monitored variables exceeded preselected operational limits to provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and nuclear system process barrier, Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Primary Containment and Reactor Vessel Isolation Control System are no longer required. Information provided on figure 7.3-3 is no longer applicable. Information regarding the Primary Containment and Reactor Vessel Isolation Control System is obsolete.	
	Figure 7.3-4a	Deleted	Previously Deleted	

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Figure 7.3-4b	Main Steam Line Isolation Valve, Schematic Control Diagram 5920-6766	Delete Figure 7.3-4b, Main Steam Line Isolation Valve, Schematic Control Diagram. The Primary Containment and Reactor Vessel Isolation Control System initiated automatic isolation of appropriate pipelines which penetrate the primary containment whenever monitored variables exceeded preselected operational limits to provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and nuclear system process barrier, Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Primary Containment and Reactor Vessel Isolation Control System are no longer required. Information provided on figure 7.3-4b is no longer applicable. Information regarding the Primary Containment and Reactor Vessel Isolation Control System is obsolete.	
Figure 7.3-5a	Nuclear Boiler Miscellaneous Systems, Functional Control Diagram 5920-612, Sh1	Delete Figure 7.3-5a, Nuclear Boiler Miscellaneous Systems, Functional Control Diagram. The Primary Containment and Reactor Vessel Isolation Control System initiated automatic isolation of appropriate pipelines which penetrate the primary containment whenever monitored variables exceeded preselected operational limits to provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and nuclear system process barrier, Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Primary Containment and Reactor Vessel Isolation Control System are no longer required. Information provided on figure 7.3-5a is no longer applicable. Information regarding the Primary Containment and Reactor Vessel Isolation Control System is obsolete.	
Figure 7.3-5b	Primary Containment and Reactor Vessel Isolation Control System, Functional Control Diagram	Delete Figure 7.3-5b, Primary Containment and Reactor Vessel Isolation Control System, Functional Control Diagram. The Primary Containment and Reactor Vessel Isolation Control System initiated automatic isolation of appropriate pipelines which penetrate the primary containment whenever monitored variables exceeded preselected operational limits to provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and nuclear system process barrier, Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Primary Containment and Reactor Vessel Isolation Control System are no longer required. Information provided on figure 7.3-5b is no longer applicable. Information regarding the Primary Containment and Reactor Vessel Isolation Control System is obsolete.	
Figure 7.3-6	Area and Compartment Leakage Detection by Temperature Measurement	Delete Figure 7.3-6, Area and Compartment Leakage Detection by Temperature Measurement. The Primary Containment and Reactor Vessel Isolation Control System initiated automatic isolation of appropriate pipelines which penetrate the primary containment whenever monitored variables exceeded preselected operational limits to provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and nuclear system process barrier, Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Primary Containment and Reactor Vessel Isolation Control System are no longer required. Information provided on figure 7.3-6 is no longer applicable. Information regarding the Primary Containment and Reactor Vessel Isolation Control System is obsolete.	
Figure 7.3-7A	Typical High Temperature Channels	Delete Figure 7.3-7A, Typical High Temperature Channels. The Primary Containment and Reactor Vessel Isolation Control System initiated automatic isolation of appropriate pipelines which penetrate the primary containment whenever monitored variables exceeded preselected operational limits to provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and nuclear system process barrier, Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Primary Containment and Reactor Vessel Isolation Control System are no longer required. Information provided on figure 7.3-7A is no longer applicable. Information regarding the Primary Containment and Reactor Vessel Isolation Control System is obsolete.	
Figure 7.3-7B	RWCS High Flow Channel	Delete Figure 7.3-7B, RWCS High Flow Channel. The Primary Containment and Reactor Vessel Isolation Control System initiated automatic isolation of appropriate pipelines which penetrate the primary containment whenever monitored variables exceeded preselected operational limits to provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and nuclear system process barrier, Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Primary Containment and Reactor Vessel Isolation Control System are no longer required. Information provided on figure 7.3-7B is no longer applicable. Information regarding the Primary Containment and Reactor Vessel Isolation Control System is obsolete.	
Figure 7.3-8	Typical Arrangement for Main Steamline Leak Detection by Flow Measurement	Delete Figure 7.3-8, Typical Arrangement for Main Steamline Leak Detection by Flow Measurement. The Primary Containment and Reactor Vessel Isolation Control System initiated automatic isolation of appropriate pipelines which penetrate the primary containment whenever monitored variables exceeded preselected operational limits to provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and nuclear system process barrier, Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Primary Containment and Reactor Vessel Isolation Control System are no longer required. Information provided on figure 7.3-8 is no longer applicable. Information regarding the Primary Containment and Reactor Vessel Isolation Control System is obsolete.	

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Figure 7.3-9	Main Steamline High Flow Channels	Delete Figure 7.3-9, Main Steamline High Flow Channels. The Primary Containment and Reactor Vessel Isolation Control System initiated automatic isolation of appropriate pipelines which penetrate the primary containment whenever monitored variables exceeded preselected operational limits to provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and nuclear system process barrier, Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Primary Containment and Reactor Vessel Isolation Control System are no longer required. Information provided on figure 7.3-9 is no longer applicable. Information regarding the Primary Containment and Reactor Vessel Isolation Control System is obsolete.	
Figure 7.3-10	RCIC Elbow Tap Arrangement for Gross Leak Detection	Delete Figure 7.3-10, RCIC Elbow Tap Arrangement for Gross Leak Detection. The Primary Containment and Reactor Vessel Isolation Control System initiated automatic isolation of appropriate pipelines which penetrate the primary containment whenever monitored variables exceeded preselected operational limits to provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and nuclear system process barrier, Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Primary Containment and Reactor Vessel Isolation Control System are no longer required. Information provided on figure 7.3-10 is no longer applicable. Information regarding the Primary Containment and Reactor Vessel Isolation Control System is obsolete.	
Figure 7.3-11	Typical Elbow Tap Arrangement for Gross Leak Detection	Delete Figure 7.3-11, Typical Elbow Tap Arrangement for Gross Leak Detection. The Primary Containment and Reactor Vessel Isolation Control System initiated automatic isolation of appropriate pipelines which penetrate the primary containment whenever monitored variables exceeded preselected operational limits to provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and nuclear system process barrier, Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Primary Containment and Reactor Vessel Isolation Control System are no longer required. Information provided on figure 7.3-11 is no longer applicable. Information regarding the Primary Containment and Reactor Vessel Isolation Control System is obsolete.	
Figure 7.3-12A	Isolation Control System, Typical Trip System B 5920-2093 Sheet 1 & 2	Delete Figure 7.3-12A, Isolation Control System, Typical Trip System. The Primary Containment and Reactor Vessel Isolation Control System initiated automatic isolation of appropriate pipelines which penetrate the primary containment whenever monitored variables exceeded preselected operational limits to provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and nuclear system process barrier, Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Primary Containment and Reactor Vessel Isolation Control System are no longer required. Information provided on figure 7.3-12A is no longer applicable. Information regarding the Primary Containment and Reactor Vessel Isolation Control System is obsolete.	
Figure 7.3-12B	Isolation Control System, Typical Trip System A 5920-2092 Sheet 1 & 2	Delete Figure 7.3-12B, Isolation Control System, Typical Trip System. The Primary Containment and Reactor Vessel Isolation Control System initiated automatic isolation of appropriate pipelines which penetrate the primary containment whenever monitored variables exceeded preselected operational limits to provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and nuclear system process barrier, Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Primary Containment and Reactor Vessel Isolation Control System are no longer required. Information provided on figure 7.3-12B is no longer applicable. Information regarding the Primary Containment and Reactor Vessel Isolation Control System is obsolete.	
Figure 7.3-12c	Isolation Control System, Typical Controls for Inboard Main Steamline Isolation Valves 5920-2094 5920-2095	Delete Figure 7.3-12c, Isolation Control System, Typical Controls for Inboard Main Steamline Isolation Valves. The Primary Containment and Reactor Vessel Isolation Control System initiated automatic isolation of appropriate pipelines which penetrate the primary containment whenever monitored variables exceeded preselected operational limits to provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and nuclear system process barrier, Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Primary Containment and Reactor Vessel Isolation Control System are no longer required. Information provided on figure 7.3-12c is no longer applicable. Information regarding the Primary Containment and Reactor Vessel Isolation Control System is obsolete.	
Figure 7.3-12d	Isolation Control System, Typical Controls for Outboard Main Steamline Isolation Valves 5920-2096	Delete Figure 7.3-12d, Isolation Control System, Typical Controls for Outboard Main Steamline Isolation Valves. The Primary Containment and Reactor Vessel Isolation Control System initiated automatic isolation of appropriate pipelines which penetrate the primary containment whenever monitored variables exceeded preselected operational limits to provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and nuclear system process barrier, Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Primary Containment and Reactor Vessel Isolation Control System are no longer required. Information provided on figure 7.3-12d is no longer applicable. Information regarding the Primary Containment and Reactor Vessel Isolation Control System is obsolete.	

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Figure 7.3-12e	Isolation Control System, Typical Controls for RHRS and Reactor Water Cleanup Valves 5920-2098	Delete Figure 7.3-12e, Isolation Control System, Typical Controls for RHRS and Reactor Water Cleanup Valves. The Primary Containment and Reactor Vessel Isolation Control System initiated automatic isolation of appropriate pipelines which penetrate the primary containment whenever monitored variables exceeded preselected operational limits to provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and nuclear system process barrier, Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Primary Containment and Reactor Vessel Isolation Control System are no longer required. Information provided on figure 7.3-12e is no longer applicable. Information regarding the Primary Containment and Reactor Vessel Isolation Control System is obsolete.	
Figure 7.3-12F	Isolation Control System, Typical Controls for RHRS Shutdown Cooling Valves 5920-2099	Delete Figure 7.3-12F, Isolation Control System, Typical Controls for RHRS Shutdown Cooling Valves. The Primary Containment and Reactor Vessel Isolation Control System initiated automatic isolation of appropriate pipelines which penetrate the primary containment whenever monitored variables exceeded preselected operational limits to provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and nuclear system process barrier, Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Primary Containment and Reactor Vessel Isolation Control System are no longer required. Information provided on figure 7.3-12F is no longer applicable. Information regarding the Primary Containment and Reactor Vessel Isolation Control System is obsolete.	
Figure 7.3-12g	Isolation Control System, Typical Controls for Reactor Water Cleanup System Valves 5920-2101	Delete Figure 7.3-12g, Isolation Control System, Typical Controls for Reactor Water Cleanup System Valves. The Primary Containment and Reactor Vessel Isolation Control System initiated automatic isolation of appropriate pipelines which penetrate the primary containment whenever monitored variables exceeded preselected operational limits to provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and nuclear system process barrier, Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Primary Containment and Reactor Vessel Isolation Control System are no longer required. Information provided on figure 7.3-12g is no longer applicable. Information regarding the Primary Containment and Reactor Vessel Isolation Control System is obsolete.	
Figure 7.3-12h	Isolation Control System, Typical Controls for RHRS Discharge to Radwaste Valves 5920-2100	Delete Figure 7.3-12h, Isolation Control System, Typical Controls for RHRS Discharge to Radwaste Valves . The Primary Containment and Reactor Vessel Isolation Control System initiated automatic isolation of appropriate pipelines which penetrate the primary containment whenever monitored variables exceeded preselected operational limits to provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and nuclear system process barrier, Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Primary Containment and Reactor Vessel Isolation Control System are no longer required. Information provided on figure 7.3-12h is no longer applicable. Information regarding the Primary Containment and Reactor Vessel Isolation Control System is obsolete.	
Figure 7.3-12i	Isolation Control System, Typical Controls for Main Steamline Drain Valves 5920-2097	Delete Figure 7.3-12i, Isolation Control System, Typical Controls for Main Steamline Drain Valves. The Primary Containment and Reactor Vessel Isolation Control System initiated automatic isolation of appropriate pipelines which penetrate the primary containment whenever monitored variables exceeded preselected operational limits to provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and nuclear system process barrier, Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Primary Containment and Reactor Vessel Isolation Control System are no longer required. Information provided on figure 7.3-12i is no longer applicable. Information regarding the Primary Containment and Reactor Vessel Isolation Control System is obsolete.	
Figure 7.3-12j	Isolation Control System, Typical Controls for Recirculation Loop Sample Valves 5920-2094	Delete Figure 7.3-12j, Isolation Control System, Typical Controls for Recirculation Loop Sample Valves. The Primary Containment and Reactor Vessel Isolation Control System initiated automatic isolation of appropriate pipelines which penetrate the primary containment whenever monitored variables exceeded preselected operational limits to provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and nuclear system process barrier, Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Primary Containment and Reactor Vessel Isolation Control System are no longer required. Information provided on figure 7.3-12j is no longer applicable. Information regarding the Primary Containment and Reactor Vessel Isolation Control System is obsolete.	
Figure 7.3-12k	Isolation Control System, Typical Controls for TIP System 5920-2094	Delete Figure 7.3-12k, Isolation Control System, Typical Controls for TIP System . The Primary Containment and Reactor Vessel Isolation Control System initiated automatic isolation of appropriate pipelines which penetrate the primary containment whenever monitored variables exceeded preselected operational limits to provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and nuclear system process barrier, Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the Primary Containment and Reactor Vessel Isolation Control System are no longer required. Information provided on figure 7.3-12k is no longer applicable. Information regarding the Primary Containment and Reactor Vessel Isolation Control System is obsolete.	

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7.4	CORE STANDBY COOLING SYSTEMS CONTROL AND INSTRUMENTATION			See Section 6.0, Core Standby Cooling Systems	
	7.4.1	Safety Objective		Delete Section 7.4.1, Core Standby Cooling Systems (CSCS) Control and Instrumentation Safety Objective. The controls and instrumentation for the Core Standby Cooling Systems initiated appropriate responses from the various cooling systems so that the fuel was adequately cooled under abnormal or accident conditions. The cooling provided by the systems restricted the release of radioactive materials from the fuel by limiting the extent of fuel damage following situations in which reactor coolant was lost from the nuclear system. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the CSCS are no longer required; the CSCS Safety Objective is not applicable since there is no core to protect. Information regarding theCore Standby Cooling Systems (CSCS) Control and Instrumentation is obsolete.	
	7.4.2	Safety Design Bases		Delete Section 7.4.2, Core Standby Cooling Systems (CSCS) Control and Instrumentation Safety Design Bases. The controls and instrumentation for the Core Standby Cooling Systems initiated appropriate responses from the various cooling systems so that the fuel was adequately cooled under abnormal or accident conditions. The cooling provided by the systems restricted the release of radioactive materials from the fuel by limiting the extent of fuel damage following situations in which reactor coolant was lost from the nuclear system. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions of the CSCSs are no longer required; the CSCS Safety Design Bases are not applicable since there is no core to protect. The CSCS functions of the associated systems have been categorized as abandoned in accordance with the associated Engineering Changes. Information regarding theCore Standby Cooling Systems (CSCS) Control and Instrumentation is obsolete.	
	7.4.3	Description			
		7.4.3.1	Identification	Delete Section 7.4.3.1, Core Standby Cooling Systems (CSCS) Controls and Instrumentation Identification. The controls and instrumentation for the Core Standby Cooling Systems are identified as that equipment required for the initiation and control of HPCIS, ADS, CSS, and the LPCI function of RHR to ensure the fuel was adequately cooled under abnormal or accident conditions. The cooling provided by the systems restricted the release of radioactive materials from the fuel by limiting the extent of fuel damage following situations in which reactor coolant was lost from the nuclear system. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions of the CSCSs are no longer required. Since the functions of the CSCSs are no longer required, the instrumentation and Controls for those systems are also no longer required. Information regarding the Core Standby Cooling Systems (CSCS) Controls and Instrumentation is obsolete.	
		7.4.3.2	HPCI Control & Instrumentation	Delete Section7.4.3.2, High Pressure Coolant Injection System Controls and Instrumentation Description. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since a LOCA can no longer occur, the High Pressure Coolant Injection System is no longer required to cool or reflood the core to limit the release of radioactiv material to the environs. Therefore, the HPCI System Controls and Instrumentation are also no longer required. Information provided in the HPCI Controls and Instrumentation description section is no longer applicable. Information regarding the HPCI system is obsolete.	
		7.4.3.3	ADS Control & Instrumentation	Delete Section 7.4.3.3, Automatic Depressurization System Control and Instrumentation Description. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since a LOCA can no longer occur and all fuel is stored in the fuel pool or on the ISFSI, the Automatic Depressurization System is no longer required to function to allow low pressure systems to inject to cool or reflood the core to limit the release of radioactive material to the environs. Therefore, the ADS Controls and Instrumentation are also no longer required.The information provided in the ADS Control and Instrumentation Description section is no longer applicable. Information regarding the ADS is obsolete.	
		7.4.3.4	Core Spray Control & Instrumentation	Delete Section 7.4.3.4, Core Spray System Control and Instrumentation Description. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fue in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since a LOCA can no longer occur and all fuel is stored in the fuel pool or on the ISFSI, the Core Spray System is no longer required to function to cool the core following a LOCA to limi the release of radioactive material to the environs. Therefore, the Core Spray System Controls and Instrumentation are also no longer required.The information provided in the Core Spray description section is no longer applicable. Information regarding the Core Spray System is obsolete.	
		7.4.3.5	LPCI Control & Instrumentation	Delete Section7.4.3.5, description of the Low Pressure Coolant Injection Control and Instrumentation function of the Residual Heat Removal System . Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since a LOCA can no longer occur and all fuel is stored in the fuel pool or on the ISFSI, the LPCI function of RHR to cool the core following a LOCA to limit the release of radioactive material to the environs is no longer required. Therefore, the LPCI System Controls and Instrumentation are also no longer required.The information provided in the LPCI description section is no longer applicable. Information regarding the LPCI function of the RHR System is obsolete.	

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7.4.4	Safety Evaluation	Delete Section 7.4.4, Core Standby Cooling Systems Instrumentation and Controls Safety Evaluation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since a LOCA can no longer occur, the Core Standby Cooling Systems are no longer required to cool or reflood the core to limit the release of radioactive material to the environs. Therefore, the CSCS Controls and Instrumentation are also no longer required. A safety evaluation of the Core Standby Cooling Sysems Controls and Instrumentation is not required. The information provided in the Safety Evaluation section is no longer applicable. Information regarding the Core Standby Cooling Systems is obsolete.	
7.4.5	Inspection and Testing	Delete Section 7.4.5, Core Standby Cooling Systems Controls and Instrumentation Inspection and Testing. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since a LOCA can no longer occur,the functions of the Core Standby Cooling Systems are not required. Consequently, demonstration of the ability of the CSCSs to meet their function through inspection and testing of the CSCS Controls and Instrumentation is not required. The information provided in the Core Standby Cooling Systems Controls and Instrumentation Inspection and testing section is no longer applicable. Information provided in this section is obsolete.	
7.4.6	References	Delete Section 7.4.6, References. Since section 7.4 has been deleted, References are no longer applicable.	
Table 7.4.1	High Pressure Coolant Injection System Instrumentation	Delete Table 7.4.1, HPCI Instrumentation. Section 7.4.3.2, HPCI System Controls and Instrumentation has been deleted since it is no longer applicable. Therefore, the table listing HPCI System instrumentation is no longer required. Information provided in the table is obsolete.	
Table 7.4.2	Automatic Depressurization System Instrumentation	Delete Table 7.4.2, ADS Instrumentation. Section 7.4.3.3, ADS Controls and Instrumentation has been deleted since it is no longer applicable. Therefore, the table listing ADS instrumentation is no longer required. Information provided in the table is obsolete.	
Table 7.4.3	Core Spray System Instrumentation	Delete Table 7.4.3, Core Spray System Instrumentation. Section 7.4.3.4, Core Spray System Controls and Instrumentation has been deleted since it is no longer applicable. Therefore, the table listing Core Spray System instrumentation is no longer required. Information provided in the table is obsolete.	
Table 7.4.4	Low Pressure Coolant Injection Instrumentation	Delete Table 7.4.4, LPCI System Instrumentation. Section 7.4.3.5, LPCI System Controls and Instrumentation has been deleted since it is no longer applicable. Therefore, the table listing LPCI System instrumentation is no longer required. Information provided in the table is obsolete.	
Figure 7.4-1A	High Pressure Coolant Injection System Arrangement G-191169	Delete Figure 7.4-1A, High Pressure Coolant Injection System Arrangement. The controls and instrumentation for the Core Standby Cooling Systems are identified as that equipment required for the initiation and control of HPCIS, ADS, CSS, and the LPCI function of RHR to ensure the fuel was adequately cooled under abnormal or accident conditions. The cooling provided by the systems restricted the release of radioactive materials from the fuel by limiting the extent of fuel damage following situations in which reactor coolant was lost from the nuclear system. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions of the CSCSs are no longer required. Since the functions of the CSCSs are no longer required, the instrumentation and Controls for those systems are also no longer required. Therefore, the information provided on Figure 7.4-1A is no longer applicable. Information regarding the Core Standby Cooling Systems (CSCS) Controls and Instrumentation is obsolete.	
Figure 7.4-1B	G-191169, Sh2 High Pressure Coolant Injection System, Turbine and Pump Unit	Delete Figure 7.4-1B, High Pressure Coolant Injection System, Turbine and Pump Unit. The controls and instrumentation for the Core Standby Cooling Systems are identified as that equipment required for the initiation and control of HPCIS, ADS, CSS, and the LPCI function of RHR to ensure the fuel was adequately cooled under abnormal or accident conditions. The cooling provided by the systems restricted the release of radioactive materials from the fuel by limiting the extent of fuel damage following situations in which reactor coolant was lost from the nuclear system. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions of the CSCSs are no longer required. Since the functions of the CSCSs are no longer required, the instrumentation and Controls for those systems are also no longer required. Therefore, the information provided on Figure 7.4-1B is no longer applicable. Information regarding the Core Standby Cooling Systems (CSCS) Controls and Instrumentation is obsolete.	
Figure 7.4-2a	5920-38 High Pressure Coolant Injection System Valves, Functional Control Diagram	Delete Figure 7.4-2a, High Pressure Coolant Injection System, Turbine and Pump Unit. The controls and instrumentation for the Core Standby Cooling Systems are identified as that equipment required for the initiation and control of HPCIS, ADS, CSS, and the LPCI function of RHR to ensure the fuel was adequately cooled under abnormal or accident conditions. The cooling provided by the systems restricted the release of radioactive materials from the fuel by limiting the extent of fuel damage following situations in which reactor coolant was lost from the nuclear system. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions of the CSCSs are no longer required. Since the functions of the CSCSs are no longer required, the instrumentation and Controls for those systems are also no longer required. Therefore, the information provided on Figure 7.4-2a is no longer applicable. Information regarding the Core Standby Cooling Systems (CSCS) Controls and Instrumentation is obsolete.	
Figure 7.4-2b	5920-39 High Pressure Coolant Injection System, Miscellaneous Components, Functional Control Diagram	Delete Figure 7.4-2b, High Pressure Coolant Injection System, Miscellaneous Components, Functional Control Diagram. The controls and instrumentation for the Core Standby Cooling Systems are identified as that equipment required for the initiation and control of HPCIS, ADS, CSS, and the LPCI function of RHR to ensure the fuel was adequately cooled under abnormal or accident conditions. The cooling provided by the systems restricted the release of radioactive materials from the fuel by limiting the extent of fuel damage following situations in which reactor coolant was lost from the nuclear system. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions of the CSCSs are no longer required. Since the functions of the CSCSs are no longer required, the instrumentation and Controls for those systems are also no longer required. Therefore, the information provided on Figure 7.4-2b is no longer applicable. Information regarding the Core Standby Cooling Systems (CSCS) Controls and Instrumentation is obsolete.	

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Figure 7.4-2c	5920-441 High Pressure Coolant Injection System, Turbine Functional Control Diagram	Delete Figure 7.4-2c, High Pressure Coolant Injection System, Turbine Functional Control Diagram. The controls and instrumentation for the Core Standby Cooling Systems are identified as that equipment required for the initiation and control of HPCIS, ADS, CSS, and the LPCI function of RHR to ensure the fuel was adequately cooled under abnormal or accident conditions. The coolin provided by the systems restricted the release of radioactive materials from the fuel by limiting the extent of fuel damage following situations in which reactor coolant was lost from the nuclear system. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions of the CSCSs are no longer required. Since the functions of the CSCSs are no longer required, the instrumentation and Controls for those systems are also no longer required. Therefore, the information provided on Figure 7.4-2c is no longer applicable. Information regarding the Core Standby Cooling Systems (CSCS) Controls and Instrumentation is obsolete.	
Figure 7.4-3	5920-611 Automatic Depressurization System, Functional Control Diagram	Delete Figure 7.4-3, Automatic Depressurization System, Functional Control Diagram. The controls and instrumentation for the Core Standby Cooling Systems are identified as that equipment required for the initiation and control of HPCIS, ADS, CSS, and the LPCI function of RHR to ensure the fuel was adequately cooled under abnormal or accident conditions. The cooling provided by the systems restricted the release of radioactive materials from the fuel by limiting the extent of fuel damage following situations in which reactor coolant was lost from the nuclear system. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions of the CSCSs are no longer required. Since the functions of the CSCSs are no longer required, the instrumentation and Controls for those systems are also no longer required. Therefore, the information provided on Figure 7.4-3 is no longer applicable. Information regarding the Core Standby Cooling Systems (CSCS) Controls and Instrumentation is obsolete.	
Figure 7.4-4	G-191168 Core Spray System Piping and Instrument Diagram	Delete Figure 7.4-4, Core Spray System Piping and Instrument Diagram. The controls and instrumentation for the Core Standby Cooling Systems are identified as that equipment required for the initiation and control of HPCIS, ADS, CSS, and the LPCI function of RHR to ensure the fuel was adequately cooled under abnormal or accident conditions. The cooling provided by the systems restricted the release of radioactive materials from the fuel by limiting the extent of fuel damage following situations in which reactor coolant was lost from the nuclear system. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions of the CSCSs are no longer required. Since the functions of the CSCSs are no longer required, the instrumentation and Controls for those systems are also no longer required. Therefore, the information provided on Figure 7.4-4 is no longer applicable. Information regarding the Core Standby Cooling Systems (CSCS) Controls and Instrumentation is obsolete.	
Figure 7.4-5	5920-37 Core Spray System, Functional Control Diagram	Delete Figure 7.4-5, Core Spray System, Functional Control Diagram. The controls and instrumentation for the Core Standby Cooling Systems are identified as that equipment required for the initiation and control of HPCIS, ADS, CSS, and the LPCI function of RHR to ensure the fuel was adequately cooled under abnormal or accident conditions. The cooling provided by the systems restricted the release of radioactive materials from the fuel by limiting the extent of fuel damage following situations in which reactor coolant was lost from the nuclear system. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions of the CSCSs are no longer required. Since the functions of the CSCSs are no longer required, the instrumentation and Controls for those systems are also no longer required. Therefore, the information provided on Figure 7.4-5 is no longer applicable. Information regarding the Core Standby Cooling Systems (CSCS) Controls and Instrumentation is obsolete.	
Figure 7.4-6	G-191172 Residual Heat Removal System, Piping and Instrumentation Diagram	Delete Figure 7.4-6, Residual Heat Removal System, Piping and Instrumentation Diagram. The controls and instrumentation for the Core Standby Cooling Systems are identified as that equipment required for the initiation and control of HPCIS, ADS, CSS, and the LPCI function of RHR to ensure the fuel was adequately cooled under abnormal or accident conditions. The coolin provided by the systems restricted the release of radioactive materials from the fuel by limiting the extent of fuel damage following situations in which reactor coolant was lost from the nuclear system. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions of the CSCSs are no longer required. Since the functions of the CSCSs are no longer required, the instrumentation and Controls for those systems are also no longer required. Therefore, the information provided on Figure 7.4-6 is no longer applicable. Information regarding the Core Standby Cooling Systems (CSCS) Controls and Instrumentation is obsolete.	
Figure 7.4-7a	5920-27 Residual Heat Removal System, LPCI Pump and Valves, Functional Control Diagram	Delete Figure 7.4-7a, Residual Heat Removal System, LPCI Pump and Valves, Functional Control Diagram. The controls and instrumentation for the Core Standby Cooling Systems are identified as that equipment required for the initiation and control of HPCIS, ADS, CSS, and the LPCI function of RHR to ensure the fuel was adequately cooled under abnormal or accident conditions. The cooling provided by the systems restricted the release of radioactive materials from the fuel by limiting the extent of fuel damage following situations in which reactor coolant was lost from the nuclear system. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions of the CSCSs are no longer required. Since the functions of the CSCSs are no longer required, the instrumentation and Controls for those systems are also no longer required. Therefore, the information provided on Figur 7.4-7a is no longer applicable. Information regarding the Core Standby Cooling Systems (CSCS) Controls and Instrumentation is obsolete.	

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	Figure 7.4-7b	5920-28 Residual Heat Removal System, LPCI Valves, Functional Control Diagram		Delete Figure 7.4-7b, Residual Heat Removal System, LPCI Valves, Functional Control Diagram. The controls and instrumentation for the Core Standby Cooling Systems are identified as that equipment required for the initiation and control of HPCIS, ADS, CSS, and the LPCI function of RHR to ensure the fuel was adequately cooled under abnormal or accident conditions. The cooling provided by the systems restricted the release of radioactive materials from the fuel by limiting the extent of fuel damage following situations in which reactor coolant was lost from the nuclear system. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions of the CSCSs are no longer required. Since the functions of the CSCSs are no longer required, the instrumentation and Controls for those systems are also no longer required. Therefore, the information provided on Figure 7.4-7b is no longer applicable. Information regarding the Core Standby Cooling Systems (CSCS) Controls and Instrumentation is obsolete.	
	Figure 7.4-7c	5920-29 Residual Heat Removal System, LPCI Miscellaneous Components Functional Control Diagram		Delete Figure 7.4-7c, Residual Heat Removal System, LPCI Miscellaneous Components Functional Control Diagram. The controls and instrumentation for the Core Standby Cooling Systems are identified as that equipment required for the initiation and control of HPCIS, ADS, CSS, and the LPCI function of RHR to ensure the fuel was adequately cooled under abnormal or accident conditions. The cooling provided by the systems restricted the release of radioactive materials from the fuel by limiting the extent of fuel damage following situations in which reactor coolant was lost from the nuclear system. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions of the CSCSs are no longer required. Since the functions of the CSCSs are no longer required, the instrumentation and Controls for those systems are also no longer required. Therefore, the information provided on Figure 7.4-7c is no longer applicable. Information regarding the Core Standby Cooling Systems (CSCS) Controls and Instrumentation is obsolete.	
	Figure 7.4-8	5920-1816 Recirculation Loop Valves, Functional Control Diagram		Delete Figure 7.4-8, Recirculation Loop Valves, Functional Control Diagram. The controls and instrumentation for the Core Standby Cooling Systems are identified as that equipment required for the initiation and control of HPCIS, ADS, CSS, and the LPCI function of RHR to ensure the fuel was adequately cooled under abnormal or accident conditions. The cooling provided by the systems restricted the release of radioactive materials from the fuel by limiting the extent of fuel damage following situations in which reactor coolant was lost from the nuclear system. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions of the CSCSs are no longer required. Since the functions of the CSCSs are no longer required, the instrumentation and Controls for those systems are also no longer required. Therefore, the information provided on Figure 7.4-8 is no longer applicable. Information regarding the Core Standby Cooling Systems (CSCS) Controls and Instrumentation is obsolete.	
7.5	NEUTRON MONITORING SYSTEM				
	7.5.1	Safety Objective		Delete Section 7.5.1, Neutron Monitoring System Safety Objective. The safety objective of the Neutron Monitoring System was to detect conditions in the core that threatened the overall integrity of the fuel barrier due to excessive power generation and to provide signals to the Reactor Protection System, so that the release of radioactive material from the fuel barrier was limited. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the Neutron Monitoring System are no longer required; the Neutron Monitoring System Safety Objective is not applicable since there is no core to protect.	
	7.5.2	Power Generation Objective		Delete Section 7.5.2, Neutron Monitoring System Power Generation Objective. The power generation objective of the Neutron Monitoring System was to provide information for the efficient, expedient operation and control of the reactor. Two specific power generation objectives of the Neutron Monitoring System were to detect conditions that could lead to local fuel damage and to provide signals that were used to prevent such damage, so that station availability was not reduced. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the Neutron Monitoring System are no longer required; the Neutron Monitoring System Power Generation Objective is not applicable since power operations can no longer occur.	
	7.5.3	Identification		Delete Section, provides only a title listing of the six major subsystems included in Neutron Monitoring .	
	7.5.4	Source Range Monitor Subsystem			
		7.5.4.1	Power Generation Design Basis	Delete Section 7.5.4.1, Source Range Monitor Subsystem, Power Generation Design Basis. The Source Range Monitor Subsystem of the Neutron Monitoring System provided nuclear core neutron flux level information and protective actions associated with certain conditions. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the Source Range Monitor Subsystem are no longer required; the Source Range Monitor Subsystem Power Generation Objective is not applicable since a nuclear core can no longer be loaded into the RPV.	
		7.5.4.2	Description	Delete Section 7.5.4.2, Source Range Monitor Subsystem, Description. The Source Range Monitor Subsystem of the Neutron Monitoring System provided nuclear core neutron flux level information and protective actions associated with certain conditions. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the Source Range Monitor Subsystem are no longer required; the information provided in the Source Range Monitor Subsystem Description Section s not applicable since a nuclear core can no longer be loaded into the RPV.	

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		7.5.4.3	Power Generation Evaluation	Delete Section 7.5.4.3, Source Range Monitor Subsystem, Power Generation Evaluation. The Source Range Monitor Subsystem of the Neutron Monitoring System provided nuclear core neutron flux level information and protective actions associated with certain conditions. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the Source Range Monitor Subsystem are no longer required; the information provided in the Source Range Monitor Subsystem Power Generation Evaluation Section s not applicable since a nuclear core can no longer be loaded into the RPV.	
		7.5.4.4	Inspection and Testing	Delete Section 7.5.4.4, Source Range Monitor Subsystem,Power Generation Evaluation. The Source Range Monitor Subsystem of the Neutron Monitoring System provided nuclear core neutron flux level information and protective actions associated with certain conditions. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the Source Range Monitor Subsystem are no longer required; the information provided in the Source Range Monitor Subsystem Power Generation Evaluation Section s not applicable since a nuclear core can no longer be loaded into the RPV.	
	7.5.5	Intermediate Range Monitor Subsystem			
		7.5.5.1	Safety Design Basis	Delete Section 7.5.5.1, Intermediate Range Monitor Subsystem Safety Design Basis. The safety design basis of the IRMS was to generate a trip signal to prevent fuel damage resulting from abnormal operational transients that occurred while operating in the intermediate power range Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the Intermediate Range Monitor Subsystem are no longer required; the Intermediate Range Monitor Subsystem Safety Design Basis is not applicable since there is no core to protect.	
		7.5.5.2	Power Generation Design Basis	Delete Section 7.5.5.2, Intermediate Range Monitor Subsystem Power Generation Design Basis. The power generation design basis of the IRMS was to generate a trip signal to block rod withdrawal if the IRMS reading exceeded a preset value or if the IRMS was not operating properly. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the Intermediate Range Monitor Subsystem are no longer required; the Intermediate Range Monitor Subsystem Power Generation Design Basis was not applicable since there is no core to protect.	
		7.5.5.3	Description	Delete Section7.5.5.3, Intermediate Range Monitor Subsystem Description. The IRMS functioned to generate a trip signal to prevent fuel damage resulting from abnormal operational transients that occurred while operating in the intermediate power range. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the protective functions of the IRMS are no longer required; the IRMS Description provided is not applicable since there is no core to protect. Information regarding the Reactor Protection System is obsolete.	
		7.5.5.4	Safety Evaluation	Delete Section7.5.5.4, Intermediate Range Monitor Subsystem Safety Evaluation. The IRMS functioned to generate a trip signal to prevent fuel damage resulting from abnormal operational transients that occurred while operating in the intermediate power range. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the IRMS are no longer required; a safety evaluation of the IRMS is not required since the functions of the IRMS are not required. Information regarding the IRMS is obsolete.	
		7.5.5.5	Power Generation Evaluation	Delete Section7.5.5.5, Intermediate Range Monitor Subsystem Power Generation Evaluation. The power generation function of the IRMS was to generate a trip signal to block rod withdrawal the IRMS reading exceeded a preset value or if the IRMS was not operating properly Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, a power generation evaluation of the IRMS is not required since the functions of the IRMS are not required and the IRMS has been categorized as abandoned in accordance with EC 48455. Information regarding the IRMS is obsolete.	
		7.5.5.6	Inspection and Testing	Delete Section 7.5.5.6, Intermediate Range Monitor Subsystem Inspection and Testing. The IRMS functioned to generate a trip signal to prevent fuel damage resulting from abnormal operational transients that occurred while operating in the intermediate power range. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, inspection and testing of the IRMS is not required since the functions of the IRMS are not required. The IRMS has been categorized as abandoned in accordance with EC 48455. Information regarding the IRMS is obsolete.	
	7.5.6	Local Power Range Monitor Subsystem			

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		7.5.6.1	Power Generation Design Basis	Delete Section 7.5.6.1, Local Power Range Monitor Subsystem Power Generation Design Basis. The LPRMS functioned to provide signals proportional to the local neutron flux at various locations within the reactor core to the Average Power Range Monitor Subsystem (APRMS), the Rod Block Monitor Subsystem, alarms, the process computer and local power indications. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the Local Power Range Monitor Subsystem are no longer required.	
		7.5.6.2	Description	Delete Section 7.5.6.2, Local Power Range Monitor Subsystem Description. The LPRMS functioned to provide signals proportional to the local neutron flux at various locations within the reactor core to the Average Power Range Monitor Subsystem (APRMS), the Rod Block Monitor Subsystem, alarms, the process computer and local power indications. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions of the LPRMS are no longer required; the LPRMS Description provided is not applicable. Information regarding the LPRMS is obsolete.	
		7.5.6.3	Power Generation Evaluation	Delete Section 7.5.6.3, Local Power Range Monitor Subsystem Power Generation Evaluation. The LPRMS functioned to provide signals proportional to the local neutron flux at various locations within the reactor core to the Average Power Range Monitor Subsystem (APRMS), the Rod Block Monitor Subsystem, alarms, the process computer and local power indications. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions of the Local Power Range Monitor Subsystem are no longer required; a power generation evaluation is not applicable.	
		7.5.6.4	Inspection and Testing	Delete Section 7.5.6.4, Local Power Range Monitor Subsystem Inspection and Testing. The LPRMS functioned to provide signals proportional to the local neutron flux at various locations within the reactor core to the Average Power Range Monitor Subsystem (APRMS), the Rod Block Monitor Subsystem, alarms, the process computer and local power indications. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, inspection and testing of the LPRMS is not required since the functions of the LPRMS are not required. Information regarding the LPRMS is obsolete.	
	7.5.7	Average Power Range Monitor Subsystem			
		7.5.7.1	Safety Design Basis	Delete Section 7.5.7.1, Average Power Range Monitor Subsystem Safety Design Basis. The APRMS functioned to generate a scram trip signal in response to average neutron flux increases resulting from abnormal operational transients in time to prevent fuel damage. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions of the Average Power Range Monitor Subsystem are no longer required; the Average Power Range Monitor Subsystem Safety Design Basis is not applicable.	
		7.5.7.2	Power Generation Design Basis	Delete Section 7.5.7.2, Average Power Range Monitor Subsystem Safety Design Basis. The APRMS functioned to generate a scram trip signal in response to average neutron flux increases resulting from abnormal operational transients in time to prevent fuel damage, provided a continuous indication of average reactor power and provided trip signal for blocking rod withdrawal. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the Average Power Range Monitor Subsystem are no longer required.	
		7.5.7.3	Description	Delete Section 7.5.7.3, Average Power Range Monitor Subsystem Description. The APRMS functioned to generate a scram trip signal in response to average neutron flux increases resulting from abnormal operational transients in time to prevent fuel damage, provided a continuous indication of average reactor power and provided trip signal for blocking rod withdrawal. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions of the APRMS are no longer required; the APRMS Description provided is not applicable. Information regarding the APRMS is obsolete.	
		7.5.7.4	Safety Evaluation	Delete Section 7.5.7.4, Average Power Range Monitor Subsystem Safety Evaluation The APRMS functioned to generate a scram trip signal in response to average neutron flux increases resulting from abnormal operational transients in time to prevent fuel damage, provided a continuous indication of average reactor power and provided trip signal for blocking rod withdrawal. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions of the APRMS are no longer required; a safety evaluation of the APRMS is not required since the functions of the APRMS are not required. Information regarding the APRMS is obsolete.	
		7.5.7.5	Power Generation Evaluation	Delete Section 7.5.7.5, Average Power Range Monitor Subsystem Safety Evaluation The APRMS functioned to generate a scram trip signal in response to average neutron flux increases resulting from abnormal operational transients in time to prevent fuel damage, provided a continuous indication of average reactor power and provided trip signal for blocking rod withdrawal. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions of the Average Power Range Monitor Subsystem are no longer required; a power generation evaluation is not applicable.	

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		7.5.7.6	Inspection and Testing	Delete Section 7.5.7.5, Average Power Range Monitor Subsystem Safety Evaluation The APRMS functioned to generate a scram trip signal in response to average neutron flux increases resulting from abnormal operational transients in time to prevent fuel damage, provided a continuous indication of average reactor power and provided trip signal for blocking rod withdrawal. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, inspection and testing of the APRMS is not required since the functions of the APRMS are not required. Information regarding the APRMS is obsolete.	
	7.5.8	Rod Block Monitor Subsystem			
		7.5.8.1	Power Generation Design Basis	Delete Section 7.5.8.1, Rod Block Monitor Subsystem Power Generation Design Basis. The RBMS functioned to prevent local fuel damage as a result of a single rod withdrawal error under the worst permitted condition of RBM bypass. RBMS also provided a signal to permit operator evaluation of the change in the local relative power level during control rod movement. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions of the Rod Block Monitor Subsystem are no longer required. The RBMS has been categorized as abandoned in accordance with EC 48455.	
		7.5.8.2	Description	Delete Section 7.5.8.2, Rod Block Monitor Subsystem Power Generation Description. The RBMS functioned to prevent local fuel damage as a result of a single rod withdrawal error under the worst permitted condition of RBM bypass. RBMS also provided a signal to permit operator evaluation of the change in the local relative power level during control rod movement. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions of the RBMS are no longer required; the RBMS Description provided is not applicable. The RBMS has been categorized as abandoned in accordance with EC 48455. Information regarding the RBMS is obsolete.	
		7.5.8.3	Power Generation Evaluation	Delete Section 7.5.8.3, Rod Block Monitor Subsystem Power Generation Evaluation The RBMS functioned to prevent local fuel damage as a result of a single rod withdrawal error under the worst permitted condition of RBM bypass. RBMS also provided a signal to permit operator evaluation of the change in the local relative power level during control rod movement. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions of the Rod Block Monitor Subsystem are no longer required; a power generation evaluation is not applicable.	
		7.5.8.4	Inspection and Testing	Delete Section 7.5.8.4, Rod Block Monitor Subsystem Inspection and Testing. The RBMS functioned to prevent local fuel damage as a result of a single rod withdrawal error under the worst permitted condition of RBM bypass. RBMS also provided a signal to permit operator evaluation of the change in the local relative power level during control rod movement. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, inspection and testing of the RBMS is not required since the functions of the RBMS are not required. Information regarding the RBMS is obsolete.	
	7.5.9	Traversing Incore Probe Subsystem			
		7.5.9.1	Power Generation Design Basis	Delete Section 7.5.9.1, Traversing Incore Probe Subsystem Power Generation Design Basis. The TIPS provided a signal proportional to the gamma flux, which was related to the axial neutron flux distribution at selected small axial intervals over the regions of the core where LPRM detector assemblies were located. This signal was high precision to allow reliable calibration of LPRM gains. The TIPS also provided accurate indication of the position of the flux measurement which allowed a pointwise or continuous measurement of the axial gamma flux distribution. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions of the TIP Subsystem are no longer required.	
		7.5.9.2	Description	Delete Section 7.5.9.2, Traversing Incore Probe Subsystem Power Generation Design Basis. The TIPS provided a signal proportional to the gamma flux, which was related to the axial neutron flux distribution at selected small axial intervals over the regions of the core where LPRM detector assemblies were located. This signal was high precision to allow reliable calibration of LPRM gains. The TIPS also provided accurate indication of the position of the flux measurement which allowed a pointwise or continuous measurement of the axial gamma flux distribution. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions of the TIPS are no longer required; the TIPS Description provided is not applicable. Information regarding the TIPS is obsolete.	
		7.5.9.3	Power Generation Evaluation	Delete Section 7.5.9.3, Traversing Incore Probe Subsystem Power Generation Design Basis. The TIPS provided a signal proportional to the gamma flux, which was related to the axial neutron flux distribution at selected small axial intervals over the regions of the core where LPRM detector assemblies were located. This signal was high precision to allow reliable calibration of LPRM gains. The TIPS also provided accurate indication of the position of the flux measurement which allowed a pointwise or continuous measurement of the axial gamma flux distribution. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions of the Traversing Incore Probe Subsystem are no longer required; a power generation evaluation is not applicable.	

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		7.5.9.4	Inspection and Testing	Delete Section 7.5.9.4, Traversing Incore Probe Subsystem Power Generation Design Basis. The TIPS provided a signal proportional to the gamma flux, which was related to the axial neutron flux distribution at selected small axial intervals over the regions of the core where LPRM detector assemblies were located. This signal was high precision to allow reliable calibration of LPRM gains. The TIPS also provided accurate indication of the position of the flux measurement which allowed a pointwise or continuous measurement of the axial gamma flux distribution. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, inspection and testing of the TIPS is not required since the functions of the TIPS are not required. Information regarding the TIPS is obsolete.	
	7.5.10	References		Delete Section 7.5.10, References. Since the neutron monitoring subsystems have all been abandoned these references no longer apply.	
	Table 7.5.1	SRM Trips		Delete Table 7.5.1, SRM Trips. Since Section 7.5.4, SRM Subsystem has been deleted, a table listing SRM trips is no longer required. This information is obsolete	
	Table 7.5.2	IRM Trips		Delete Table 7.5.2, IRM Trips. Since Section 7.5.5, IRM Subsystem has been deleted, a table listing IRM trips is no longer required. This information is obsolete	
	Table 7.5.3	LPRM Trips		Delete Table 7.5.3 LPRM Trips. Since Section 7.5.6, LPRM Subsystem has been deleted, a table listing LPRM trips is no longer required. This information is obsolete	
	Table 7.5.4	APRM Trips		Delete Table 7.5.4, APRM Trips. Since Section 7.5.7, APRM Subsystem has been deleted, a table listing APRM trips is no longer required. This information is obsolete	
	Table 7.5.5	RBM Trips		Delete Table 7.5.5, RBM Trips. Since Section 7.5.8, RBM Subsystem has been deleted, a table listing RBM trips is no longer required. This information is obsolete	
	Figure 7.5-1	5920-270 Startup Range Neutron Monitor, Instrument Electrical Diagram		Delete Figure 7.5-1, Startup Range Neutron Monitor, Instrument Electrical Diagram. The Neutron Monitoring System detected conditions in the core that threatened the overall integrity of the fuel barrier due to excessive power generation and provided signals to the Reactor Protection System, so that the release of radioactive material from the fuel barrier was limited. The Neutron Monitoring System also provided information for the efficient, expedient operation and control of the reactor. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the Neutron Monitoring System are no longer required, the information provided on Figure 7.5-1 is no longer applicable. Information regarding the Neutron Monitoring System is obsolete since there is no core to operate and protect.	
	Figure 7.5-2	5920-495 SRM/IRM Neutron Monitoring Unit		Delete Figure 7.5-2, SRM/IRM Neutron Monitoring Unit. The Neutron Monitoring System detected conditions in the core that threatened the overall integrity of the fuel barrier due to excessive power generation and provided signals to the Reactor Protection System, so that the release of radioactive material from the fuel barrier was limited. The Neutron Monitoring System also provided information for the efficient, expedient operation and control of the reactor. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the Neutron Monitoring System are no longer required, the information provided on Figure 7.5-2 is no longer applicable. Information regarding the Neutron Monitoring System is obsolete since there is no core to operate and protect.	
	Figure 7.5-3a	Detector Drive System Schematic		Delete Figure 7.5-3a, Detector Drive System Schematic. The Neutron Monitoring System detected conditions in the core that threatened the overall integrity of the fuel barrier due to excessive power generation and provided signals to the Reactor Protection System, so that the release of radioactive material from the fuel barrier was limited. The Neutron Monitoring System also provided information for the efficient, expedient operation and control of the reactor. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the Neutron Monitoring System are no longer required, the information provided on Figure 7.5-3a is no longer applicable. Information regarding the Neutron Monitoring System is obsolete since there is no core to operate and protect.	
	Figure 7.5-3b	5920-1812 SRM/IRM Detector Drive System, Functional Control Diagram		Delete Figure 7.5-3b, SRM/IRM Detector Drive System, Functional Control Diagram. The Neutron Monitoring System detected conditions in the core that threatened the overall integrity of the fuel barrier due to excessive power generation and provided signals to the Reactor Protection System, so that the release of radioactive material from the fuel barrier was limited. The Neutron Monitoring System also provided information for the efficient, expedient operation and control of the reactor. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the Neutron Monitoring System are no longer required, the information provided on Figure 7.5-3b is no longer applicable. Information regarding the Neutron Monitoring System is obsolete since there is no core to operate and protect.	

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Figure 7.5-4	Functional Block Diagram of SRM Channel	Delete Figure 7.5-4, Functional Block Diagram of SRM Channel. The Neutron Monitoring System detected conditions in the core that threatened the overall integrity of the fuel barrier due to excessive power generation and provided signals to the Reactor Protection System, so that the release of radioactive material from the fuel barrier was limited. The Neutron Monitoring System also provided information for the efficient, expedient operation and control of the reactor. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the Neutron Monitoring System are no longer required, the information provided on Figure 7.5-4 is no longer applicable. Information regarding the Neutron Monitoring System is obsolete since there is no core to operate and protect.	
Figure 7.5-5	5920-1807 Neutron Monitoring System, Functional Control Diagram	Delete Figure 7.5-5, Neutron Monitoring System, Functional Control Diagram. The Neutron Monitoring System detected conditions in the core that threatened the overall integrity of the fuel barrier due to excessive power generation and provided signals to the Reactor Protection System, so that the release of radioactive material from the fuel barrier was limited. The Neutron Monitoring System also provided information for the efficient, expedient operation and control of the reactor. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the Neutron Monitoring System are no longer required, the information provided on Figure 7.5-5 is no longer applicable. Information regarding the Neutron Monitoring System is obsolete since there is no core to operate and protect.	
Figure 7.5-6	Source Range Monitoring System Core Locations	Delete Figure 7.5-6, Source Range Monitoring System Core Locations. The Neutron Monitoring System detected conditions in the core that threatened the overall integrity of the fuel barrier due to excessive power generation and provided signals to the Reactor Protection System, so that the release of radioactive material from the fuel barrier was limited. The Neutron Monitoring System also provided information for the efficient, expedient operation and control of the reactor. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the Neutron Monitoring System are no longer required, the information provided on Figure 7.5-6 is no longer applicable. Information regarding the Neutron Monitoring System is obsolete since there is no core to operate and protect.	
Figure 7.5-7	Functional Block Diagram of IRM Channel	Delete Figure 7.5-7, Functional Block Diagram of IRM Channel. The Neutron Monitoring System detected conditions in the core that threatened the overall integrity of the fuel barrier due to excessive power generation and provided signals to the Reactor Protection System, so that the release of radioactive material from the fuel barrier was limited. The Neutron Monitoring System also provided information for the efficient, expedient operation and control of the reactor. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the Neutron Monitoring System are no longer required, the information provided on Figure 7.5-7 is no longer applicable. Information regarding the Neutron Monitoring System is obsolete since there is no core to operate and protect.	
Figure 7.5-8	IRM Locations	Delete Figure 7.5-8, IRM Locations. The Neutron Monitoring System detected conditions in the core that threatened the overall integrity of the fuel barrier due to excessive power generation and provided signals to the Reactor Protection System, so that the release of radioactive material from the fuel barrier was limited. The Neutron Monitoring System also provided information for the efficient, expedient operation and control of the reactor. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the Neutron Monitoring System are no longer required, the information provided on Figure 7.5-8 is no longer applicable. Information regarding the Neutron Monitoring System is obsolete since there is no core to operate and protect.	
Figure 7.5-9	Deleted	Previously Deleted	
Figure 7.5-10	Deleted	Previously Deleted	
Figure 7.5-11	5920-271 Power Range Neutron Monitoring, Instrument Electrical Diagram	Delete Figure 7.5-11, Power Range Neutron Monitoring, Instrument Electrical Diagram. The Neutron Monitoring System detected conditions in the core that threatened the overall integrity of the fuel barrier due to excessive power generation and provided signals to the Reactor Protection System, so that the release of radioactive material from the fuel barrier was limited. The Neutron Monitoring System also provided information for the efficient, expedient operation and control of the reactor. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the Neutron Monitoring System are no longer required, the information provided on Figure 7.5-11 is no longer applicable. Information regarding the Neutron Monitoring System is obsolete since there is no core to operate and protect.	

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Figure 7.5-12	LPRM Locations	Delete Figure 7.5-12, LPRM Locations. The Neutron Monitoring System detected conditions in the core that threatened the overall integrity of the fuel barrier due to excessive power generation and provided signals to the Reactor Protection System, so that the release of radioactive material from the fuel barrier was limited. The Neutron Monitoring System also provided information for the efficient, expedient operation and control of the reactor. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the Neutron Monitoring System are no longer required, the information provided on Figure 7.5-12 is no longer applicable. Information regarding the Neutron Monitoring System is obsolete since there is no core to operate and protect.	
Figure 7.5-13	5920-496 Power Range Neutron Monitoring Unit	Delete Figure 7.5-13, Power Range Neutron Monitoring Unit. The Neutron Monitoring System detected conditions in the core that threatened the overall integrity of the fuel barrier due to excessive power generation and provided signals to the Reactor Protection System, so that the release of radioactive material from the fuel barrier was limited. The Neutron Monitoring System also provided information for the efficient, expedient operation and control of the reactor. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the Neutron Monitoring System are no longer required, the information provided on Figure 7.5-13 is no longer applicable. Information regarding the Neutron Monitoring System is obsolete since there is no core to operate and protect.	
Figure 7.5-14a	LPRM to APRM Assignment Scheme (System A)	Delete Figure 7.5-14a, LPRM to APRM Assignment Scheme (System A). The Neutron Monitoring System detected conditions in the core that threatened the overall integrity of the fuel barrier due to excessive power generation and provided signals to the Reactor Protection System, so that the release of radioactive material from the fuel barrier was limited. The Neutron Monitoring System also provided information for the efficient, expedient operation and control of the reactor. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the Neutron Monitoring System are no longer required, the information provided on Figure 7.5-14a is no longer applicable. Information regarding the Neutron Monitoring System is obsolete since there is no core to operate and protect.	
Figure 7.5-14b	LPRM to APRM Assignment Scheme (System B)	Delete Figure 7.5-14b, LPRM to APRM Assignment Scheme (System B). The Neutron Monitoring System detected conditions in the core that threatened the overall integrity of the fuel barrier due to excessive power generation and provided signals to the Reactor Protection System, so that the release of radioactive material from the fuel barrier was limited. The Neutron Monitoring System also provided information for the efficient, expedient operation and control of the reactor. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the Neutron Monitoring System are no longer required, the information provided on Figure 7.5-14b is no longer applicable. Information regarding the Neutron Monitoring System is obsolete since there is no core to operate and protect.	
Figure 7.5-15	APRM Tracking, Reduction in Power By Flow Control	Delete Figure 7.5-15, APRM Tracking, Reduction in Power By Flow Control. The Neutron Monitoring System detected conditions in the core that threatened the overall integrity of the fuel barrier due to excessive power generation and provided signals to the Reactor Protection System, so that the release of radioactive material from the fuel barrier was limited. The Neutron Monitoring System also provided information for the efficient, expedient operation and control of the reactor. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the Neutron Monitoring System are no longer required, the information provided on Figure 7.5-15 is no longer applicable. Information regarding the Neutron Monitoring System is obsolete since there is no core to operate and protect.	
Figure 7.5-16	APRM Tracking With On Limits Control Rod Withdrawal	Delete Figure 7.5-16, APRM Tracking With On Limits Control Rod Withdrawal. The Neutron Monitoring System detected conditions in the core that threatened the overall integrity of the fuel barrier due to excessive power generation and provided signals to the Reactor Protection System, so that the release of radioactive material from the fuel barrier was limited. The Neutron Monitoring System also provided information for the efficient, expedient operation and control of the reactor. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the Neutron Monitoring System are no longer required, the information provided on Figure 7.5-16 is no longer applicable. Information regarding the Neutron Monitoring System is obsolete since there is no core to operate and protect.	

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Figure 7.5-17	Assignment of LPRM Assemblies to RBM's	Delete Figure 7.5-17, Assignment of LPRM Assemblies to RBM's. The Neutron Monitoring System detected conditions in the core that threatened the overall integrity of the fuel barrier due to excessive power generation and provided signals to the Reactor Protection System, so that the release of radioactive material from the fuel barrier was limited. The Neutron Monitoring System also provided information for the efficient, expedient operation and control of the reactor. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the Neutron Monitoring System are no longer required, the information provided on Figure 7.5-17 is no longer applicable. Information regarding the Neutron Monitoring System is obsolete since there is no core to operate and protect.	
Figure 7.5-18	RBM Channel A+C Response to Control Rod Motion	Delete Figure 7.5-18, RBM Channel A+C Response to Control Rod Motion. The Neutron Monitoring System detected conditions in the core that threatened the overall integrity of the fuel barrier due to excessive power generation and provided signals to the Reactor Protection System, so that the release of radioactive material from the fuel barrier was limited. The Neutron Monitoring System also provided information for the efficient, expedient operation and control of the reactor. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the Neutron Monitoring System are no longer required, the information provided on Figure 7.5-18 is no longer applicable. Information regarding the Neutron Monitoring System is obsolete since there is no core to operate and protect.	
Figure 7.5-19	RBM Channel B+D Response to Control Rod Motion	Delete Figure 7.5-19, RBM Channel B+D Response to Control Rod Motion. The Neutron Monitoring System detected conditions in the core that threatened the overall integrity of the fuel barrier due to excessive power generation and provided signals to the Reactor Protection System, so that the release of radioactive material from the fuel barrier was limited. The Neutron Monitoring System also provided information for the efficient, expedient operation and control of the reactor. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the Neutron Monitoring System are no longer required, the information provided on Figure 7.5-19 is no longer applicable. Information regarding the Neutron Monitoring System is obsolete since there is no core to operate and protect.	
Figure 7.5-20	Assignment of LPRM Strings to TIP Machines	Delete Figure 7.5-20, Assignment of LPRM Strings to TIP Machines. The Neutron Monitoring System detected conditions in the core that threatened the overall integrity of the fuel barrier due to excessive power generation and provided signals to the Reactor Protection System, so that the release of radioactive material from the fuel barrier was limited. The Neutron Monitoring System also provided information for the efficient, expedient operation and control of the reactor. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the Neutron Monitoring System are no longer required, the information provided on Figure 7.5-20 is no longer applicable. Information regarding the Neutron Monitoring System is obsolete since there is no core to operate and protect.	
Figure 7.5-21	Traversing Incore Probe Subsystem Block Diagram	Delete Figure 7.5-21, Traversing Incore Probe Subsystem Block Diagram. The Neutron Monitoring System detected conditions in the core that threatened the overall integrity of the fuel barrier due to excessive power generation and provided signals to the Reactor Protection System, so that the release of radioactive material from the fuel barrier was limited. The Neutron Monitoring System also provided information for the efficient, expedient operation and control of the reactor. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the Neutron Monitoring System are no longer required, the information provided on Figure 7.5-21 is no longer applicable. Information regarding the Neutron Monitoring System is obsolete since there is no core to operate and protect.	
Figure 7.5-22	5920-9365 Gamma Traversing Incore Probe Assembly	Delete Figure 7.5-22, Gamma Traversing Incore Probe Assembly. The Neutron Monitoring System detected conditions in the core that threatened the overall integrity of the fuel barrier due to excessive power generation and provided signals to the Reactor Protection System, so that the release of radioactive material from the fuel barrier was limited. The Neutron Monitoring System also provided information for the efficient, expedient operation and control of the reactor. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the Neutron Monitoring System are no longer required, the information provided on Figure 7.5-22 is no longer applicable. Information regarding the Neutron Monitoring System is obsolete since there is no core to operate and protect.	

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	Figure 7.5-23a	5920-426 Neutron Monitoring System Arrangement	Delete Figure 7.5-23a, Neutron Monitoring System Arrangement. The Neutron Monitoring System detected conditions in the core that threatened the overall integrity of the fuel barrier due to excessive power generation and provided signals to the Reactor Protection System, so that the release of radioactive material from the fuel barrier was limited. The Neutron Monitoring System also provided information for the efficient, expedient operation and control of the reactor. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the Neutron Monitoring System are no longer required, the information provided on Figure 7.5-23a is no longer applicable. Information regarding the Neutron Monitoring System is obsolete since there is no core to operate and protect.	
	Figure 7.5-23b	5920-427 Neutron Monitoring System Arrangement	Delete Figure 7.5-23b, Neutron Monitoring System Arrangement. The Neutron Monitoring System detected conditions in the core that threatened the overall integrity of the fuel barrier due to excessive power generation and provided signals to the Reactor Protection System, so that the release of radioactive material from the fuel barrier was limited. The Neutron Monitoring System also provided information for the efficient, expedient operation and control of the reactor. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the Neutron Monitoring System are no longer required, the information provided on Figure 7.5-23b is no longer applicable. Information regarding the Neutron Monitoring System is obsolete since there is no core to operate and protect.	
	Figure 7.5-24	5920-1813 Traversing Incore Probe, Functional Control Diagram	Delete Figure 7.5-24, Traversing Incore Probe, Functional Control Diagram. The Neutron Monitoring System detected conditions in the core that threatened the overall integrity of the fuel barrier due to excessive power generation and provided signals to the Reactor Protection System, so that the release of radioactive material from the fuel barrier was limited. The Neutron Monitoring System also provided information for the efficient, expedient operation and control of the reactor. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the Neutron Monitoring System are no longer required, the information provided on Figure 7.5-24 is no longer applicable. Information regarding the Neutron Monitoring System is obsolete since there is no core to operate and protect.	
	Figure 7.5-25	Ranges of Neutron Monitoring System	Delete Figure 7.5-25, Ranges of Neutron Monitoring System. The Neutron Monitoring System detected conditions in the core that threatened the overall integrity of the fuel barrier due to excessive power generation and provided signals to the Reactor Protection System, so that the release of radioactive material from the fuel barrier was limited. The Neutron Monitoring System also provided information for the efficient, expedient operation and control of the reactor. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the Neutron Monitoring System are no longer required, the information provided on Figure 7.5-25 is no longer applicable. Information regarding the Neutron Monitoring System is obsolete since there is no core to operate and protect.	
	Figure 7.5-26	IRM Circuit Arrangement for Reactor Protection System Input	Delete Figure 7.5-26, IRM Circuit Arrangement for Reactor Protection System Input. The Neutron Monitoring System detected conditions in the core that threatened the overall integrity of the fuel barrier due to excessive power generation and provided signals to the Reactor Protection System, so that the release of radioactive material from the fuel barrier was limited. The Neutron Monitoring System also provided information for the efficient, expedient operation and control of the reactor. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the Neutron Monitoring System are no longer required, the information provided on Figure 7.5-26 is no longer applicable. Information regarding the Neutron Monitoring System is obsolete since there is no core to operate and protect.	
	Figure 7.5-27	Typical APRM Circuit Arrangement for Reactor Protection System Input	Delete Figure 7.5-27, Typical APRM Circuit Arrangement for Reactor Protection System Input. The Neutron Monitoring System detected conditions in the core that threatened the overall integrity of the fuel barrier due to excessive power generation and provided signals to the Reactor Protection System, so that the release of radioactive material from the fuel barrier was limited. The Neutron Monitoring System also provided information for the efficient, expedient operation and control of the reactor. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions of the Neutron Monitoring System are no longer required, the information provided on Figure 7.5-27 is no longer applicable. Information regarding the Neutron Monitoring System is obsolete since there is no core to operate and protect.	
7.6	REFUELING INTERLOCKS			

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	7.6.1	Power Generation Objective	Delete Section 7.6.1, Refueling Interlocks Power Generation Objective. The refueling interlocks prevented an inadvertent criticality during refueling operations by reinforcing operational procedures that prohibited taking the reactor critical under certain situations encountered during refueling operations by restricting the movement of control rods and the operation of refueling equipment. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, refueling operations will no longer occur and protective functions to prevent an inadvertent criticality during refueling operations are no longer required. Consequently, the preventative functions provided by the Refueling Interlocks are no longer required. Refueling Interlocks information is obsolete.	
	7.6.2	Power Generation Design Bases	Delete Section 7.6.2, Refueling Interlocks Power Generation Design Bases. The refueling interlocks prevented an inadvertent criticality during refueling operations by reinforcing operational procedures that prohibited taking the reactor critical under certain situations encountered during refueling operations by restricting the movement of control rods and the operation of refueling equipment. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, refueling operations will no longer occur and protective functions to prevent an inadvertent criticality during refueling operations are no longer required. Consequently, the preventative functions provided by the Refueling Interlocks are no longer required. Refueling Interlocks information is obsolete.	
	7.6.3	Description	Delete Section 7.6.3, Refueling Interlocks Description. The refueling interlocks prevented an inadvertent criticality during refueling operations by reinforcing operational procedures that prohibited taking the reactor critical under certain situations encountered during refueling operations by restricting the movement of control rods and the operation of refueling equipment. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, refueling operations will no longer occur and protective functions to prevent an inadvertent criticality during refueling operations are no longer required. Consequently, the preventative functions provided by the Refueling Interlocks are no longer required. The information provided in the Refueling Interlocks Description Section is no longer applicable. Refueling Interlocks information is obsolete.	
	7.6.4	Inspection and Testing	Delete Section 7.6.4, Refueling Interlocks Inspection and Testing. The refueling interlocks prevented an inadvertent criticality during refueling operations by reinforcing operational procedures that prohibited taking the reactor critical under certain situations encountered during refueling operations by restricting the movement of control rods and the operation of refueling equipment. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, refueling operations will no longer occur and protective functions to prevent an inadvertent criticality during refueling operations are no longer required. Consequently, the preventative functions provided by the Refueling Interlocks are no longer required, Therefore, inspection and testing of the refueling interlocks is no longer required. Refueling Interlocks information is obsolete.	
	Table 7.6.1	Refueling Intelock Effectiveness	Delete Table 7.6.1, Refueling Interlock Effectiveness. Since it is no longer possible to load a nuclear core, refueling operations will no longer occur and protective functions to prevent an inadvertent criticality during refueling operations are no longer required. Consequently, the preventative functions provided by the Refueling Interlocks are no longer required. Therefore, a table listing the effectiveness of the refueling interlocks is no longer required. Refueling Interlocks information is obsolete.	
	Figure 7.6-1	Refueling Interlocks	Delete Figure 7.6-1, Refueling Interlocks. The refueling interlocks prevented an inadvertent criticality during refueling operations by reinforcing operational procedures that prohibited taking the reactor critical under certain situations encountered during refueling operations by restricting the movement of control rods and the operation of refueling equipment. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, refueling operations will no longer occur and protective functions to prevent an inadvertent criticality during refueling operations are no longer required. Consequently, the preventative functions provided by the Refueling Interlocks are no longer required. The information provided on Figure 7.6-1 is no longer applicable. Refueling Interlocks information is obsolete.	
	Figure 7.6-2A	Control Rod Drive Hydraulic System, Refueling Interlocks, Functional Control Diagram 5920-1638	Delete Figure 7.6-2A, Control Rod Drive Hydraulic System, Refueling Interlocks, Functional Control Diagram. The refueling interlocks prevented an inadvertent criticality during refueling operations by reinforcing operational procedures that prohibited taking the reactor critical under certain situations encountered during refueling operations by restricting the movement of control rods and the operation of refueling equipment. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, refueling operations will no longer occur and protective functions to prevent an inadvertent criticality during refueling operations are no longer required. Consequently, the preventative functions provided by the Refueling Interlocks are no longer required. The information provided on Figure 7.6-2A is no longer applicable. Refueling Interlocks information is obsolete.	
	Figure 7.6-2B	Control Rod Drive Hydraulic System, Refueling Interlocks, Functional Control Diagram 5920-1639	Delete Figure 7.6-2B, Control Rod Drive Hydraulic System, Refueling Interlocks, Functional Control Diagram. The refueling interlocks prevented an inadvertent criticality during refueling operations by reinforcing operational procedures that prohibited taking the reactor critical under certain situations encountered during refueling operations by restricting the movement of control rods and the operation of refueling equipment. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, refueling operations will no longer occur and protective functions to prevent an inadvertent criticality during refueling operations are no longer required. Consequently, the preventative functions provided by the Refueling Interlocks are no longer required. The information provided on Figure 7.6-2B is no longer applicable. Refueling Interlocks information is obsolete.	

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7.7	REACTOR MANUAL CONTROL SYSTEM			The Reactor Manual Control System provided the operator with the means to make changes in nuclear reactivity so that reactor power level and power distribution could be controlled. The system allowed the operator to manipulate control rods.	
	7.7.1	Power Generation Objective		Delete Section 7.7.1, Reactor Manual Control System Power Generation Objective. The Reactor Manual Control System provided the operator with the means to make changes in nuclear reactivity so that reactor power level and power distribution could be controlled. The system allowed the operator to manipulate control rods. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur, it is no longer possible to change core nuclear reactivity and control rod manipulation will no longer occur. Consequently, the functions provided by the Reactor Manual Control System are no longer required. Reactor Manual Control System information is obsolete.	
	7.7.2	Safety Design Bases		Delete Section 7.7.2, Reactor Manual Control System Safety Design Bases. The Reactor Manual Control System provided the operator with the means to make changes in nuclear reactivity so that reactor power level and power distribution could be controlled. The system allowed the operator to manipulate control rods. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur, it is no longer possible to change core nuclear reactivity and control rod manipulation will no longer occur. Consequently, the functions provided by the Reactor Manual Control System are no longer required. Safety Design Bases for the Reactor Manual Control System are no longer applicable. Reactor Manual Control System information is obsolete.	
	7.7.3	Power Generation Design Bases		Delete Section 7.7.3, Reactor Manual Control System Power Generation Design Bases. The Reactor Manual Control System provided the operator with the means to make changes in nuclear reactivity so that reactor power level and power distribution could be controlled. The system allowed the operator to manipulate control rods. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur, it is no longer possible to change core nuclear reactivity and control rod manipulation will no longer occur. Consequently, the functions provided by the Reactor Manual Control System are no longer required. Power Generation Design Bases for the Reactor Manual Control System are no longer applicable. Reactor Manual Control System information is obsolete.	
	7.7.4	Description			
		7.7.4.1	Identification	Delete Section 7.7.4.1, Reactor Manual Control System Description, Identification. The Reactor Manual Control System provided the operator with the means to make changes in nuclear reactivity so that reactor power level and power distribution could be controlled. The system allowed the operator to manipulate control rods. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions provided by the Reactor Manual Control System are no longer required; the information provided to identify the components of the Reactor Manual Control System is no l onger applicable. Reactor Manual Control System information is obsolete.	
		7.7.4.2	Operation	Delete Section 7.7.4.2, Reactor Manual Control System Description, Operation. The Reactor Manual Control System provided the operator with the means to make changes in nuclear reactivity so that reactor power level and power distribution could be controlled. The system allowed the operator to manipulate control rods. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions provided by the Reactor Manual Control System are no longer required; the information provided to describe the operation of the Reactor Manual Control System is no longer applicable. Reactor Manual Control System information is obsolete.	
		7.7.4.3	Rod Block Interlocks	Delete Section 7.7.4.3, Reactor Manual Control System Description, Rod Block Interlocks. The Reactor Manual Control System provided the operator with the means to make changes in nuclear reactivity so that reactor power level and power distribution could be controlled. The system allowed the operator to manipulate control rods. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions provided by the Reactor Manual Control System, including rod blocks, are no longer required; the information provided to describe the management of control rod blocks by the Reactor Manual Control System is no longer applicable. Reactor Manual Control System information is obsolete.	
		7.7.4.4	Instrumentation	Delete Section 7.7.4.4, Reactor Manual Control System Description, Instrumentation. The Reactor Manual Control System provided the operator with the means to make changes in nuclear reactivity so that reactor power level and power distribution could be controlled. The system allowed the operator to manipulate control rods. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions provided by the Reactor Manual Control System, including rod blocks, are no longer required; the information provided to describe the management of control rod blocks by the Reactor Manual Control System is no longer applicable. Reactor Manual Control System information is obsolete.	

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	7.7.5	Safety Evaluation	Delete Section 7.7.5, Reactor Manual Control System Safety Evaluation. The Reactor Manual Control System provided the operator with the means to make changes in nuclear reactivity so tha reactor power level and power distribution could be controlled. The system allowed the operator to manipulate control rods. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions of the Reactor Manual Control System are no longer required; a safety evaluation of the Reactor Manual Control System is not required since the functions of the RMCS are not required. Information regarding the RMCS is obsolete.	
	7.7.6	Inspection and Testing	Delete Section 7.7.6, Reactor Manual Control System Safety Evaluation. The Reactor Manual Control System provided the operator with the means to make changes in nuclear reactivity so tha reactor power level and power distribution could be controlled. The system allowed the operator to manipulate control rods. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions provided by the Reactor Manual Control System are no longer required. Therefore, inspection and testing of the Reactor Manual Control System is no longer required. Reactor Manual Control System information is obsolete.	
	Table 7.7.1	Reactor Manual Control System Instrumentation	Delete Table 7.7.1, Reactor Manual Control System Instrumentation. Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions provided by the Reactor Manual Control System are no longer required. Therefore, a table listing the Reactor Manual Control System Instrumentation is no longer required. Reactor Manual Control System information is obsolete.	
	Figure 7.7-1a	5920-307 Control Rod Drive Hydraulic System, Functional Control Diagram	Delete Figure 7.7-1a, Control Rod Drive Hydraulic System, Functional Control Diagram. The Reactor Manual Control System provided the operator with the means to make changes in nuclear reactivity so that reactor power level and power distribution could be controlled. The system allowed the operator to manipulate control rods. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the Reactor Manual Control System are no longer required. The information provided on Figure 7.7-1a is no longer applicable. Reactor Manual Control System information is obsolete.	
	Figure 7.7-1b	5920-308 Control Rod Drive Hydraulic System, Functional Control Diagram	Delete Figure 7.7-1b, Control Rod Drive Hydraulic System, Functional Control Diagram. The Reactor Manual Control System provided the operator with the means to make changes in nuclear reactivity so that reactor power level and power distribution could be controlled. The system allowed the operator to manipulate control rods. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the Reactor Manual Control System are no longer required. The information provided on Figure 7.7-1b is no longer applicable. Reactor Manual Control System information is obsolete.	
	Figure 7.7-1c	5920-1636 Control Rod Drive Hydraulic System, Functional Control Diagram	Delete Figure 7.7-1c, Control Rod Drive Hydraulic System, Functional Control Diagram. The Reactor Manual Control System provided the operator with the means to make changes in nuclear reactivity so that reactor power level and power distribution could be controlled. The system allowed the operator to manipulate control rods. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the Reactor Manual Control System are no longer required. The information provided on Figure 7.7-1c is no longer applicable. Reactor Manual Control System information is obsolete.	
	Figure 7.7-1d	5920-1637 Control Rod Drive Hydraulic System, Functional Control Diagram	Delete Figure 7.7-1d, Control Rod Drive Hydraulic System, Functional Control Diagram. The Reactor Manual Control System provided the operator with the means to make changes in nuclear reactivity so that reactor power level and power distribution could be controlled. The system allowed the operator to manipulate control rods. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the Reactor Manual Control System are no longer required. The information provided on Figure 7.7-1d is no longer applicable. Reactor Manual Control System information is obsolete.	
	Figure 7.7-1e	5920-1638 Control Rod Drive Hydraulic System, Functional Control Diagram	Delete Figure 7.7-1e, Control Rod Drive Hydraulic System, Functional Control Diagram. The Reactor Manual Control System provided the operator with the means to make changes in nuclear reactivity so that reactor power level and power distribution could be controlled. The system allowed the operator to manipulate control rods. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the Reactor Manual Control System are no longer required. The information provided on Figure 7.7-1e is no longer applicable. Reactor Manual Control System information is obsolete.	

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	Figure 7.7-2	Reactor Control Board	Delete Figure 7.7-2, Reactor Control Board. The Reactor Manual Control System provided the operator with the means to make changes in nuclear reactivity so that reactor power level and power distribution could be controlled. The system allowed the operator to manipulate control rods. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the Reactor Manual Control System are no longer required. The information provided on Figure 7.7-2 is no longer applicable. Reactor Manual Control System information is obsolete.	
	Figure 7.7-3	RBM Flow Variance Circuit, Schematic	Delete Figure 7.7-3, RBM Flow Variance Circuit, Schematic. The Reactor Manual Control System provided the operator with the means to make changes in nuclear reactivity so that reactor power level and power distribution could be controlled. The system allowed the operator to manipulate control rods. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the Reactor Manual Control System are no longer required. The information provided on Figure 7.7-3 is no longer applicable. Reactor Manual Control System information is obsolete.	
	Figure 7.7-4	Input Signals to Four-Rod Display	Delete Figure 7.7-4, Input Signals to Four-Rod Display. The Reactor Manual Control System provided the operator with the means to make changes in nuclear reactivity so that reactor power level and power distribution could be controlled. The system allowed the operator to manipulate control rods. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the Reactor Manual Control System are no longer required. The information provided on Figure 7.7-4 is no longer applicable. Reactor Manual Control System information is obsolete.	
	Figure 7.7-5	Control Rod Positions Display Typical Process Computer Printout	Delete Figure 7.7-5, Control Rod Positions Display Typical Process Computer Printout. The Reactor Manual Control System provided the operator with the means to make changes in nuclear reactivity so that reactor power level and power distribution could be controlled. The system allowed the operator to manipulate control rods. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the Reactor Manual Control System are no longer required. The information provided on Figure 7.7-5 is no longer applicable. Reactor Manual Control System information is obsolete.	
7.8	REACTOR VESSEL INSTRUMENTATION		Reactor vessel instrumentation monitored and transmitted information concerning key reactor vessel operating parameters during planned operations to ensure that sufficient control of these parameters was possible in order to avoid a release of radioactive material to the environs such that the limits to 10CFR20 were exceeded, avoid nuclear system stress in excess of that allowed by applicable industry codes, and avoid any operating conditions not considered by station safety analyses.	
	7.8.1	Power Generation Objective	Delete Section 7.8.1, Reactor Vessel Instrumentation, Power Generation Objective. Reactor vessel instrumentation monitored and transmitted information concerning key reactor vessel operating parameters during planned operations to ensure that sufficient control of these parameters was possible in order to avoid a release of radioactive material to the environs such that the limits to 10CFR20 were exceeded, avoid nuclear system stress in excess of that allowed by applicable industry codes, and avoid any operating conditions not considered by station safety analyses. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the Reactor Vessel Instrumentation are no longer required; the RVI Power Generation Objective is no longer applicable. Reactor Vessel Instrumentation information is obsolete.	
	7.8.2	Safety Objective	Delete Section 7.8.2, Reactor Vessel Instrumentation, Safety Objective. Reactor vessel instrumentation monitored and transmitted information concerning key reactor vessel operating parameters during planned operations to ensure that sufficient control of these parameters was possible in order to avoid a release of radioactive material to the environs such that the limits to 10CFR20 were exceeded, avoid nuclear system stress in excess of that allowed by applicable industry codes, and avoid any operating conditions not considered by station safety analyses. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the Reactor Vessel Instrumentation are no longer required; the RVI Safety Objective is no longer applicable. Reactor Vessel Instrumentation information is obsolete.	

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	7.8.3	Power Generation Design Bases		Delete Section 7.8.3, Reactor Vessel Instrumentation, Power Generation Design Bases. Reactor vessel instrumentation monitored and transmitted information concerning key reactor vessel operating parameters during planned operations to ensure that sufficient control of these parameters was possible in order to avoid a release of radioactive material to the environs such that the limits to 10CFR20 were exceeded, avoid nuclear system stress in excess of that allowed by applicable industry codes, and avoid any operating conditions not considered by station safety analyses. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the Reactor Vessel Instrumentation are no longer required; the RVI Power Generation Design Bases are no longer applicable. Reactor Vessel Instrumentation information is obsolete.	
	7.8.4	Safety Design Bases		Delete Section 7.8.4, Reactor Vessel Instrumentation, Safety Design Bases. Reactor vessel instrumentation monitored and transmitted information concerning key reactor vessel operating parameters during planned operations to ensure that sufficient control of these parameters was possible in order to avoid a release of radioactive material to the environs such that the limits t 10CFR20 were exceeded, avoid nuclear system stress in excess of that allowed by applicable industry codes, and avoid any operating conditions not considered by station safety analyses. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the Reactor Vessel Instrumentation are no longer required; the RVI Safety Design Bases are no longer applicable. Reactor Vessel Instrumentation information is obsolete.	
	7.8.5	Description			
		7.8.5.1	Reactor Vessel Surface Temperature	Delete Section 7.8.5.1, Reactor Vessel Instrumentation, Reactor Vessel Surface Temperature. Reactor vessel instrumentation monitored and transmitted information concerning key reactor vessel operating parameters during planned operations to ensure that sufficient control of these parameters was possible in order to avoid a release of radioactive material to the environs such that the limits to 10CFR20 were exceeded, avoid nuclear system stress in excess of that allowed by applicable industry codes, and avoid any operating conditions not considered by station safety analyses. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions and information provided by the Reactor Vessel Surface Temperature Instrumentation are no longer required; the information provided to describe the Reactor Vessel Surface Temperature Instrumentation is no longer applicable. Reactor Vessel Instrumentation information is obsolete.	
		7.8.5.2	Reactor Vessel Water Level	Delete Section 7.8.5.2, Reactor Vessel Instrumentation, Reactor Vessel Water Level. Reactor vessel instrumentation monitored and transmitted information concerning key reactor vessel operating parameters during planned operations to ensure that sufficient control of these parameters was possible in order to avoid a release of radioactive material to the environs such that the limits to 10CFR20 were exceeded, avoid nuclear system stress in excess of that allowed by applicable industry codes, and avoid any operating conditions not considered by station safety analyses. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions and information provided by the Reactor Vessel Water Level Instrumentation are no longer required; the information provided to describe the Reactor Vessel Water Level Instrumentation is no longer applicable. Reactor Vessel Instrumentation information is obsolete.	
		7.8.5.3	Reactor Vessel Coolant Flow Rates and Differential Temperatures	Delete Section 7.8.5.3, Reactor Vessel Instrumentation, Reactor Vessel Coolant Flow Rates and Differential Temperatures. Reactor vessel instrumentation monitored and transmitted information concerning key reactor vessel operating parameters during planned operations to ensure that sufficient control of these parameters was possible in order to avoid a release of radioactive material to the environs such that the limits to 10CFR20 were exceeded, avoid nuclear system stress in excess of that allowed by applicable industry codes, and avoid any operating conditions not considered by station safety analyses. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions and information provided by the Reactor VesselCoolant Flow Rates and Differential Temperatures Instrumentation are no longer required; the information provided to describe the Reactor Vessel Coolant Flow Rates and Differential Temperatures instrumentation is no l onger applicable. Reactor Vessel Instrumentation information is obsolete.	
		7.8.5.4	Reactor Vessel Internal Pressure	Delete Section 7.8.5.4, Reactor Vessel Instrumentation, Reactor Vessel Internal Pressure. Reactor vessel instrumentation monitored and transmitted information concerning key reactor vessel operating parameters during planned operations to ensure that sufficient control of these parameters was possible in order to avoid a release of radioactive material to the environs such that the limits to 10CFR20 were exceeded, avoid nuclear system stress in excess of that allowed by applicable industry codes, and avoid any operating conditions not considered by station safety analyses. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions and information provided by the Reactor Vessel Internal Pressure are no longer required; the information provided to describe the Reactor Vessel Internal Pressure instrumentation is no longer applicable. Reactor Vessel Instrumentation information is obsolete.	

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		7.8.5.5	Reactor Vessel Top Head Flange Leak Detection	Delete Section 7.8.5.5, Reactor Vessel Instrumentation, Reactor Vessel Top Head Flange Leak Detection. Reactor vessel instrumentation monitored and transmitted information concerning key reactor vessel operating parameters during planned operations to ensure that sufficient control of these parameters was possible in order to avoid a release of radioactive material to the environs such that the limits to 10CFR20 were exceeded, avoid nuclear system stress in excess of that allowed by applicable industry codes, and avoid any operating conditions not considered by station safety analyses. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions and information provided by the Reactor Vessel Top Head Flange Leak Detection instrumentation are no longer required; the information provided to describe the Reactor Vessel Top Head Flange Leak Detection instrumentation is no longer applicable. Reactor Vessel Instrumentation information is obsolete.	
		7.8.5.6	Reference Leg Temperature Monitoring	Delete Section 7.8.5.6, Reactor Vessel Instrumentation,Reference Leg Temperature Monitoring. Reactor vessel instrumentation monitored and transmitted information concerning key reactor vessel operating parameters during planned operations to ensure that sufficient control of these parameters was possible in order to avoid a release of radioactive material to the environs such that the limits to 10CFR20 were exceeded, avoid nuclear system stress in excess of that allowed by applicable industry codes, and avoid any operating conditions not considered by station safety analyses. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions and information provided by the Reference Leg Temperature Monitoring instrumentation are no longer required; the information provided to describe the Reference Leg Temperature Monitoring instrumentation is no longer applicable. Reactor Vessel Instrumentation information is obsolete.	
		7.8.5.7	Reference Leg Back Fill System	Delete Section 7.8.5.7, Reactor Vessel Instrumentation, Reference Leg Back Fill System. The Back Fill System ensured that the reactor water level reference leg fluid for selected condensing chambers did not become saturated with noncondensible gases. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions provided by the Reference Leg Back Fill System are no longer required; the information provided to describe the Reference Leg Backfill System is no longer applicable. Reactor Vessel Instrumentation information is obsolete.	
		7.8.5.8	Safety Valve Leak Detection	Delete Section 7.8.5.8, Reactor Vessel Instrumentation, Safety Valve Leak Detection. Reactor vessel instrumentation monitored and transmitted information concerning key reactor vessel operating parameters during planned operations to ensure that sufficient control of these parameters was possible in order to avoid a release of radioactive material to the environs such that the limits to 10CFR20 were exceeded, avoid nuclear system stress in excess of that allowed by applicable industry codes, and avoid any operating conditions not considered by station safety analyses. An acoustic accelerometer installed on each of the reactor vessel safety valves provided indirect indication of valve position. Leakage was also detected by temperature elements mounted on the relief valve tailpipes. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.Therefore, the functions and information provided by the Safety Valve Leak Detection instrumentation are no longer required; the information provided to describe the Safety Valve Leak Detection instrumentation is no longer applicable. Reactor Vessel Instrumentation information is obsolete.	
	7.8.6	Safety Evaluation		Delete Section 7.8.6, Reactor Vessel Instrumentation, Safety Evaluation. Reactor vessel instrumentation monitored and transmitted information concerning key reactor vessel operating parameters during planned operations to ensure that sufficient control of these parameters was possible in order to avoid a release of radioactive material to the environs such that the limits to 10CFR20 were exceeded, avoid nuclear system stress in excess of that allowed by applicable industry codes, and avoid any operating conditions not considered by station safety analyses. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by Reactor Vessel Instrumentation are no longer required; a safety evaluation of the Reactor Vessel Instrumentation is not required since the functions and information provided by the Reactor Vessel Instrumentation are not required. Information regarding Reactor Vessel Instrumentation is obsolete.	
	7.8.7	Inspection and Testing		Delete Section 7.8.7, Reactor Vessel Instrumentation, Inspection and Testing. Reactor vessel instrumentation monitored and transmitted information concerning key reactor vessel operating parameters during planned operations to ensure that sufficient control of these parameters was possible in order to avoid a release of radioactive material to the environs such that the limits to 10CFR20 were exceeded, avoid nuclear system stress in excess of that allowed by applicable industry codes, and avoid any operating conditions not considered by station safety analyses. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions provided by the Reactor Manual Control System are no longer required. Therefore, inspection and testing of the Reactor Manual Control System is no longer required. Reactor Manual Control System information is obsolete.	
	Table 7.8.1	Reactor Vessel Instrumentation		Delete Table 7.8.1, Reactor Vessel Instrumentation. Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions and information provided by Reactor Vessel Instrumentation are no longer required. Therefore, a table listing the Reactor Vessel Instrumentation is no longer required. Reactor Vessel Instrumentation information is obsolete.	

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	Figure 7.8-1	G-191167 Nuclear System Pressure Relief System, Piping and Instrumentation Diagram	Delete Figure 7.8-1, Nuclear System Pressure Relief System, Piping and Instrumentation Diagram. Reactor vessel instrumentation monitored and transmitted information concerning key reactor vessel operating parameters during planned operations to ensure that sufficient control of these parameters was possible in order to avoid a release of radioactive material to the environs such that the limits to 10CFR20 were exceeded, avoid nuclear system stress in excess of that allowed by applicable industry codes, and avoid any operating conditions not considered by station safety analyses. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the Reactor Vessel Instrumentation are no longer required. The Reactor Vessel Indication information provided on Figure 7.8-1 is no longer applicable. Reactor Vessel Instrumentation information is obsolete.	
	Figure 7.8-2a	G-191267, Sh1 Reactor Vessel Instrumentation Piping and Instrumentation Diagram	Delete Figure 7.8-2a, Reactor Vessel Instrumentation Piping and Instrumentation Diagram. Reactor vessel instrumentation monitored and transmitted information concerning key reactor vessel operating parameters during planned operations to ensure that sufficient control of these parameters was possible in order to avoid a release of radioactive material to the environs such that the limits to 10CFR20 were exceeded, avoid nuclear system stress in excess of that allowed by applicable industry codes, and avoid any operating conditions not considered by station safety analyses. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the Reactor Vessel Instrumentation are no longer required. The Reactor Vessel Indication information provided on Figure 7.8-2a is no longer applicable. Reactor Vessel Instrumentation information is obsolete.	
	Figure 7.8-2b	G-191267, Sh2 Reactor Vessel Instrumentation Piping and Instrumentation Diagram	Delete Figure 7.8-2b, Reactor Vessel Instrumentation Piping and Instrumentation Diagram. Reactor vessel instrumentation monitored and transmitted information concerning key reactor vessel operating parameters during planned operations to ensure that sufficient control of these parameters was possible in order to avoid a release of radioactive material to the environs such that the limits to 10CFR20 were exceeded, avoid nuclear system stress in excess of that allowed by applicable industry codes, and avoid any operating conditions not considered by station safety analyses. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the Reactor Vessel Instrumentation are no longer required. The Reactor Vessel Indication information provided on Figure 7.8-2b is no longer applicable. Reactor Vessel Instrumentation information is obsolete.	
	Figure 7.8-3	Reactor Vessel Thermocouple Locations	Delete Figure 7.8-3, Reactor Vessel Thermocouple Locations. Reactor vessel instrumentation monitored and transmitted information concerning key reactor vessel operating parameters during planned operations to ensure that sufficient control of these parameters was possible in order to avoid a release of radioactive material to the environs such that the limits to 10CFR20 were exceeded, avoid nuclear system stress in excess of that allowed by applicable industry codes, and avoid any operating conditions not considered by station safety analyses. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the Reactor Vessel Instrumentation are no longer required. The Reactor Vessel thermocouple location information provided on Figure 7.8-3 is no longer applicable. Reactor Vessel Instrumentation information is obsolete.	
7.9	RECIRCULATION FLOW CONTROL SYSTEM		The Recirculation Flow Control System controlled reactor power level over a limited range by controlling the flow rate of the reactor recirculating water.	
	7.9.1	Power Generation Objective	Delete Section 7.9.1, Recirculation Flow Control System, Power Generation Objective. The Recirculation Flow Control System functioned to control reactor power level over a limited range by controlling the flow rate of the reactor recirculating water. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the Recirculation Flow Control System are no longer required; the Recirculation Flow Control System Power Generation Objective is no longer applicable. Recirculation Flow Control System information is obsolete.	
	7.9.2	Power Generation Design Bases	Delete Section 7.9.2, Recirculation Flow Control System, Power Generation Design Bases. The Recirculation Flow Control System functioned to control reactor power level over a limited range by controlling the flow rate of the reactor recirculating water. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the Recirculation Flow Control System are no longer required; the Recirculation Flow Control System Power Generation Design Bases are no longer applicable. Recirculation Flow Control System information is obsolete.	

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	7.9.3	Safety Design Bases		Delete Section 7.9.3, Recirculation Flow Control System, Safety Design Bases. The Recirculation Flow Control System functioned to control reactor power level over a limited range by controlling the flow rate of the reactor recirculating water. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the Recirculation Flow Control System are no longer required; the Recirculation Flow Control System Safety Design Bases are no longer applicable. Recirculation Flow Control System information is obsolete.	
	7.9.4	Description			
		7.9.4.1	General	Delete Section 7.9.4.1, Recirculation Flow Control System, General Description. The Recirculation Flow Control System functioned to control reactor power level over a limited range by controlling the flow rate of the reactor recirculating water. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the Recirculation Flow Control System are no longer required; the information provided in the General Description Section of the Recirculation Flow Control System is no longer applicable. Recirculation Flow Control System information is obsolete.	
		7.9.4.2	Motor Generator Set	Delete Section 7.9.4.2, Recirculation Flow Control System, Motor Generator Set. The Recirculation Flow Control System functioned to control reactor power level over a limited range by controlling the flow rate of the reactor recirculating water. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the Recirculation Flow Control System are no longer required; the description of the Recirculation Flow Control System Motor Generator Sets provided is no longer applicable. Recirculation Flow Control System information is obsolete.	
		7.9.4.3	Speed Control Components	Delete Section 7.9.4.3, Recirculation Flow Control System, Speed Control Components. The Recirculation Flow Control System functioned to control reactor power level over a limited range by controlling the flow rate of the reactor recirculating water. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the Recirculation Flow Control System are no longer required; the description of the Recirculation Flow Control System Speed Control Components provided is no longer applicable. Recirculation Flow Control System information is obsolete.	
		7.9.4.4	System Operation	Delete Section 7.9.4.4, Recirculation Flow Control System, System Operation. The Recirculation Flow Control System functioned to control reactor power level over a limited range by controlling the flow rate of the reactor recirculating water. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the Recirculation Flow Control System are no longer required; the description of the Recirculation Flow Control System Operation provided is no longer applicable. Recirculation Flow Control System information is obsolete.	
	7.9.5	Safety Evaluation		Delete Section 7.9.5, Recirculation Flow Control System, Safety Evaluation. The Recirculation Flow Control System functioned to control reactor power level over a limited range by controlling the flow rate of the reactor recirculating water. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the Recirculation Flow Control System are no longer required; a safety evaluation of the Recirculation Flow Control SystemSystem is no longer applicable. Recirculation Flow Control System information is obsolete.	
	7.9.6	Inspection and Testing		Delete Section 7.9.6, Recirculation Flow Control System, Inspection and Testing. The Recirculation Flow Control System functioned to control reactor power level over a limited range by controlling the flow rate of the reactor recirculating water. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the Recirculation Flow Control System are no longer required; Inspection and testing of the Recirculation Flow Control System System components is no longer required. Recirculation Flow Control System information is obsolete.	
	7.9.7	References		Delete Section 7.9.7, References. Since the functions provided by the Recirculation Flow Control System are no longer required, references are no longer applicable.	
	Figure 7.9-1	Recirculation Flow Control		Delete Figure 7.9-1, Recirculation Flow Control. The Recirculation Flow Control System functioned to control reactor power level over a limited range by controlling the flow rate of the reactor recirculating water. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the Recirculation Flow Control are no longer required. The information provided on Figure 7.9-1 regarding Recirculation Flow Control is no longer applicable. Recirculation Flow Control System information is obsolete.	
	Figure 7.9-2	Deleted		Previously Deleted	

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	Figure 7.9-3	5920-279 Recirculation Flow Control System, Instrument Electrical Diagram		Delete Figure 7.9-3, Recirculation Flow Control System, Instrument Electrical Diagram. The Recirculation Flow Control System functioned to control reactor power level over a limited range by controlling the flow rate of the reactor recirculating water. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the Recirculation Flow Control are no longer required. The information provided on Figure 7.9-3 regarding Recirculation Flow Control is no longer applicable. Recirculation Flow Control System information is obsolete.	
	Figure 7.9-4	5920-1034 Recirculation Flow Control System, Speed Control System Instrument Electrical Diagram		Delete Figure 7.9-4, Recirculation Flow Control System, Speed Control System Instrument Electrical Diagram. The Recirculation Flow Control System functioned to control reactor power level over a limited range by controlling the flow rate of the reactor recirculating water. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the Recirculation Flow Control are no longer required. The information provided on Figure 7.9-4 regarding Recirculation Flow Control is no longer applicable. Recirculation Flow Control System information is obsolete.	
	Figure 7.9-5	Deleted		Previously Deleted	
7.10	FEEDWATER CONTROL SYSTEM			The Feedwater Control System functioned to maintain a pre established water level in the reactor vessel during planned operation.	
	7.10.1	Power Generation Objective		Delete Section 7.10.1, Feedwater Control System Power Generation Objective. The Feedwater Control System functioned to maintain a pre established water level in the reactor vessel during planned operation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the Feedwater Control System are no longer required; The Feedwater Control System power generation objective is no longer applicable. Feedwater Control System information is obsolete.	
	7.10.2	Power Generation Design Bases		Delete Section 7.10.2, Feedwater Control System Power Generation Design Bases. The Feedwater Control System functioned to maintain a pre established water level in the reactor vessel during planned operation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the Feedwater Control System are no longer required; The Feedwater Control System power generation design bases are no longer applicable. Feedwater Control System information is obsolete.	
	7.10.3	Description			
		7.10.3.1	Reactor Vessel Water Level Measurement	Delete Section 7.10.3.1, Feedwater Control System Description of Reactor Vessel Water Level Measurement. The Feedwater Control System functioned to maintain a pre established water level in the reactor vessel during planned operation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the Feedwater Control System are no longer required; Reactor Vessel Water Level Measurement for Feedwater Control is no longer required. Feedwater Control System information is obsolete.	
		7.10.3.2	Steam Flow Measurement	Delete Section 7.10.3.2, Feedwater Control System Description of Steam Flow Measurement. The Feedwater Control System functioned to maintain a pre established water level in the reactor vessel during planned operation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the Feedwater Control System are no longer required; Steam Flow Measurement for Feedwater Control is no longer required. Feedwater Control System information is obsolete.	
		7.10.3.3	Feedwater Flow Measurement	Delete Section 7.10.3.3, Feedwater Control System Description of Feedwater Flow Measurement. The Feedwater Control System functioned to maintain a pre established water level in the reactor vessel during planned operation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the Feedwater Control System are no longer required; Feedwater Flow Measurement for Feedwater Control is no longer required. Feedwater Control System information is obsolete.	
		7.10.3.4	Feedwater Control Signal	Delete Section 7.10.3.4, Feedwater Control System Description of the Feedwater Flow Control Signal. The Feedwater Control System functioned to maintain a pre established water level in the reactor vessel during planned operation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the Feedwater Control System are no longer required; the description of the Feedwater Control Signal for Feedwater Control is no longer required. Feedwater Control System information is obsolete.	

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		7.10.3.5	Feedwater Valve Control	Delete Section 7.10.3.5, Feedwater Control System Description of Feedwater Valve Control. The Feedwater Control System functioned to maintain a pre established water level in the reactor vessel during planned operation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the Feedwater Control System are no longer required; the description of Feedwater Valve Control is no longer required. Feedwater Control System information is obsolete.	
		7.10.3.6	Reactor Vessel Pressure Measurement	Delete Section 7.10.3.6, Feedwater Control System Description of Reactor Vessel Pressure Measurement. The Feedwater Control System functioned to maintain a pre established water level in the reactor vessel during planned operation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the Feedwater Control System are no longer required; Reactor Vessel Pressure Measurement for Feedwater Control is no longer required. Feedwater Control System information is obsolete.	
	7.10.4	Inspection and Testing		Delete Section 7.10.4, Feedwater Control System Inspection and Testing. The Feedwater Control System functioned to maintain a pre established water level in the reactor vessel during planned operation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the Feedwater Control System are no longer required; Feedwater Control System Inspection and Testing is no longer required. Feedwater Control System information is obsolete.	
	Figure 7.10-1	5920-204 Feedwater Control System, Instrument Electrical Diagram		Delete Figure 7.10-1, Feedwater Control System, Instrument Electrical Diagram. The Feedwater Control System functioned to maintain a pre established water level in the reactor vessel during planned operation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the Feedwater Control System are no longer required. The Feedwater Control System related information provided on Figure 7.10-1 is no longer applicable. Feedwater Control System information is obsolete.	
7.11	PRESSURE REGULATOR AND TURBINE GENERATOR CONTROL			The Pressure Regulator and Turbine Generator Control System functioned to control steam flow and pressure to the turbine and to protect the turbine from overpressure or excessive speed, with controls integrated with those of the nuclear systems for an efficient, reliable power generation unit.	
	7.11.1	Power Generation Objective		Delete section 7.11.1, Pressure Regulator and Turbine Generator Control System Power Generation Objective. The Pressure Regulator and Turbine Generator Control System functioned to control steam flow and pressure to the turbine and to protect the turbine from overpressure or excessive speed, with controls integrated with those of the nuclear systems for an efficient, reliable power generation unit. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the Pressure Regulator and Turbine Generator Control System are no longer required; the Pressure Regulator and Turbine Generator Control System power generation objective is no longer applicable. Pressure Regulator and Turbine Generator Control System information is obsolete.	
	7.11.2	Power Generation Design Bases		Delete section 7.11.2, Pressure Regulator and Turbine Generator Control System Power Generation Design Bases The Pressure Regulator and Turbine Generator Control System functioned to control steam flow and pressure to the turbine and to protect the turbine from overpressure or excessive speed, with controls integrated with those of the nuclear systems for an efficient, reliable power generation unit. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the Pressure Regulator and Turbine Generator Control System are no longer required; the Pressure Regulator and Turbine Generator Control System power generation design bases are no longer applicable. The Pressure Regulator and Turbine Generator Control System information is obsolete.	
	7.11.3	Description			
		7.11.3.1	General	Delete section 7.11.3.1, Pressure Regulator and Turbine Generator Control System General Description. The Pressure Regulator and Turbine Generator Control System functioned to control steam flow and pressure to the turbine and to protect the turbine from overpressure or excessive speed, with controls integrated with those of the nuclear systems for an efficient, reliable power generation unit. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the Pressure Regulator and Turbine Generator Control System are no longer required; the information provided in the Pressure Regulator and Turbine Generator Control System General Description is no longer applicable. Pressure Regulator and Turbine Generator Control System information is obsolete.	
		7.11.3.2	Pressure Regulator	Delete section 7.11.3.2, Pressure Regulator and Turbine Generator Control System Pressure Regulator Description. The Pressure Regulator and Turbine Generator Control System functioned to control steam flow and pressure to the turbine and to protect the turbine from overpressure or excessive speed, with controls integrated with those of the nuclear systems for an efficient, reliable power generation unit. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the Pressure Regulator and Turbine Generator Control System are no longer required; the information provided in the Pressure Regulator Description section is no longer applicable. Pressure Regulator and Turbine Generator Control System information is obsolete.	

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		7.11.3.3	Turbine Generator Controls	Delete section 7.11.3.3, Pressure Regulator and Turbine Generator Control System Pressure Regulator Description. The Pressure Regulator and Turbine Generator Control System functioned to control steam flow and pressure to the turbine and to protect the turbine from overpressure or excessive speed, with controls integrated with those of the nuclear systems for an efficient, reliable power generation unit. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the Pressure Regulator and Turbine Generator Control System are no longer required; the information provided in the Pressure Regulator Description section is no longer applicable. Pressure Regulator and Turbine Generator Control System information is obsolete.			
		7.11.3.4	Turbine Bypass System Control	Delete section 7.11.3.4, Pressure Regulator and Turbine Bypass System Control. The Pressure Regulator and Turbine Generator Control System functioned to control steam flow and pressure to the turbine and to protect the turbine from overpressure or excessive speed, with controls integrated with those of the nuclear systems for an efficient, reliable power generation unit. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the Pressure Regulator and Turbine Generator Control System are no longer required; the information provided in the Turbine Bypass System Control Description section is no longer applicable. Pressure Regulator and Turbine Generator Control System information is obsolete.			
	7.11.4	Inspection and Testing		Delete section 7.11.4, Pressure Regulator and Turbine Generator Control System Inspection and Testing. The Pressure Regulator and Turbine Generator Control System functioned to control steam flow and pressure to the turbine and to protect the turbine from overpressure or excessive speed, with controls integrated with those of the nuclear systems for an efficient, reliable power generation unit. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the Pressure Regulator and Turbine Generator Control System are no longer required; testing of the Pressure Regulator and Turbine Generator Control System components is no longer required. Pressure Regulator and Turbine Generator Control System information is obsolete.			
7.12	PROCESS RADIATION MONITORING SYSTEM					4.7.1	PROCESS RADIATION MONITORING SYSTEM
	7.12.1	Main Steam Line Radiation Monitoring System		The Main Steam Line Monitoring System monitored for the gross release of fission products from the fuel and, upon indication of such failure, initiated appropriate action to contain the released fission products.			
		7.12.1.1	Safety Objective	Delete section 7.12.1.1, Main Steam Line Radiation Monitoring System, Safety Objective. The Main Steam Line Monitoring System monitored for the gross release of fission products from the fuel and, upon indication of such failure, initiated appropriate action to contain the released fission products. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the Main Steam Line Radiation Monitoring System are no longer required; the Main Steam Line Radiation Monitoring System Safety Objective is no longer applicable. Main Steam Line Radiation Monitoring System information is obsolete.			
		7.12.1.2	Safety Design Basis	Delete section 7.12.1.2, Main Steam Line Radiation Monitoring System, Safety Design Basis. The Main Steam Line Monitoring System monitored for the gross release of fission products from the fuel and, upon indication of such failure, initiated appropriate action to contain the released fission products. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the Main Steam Line Radiation Monitoring System are no longer required; the Main Steam Line Radiation Monitoring System Safety Design Basis is no longer applicable. Main Steam Line Radiation Monitoring System information is obsolete.			
		7.12.1.3	Description	Delete section 7.12.1.3, Main Steam Line Radiation Monitoring System, Description. The Main Steam Line Monitoring System monitored for the gross release of fission products from the fuel and, upon indication of such failure, initiated appropriate action to contain the released fission products. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the Main Steam Line Radiation Monitoring System functions and information outlined in the Description section are no longer applicable. Main Steam Line Radiation Monitoring System information is obsolete.			
		7.12.1.4	Safety Evaluation	Delete section 7.12.1.4, Main Steam Line Radiation Monitoring System, Safety Design Basis. The Main Steam Line Monitoring System monitored for the gross release of fission products from the fuel and, upon indication of such failure, initiated appropriate action to contain the released fission products. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the Main Steam Line Radiation Monitoring System are no longer required; the Main Steam Line Radiation Monitoring System Safety Evaluation is no longer applicable. Main Steam Line Radiation Monitoring System information is obsolete.			

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UFSAR Section				FSAR Conversion to DSAR Change Summary	DSAR Section				
		7.12.1.5	Inspection and Testing	Delete section 7.12.1.5, Main Steam Line Radiation Monitoring System, Safety Design Basis. The Main Steam Line Monitoring System monitored for the gross release of fission products from the fuel and, upon indication of such failure, initiated appropriate action to contain the released fission products. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the Main Steam Line Radiation Monitoring System are no longer required; Inspection and testing of the the Main Steam Line Radiation Monitoring System components is no longer required. Main Steam Line Radiation Monitoring System information is obsolete.					
	7.12.2	Off Gas Radiation Monitoring System		The Off Gas Radiation Monitoring System indicated when limits for the release of radioactive material to the environs were approached and initiated appropriate control of offgas so that the limits were not exceeded.					
		7.12.2.1	Power Generation Objective	Delete Section 7.12.2.1, Off Gas Radiation Monitoring System Power Generation Objective. The Off Gas Radiation Monitoring System indicated when limits for the release of radioactive material to the environs were approached and initiated appropriate control of offgas so that the limits were not exceeded. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the Off Gas Radiation Monitoring System are no longer required; the Off Gas Radiation Monitoring System Power Generation Objective is no longer applicable. Off Gas Radiation Monitoring System information is obsolete.					
		7.12.2.2	Power Generation Design Basis	Delete Section 7.12.2.2, Off Gas Radiation Monitoring System Power Generation Design Basis. The Off Gas Radiation Monitoring System indicated when limits for the release of radioactive material to the environs were approached and initiated appropriate control of offgas so that the limits were not exceeded. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the Off Gas Radiation Monitoring System are no longer required; the Off Gas Radiation Monitoring System Power Generation Design Basis is no longer applicable. Off Gas Radiation Monitoring System information is obsolete.					
		7.12.2.3	Description	Delete Section 7.12.2.3, Off Gas Radiation Monitoring System Power Generation Description. The Off Gas Radiation Monitoring System indicated when limits for the release of radioactive material to the environs were approached and initiated appropriate control of offgas so that the limits were not exceeded. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the Off Gas Radiation Monitoring System are no longer required; Information regarding the Off Gas Radiation Monitoring System provided in the Description section is no longer applicable. Off Gas Radiation Monitoring System information is obsolete.					
		7.12.2.4	Inspection and Testing	Delete Section 7.12.2.2, Off Gas Radiation Monitoring System Inspection and Testing. The Off Gas Radiation Monitoring System indicated when limits for the release of radioactive material to the environs were approached and initiated appropriate control of offgas so that the limits were not exceeded. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the Off Gas Radiation Monitoring System are no longer required; Inspection and testing of the Off Gas Radiation Monitoring System is not required. Off Gas Radiation Monitoring System information is obsolete.					
	7.12.3	Plant Stack Radiation Monitor System					4.7.1.1	Plant Stack Radiation Monitor System	
		7.12.3.1	Power Generation Objective	Modify Section 7.12.3.1, Plant Stack Radiation Monitor System, Power Generation Objective to remove reference to operating and accident conditions. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.				4.7.1.1.1	Objective
		7.12.3.2	Power Generation Design Basis	Modify Section 7.12.3.1, Plant Stack Radiation Monitor System, Power Generation Design Basis to remove reference to operating and accident conditions. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, release rate values above operational release rate limits down to the release rates normally encountered during high power operation can no longer occur.				4.7.1.1.2	Design Basis
		7.12.3.3	Description	No changes				4.7.1.1.3	Description
		7.12.3.4	Inspection and Testing	Modify Section 7.12.3.4, Plant Stack Radiation Monitor System, Inspection and Testing to remove reference to testing required by Technical Specifications. All Plant Stack Radiation Monitoring System testing requirements have been removed from the Technical Specifications.				4.7.1.1.4	Inspection and Testing
	7.12.4	Process Liquid Radiation Monitoring System					4.7.1.2	Process Liquid Radiation Monitoring System	
		7.12.4.1	Power Generation Objective	Modify Section 7.12.4.1, Process Liquid Radiation Monitor System, Power Generation Objective to remove reference to operating and accident conditions.				4.7.1.2.1	Objective
		7.12.4.2	Power Generation Design Basis	Modify Section 7.12.4.2, Process Liquid Radiation Monitor System, Power Generation Design Basis to remove reference to operating and accident conditions.				4.7.1.2.2	Design Basis

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UFSAR Section				FSAR Conversion to DSAR Change Summary	DSAR Section				
		7.12.4.3	Description	Modify Section 7.12.4.3, Process Liquid Radiation Monitor System, Description to remove reference to operating and accident conditions, the Reactor Building Closed Cooling Water System and the RHR Service Water System. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by RBCCW and RHR Service Water are no longer required. Consequently, process radiation monitoring of those systems is no longer required and has been removed. Information regarding the RBCCW and RHR process radiation monitors is obsolete.				4.7.1.2.3	Description
		7.12.4.4	Inspection and Testing	No Changes				4.7.1.2.4	Inspection and Testing
	7.12.5	Reactor Building Ventilation Radiation Monitoring System					4.7.1.3	Reactor Building Ventilation Radiation Monitoring System	
		7.12.5.1	Safety Objective	Modified section 7.12.5.1 to delete "Safety" from the section title, and to remove reference to initiation of isolation and trip signals based on RBV high radiation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. License Amendment 262, approved by the NRC in February 2015, eliminated operability requirements for secondary containment, the Standby Gas Treatment (SGT) system, and the reactor building ventilation isolation and the SGT system initiation instrumentation when handling irradiated fuel or a fuel cask. The changes require the subject systems to be operable only during movement of "recently" irradiated fuel assemblies in secondary containment and during operations with the potential to drain the reactor vessel. The period of sufficient radioactive decay was determined to be 13 days. Since VYNPS certified permanent defueling in January 2015, all fuel in the fuel pool is considered sufficiently decayed. . Consequently, automatic protective actions based on Reactor Building Ventilation Radiation Levels are no longer required. Reactor Building Vent Rad monitoring will provide only alarm functions.				4.7.1.3.1	Objective
		7.12.5.2	Safety Design Basis	Modified section 7.12.5.2 to delete "Safety" from the section title, and to remove reference to initiation of isolation and trip signals based on RBV high radiation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. License Amendment 262, approved by the NRC in February 2015, eliminated operability requirements for secondary containment, the Standby Gas Treatment (SGT) system, and the reactor building ventilation isolation and the SGT system initiation instrumentation when handling irradiated fuel or a fuel cask. The changes require the subject systems to be operable only during movement of "recently" irradiated fuel assemblies in secondary containment and during operations with the potential to drain the reactor vessel. The period of sufficient radioactive decay was determined to be 13 days. Since VYNPS certified permanent defueling in January 2015, all fuel in the fuel pool is considered sufficiently decayed. . Consequently, automatic protective actions based on Reactor Building Ventilation Radiation Levels are no longer required. Reactor Building Vent Rad monitoring will provide only alarm functions.				4.7.1.3.2	Design Basis
		7.12.5.3	Description	Modified section 7.12.5.3 to remove reference to initiation of isolation and trip signals based on RBV high radiation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. License Amendment 262, approved by the NRC in February 2015, eliminated operability requirements for secondary containment, the Standby Gas Treatment (SGT) system, and the reactor building ventilation isolation and the SGT system initiation instrumentation when handling irradiated fuel or a fuel cask. The changes require the subject systems to be operable only during movement of "recently" irradiated fuel assemblies in secondary containment and during operations with the potential to drain the reactor vessel. The period of sufficient radioactive decay was determined to be 13 days. Since VYNPS certified permanent defueling in January 2015, all fuel in the fuel pool is considered sufficiently decayed. Consequently, automatic protective actions based on Reactor Building Ventilation Radiation Levels are no longer required. Reactor Building Vent Rad monitoring will provide only alarm functions.				4.7.1.3.3	Description
		7.12.5.4	Safety Evaluation	Modified section 7.12.5.4 to delete "Safety" from the section title, and to remove reference to initiation of isolation and trip signals based on RBV high radiation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Analysis has demonstrated that secondary containment is no longer required to mitigate the consequences of the remaining design basis accident, a fuel handling accident. Consequently, automatic protective actions based on Reactor Building Ventilation Radiation Levels are no longer required. Reactor Building Vent Rad monitoring will provide only alarm functions.				4.7.1.3.4	Evaluation
		7.12.5.5	Inspection and Testing	Modified Section 7.12.5.5 to delete reference to Technical Specifications since VY will never be in a mode in which the surveillance requirements will apply.				4.7.1.3.5	Inspection and Testing
	7.12.6	Containment High Range Radiation Monitoring System		The Containment High Range Radiation Monitoring System monitored the containment atmosphere for the gross release of fission products during a LOCA/HELB.					
		7.12.6.1	Safety Objective	Delete Section 7.12.6.1, Containment High Range Radiation Monitoring System, Safety Objective. The Containment High Range Radiation Monitoring System monitored the containment atmosphere for the gross release of fission products during a LOCA/HELB. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the Containment High Range Radiation Monitoring System are no longer required; the Containment High Range Radiation Monitoring System Safety Objective is no longer applicable. Containment High Range Radiation Monitoring System information is obsolete.					

VY UFSAR					VY DSAR				
UFSAR Section				FSAR Conversion to DSAR Change Summary	DSAR Section				
		7.12.6.2	Safety Design Basis	Delete Section 7.12.6.2, Containment High Range Radiation Monitoring System, Safety Design Basis. The Containment High Range Radiation Monitoring System monitored the containment atmosphere for the gross release of fission products during a LOCA/HELB. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the Containment High Range Radiation Monitoring System are no longer required; the Containment High Range Radiation Monitoring System Safety Design Basis is no longer applicable. Containment High Range Radiation Monitoring System information is obsolete.					
		7.12.6.3	Description	Delete Section 7.12.6.3, Containment High Range Radiation Monitoring System, Description. The Containment High Range Radiation Monitoring System monitored the containment atmosphere for the gross release of fission products during a LOCA/HELB. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the Containment High Range Radiation Monitoring System are no longer required; the information regarding the Containment High Range Radiation Monitoring System provided in the Description section is no longer applicable. Containment High Range Radiation Monitoring System information is obsolete.					
		7.12.6.4	Safety Evaluation	Delete Section 7.12.6.4, Containment High Range Radiation Monitoring System, Safety Design Basis. The Containment High Range Radiation Monitoring System monitored the containment atmosphere for the gross release of fission products during a LOCA/HELB. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the Containment High Range Radiation Monitoring System are no longer required; a Containment High Range Radiation Monitoring System Safety evaluation is no longer required. Containment High Range Radiation Monitoring System information is obsolete.					
		7.12.6.5	Inspection and Testing	Delete Section 7.12.6.5, Containment High Range Radiation Monitoring System, Inspection and Testing. The Containment High Range Radiation Monitoring System monitored the containment atmosphere for the gross release of fission products during a LOCA/HELB. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the Containment High Range Radiation Monitoring System are no longer required; Inspection and Testing of the Containment High Range Radiation Monitoring System is no longer required. Containment High Range Radiation Monitoring System information is obsolete.					
	Table 7.12.1	Process Radiation Monitoring System Characteristics		Modified to delete Main Steam Line, Main Condenser AE Offgas, Augmented Offgas, and Containment High Range Rad monitors. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the listed radiation monitors are no longer required. Modify Table 7.12.1 to delete sensitivity column. Per technical review, information provided in this column is not meaningful and is not required.		Table 4.7.1.1	Process Radiation Monitoring System Characteristics		
	Table 7.12.2	Process Radiation Monitoring Environmentatal and Power Supply Design Conditions		No changes		Table 4.7.1.2	Process Radiation Monitoring Environmental and Power Supply Design Conditions		
	Table 7.12.3	Plant Stack Radiation Monitor System Characteristics		No changes		Table 4.7.1.3	Plant Stack Radiation Monitor System Characteristics		
	Figure 7.12-1	5920-00526 Process Radiation Monitoring System Instrumentation Diagram		Modify to delete Main Steam Line, Main Condenser AE Offgas, Augmented Offgas, and Containment High Range Rad monitors. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the listed radiation monitors are no longer required.		Figure 4.7.1-1	5920-00526 Process Radiation Monitoring System Instrumentation Diagram		
	Figure 7.12-2	5920-3994 Plant Stack Radiation Monitoring System Diagram		No changes		Figure 4.7.1-2	5920-03994 Plant Stack Radiation Monitoring System Diagram		
7.13	AREA RADIATION MONITORING SYSTEM					4.7.2	AREA RADIATION MONITORING SYSTEM		
	7.13.1	Power Generation Objective		Modified Section 7.13.1 to delete the term "Power Generation" from the section title. Since the term "Power Generation" is based in the classification process described in section 1.4.1 which is no longer used, the term is no longer required and is not applicable to a permanently defueled station.			4.7.2.1	Objective	
	7.13.2	Power Generation Design Bases		Modified Section 7.13.2 to delete the term "Power Generation" from the section title. Since the term "Power Generation" is based in the classification process described in section 1.4.1 which is no longer used, the term is no longer required and is not applicable to a permanently defueled station. Section wording modified to remove excess specificity.			4.7.2.2	Design Bases	
	7.13.3	Description					4.7.2.3	Description	
		7.13.3.1	Monitors	Modify Section 7.13.3.1, Monitors, Paragraph 2, Area Airborne Radiation Monitoring Systems to delete all reference to the Reactor Containment Airborne Radiation Monitoring System. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Primary Containment System are no longer required. Therefore the functions and information provided by the Reactor Containment Airborne Radiation Monitoring System are no longer applicable and are obsolete.				4.7.2.3.1	Monitors
		7.13.3.2	Locations	No Changes				4.7.2.3.2	Locations

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UFSAR Section		FSAR Conversion to DSAR Change Summary	DSAR Section	
7.13.4	Inspection and Testing	Modify Section 7.13.4, Inspection and Testing, to delete all reference to the Reactor Containment Airborne Radiation Monitoring System. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Primary Containment System are no longer required. Therefore the functions and information provided by the Reactor Containment Airborne Radiation Monitoring System are no longer applicable and are obsolete.	4.7.2.4	Inspection and Testing
Table 7.13.1	Area Radiation Monitoring System, Environmental and Power Supply Design Conditions	Modify Table 7.13.1, to delete reference to the Vermont Yankee Environmental Qualification Program. Following certification of permanent defueling the protection provided by the VY EQ program is no longer required and is obsolete.	Table 4.7.2.1	Area Radiation Monitoring System, Environmental and Power Supply Design Conditions
Table 7.13.2	Locations of Area Radiation Monitors	No Changes	Table 4.7.2.2	Locations of Area Radiation Monitors
Table 7.13.3	Reactor Containment and Reactor Building Area Airborne Radiation Monitoring System	Modify Table 7.13.3 to delete all reference to the Reactor Containment Airborne Radiation Monitoring System. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Primary Containment System are no longer required. Therefore the functions and information provided by the Reactor Containment Airborne Radiation Monitoring System are no longer applicable and are obsolete.	Table 4.7.2.3	Reactor Building Area Airborne Radiation Monitoring System
Table 7.13.4	Technical Support Center Area Airborne Radiation Monitoring System	No Changes	Table 4.7.2.4	Technical Support Center Area Airborne Radiation Monitoring System
Figure 7.13-1	5920-430 Sheet 1 Area Radiation Monitoring System, Functional Block Diagram	No Changes	Figure 4.7.2-1	5920-430 Sheet 1 Area Radiation Monitoring System, Functional Block Diagram
Figure 7.13-2	Reactor Building Area Airborne Radiation Monitoring System	No Changes	Figure 4.7.2-2	Reactor Building Area Airborne Radiation Monitoring System
Figure 7.13-3	Reactor Containment Area Airborne Radiation Monitoring System	Delete Figure 7.13-3. All reference to the Reacto Containment Airborne Radiation Monitoring System has been deleted. System functions are no longer required.		
7.14	HEALTH PHYSICS INSTRUMENTATION	Editorial and section title change only	4.3	HEALTH PHYSICS INSTRUMENTATION
7.14.1	Power Generation Objective	Editorial and section title change only	4.3.1	Objective
7.14.2	Description	Editorial and section title change only	4.3.2	Description
7.14.2.1	Portable Instrumentation	Editorial and section title change only	4.3.2	Portable Instrumentation
7.14.2.2	Fixed and Laboratory Instrumentation	Editorial and section title change only	4.3.2	Fixed and Laboratory Instrumentation
7.15	PROCESS COMPUTER SYSTEM		3.3.11	PROCESS COMPUTER SYSTEM
7.15.1	Power Generation Objective	Modify Section 7.15.1, Process Computer System, Power Generation Objective to remove reference to core thermal hydraulic condition determination. Delete reference to ERDS. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, core thermal-hydraulic conditions no longer exist. Additionally, ERDS is no longer required, refer to FCR 27/017.	3.3.11.1	Objectives
7.15.2	Power Generation Design Bases	Modify Section 7.15.2, Process Computer System, Power Generation Design Bases to remove reference to core power and thermal hydraulic condition determination, EOP entry condition determination, margin and display plot determination, critical plant parameters and ERDS. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, core power, thermal-hydraulic conditions, EOP entry conditions, and critical plant parameters no longer exist. Additionally, ERDS is no longer required, refer to FCR 27/017.	3.3.11.2	Design Bases
7.15.3	Description		3.3.11.3	Description
7.15.3.1	Computer System Components	No changes	3.3.11.3.1	Computer System Components

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UFSAR Section				FSAR Conversion to DSAR Change Summary	DSAR Section		
		7.15.3.2	Reactor Core Performance Function	Delete section 7.15.3.2, Reactor Core Performance Function. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, reactor core performance can no longer be determined and the performance detemination function of the Process Computer is no longer required.			
		7.15.3.3	Rod Worth Minimizer Function	Delete section 7.15.3.3, Rod Worth Minimizer Function. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the Rod Worth Minimizer are no longer required. Rod Worth Minimizer information is obsolete.			
		7.15.3.4	Monitor Alarm and Logging Functions	Modify Section 7.15.3.4, Monitor Alarm and Logging Functions to delete reference to Post-Mortem Logging Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the function and information provided by the post-mortem logging feature are no longer required.		3.3.11.3.2	Monitor Alarm and Logging Functions
	7.15.4	Inspection and Testing		No Changes	3.3.11.4	Inspection and Testing	
	7.15.5	Cyber Security		No Changes	3.3.11.5	Cyber Security	
				Added to describe Process Computer Data feeds to the Plant Data Server.	3.3.11.6	Process Computer Data Feed to the Plant Data Server (PDS)	
	Table 7.15.1	Instrumentation Input Summary		Delete Table 7.15.1, PPC Instrumentation Input Summary. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the PPC inputs listed in the table are no longer valid.			
	Table 7.15.2	INSTRUMENTATION OUTPUT SUMMARY SIGNAL OUTPUT DESCRIPTION		Delete Table 7.15.2, INSTRUMENTATION OUTPUT SUMMARY SIGNAL OUTPUT DESCRIPTION . Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the TIP, RPIS and RWM outputs listed in table 7.15.2 are no longer valid. The remaining four outputs are not significant and do not warrant listing on a table.			
	Figure 7.15-1	5920-1640 Rod Worth Minimizer, Functional Control Diagram		Delete Figure 7.15-1, Rod Worth Minimizer Functional Control Diagram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the Rod Worth Minimizer are no longer required. The information provided on Figure 7.15-1 is not applicable. Rod Worth Minimizer information is obsolete.			
7.16	Nuclear System Stability Analysis			The Nuclear System Stability Analysis demonstrated that stability-related neutron flux oscillations will be prevented or detected and suppressed prior to exceeding specified acceptable fuel design limits.			
	7.16.1	Safety Objective		Delete Section 7.16.1, Nuclear System Stability Analysis, Safety Objective. The Nuclear System Stability Analysis demonstrated that stability-related neutron flux oscillations will be prevented or detected and suppressed prior to exceeding specified acceptable fuel design limits. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, core power oscillations cannot occur; the functions and information provided by the Nuclear System Stability Analysis are no longer required. Therefore, the Nuclear System Stability Analysis Safety Objective is no longer applicable. Nuclear System Stability Analysis information is obsolete.			
	7.16.2	Safety Design Basis		Delete Section 7.16.2, Nuclear System Stability Analysis, Safety Design Basis The Nuclear System Stability Analysis demonstrated that stability-related neutron flux oscillations will be prevented or detected and suppressed prior to exceeding specified acceptable fuel design limits. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, core power oscillations cannot occur; the functions and information provided by the Nuclear System Stability Analysis are no longer required. Therefore, the Nuclear System Stability Analysis Safety Design Basis is no longer applicable. Nuclear System Stability Analysis information is obsolete.			
	7.16.3	Power Generation Design Basis		Delete Section 7.16.3, Nuclear System Stability Analysis, Power Generation Design Basis The Nuclear System Stability Analysis demonstrated that stability-related neutron flux oscillations will be prevented or detected and suppressed prior to exceeding specified acceptable fuel design limits. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, core power oscillations cannot occur; the functions and information provided by the Nuclear System Stability Analysis are no longer required. Therefore, the Nuclear System Stability Analysis Power Generation Design Basis is no longer applicable. Nuclear System Stability Analysis information is obsolete.			

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	7.16.4	Conformance With Safety and Power Generation Design Basis	Delete Section 7.16.4, Nuclear System Stability Analysis, Conformance With Safety and Power Generation Design Basis. The Nuclear System Stability Analysis demonstrated that stability-related neutron flux oscillations will be prevented or detected and suppressed prior to exceeding specified acceptable fuel design limits. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, core power oscillations cannot occur; demonstration of Conformance With the Safety and Power Generation Design Basis of the Nuclear System Stability Analysis is no longer required. Nuclear System Stability Analysis information is obsolete.	
	7.16.5	Description and Performance Analysis	Delete Section 7.16.5, Nuclear System Stability Analysis, Description and Performance Analysis. The Nuclear System Stability Analysis demonstrated that stability-related neutron flux oscillations will be prevented or detected and suppressed prior to exceeding specified acceptable fuel design limits. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, core power oscillations cannot occur; a description of core oscillations and a performance analysis demonstrating adequacy of prevention and detection of core oscillations at Vermont Yankee is no longer required. Nuclear System Stability Analysis information is obsolete.	
	7.16.6	Conclusions	Delete Section 7.16.6, Nuclear System Stability Analysis, Conclusions. The Nuclear System Stability Analysis demonstrated that stability-related neutron flux oscillations will be prevented or detected and suppressed prior to exceeding specified acceptable fuel design limits. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, core power oscillations cannot occur; a conclusion which states that VY is adequately protected from core oscillations and their consequences is no longer required. Nuclear System Stability Analysis information is obsolete.	
	7.16.7	References	Delete Section 7.16.5, Nuclear System Stability Analysis, Description and Performance Analysis. Since core power oscillations can no longer occur, the references provided in section 7.16.7 are no longer applicable.	
	Figure 7.16-1	BWROG Stability Criteria Map	Delete Figure 7.16-1, BWROG Stability Criteria Map. The Nuclear System Stability Analysis demonstrated that stability-related neutron flux oscillations will be prevented or detected and suppressed prior to exceeding specified acceptable fuel design limits. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, core power oscillations cannot occur; the functions and information provided by the Nuclear System Stability Analysis are no longer required. Therefore, the information provided on Figure 7.16-1 is no longer applicable. Nuclear System Stability Analysis information is obsolete.	
7.17	STANDBY GAS TREATMENT SYSTEM CONTROL AND INSTRUMENTATION			
	7.17.1	Safety Objective	Delete Section 7.17.1, Standby Gas Treatment System Control and Instrumentation Safety Objective. The controls and instrumentation for the Standby Gas Treatment System (SGTS) initiated appropriate responses from the SGTS and associated ventilating system so that the minimum safety requirements in the Reactor Building could be met. The Standby Gas Treatment System limited airborne fission product release to the environment by the use of high efficiency filters and by the maintenance of a negative pressure in the Reactor Building. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Analysis has demonstrated that the Standby Gas Treatment System is no longer required to mitigate the consequences of the remaining design basis accident, a fuel handling accident. Since the Standby Gas Treatment System is no longer required, the functions and information provided by the SBGT Controls and Instrumentation are also no longer required. Consequently, the Standby Gas Treatment System Controls and Instrumentation Safety Objective is not applicable. SBGT Controls and Instrumentation information is obsolete.	
	7.17.2	Safety Design Basis	Delete Section7.17.2, Standby Gas Treatment System Control and Instrumentation Safety Design Basis The controls and instrumentation for the Standby Gas Treatment System (SGTS) initiated appropriate responses from the SGTS and associated ventilating system so that the minimum safety requirements in the Reactor Building could be met. The Standby Gas Treatment System limited airborne fission product release to the environment by the use of high efficiency filters and by the maintenance of a negative pressure in the Reactor Building. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Analysis has demonstrated that the Standby Gas Treatment System is no longer required to mitigate the consequences of the remaining design basis accident, a fuel handling accident. Since the Standby Gas Treatment System is no longer required, the functions and information provided by the SBGT Controls and Instrumentation are also no longer required. Consequently, the Standby Gas Treatment System Controls and Instrumentation Safety Design Basis is not applicable. SBGT Controls and Instrumentation information is obsolete.	

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	7.17.3	Description	Delete Section7.17.3, Standby Gas Treatment System Control and Instrumentation Safety Design Basis The controls and instrumentation for the Standby Gas Treatment System (SGTS) initiated appropriate responses from the SGTS and associated ventilating system so that the minimum safety requirements in the Reactor Building could be met. The Standby Gas Treatment System limited airborne fission product release to the environment by the use of high efficiency filters and by the maintenance of a negative pressure in the Reactor Building. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Analysis has demonstrated that the Standby Gas Treatment System is no longer required to mitigate the consequences of the remaining design basis accident, a fuel handling accident. Since the Standby Gas Treatment System is no longer required, the functions and information provided by the SBGT Controls and Instrumentation are also no longer required. Consequently, the information provided in the Standby Gas Treatment System Controls and Instrumentation Description section is not applicable. SBGT Controls and Instrumentation information is obsolete.	
	7.17.4	Safety Evaluation	Delete Section7.17.4, Standby Gas Treatment System Control and Instrumentation Safety Evaluation. The controls and instrumentation for the Standby Gas Treatment System (SGTS) initiated appropriate responses from the SGTS and associated ventilating system so that the minimum safety requirements in the Reactor Building could be met. The Standby Gas Treatment System limited airborne fission product release to the environment by the use of high efficiency filters and by the maintenance of a negative pressure in the Reactor Building. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Analysis has demonstrated that the Standby Gas Treatment System is no longer required to mitigate the consequences of the remaining design basis accident, a fuel handling accident. Since the Standby Gas Treatment System is no longer required, the functions and information provided by the SBGT Controls and Instrumentation are also no longer required. Consequently, a safety evaluation of the Standby Gas Treatment System Controls and Instrumentation is not required. SBGT Controls and Instrumentation information is obsolete.	
	7.17.5	Inspection and Testing	Delete Section7.17.5, Standby Gas Treatment System Control and Instrumentation Inspection and Testing. The controls and instrumentation for the Standby Gas Treatment System (SGTS) initiated appropriate responses from the SGTS and associated ventilating system so that the minimum safety requirements in the Reactor Building could be met. The Standby Gas Treatment System limited airborne fission product release to the environment by the use of high efficiency filters and by the maintenance of a negative pressure in the Reactor Building. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Analysis has demonstrated that the Standby Gas Treatment System is no longer required to mitigate the consequences of the remaining design basis accident, a fuel handling accident. Since the Standby Gas Treatment System is no longer required, the functions and information provided by the SBGT Controls and Instrumentation are also no longer required. Consequently, Inspection and Testing of the Standby Gas Treatment System Controls and Instrumentation is not required. SBGT Controls and Instrumentation information is obsolete.	
	Table 7.17.1	Standby Gas Treatment System Instrumentation	Delete Table 7.17.1, Standby Gas Treatment System Instrumentation. The controls and instrumentation for the Standby Gas Treatment System (SGTS) initiated appropriate responses from the SGTS and associated ventilating system so that the minimum safety requirements in the Reactor Building could be met. The Standby Gas Treatment System limited airborne fission product release to the environment by the use of high efficiency filters and by the maintenance of a negative pressure in the Reactor Building. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Analysis has demonstrated that the Standby Gas Treatment System is no longer required to mitigate the consequences of the remaining design basis accident, a fuel handling accident. Since the Standby Gas Treatment System is no longer required, the functions and information provided by the SBGT Controls and Instrumentation are also no longer required. Consequently, a table of Standby Gas Treatment System Instrumentation is not required. SBGT Controls and Instrumentation information is obsolete.	
	Figure 7.17-1	G-191646 Standby Gas Treatment System Flow Diagram	Delete Figure 7.17-1, Standby Gas Treatment System Flow Diagram. The controls and instrumentation for the Standby Gas Treatment System (SGTS) initiated appropriate responses from the SGTS and associated ventilating system so that the minimum safety requirements in the Reactor Building could be met. The Standby Gas Treatment System limited airborne fission product release to the environment by the use of high efficiency filters and by the maintenance of a negative pressure in the Reactor Building. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Analysis has demonstrated that the Standby Gas Treatment System is no longer required to mitigate the consequences of the remaining design basis accident, a fuel handling accident. Since the Standby Gas Treatment System is no longer required, the functions and information provided by the SBGT Controls and Instrumentation are also no longer required. Consequently, the information provided on Figure 7.17-1 is not applicable. SBGT Controls and Instrumentation information is obsolete.	

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	Figure 7.17-2	G-191646 Standby Gas Treatment System Logic Diagram	Delete Figure 7.17-2, Standby Gas Treatment System Logic Diagram. The controls and instrumentation for the Standby Gas Treatment System (SGTS) initiated appropriate responses from the SGTS and associated ventilating system so that the minimum safety requirements in the Reactor Building could be met. The Standby Gas Treatment System limited airborne fission product release to the environment by the use of high efficiency filters and by the maintenance of a negative pressure in the Reactor Building. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Analysis has demonstrated that the Standby Gas Treatment System is no longer required to mitigate the consequences of the remaining design basis accident, a fuel handling accident. Since the Standby Gas Treatment System is no longer required, the functions and information provided by the SBGT Controls and Instrumentation are also no longer required. Consequently, the information provided on Figure 7.17-2 is not applicable. SBGT Controls and Instrumentation information is obsolete.	
7.18	ANTICIPATED TRANSIENT WITHOUT SCRAM PREVENTION/MITIGATION SYSTEM		The ATWS Prevention/Mitigation System provided a means to reduce the probability of a failure to scram or mitigated the consequences in the unlikely event that an anticipated transient occured and control rods failed to insert.	
	7.18.1	Safety Objective	Delete Section 7.18.1, Anticipated Transient Without Scram Prevention/Mitigation System, Safety Objective. The ATWS Prevention/Mitigation System provided a means to reduce the probability of a failure to scram or mitigated the consequences in the unlikely event that an anticipated transient occured and control rods failed to insert. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the Anticipated Transient Without Scram Prevention/Mitigation System are no longer required; the Anticipated Transient Without Scram Prevention/Mitigation System Safety Objective is no longer applicable Anticipated Transient Without Scram Prevention/Mitigation System information is obsolete.	
	7.18.2	Safety Design Bases	Delete Section 7.18.2, Anticipated Transient Without Scram Prevention/Mitigation System, Safety Design Bases. The ATWS Prevention/Mitigation System provided a means to reduce the probability of a failure to scram or mitigated the consequences in the unlikely event that an anticipated transient occured and control rods failed to insert. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the Anticipated Transient Without Scram Prevention/Mitigation System are no longer required; the Anticipated Transient Without Scram Prevention/Mitigation System Safety Design Bases are no longer applicable. Anticipated Transient Without Scram Prevention/Mitigation System information is obsolete.	
	7.18.3	Description	Delete Section 7.18.3, Anticipated Transient Without Scram Prevention/Mitigation System, Description. The ATWS Prevention/Mitigation System provided a means to reduce the probability of a failure to scram or mitigated the consequences in the unlikely event that an anticipated transient occured and control rods failed to insert. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the Anticipated Transient Without Scram Prevention/Mitigation System are no longer required; the description of the Anticipated Transient Without Scram Prevention/Mitigation System Safety Design Bases is no longer applicable. Anticipated Transient Without Scram Prevention/Mitigation System information is obsolete.	
	7.18.4	Safety Evaluation	Delete Section 7.18.4, Anticipated Transient Without Scram Prevention/Mitigation System, Safety Evaluation. The ATWS Prevention/Mitigation System provided a means to reduce the probability of a failure to scram or mitigated the consequences in the unlikely event that an anticipated transient occured and control rods failed to insert. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the Anticipated Transient Without Scram Prevention/Mitigation System are no longer required; a safety evaluation of the Anticipated Transient Without Scram Prevention/Mitigation System Safety Design Bases is no longer required. Anticipated Transient Without Scram Prevention/Mitigation System information is obsolete.	
	7.18.5	Inspection and Testing	Delete Section 7.18.5, Anticipated Transient Without Scram Prevention/Mitigation System, Inspection and Testing. The ATWS Prevention/Mitigation System provided a means to reduce the probability of a failure to scram or mitigated the consequences in the unlikely event that an anticipated transient occured and control rods failed to insert. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the Anticipated Transient Without Scram Prevention/Mitigation System are no longer required; Inspection and Testing of the Anticipated Transient Without Scram Prevention/Mitigation System is no longer required. Anticipated Transient Without Scram Prevention/Mitigation System information is obsolete.	
	Table 7.18.1	Anticipated Transient Without Scram Prevention/Mitigation System Instrumentation	Delete Table 7.18.1, Anticipated Transient Without Scram Prevention/Mitigation System Instrumentation. Since Therefore, the functions and information provided by the Anticipated Transient Without Scram Prevention/Mitigation System are no longer required; a table listing the ATWS Prevention/Mitigation System Instrumentation is no longer required. Anticipated Transient Without Scram Prevention/Mitigation System information is obsolete.	

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	Figure 7.18-1a	RPT and ARI Trip Logic	Delete Figure 7.18-1a, RPT and ARI Trip Logic. The ATWS Prevention/Mitigation System provided a means to reduce the probability of a failure to scram or mitigated the consequences in the unlikely event that an anticipated transient occured and control rods failed to insert. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the Anticipated Transient Without Scram Prevention/Mitigation System are no longer required. The information provided on Figure 7.18-1a regarding the Anticipated Transient Without Scram Prevention/Mitigation System is no longer applicable. Anticipated Transient Without Scram Prevention/Mitigation System information is obsolete.					
	Figure 7.18-1b	RPT and ARI Trip Actuators	Delete Figure 7.18-1b, RPT and ARI Trip Actuators. The ATWS Prevention/Mitigation System provided a means to reduce the probability of a failure to scram or mitigated the consequences in the unlikely event that an anticipated transient occured and control rods failed to insert. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions and information provided by the Anticipated Transient Without Scram Prevention/Mitigation System are no longer required. The information provided on Figure 7.18-1b regarding the Anticipated Transient Without Scram Prevention/Mitigation System is no longer applicable. Anticipated Transient Without Scram Prevention/Mitigation System information is obsolete.					
8.0	**Station Electrical Power Systems**		Placeholder section containing no information		3.3.3	Electrical Power Systems		
8.1	Station Electrical Systems		Modify Section 8.1, Station Electrical Systems Summary Description to eliminate startup, operational and accident references. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, electrical alignments for those conditions are no longer required. This summary level information was consolidated with UFSAR Plant Summary Section 1.6			1.3.5.1	Electrical Power Systems	
	Figure 8.1-1	G-191298, Sh.3 Main One-Line Diagram	No Change		Figure 3.3.3-1	G-191298, Sh.3 Main One-Line Diagram		
	Figure 8.1-2	G-191299 4160 and 480 Volt Auxiliary One-Line Diagram	No Change		Figure 3.3.3-2	G-191299 4160 and 480 Volt Auxiliary One-Line Diagram		
8.2	STATION MAIN GENERATOR UNIT							
	8.2.1	Power Generation Objective	Delete section 8.2.1, Station Main Generator Power Generation Objective. The station main generator provided a reliable source of electrical power to the 345 kV grid and the station Auxiliary Power System. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, the functions provided by the Main Generator are no longer required. The Main Generator Power Generation objective is not applicable. Main Generator information is obsolete.					
	8.2.2	Power Generation Design Basis	Delete section 8.2.2, Station Main Generator Power Generation Design Basis, The station main generator provided a reliable source of electrical power to the 345 kV grid and the station Auxiliary Power System. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, the functions provided by the Main Generator are no longer required. The Main Generator Power Generation design basis is no applicable. Main Generator information is obsolete.					
	8.2.3	Description	Delete section 8.2.3, Station Main Generator Description, The station main generator provided a reliable source of electrical power to the 345 kV grid and the station Auxiliary Power System. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, the information provided by the Main Generator description section is no longer applicable. Main Generator information is obsolete.					
	8.2.4	Inspection and Testing	Delete section 8.2.4, Station Main Generator Inspection and Testing The station main generator provided a reliable source of electrical power to the 345 kV grid and the station Auxiliary Power System. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, the inspection and testing of the Main Generator is no longer required. Main Generator information is obsolete.					
8.3	STATION TRANSMISSION SYSTEM					3.3.3.1	Transmission System	
	8.3.1	Power Generation Objective	Modify Section 8.3.1 , Station Transmission System Power Generation Objective to remove discussions of 345 kV transmission lines, and reword to describe station power supply via the 115 kV startup transformers. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Consequently, the station will no longer be generating and supplying power to the 345 kV transmission system. The station is isolated from the 345 kV transmission system, Station Power is supplied via the 115 kV system.				3.3.3.1.1	Objective

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	8.3.2	Power Generation Design Basis		Modify Section 8.3.2 , Station Transmission System Power Generation Design Basis to remove discussions of 345 kV transmission lines, and reword to describe station power supply via the 11 kV startup transformers. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Consequently, the station will no longer be generating and supplying power to the 345 kV transmission system. The station is isolated from the 345 kV transmission system, Station Power is supplied via the 115 kV system.				3.3.3.1.2	Design Basis	
	8.3.3	Description		Modify Section 8.3.3 , Station Transmission System Description to remove discussions of 345 kV transmission lines, and reword to describe station power supply via the 115 kV startup transformers. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Consequently, the station will no longer be generating and supplying power to the 345 kV transmission system. The station is isolated from the 345 kV transmission system, Station Power is supplied via the 115 kV system.				3.3.3.1.3	Description	
	8.3.4	Evaluation of System Protection		Modify Section 8.3.4 , Station Transmission System Description to remove discussions of 345 kV transmission lines, and reword to describe station power supply via the 115 kV startup transformers. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Consequently, the station will no longer be generating and supplying power to the 345 kV transmission system. The station is isolated from the 345 kV transmission system, Station Power is supplied via the 115 kV system.				3.3.3.1.4	Evaluation of System Protection	
	8.3.5	Technical Specifications		Delete Section 8.3.5 , Station Transmission System Technical Specification Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Consequently, the station will no longer be generating and supplying power to the 345 kV transmission system. The station is isolated from the 345 kV transmission system, Station Power is supplied via the 115 kV system.						
	Figure 8.3-1	Transmission Lines River Crossing		Delete Figure 8.3-1 , Transmission Lines River Crossing. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Consequently, the station will no longer be generating and supplying power to the 345 kV transmission system. The station is isolated from the 345 kV transmission system, Station Power is supplied via the 115 kV system.						
	Figure 8.3-2	G-191298 sheet 3, MAIN ONE LINE WIRING DIAGRAM		No Changes		Figure 3.3.3-1	G-191298 sheet 3, MAIN ONE LINE WIRING DIAGRAM			
8.4	STATION AUXILIARY POWER SYSTEMS						3.3.3.2	Auxiliary Power Systems		
	8.4.1	Power Generation Objective		Modify Section 8.4.1, Station Auxiliary Power System, Power Generation Objective. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, the main generator will no longer be operated, the station auxiliary power system will be powered from the startup transformers. All reference to operational conditions has been deleted. The power generation Objective section has been modified to remove the term "Power Generation", and the objective has been reworded to reflect the safe storage and handling of irradiated fuel.				3.3.3.2.1	Objective	
	8.4.2	Power Generation Design Basis		Modify Section 8.4.2, Station Auxiliary Power System, Power Generation Design Basis. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, the main generator will no longer be operated, the station auxiliary power system will be powered from the startup transformers. All reference to operational conditions has been deleted. The power generation Design Basis section has been modified to remove the term "Power Generation", and the design basis section has been reworded to reflect the safe storage and handling of irradiated fuel.				3.3.3.2.2	Design Basis	
	8.4.3	Safety Objective		Delete section 8.4.3, Station Auxiliary Power System, Safety Objective Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the Station Auxiliary Power System Safety Objective is no longer applicable.						
	8.4.4	Safety Design Basis		Delete section 8.4.4, Station Auxiliary Power System, Safety Design Basis Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the Station Auxiliary Power System Safety Design Basis is no longer applicable.						
	8.4.5	Description						3.3.3.2.3	Description	
		8.4.5.1	4160 V Switchgear	Modify Section 8.4.5.1, Station Auxiliary Power System, 4260 V Switchgear to remove reference to operating and accident conditions, reflect that the auxiliary power system can no longer be supplied by the unit auxiliary transformer, and note the station auxiliary power system function is now to supply power for the safe storage and handling of irradiated fuel. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.					3.3.3.2.3.1	4160 V Switchgear
		8.4.5.2	480 V Buses	Modify Section 8.4.5.2, Station Auxiliary Power System, 480 V Busses to remove reference to operating and accident conditions, reflect that the auxiliary power system can no longer be supplied by the unit auxiliary transformer, and note the station auxiliary power system function is now to supply power for the safe storage and handling of irradiated fuel. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.					3.3.3.2.3.2	480 V Buses

VY UFSAR		FSAR Conversion to DSAR Change Summary	VY DSAR	
UFSAR Section			DSAR Section	
8.4.5.3	120/240 V Instrumentation Distribution System	Modify Section 8.4.5.3, Station Auxiliary Power System, 120/240 V Instrumentation Distribution System to remove reference to electrical loads which are no longer required and remove reference to the MG set. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, many of the loads supplied by the 120/240 Instrumentation System are no longer required. Additionally, the functions provided by the MG set are no longer required. That information is historical and obsoleted and may be deleted.	3.3.3.2.3.3	120/240 V Instrumentation Distribution System
8.4.6	Cable Installation and Separation Criteria	Modify Section 8.4.6, Station Auxiliary Power System, Cable Installation and Separation Criteria to remove reference to cable separation criteria for redundant safety systems. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions provided by the Reactor Protection System, the Primary Containment Isolation System, the Engineered Safety Feature and ECCS, Primary Containment Electrical Penetrations, and Drywell Separation criteria are no longer applicable. Therefore, the Cable Installation and Separation Criteria section has been modified to remove reference to those areas, as well as reference to operating conditions.	3.3.3.2.4	Cable Installation and Separation Criteria
8.4.7	Safety Evaluation	Delete Section 8.4.7, Station Auxiliary Power System, Safety Evaluation Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2).Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, a safety evaluation of the Station Auxiliary Power System is not required.		
8.4.8	Inspection and Testing	Modify Section 8.4.8, Station Auxiliary Power System, Inspection and Testing to remove reference to operating conditions Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2).Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.	3.3.3.2.5	Inspection and Testing
8.4.9	Technical Specifications	Delete Section 8.4.9, Station Auxiliary Power System, Technical Specifications. All TS surveillances related to the Station Auxiliary Power System have been deleted in the Defueled TS submitta to the NRC		
Figure 8.4-1	G-191299 Emergency 4160 Volt Auxiliary One-Line Diagram	No changes	Figure 3.3.3-2	G-191299 Balance of Plant 4160 Volt Auxiliary One-Line Diagram
Figure 8.4-2	G-191299 Balance of Plant 4160 Volt Auxiliary One-Line Diagram	No changes	Figure 3.3.3-2	G-191299 Balance of Plant 4160 Volt Auxiliary One-Line Diagram
Figure 8.4-3	G-191300, Sh1 Emergency 480 Volt Auxiliary One-Line Diagram (SI)	No changes	Figure 3.3.3-3	G-191300, Sh1 Emergency 480 Volt Auxiliary One-Line Diagram (SI)
Figure 8.4-4	G-191300, Sh2 Emergency 480 Volt Auxiliary One-Line Diagram (SI)	No changes	Figure 3.3.3-4	G-191300, Sh2 Emergency 480 Volt Auxiliary One-Line Diagram (SI)
Figure 8.4-5	G-191301, Sh1 Emergency 480 Volt Auxiliary One-Line Diagram (SII)	No changes	Figure 3.3.3-5	G-191301, Sh1 Emergency 480 Volt Auxiliary One-Line Diagram (SII)
Figure 8.4-6	G-191301, Sh2 Emergency 480 Volt Auxiliary One-Line Diagram (SII)	No changes	Figure 3.3.3-6	G-191301, Sh2 Emergency 480 Volt Auxiliary One-Line Diagram (SII)
Figure 8.4-7	G-191300, Sh2 G-191301, Sh2 Emergency 480 Volt Auxiliary One-Line Diagram (SI & SII)	Delete Figure 8.4-7, Duplicates other figures in this section		
Figure 8.4-8	G-191372, Sh4 G-191372, Sh5 120/240 Volt Instrumentation One-Line Diagram	No changes	Figure 3.3.3-10	G-191372, Sh4 120/240 Volt Instrumentation One-Line Diagram
Figure 8.4-8	G-191372, Sh4 G-191372, Sh5 120/240 Volt Instrumentation One-Line Diagram	No changes	Figure 3.3.3-11	G-191372, Sh5 One Line Diagram, +24 Volt DC Power System
8.5	STATION STANDBY DIESEL-GENERATOR SYSTEMS		3.3.3.3	STANDBY DIESEL-GENERATOR SYSTEMS
8.5.1	Safety Objective	Modify Section 8.5.1, Station Standby Diesel-Generator Systems Safety Objective to remove reference to operating and accident conditions, and reflect that the diesel-generators now supply back-up power only to ensure the safe storage and handling of irradiated fuel. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the Diesel-Generators are only required to supply backup power to ensure the safe storage of irradiated fuel.	3.3.3.3.1	Objective
8.5.2	Safety Design Bases	Modify Section 8.5.2, Station Standby Diesel-Generator Systems Safety Design Bases to remove reference to operating and accident conditions, and reflect that the diesel-generators now supply back-up power only to ensure the safe storage and handling of irradiated fuel. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the Diesel-Generators are only required to supply backup power to ensure the safe storage of irradiated fuel.	3.3.3.3.2	Design Bases

VY UFSAR				VY DSAR				
UFSAR Section			FSAR Conversion to DSAR Change Summary	DSAR Section				
	8.5.3	Description	Modify Section 8.5.3, Station Standby Diesel-Generator Systems Description to remove reference to operating and accident conditions, and reflect that the diesel-generators now supply back-up power only to ensure the safe storage and handling of irradiated fuel. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the Diesel-Generators are only required to supply backup power to ensure the safe storage of irradiated fuel.				3.3.3.3.3	Description
	8.5.4	Safety Evaluation	Delete Section 8.5.4, Station Standby Diesel-Generator Systems Safety Evaluation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the safety functions of the Standby Diesel-Generators are no longer required. Consequently, a safety evaluation of the Standby Diesel-Generator Systems is no longer reqired.					
	8.5.5	Loss of Diesel Generator	Delete Section 8.5.5, Loss of Diesel Generators. Availibility of an alternate power supply alignment to busses 3 and 4 is addressed in section 8.4.5.1, 4160 VAC Busses. There are no safety consequences associate with the loss of a backup diesel-generator in the permanently defueled condition. Adequate mitigation strategies exist to ensure spent fuel temperature and inventory are maintained.					
	8.5.6	Inspection and Testing	Modify Section 8.5.4, Station Standby Diesel-Generator Systems Inspection and Testing to remove reference to operating and accident conditions, and reflect that the diesel-generators now supply back-up power only to ensure the safe storage and handling of irradiated fuel. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the Diesel-Generators are only required to supply backup power to ensure the safe storage of irradiated fuel. Standby Diesel-Generator Inspection and Testing will be commeasurate with the function of the Diesels.				3.3.3.3.4	Inspection and Testing
	8.5.7	Technical Specifications	Delete Section 8.5.7, Standby Diesel-Generator Systems Technical Specifications. VY will never be in a mode which will require the standby diesel-generators to be operable by the Technica Specifications.					
	Table 8.5.1A	Diesel Generator DG-1-1A Loading - Loss of AC and Maximum Hypothetical Accident	Delete Table 8.5.1A, Diesel Generator DG-1-1A Loading - Loss of AC and Maximum Hypothetical Accident. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the projected diesel generator loading in this table is no longer required since the maximum hypothetical accident is no longer possible.					
	Table 8.5.1B	Diesel Generator DG-1-1B Loading - Loss of AC and Maximum Hypothetical Accident	Delete Table 8.5.1B, Diesel Generator DG-1-1B Loading - Loss of AC and Maximum Hypothetical Accident. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the projected diesel generator loading in this table is no longer required since the maximum hypothetical accident is no longer possible.					
	Table 8.5.2A	Diesel Generator DG-1-1A Loading - Loss of AC	Delete table 8.5.2A. The table provided a listing of loads which would automatically start and load on each diesel generator following specific plant initiating conditions. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, when required, the Diesel-Generators will supply sufficient backup power required to ensure the safe storage of irradiated fuel. That loading is a small percentage of the Diesel Generator loading which was required following a loss of off-site power and a loss of coolant or other design basis accident. Consequently, the information provide on table 8.5.2A is historical and obsolete. As such, this table may be deleted.					
	Table 8.5.2B	Diesel Generator DG-1-1B Loading - Loss of AC	Delete table 8.5.2B. The table provided a listing of loads which would automatically start and load on each diesel generator following specific plant initiating conditions. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, when required, the Diesel-Generators will supply sufficient backup power required to ensure the safe storage of irradiated fuel. That loading is a small percentage of the Diesel Generator loading which was required following a loss of off-site power and a loss of coolant or other design basis accident. Consequently, the information provide on table 8.5.2B is historical and obsolete. As such, this table may be deleted.					
	Table 8.5.3A	Diesel Generator DG-1-1A Loading - Loss of AC and Vernon Pond	Delete Table 8.5.3A, Diesel Generator DG-1-1A Loading - Loss of AC and Vernon Pond. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since SW was lost during a loss of the Vernon Pond, DG cooling was provide by RHRSW. RHR SW is not required in the permanently defueled state, therefore, cooling to the DGs is not available during a loss of the Vernon Pond. Other power sources are available and strategies have been developed to maintain SFP inventory and cooling.Therefore, the projected diesel generator loading during a loss of the Vernon Pond is no longer required.					

VY UFSAR				VY DSAR				
UFSAR Section			FSAR Conversion to DSAR Change Summary	DSAR Section				
	Table 8.5.3B	Diesel Generator DG-1-1B Loading - Loss of AC and Vernon Pond	Delete Table 8.5.3B, Diesel Generator DG-1-1B Loading - Loss of AC and Vernon Pond. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since SW was lost during a loss of the Vernon Pond, DG cooling was provide by RHRSW. RHR SW is not required in the permanently defueled state, therefore, cooling to the DGs is not available during a loss of the Vernon Pond. Other power sources are available and strategies have been developed to maintain SFP inventory and cooling.Therefore, the projected diesel generator loading during a loss of the Vernon Pond is no longer required.					
	Figure 8.5-1	Fuel Oil System	Delete Figure 8.5-1, Fuel Oil System. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the Diesel-Generators are only required to supply backup power as necessary to ensure the safe storage of irradiated fuel. Therefore, a drawing of the fuel oil system is no longer required to demonstrate that the Diesel-Generators are capable of performing their intended function.					
8.6	125 VOLT BATTERY SYSTEM					3.3.3.4	125 VOLT BATTERY SYSTEM	
	8.6.1	Safety Objective	Modify Section 8.6.1, 125 Volt Battery System, Safety Objective to remove reference to operating and accident conditions, and reflect that the 125 V DC system now supplies power only to ensure the safe storage and handling of irradiated fuel. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the safety functions of the 125 V DC system is no longer required Consequently, the word "Safety" has been deleted from the title and the text and the word "emergency" has been deleted from the text.				3.3.3.4.1	Objective
	8.6.2	Safety Design Basis	Modify Section 8.6.2, 125 Volt Battery System, Safety Design Basis to remove reference to operating and accident conditions, and reflect that the 125 V DC system now supplies power only to ensure the safe storage and handling of irradiated fuel. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the safety design bases of the 125 V DC system is no longer applicable. Consequently, the word "Safety" has been deleted from the title and reference to redundancy, independence and single failure criteria have been deleted from the text.				3.3.3.4.2	Design Basis
	8.6.3	Description	Modify Section 8.6.3, 125 Volt Battery System, Description to remove reference to operating and accident conditions, and reflect that the 125 V DC system now supplies power only to ensure the safe storage and handling of irradiated fuel. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the performance requirements of the 125 V DC system have changed to essentially providing control power for various station components. Consequently, reference to safety related classification and discharge rates and times have been deleted from the text. Additionally, the functions provided by station battery AS-1 are no longer required. Therefore, references to station battery AS-1 have been deleted.				3.3.3.4.3	Description
	8.6.4	Safety Evaluation	Modify Section 8.6.4, 125 Volt Battery System, Safety Evaluation, to remove reference to operating and accident conditions, and reflect that the 125 V DC system now supplies power only to ensure the safe storage and handling of irradiated fuel. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the safety functions of the 125 V DC system is no longer required Consequently, the word "Safety" has been deleted from the title. Additionally, reference to safety related classification and discharge rates and times have been deleted from the text. Also, the functions provided by station battery AS-1 are no longer required. Therefore, references to station battery AS-1 have been deleted.				3.3.3.4.4	Evaluation
	8.6.5	Additional DC Systems	No Changes				3.3.3.4.5	Additional DC Systems
	8.6.6	Inspection and Testing	No Changes				3.3.3.4.6	Inspection and Testing
	8.6.7	Technical Specifications	Delete Section 8.6.7, 125 Volt Battery System, Technical Specifications. VY will never be in a mode which will require the battery system to be operable by the Technical Specifications.					
	Figure 8.6-1	G-191372, Shs 1,2,3 125 V DC One-Line Diagram	No Changes		Figure 3.3.3-7 3.3.3-8 3.3.3-9	G-191372, Shs 1,2,3 125 V DC One-Line Diagram		
8.7	+24 V DC Power Sustem					3.3.3.5	+24 V DC Power System	
	8.7.1	Power Generation Objective	Modify Section 8.7.1, +24 Volt DC Power System, Power Generation Objective to remove reference to operating and accident conditions, and reflect that the +24 Volt DC Power System now supplies power only to ensure the safe storage and handling of irradiated fuel. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since there is no core to protect, the neutron monitoring systems are no longer required . Therefore, power to the neutron monitoring systems from the +24 V DC system is no longer required. Reference to power supplied to the neutron monitoring systems has been deleted from this section.				3.3.3.5.1	Objective

VY UFSAR				VY DSAR	
UFSAR Section			FSAR Conversion to DSAR Change Summary	DSAR Section	
	8.7.2	Power Generation Design Basis	Modify Section 8.7.2, +24 Volt DC Power System, Power Generation Design Basis to remove reference to operating and accident conditions, and reflect that the +24 Volt DC Power System now supplies power only to ensure the safe storage and handling of irradiated fuel. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible.	3.3.3.5.2	Design Basis
	8.7.3	Description	Modify Section 8.7.3, +24 Volt DC Power System, Description, to remove reference to operating and accident conditions, and reflect that the +24 Volt DC Power System now supplies power only to ensure the safe storage and handling of irradiated fuel. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since there is no core to protect, the neutron monitoring systems are no longer required . Therefore, power to the neutron monitoring systems from the +24 V DC system is no longer required. Reference to power supplied to the neutron monitoring systems has been deleted from this section. Additionally, reference to discharge rates and times have been deleted from the text.	3.3.3.5.3	Description
	8.7.4	Inspection and Testing	No changes	3.3.3.5.4	Inspection and Testing
	8.7.5	Technical Specifications	Delete Section 8.7.5, 125 Volt Battery System, Technical Specifications. VY will never be in a mode which will require the battery system to be operable by the Technical Specifications.		
	Figure 8.7-1	G-191372, Sh5 One Line Diagram, +24 Volt DC Power System	No changes	Figure 3.3.3-11	G-191372, Sh5 One Line Diagram, +24 Volt DC Power System
8.8	ECCS 24 VOLT DC POWER SYSTEM				
	8.8.1	Safety Objective	Delete Section 8.8.1, ECCS 24 Volt DC Power System Safety Objective. The ECCS 24 V DC Power System provided a supply of 24 V dc power for the operation of the ECCS Analog Trip System. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2).Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the ECCS 24 Volt DC Power System are no longer required. The safety objective is no longer applicable. ECCS 24 V DC Power System information is obsolete.		
	8.8.2	Safety Design Basis	Delete Section 8.8.2, ECCS 24 Volt DC Power System Safety Design Bases. The ECCS 24 V DC Power System provided a supply of 24 V dc power for the operation of the ECCS Analog Trip System. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2).Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the ECCS 24 Volt DC Power System are no longer required. The safety design bases are no longer applicable. ECCS 24 V DC Power System information is obsolete.		
	8.8.3	Description	Delete Section 8.8.3, ECCS 24 Volt DC Power System Description. The ECCS 24 V DC Power System provided a supply of 24 V dc power for the operation of the ECCS Analog Trip System. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2).Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the ECCS 24 Volt DC Power System are no longer required. The information provided in the description section is no longer applicable. ECCS 24 V DC Power System information is obsolete.		
	8.8.4	Safety Evaluation	Delete Section 8.8.4, ECCS 24 Volt DC Power System Safety Evaluation. The ECCS 24 V DC Power System provided a supply of 24 V dc power for the operation of the ECCS Analog Trip System. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2).Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the ECCS 24 Volt DC Power System are no longer required. A safety evaluaion of the ECCS 24 Volt DC Powr System is no longer required. ECCS 24 V DC Power System information is obsolete.		
	8.8.5	Inspection and Testing	Delete Section 8.8.5, ECCS 24 Volt DC Power System Safety Evaluation. The ECCS 24 V DC Power System provided a supply of 24 V dc power for the operation of the ECCS Analog Trip System. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2).Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Thererfore, the functions provided by the ECCS 24 Volt DC Power System are no longer required. Inspection and Testing of the ECCS 24 Volt DC Power System is no longer required. ECCS 24 V DC Power System information is obsolete.		
	Figure 8.8-1	One Line Diagram ECCS 24 Volt DC Power System G-191297	Delete Section 8.8-1,One Line Diagram ECCS 24 Volt DC Power System. The ECCS 24 V DC Power System provided a supply of 24 V dc power for the operation of the ECCS Analog Trip System. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2).Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the ECCS 24 Volt DC Power System are no longer required. The information provided on figure 8.8-1 is no longer applicable. ECCS 24 V DC Power System information is obsolete.		
9.0	**Radioactive Waste Systems**				

VY UFSAR					**VY DSAR**				
UFSAR Section				FSAR Conversion to DSAR Change Summary	DSAR Section				
9.1	SUMMARY DESCRIPTION			Modified Section 9.1, Radwaste Summary Description as follows: 1. Deleted "Station Operation", inserted "safe storage and handling of irradiated fuel" to reflect the post defueling function. 2. Removed unnecessary specifity from the first paragraph. 3. Changed "Station" to "Facility" to reflect post defueling function 4. Deleted reference to the Gaseous Radwaste System. Following permanent defueling, the air ejectors, vacuum pump, gland seal condenser and the Standby Gas Treatment System will no longer be operated. Ventilation system information is provided in the appropriate ventilation system sections. 5. Removed statement regarding continued compliance with annual dose limits with 1912 MWth rating, no longer applicable post defueling.		1.3.3	Radioactive Waste Management		
9.2	LIQUID RADWASTE SYSTEM					4.5.2	Liquid Radwaste System		
	9.2.1	Power Generation Objective		Deleted "Power Generation" from the section title. Since the classification process described in section 1.4.1 is no longer used, the classification basis is no longer required and is not applicabl to a permanently defueled station.			4.5.2.1	Objective	
	9.2.2	Power Generation Design Basis		Deleted "Power Generation" from the section title. Since the classification process described in section 1.4.1 is no longer used, the classification basis is no longer required and is not applicabl to a permanently defueled station.			4.5.2.2	Design Basis	
	9.2.3	Safety Design Basis		Deleted Section 9.2.3, Safety Design basis, moved information to Section 9.2.2, Design Bases.					
	9.2.4	Description					4.5.2.3	Description	
		9.2.4.1	High Purity Wastes	Editorial Changes only				4.5.2.3.1	High Purity Wastes
		9.2.4.2	Low Purity Wastes	Editorial Changes only				4.5.2.3.2	Low Purity Wastes
		9.2.4.3	Chemical Wastes	Editorial Changes only				4.5.2.3.3	Chemical Wastes
		9.2.4.4	Detergent Wastes	Editorial Changes only				4.5.2.3.4	Detergent Wastes
	9.2.5	Safety Evaluation		Editorial Changes only			4.5.2.4	Evaluation	
	9.2.6	Inspection and Testing		No Changes			4.5.2.5	Inspection and Testing	
	Table 9.2.1	Vermont Yankee Radioactive Liquid Waste Processing Parameters		No Changes		Table 4.5.2.1	Vermont Yankee Radioactive Liquid Waste Processing Parameters		
	Table 9.2.2	Vermont Yankee Liquid Radwaste System Tank Capacities		No Changes		Table 4.5.2.2	Vermont Yankee Liquid Radwaste System Tank Capacities		
	Table 9.2.3	Vermont Yankee Liquid Effluents		No Changes		Table 4.5.2.3	Vermont Yankee Liquid Effluents		
	Table 9.2.4	Activity Input to Liquid Radwaste System (Ci/yr)		No Changes		Table 4.5.2.4	Activity Input to Liquid Radwaste System (Ci/yr)		
	Table 9.2.5	Radionuclide Discharge Concentrations		No Changes		Table 4.5.2.5	Radionuclide Discharge Concentrations		
	Figure 9.2-1a	G-191151 Radioactive Waste Building General Arrangement		No Changes		Figure 4.5.2-1	G-191151 Radioactive Waste Building General Arrangement		
	Figure 9.2-1b	G-191152 Radioactive Waste Building General Arrangement		No Changes		Figure 4.5.2-2	G-191152 Radioactive Waste Building General Arrangement		
	Figure 9.2-2	5920-644, Sh2 Radwaste System, Process Diagram		No Changes		Figure 4.5.2-3	5920-644, Sh2 Radwaste System, Process Diagram		
	Figure 9.2-3	G-191177, Sh1 Liquid Radwaste System		No Changes		Figure 4.5.2-4	G-191177, Sh1 Liquid Radwaste System		
	Figure 9.2-4	G-191177, Sh2 Liquid Radwaste System		No Changes		Figure 4.5.2-5	G-191177, Sh2 Liquid Radwaste System		
	Figure 9.2-5	G-191177, Sh3 Liquid Radwaste System		No Changes		Figure 4.5.2-6	G-191177, Sh3 Liquid Radwaste System		
	Figure 9.2-6	G-191177, Sh4 Liquid Radwaste System		No Changes		Figure 4.5.2-7	G-191177, Sh4 Liquid Radwaste System		
	Figure 9.2-7	Radwaste Area - Plan View		No Changes		Figure 4.5.2-8	Radwaste Area - Plan View		
9.3	SOLID RADWASTE SYSTEM					4.6.1	SOLID RADWASTE SYSTEM		
	9.3.1	Power Generation Objective		Deleted "Power Generation" from the section title. Since the classification process described in section 1.4.1 is no longer used, the classification basis is no longer required and is not applicabl to a permanently defueled station.			4.6.1.1	Objective	
	9.3.2	Power Generation Design Basis		Deleted "Power Generation" from the section title. Since the classification process described in section 1.4.1 is no longer used, the classification basis is no longer required and is not applicabl to a permanently defueled station.			4.6.1.2	Design Basis	
	9.3.3	Description					4.6.1.3	Description	
		9.3.3.1	General	No change				4.6.1.2.1	General

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		9.3.3.2	Wet Wastes	No change				4.6.1.2.2	Wet Wastes
		9.3.3.3	Dry Wastes	No change				4.6.1.2.3	Dry Wastes
		9.3.3.4	Waste Storage Facility	Statement regarding temporary storage of low level waste modified to delete reference to North Warehouse to accommodate EC 47799. Reference to weather tight shipping containers deleted to remove excess detail. Section deleted in accordance with EC 54271. Waste storage facility has been demolished.					
	9.3.4	Inspection and Testing		No change			4.6.1.4	Inspection and Testing	
	Table 9.3.1	Solid Radwaste Annual Disposal History		No change		Table 4.6.1	Solid Radwaste Annual Disposal History		
9.4	GASEOUS RADWASTE SYSTEM								
	9.4.1	Power Generation Objective		Delete Section 9.4.1, Gaseous Radwaste System Power Generation Objective. The Gaseous Radwaste System collected, processed, and discharged radioactive gaseous wastes to the atmosphere through the plant stack during normal operation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, radioactive gaseous waste will no longer be produced from either station operation or core related design basis accidents. Additionally, analysis has demonstrated that secondary containment and the Standby Gas Treatment System are no longer required to mitigate the consequences of the remaining design basis accident, a fuel handling accident. Therefore, the Gaseous Radwaste System power generation objective is no longer applicable. Gaseous Radwaste System information is obsolete.					
	9.4.2	Safety Objective		Delete Section 9.4.2, Gaseous Radwaste System Safety Objective. The Gaseous Radwaste System collected, processed, and discharged radioactive gaseous wastes to the atmosphere through the plant stack during normal operation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, radioactive gaseous waste will no longer be produced from either station operation or core related design basis accidents. Additionally, analysis has demonstrated that secondary containment and the Standby Gas Treatment System is no longer required to mitigate the consequences of the remaining design basis accident, a fuel handling accident. Therefore, the Gaseous Radwaste System safety objective is no longer applicable. Gaseous Radwaste System information is obsolete.					
	9.4.3	Power Generation Design Basis		Delete Section 9.4.3, Gaseous Radwaste System Power Generation Design Bases. The Gaseous Radwaste System collected, processed, and discharged radioactive gaseous wastes to the atmosphere through the plant stack during normal operation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, radioactive gaseous waste will no longer be produced from either station operation or core related design basis accidents. Additionally, analysis has demonstrated that secondary containment and the Standby Gas Treatment System are no longer required to mitigate the consequences of the remaining design basis accident, a fuel handling accident. Therefore, the Gaseous Radwaste System power generation design bases are no longer applicable. Gaseous Radwaste System information is obsolete.					
	9.4.4	Safety Design Basis		Delete Section 9.4.4, Gaseous Radwaste System Safety Design Bases. The Gaseous Radwaste System collected, processed, and discharged radioactive gaseous wastes to the atmosphere through the plant stack during normal operation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, radioactive gaseous waste will no longer be produced from either station operation or core related design basis accidents. Additionally, analysis has demonstrated that secondary containment and the Standby Gas Treatment System are no longer required to mitigate the consequences of the remaining design basis accident, a fuel handling accident. Therefore, the Gaseous Radwaste System Safety Design Bases are no longer applicable. Gaseous Radwaste System information is obsolete.					
	9.4.5	Description		Delete Section 9.4.5, Gaseous Radwaste System Description. The Gaseous Radwaste System collected, processed, and discharged radioactive gaseous wastes to the atmosphere through the plant stack during normal operation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, radioactive gaseous waste will no longer be produced from either station operation or core related design basis accidents. Additionally, analysis has demonstrated that secondary containment and the Standby Gas Treatment System are no longer required to mitigate the consequences of the remaining design basis accident, a fuel handling accident.Further, the incineration of slightly radioactive oil will no longer be performed. Therefore, information provided in the Gaseous Radwaste System Description Section is no longer applicable. Gaseous Radwaste System information is obsolete.					

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	9.4.6	Safety Evaluation	Delete Section 9.4.6, Gaseous Radwaste System Safety Evaluation. The Gaseous Radwaste System collected, processed, and discharged radioactive gaseous wastes to the atmosphere through the plant stack during normal operation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, radioactive gaseous waste will no longer be produced from either station operation or core related design basis accidents. Additionally, analysis has demonstrated that secondary containment and the Standby Gas Treatment System are no longer required to mitigate the consequences of the remaining design basis accident, a fuel handling accident. Therefore, a safety evaluation of the Gaseous Radwaste System is no longer required. Gaseous Radwaste System information is obsolete.	
	9.4.7	Inspection and Testing	Delete Section 9.4.7, Gaseous Radwaste System Inspection and Testing The Gaseous Radwaste System collected, processed, and discharged radioactive gaseous wastes to the atmosphere through the plant stack during normal operation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, radioactive gaseous waste will no longer be produced from either station operation or core related design basis accidents. Additionally, analysis has demonstrated that secondary containment and the Standby Gas Treatment System are no longer required to mitigate the consequences of the remaining design basis accident, a fuel handling accident. Therefore, inspection and testing of the Gaseous Radwaste System is no longer required. Gaseous Radwaste System information is obsolete.	
	9.4.8	REFERENCES	Delete Section 9.4.8, Gaseous Radwaste System References. The Gaseous Radwaste System collected, processed, and discharged radioactive gaseous wastes to the atmosphere through the plant stack during normal operation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, radioactive gaseous waste will no longer be produced from either station operation or core related design basis accidents. Additionally, analysis has demonstrated that secondary containment and the Standby Gas Treatment System are no longer required to mitigate the consequences of the remaining design basis accident, a fuel handling accident. Therefore, Gaseous Radwaste System references are no longer applicable. Gaseous Radwaste System information is obsolete.	
	Table 9.4.1	Air Ejector Isotope and Site Boundary Release Rates	Delete Table 9.4.1, Air Ejector Isotope and Site Boundary Release Rates. The Gaseous Radwaste System collected, processed, and discharged radioactive gaseous wastes to the atmosphere through the plant stack during normal operation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, radioactive gaseous waste will no longer be produced from either station operation or core related design basis accidents. Additionally, analysis has demonstrated that secondary containment and the Standby Gas Treatment System are no longer required to mitigate the consequences of the remaining design basis accident, a fuel handling accident. Therefore, the information provided in Table 9.4.1, Air Ejector Isotope and Site Boundary Release Rates is no longer applicable. Gaseous Radwaste System information is obsolete.	
	Figure 9.4-1	VY-E-75-001 Advanced Off Gas System Piping and Instrumentation Diagram	Delete Figure 9.4-1, Advanced Off Gas System Piping and Instrumentation Diagram. The Gaseous Radwaste System collected, processed, and discharged radioactive gaseous wastes to the atmosphere through the plant stack during normal operation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, radioactive gaseous waste will no longer be produced from either station operation or core related design basis accidents. Additionally, analysis has demonstrated that secondary containment and the Standby Gas Treatment System are no longer required to mitigate the consequences of the remaining design basis accident, a fuel handling accident. Further, the incineration of slightly radioactive oil will no longer be performed. Therefore, information provided on figure 9.4-1 is no longer applicable. Gaseous Radwaste System information is obsolete.	
	Figure 9.4-2	G-191162, Sh3 Off Gas System	Delete Figure 9.4-2, Off Gas System. The Gaseous Radwaste System collected, processed, and discharged radioactive gaseous wastes to the atmosphere through the plant stack during normal operation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, radioactive gaseous waste will no longer be produced from either station operation or core related design basis accidents. Additionally, analysis has demonstrated that secondary containment and the Standby Gas Treatment System are no longer required to mitigate the consequences of the remaining design basis accident, a fuel handling accident. Further, the incineration of slightly radioactive oil will no longer be performed. Therefore, information provided on figure 9.4-2 is no longer applicable. Gaseous Radwaste System information is obsolete.	

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10.0	**Station Auxiliary Systems**							
10.1	SUMMARY DESCRIPTION		Delete Section 10.1. An introduction to auxiliary systems is not required in the DSAR. No meaningful information is provided in this section.					
10.2	NEW FUEL STORAGE RACKS							
	10.2.1	Power Generation Objective	Delete Section 10.2.1, New Fuel Storage Racks, Power Generation Objective. The new fuel storage racks provided a safe, specially designed storage place for new fuel assemblies or bundles. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore,new fuel will no longer be received on the station. Consequently, the functions provided by the new fuel storage racks are no longer required. The New Fuel Storage Rack Power Generation Objective is no longer applicable. New Fuel Storage Rack information is obsolete.					
	10.2.2	Power Generation Design Bases	Delete Section 10.2.2, New Fuel Storage Racks, Power Generation Design Bases. The new fuel storage racks provided a safe, specially designed storage place for new fuel assemblies or bundles. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore,new fuel will no longer be received on the station. Consequently, the functions provided by the new fuel storage racks are no longer required. The New Fuel Storage Rack Power Generation Design Bases are no longer applicable. New Fuel Storage Rack information is obsolete.					
	10.2.3	Safety Design Basis	Delete Section 10.2.3, New Fuel Storage Racks, Safety Design Basis. The new fuel storage racks provided a safe, specially designed storage place for new fuel assemblies or bundles. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore,new fuel will no longer be received on the station. Consequently, the functions provided by the new fuel storage racks are no longer required. The New Fuel Storage Rack Safety Design Basis is no longer applicable. New Fuel Storage Rack information is obsolete.					
	10.2.4	Description	Delete Section 10.2.4, New Fuel Storage Racks, Description. The new fuel storage racks provided a safe, specially designed storage place for new fuel assemblies or bundles. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore,new fuel will no longer be received on the station. Consequently, the information provided by the new fuel storage racks description section is no longer applicable. New Fuel Storage Rack information is obsolete.					
	10.2.5	Safety Evaluation	Delete Section 10.2.5, New Fuel Storage Racks, Safety Evaluation. The new fuel storage racks provided a safe, specially designed storage place for new fuel assemblies or bundles. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, new fuel will no longer be received on the station. Consequently, a safety evaluation of the new fuel storage racks is no longer required. New Fuel Storage Rack information is obsolete.					
	10.2.6	Inspection and Testing	Delete Section 10.2.6, New Fuel Storage Racks, Inspection and Testing. The new fuel storage racks provided a safe, specially designed storage place for new fuel assemblies or bundles. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, new fuel will no longer be received on the station. Consequently, inspection and testing of the new fuel storage racks is no longer required. New Fuel Storage Rack information is obsolete.					
	Figure 10.2-1	New Fuel Storage Rack	Delete Figure 10.2-1, New Fuel Storage Rack. The new fuel storage racks provided a safe, specially designed storage place for new fuel assemblies or bundles. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore,new fuel will no longer be received on the station. Consequently, the information provided on figure 10.2-1 is no longer applicable. New Fuel Storage Rack information is obsolete.					
10.3	SPENT FUEL STORAGE					3.3.1.2	SPENT FUEL STORAGE	
	10.3.1	Power Generation Objective	Reword title to "Objective" delete "Power Generation" since power generation can no longer occur following certification of permanent defueling.				3.3.1.2.1	Objective
	10.3.2	Power Generation Design Basis	Reword title to "Design Basis", delete "Power Generation" since power generation can no longer occur following certification of permanent defueling. Include design bases from section 10.3.3 Safety Design Basis.				3.3.1.2.2	Design Basis
	10.3.3	Safety Design Basis	Move Safety Design Basis information into section 10.3.2, Design Basis, Delete this section.					
	10.3.4	Description	Numbering Changes only				3.3.1.2.3	Description
	10.3.5	Safety Evaluation	Deleted reference to an analysis which was performed to verify that the NFPCS, augmented by RHR as required, can maintain SFP temperature in required range under all operating conditions. NFPCS abandoned under EC 50085 following permanent defueling. FP will be cooled by SBFPCS. Numbering Changes as a consequence of combining design basis sections. Information added to document compliance with NUREG 1738 IDCs and SDAs				3.3.1.2.4	Safety Evaluation

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	10.3.6		Inspection and Testing	Numbering Changes as a consequence of combining design basis sections.	3.3.1.2.5	Inspection and Testing
				References section added to point to NUREG 1738, Reg Guide 1,174, and BVY 14-009, Compliance with IDCs and SDAs	3.3.1.5	References
	Figure 10.3-1		Fuel Storage - Arrangement	No Changes	Figure 3.3.1-1	Fuel Storage - Arrangement
	Figure 10.3-2		5920-6893 Plan - Fuel Storage Rack Arrangement	No Changes	Figure 3.3.1-2	5920-6893 Plan - Fuel Storage Rack Arrangement
	Figure 10.3-3		5920-6893 Spent Fuel Storage Rack Schematic	No Changes	Figure 3.3.1-3	5920-6893 Spent Fuel Storage Rack Schematic
	Figure 10.3-4A		NES Fuel Storage Rack Assembly (Partial)	No Changes	Figure 3.3.1-4	NES Fuel Storage Rack Assembly (Partial)
	Figure 10.3-4B		Holtec Fuel Storage Rack Assembly (Partial)	No Changes	Figure 3.3.1-5	Holtec Fuel Storage Rack Assembly (Partial)
10.4	TOOLS AND SERVICING EQUIPMENT				3.3.1.4	TOOLS AND SERVICING EQUIPMENT
	10.4.1		Power Generation Objective	Delete Section 10.4.1, Power Generation Objective. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, servicing of reactor systems will no longer be required. Any reactor related activities will occur as part of decommissioning activities as outlined in the PSDAR.		
	10.4.2		Safety Objective	Reword title to "Objective", delete "Safety". Numbering Changes as a result of deleting Power Generation Objective Section	3.3.1.4.1	Objective
	10.4.3		Safety Design Basis	Reword title to "Design Basis", delete "Safety". Changed "refueling accident" to "fuel handling accident". Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Consequently, a refueling accident can no longer occur. Numbering Changes as a result of deleting Power Generation Objective Section	3.3.1.4.2	Design Basis
	10.4.4		Description	Numbering Changes as a result of deleting Power Generation Objective Section	3.3.1.4.3	Description
		10.4.4.1	Introduction	Deleted references to reactor general servicing requirements. Servicing of reactor systems will no longer be required. Any reactor related activities will occur as part of decommissionin activities as outlined in the PSDAR. Deleted reference to flow chart outlining activities in a normal refueling outage, since post defueling, refueling outages will no longer occur. Numbering Changes as a result of deleting Power Generation Objective Section	3.3.1.4.3.1	Introduction
		10.4.4.2	Fuel Servicing Equipment	Editorial comments to improve grammar. Removed the reference to "operating" personnel. Removed references to the new fuel inspection stand functions, since new fuel will no longer be received at the station. Removed the description of typical channeling proceure for swapping an irradiated fuel channel to a new fuel bundle, since new fuel will no longer be received at the station. Additionally, channeling activites are procedurally covered and should not be described in the FSAR or DSAR. Numbering Changes as a result of deleting Power Generation Objective Section	3.3.1.4.3.2	Fuel Servicing Equipment
		10.4.4.3	Servicing Aids	Deleted reference to reasons for using a portable underwater television camera and monitor. This is detailed information not required to be in the DSAF Numbering Changes as a result of deleting Power Generation Objective Section	3.3.1.4.3.3	Servicing Aids
		10.4.4.4	Reactor Vessel Servicing Equipment	Delete section 10.4.4.4, Reactor Vessel Servicing Equipment. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, servicing of reactor systems will no longer be required. Any reactor related activities will occur as part of decommissioning activities as outlined in the PSDAR.		
		10.4.4.5	In-Vessel Servicing Equipment	Delete section 10.4.4.5, In-Vessel Servicing Equipment. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, servicing of reactor systems will no longer be required. Any reactor related activities will occur as part of decommissioning activities as outlined in the PSDAR.		
		10.4.4.6	Refueling Equipment	Re-titled "Refueling Equipment" to "Fuel Handling Equipment" since following certification of permanent defueling, refueling operations will no longe occu Reworded section to remove all reference to fuel handling in the reactor vessel and fuel transfer from the vessel to the pool. Numbering Changes as a result of deleting sections.	3.3.1.4.3.4	Fuel Handling Equipment
		10.4.4.7	Storage Equipment	Deleted reference to new fuel storage racks. Numbering Changes as a result of deleting sections.	3.3.1.4.3.5	Storage Equipment

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	10.4.4.8	Under Reactor Vessel Servicing Equipment	Delete section 10.4.4.8, Under Reactor Vessel Servicing Equipment. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, servicing under reactor vessel systems will no longer be required. Any reactor related activities will occur as part of decommissioning activities as outlined in the PSDAR.		
	10.4.4.9	Storage Pit	Delete section 10.4.4.9, Storage Pit. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the functions provided by the storage pit will no longer be required. Any reactor related activities will occur as part of decommissioning activities as outlined in the PSDAR. Numbering Changes as a result of deleting sections.		
10.4.5	Safety Evaluation		Changed "refueling accident" to "fuel handling accident". Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAV 10CFR50.82(a)(2). Consequently, a refueling accident can no longer occur. Numbering Changes as a result of deleting sections.	3.3.1.4.4	Evaluation
10.4.6	REFERENCES		Delete this section, single reference incorporated into text body		
Table 10.4.1	Tools and Servicing Equipment		Modify Table 10.4.1, Tools and Servicing Equipment. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, Reactor Vessel Servicing, in-vessel servicing, under vessel servicing and control rod drive hydraulic system servicing activities will no longer be conducted. Additionally, since all fuel has been moved to the spent fuel pool, the cattle chute will also no longer be required. Table 10.4-1 has been modified to reflect these changes. Any reactor related activities will occur as part of decommissioning activities as outlined in the PSDAR. Contents of the table pertinent to the permanently defueled condition have been moved into the body of the text, table has been deleted		
Figure 10.4-1	Typ. Plant Refueling Outage Flow Diagram		Delete Figure 10.4-1,Typ. Plant Refueling Outage Flow Diagram. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, plant refueling outages will no longer occur. The information provided in Figure 10.4-1 is not applicable. Plant refueling outage flow diagram information is obsolete. Any reactor related activities will occur as part of decommissioning activities as outlined in the PSDAR.		
Figure 10.4-2	5920-2429 Channel Gauging Fixture		Delete Figure 10.4-2, Channel Gauging Fixture. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, Fuel channeling operations and activities which require use of the channel gauging fixture will no longer be conducted. Information regarding the channel gauging fixture provided on figure 10.4-2 is not applicable and is obsolete.		
Figure 10.4-3	5920-2417 Incore Guide Tube		Delete Figure 10.4-3, Incore Guide Tube. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, Reactor Vessel Servicing, in-vessel servicing, under vessel servicing and control rod drive hydraulic system servicing activities will no longer be conducted. Information regarding incore guide tube tools provided on figure 10.4-3 is not applicable and is obsolete. Any reactor related activities will occur as part of decommissioning activities as outlined in the PSDAR.		
Figure 10.4-4	Incore Housing		Delete Figure 10.4-4, Incore Housing. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, Reactor Vessel Servicing, in-vessel servicing, under vessel servicing and control rod drive hydraulic system servicing activities will no longer be conducted. Information regarding incore housing tools provided on figure 10.4-4 is not applicable and is obsolete. Any reactor related activities will occur as part of decommissioning activities as outlined in the PSDAR.		
Figure 10.4-5	5920-2422 In-Core Guide Tube Seal		Delete Figure 10.4-5, In-Core Guide Tube Seal. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, Reactor Vessel Servicing, in-vessel servicing, under vessel servicing and control rod drive hydraulic system servicing activities will no longer be conducted. Information regarding Incore Guide Tube Seal tools provided on figure 10.4-5 is not applicable and is obsolete. Any reactor related activities will occur as part of decommissioning activities as outlined in the PSDAR.		
Figure 10.4-6	In-Core Flange Seal Test Plug		Delete Figure 10.4-6, In-Core Flange Seal Test Plug. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, Reactor Vessel Servicing, in-vessel servicing, under vessel servicing and control rod drive hydraulic system servicing activities will no longer be conducted. Information regarding In-Core Flange Seal Test Plug tools provided on figure 10.4-6 is not applicable and is obsolete. Any reactor related activities will occur as part of decommissioning activities as outlined in the PSDAR.		

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	Figure 10.4-7	5920-2419 Fuel Bail Cleaner	Delete Figure 10.4-7, Fuel Bail Cleaner. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, operations and activities which require use of the fuel bail cleaner will no longer be conducted. Information regarding the fuel bail cleaner provided on figure 10.4-7 is not applicable and is obsolete.					
	Figure 10.4-8	5920-2424 Thermal Sleeve Installation Tool	Delete Figure 10.4-8, Thermal Sleeve Installation Tool. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, Reactor Vessel Servicing, in-vessel servicing, under vessel servicing and control rod drive hydraulic system servicing activities will no longer be conducted. Information regarding Thermal Sleeve Installation Tools provided on figure 10.4-8 is not applicable and is obsolete. Any reactor related activities will occur as part of decommissioning activities as outlined in the PSDAR.					
	Figure 10.4-9	5920-2407 Thermal Sleeve	Delete Figure 10.4-9, Thermal Sleeve. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, Reactor Vessel Servicing, in-vessel servicing, under vessel servicing and control rod drive hydraulic system servicing activities will no longer be conducted. Information regarding the Thermal Sleeve provided on figure 10.4-9 is not applicable and is obsolete. An reactor related activities will occur as part of decommissioning activities as outlined in the PSDAR.					
10.5	FUEL POOL COOLING AND DEMINERALIZER SYSTEM					3.3.1.3	STANDBY FUEL POOL COOLING AND DEMINERALIZER SYSTEM	
	10.5.1	Power Generation Objective	Deleted "Power Generation" from the section title. Since the classification process described in UFSAR section 1.4.1 is no longer used, the classification basis is no longer required and is no applicable to a permanently defueled station. Section wording has been modified to reflect alignment changes in the fuel pool cooling system; the Normal Fuel Pool Cooling system has been abandoned, all fuel pool cooling is performed by the Standby Fuel Pool Cooling system. Demineralizers which are part of the installed FPC system have been abandoned, pool clarity is maintained by filtration and demineralizer units installed in the pool.				3.3.1.3.1	Objective
	10.5.2	Safety Objective	Delete section 10.5.2, Safety Objective. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the need to maintain the Reactor Building environment withi the bounding limits of the environmental qualification program for electrical equipment no longer exists. Therefore, the safety objective is no longer applicable.					
	10.5.3	Power Generation Design Basis	Deleted "Power Generation" from the section title. Since the classification process described in UFSAR section 1.4.1 is no longer used, the classification basis is no longer required and is no applicable to a permanently defueled station. Section wording has been modified to reflect alignment changes in the fuel pool cooling system; the Normal Fuel Pool Cooling system has been abandoned, all fuel pool cooling is performed by the Standby Fuel Pool Cooling system. Demineralizers which are part of the installed FPC system have been abandoned, pool clarity is maintained by filtration and demineralizer units installed in the pool.				3.3.1.3.2	Design Basis
	10.5.4	Safety Design Basis	Delete section 10.5.4, Safety Design Basis. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the need to maintain the Reactor Building environment withi the bounding limits of the environmental qualification program for electrical equipment no longer exists. Therefore, the safety design basis is no longer applicable.					
	10.5.5	Description	Section 10.5.5, Description, wording has been modified to reflect alignment changes in the fuel pool cooling system; the Normal Fuel Pool Cooling system has been abandoned, all fuel pool cooling is performed by the Standby Fuel Pool Cooling system. The SFPCS has been reclassified to non-safety related, but remains seismic class 1. Demineralizers which are part of the installed FPC system have been abandoned, pool clarity is maintained by filtration and demineralizer units installed in the pool.				3.3.1.3.3	Description
	10.5.6	Safety Evaluation	Modify Section 10.5.6 to remove the term "safety" from the section title. Since the classification process described in UFSAR section 1.4.1 is no longer used, the classification basis is no longe required and is not applicable to a permanently defueled station. Wording was added to the section to reflect that the SFP heat load has decayed to within the capacity of one train of SFPCS 40 days after shutdown with the core discharged into the pool and all fuel remaining in the pool which was present just prior to shutdown.				3.3.1.3.4	Evaluation
	10.5.7	Inspection and Testing	Modify Section 10.5.7, Inspection and Testing. There are no SFPCS testing requirements in the Technical Specifications. Additionally, at least one train of the SFPCS will be in operation at al times. Therefore, no special testing or inspection is required.				3.3.1.3.5	Inspection and Testing
	Table 10.5.1	Heat Removal Capacities	Deleted Table 10.5.1, Heat Removal Capacities. The following changes have occurred in the Fuel Pool Cooling alignment: the Normal Fuel Pool Cooling system has been permanently removed from service, all fuel pool cooling is performed by the Standby Fuel Pool Cooling system. Demineralizers which are part of the installed FPC system have been permanently removed from service, pool clarity is maintained by filtration and demineralizer units installed in the pool. Consequently, the information regarding heat removal capacities provided by table 10.5.1 is no longer applicable.					
	Table 10.5.2	Deleted	Previously Deleted					

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	Table 10.5.3	Comparison of Heat Loads to Heat Removal Capacities with SFP at Capacity	Deleted Table 10.5.3,Comparison of Heat Loads to Heat Removal Capacities with SFP at Capacity. The following changes have occurred in the Fuel Pool Cooling alignment: the Normal Fuel Pool Cooling system has been permanently removed from service, all fuel pool cooling is performed by the Standby Fuel Pool Cooling system. Demineralizers which are part of the installed FPC system have been permanently removed from service, pool clarity is maintained by filtration and demineralizer units installed in the pool. Consequently, the information regarding heat removal capacities provided by table 10.5.3 is no longer applicable.				
	Table 10.5.4	Fuel Pool Cooling and Demineralizer System - System Specifications	Deleted Table 10.5.4, Fuel Pool Cooling and Demineralizer System - System Specifications. The following changes have occurred in the Fuel Pool Cooling alignment: the Normal Fuel Pool Cooling system has been permanently removed from service, all fuel pool cooling is performed by the Standby Fuel Pool Cooling system. Demineralizers which are part of the installed FPC system have been permanently removed from service, pool clarity is maintained by filtration and demineralizer units installed in the pool. Consequently, the information regarding system specifications provided by table 10.5.4 is no longer applicable.				
	Figure 10.5-1a	G-191173, Sh1 Fuel Pool Cooling System	No Changes		Figure 3.3.1-6	G-191173, Sh1 Fuel Pool Cooling System	
	Figure 10.5-1b	G-191173, Sh2 Fuel Pool Cooling System	No Changes		Figure 3.3.1-7	G-191173, Sh2 Fuel Pool Cooling System	
	Figure 10.5-2	G-191177, Sh4 Fuel Pool Filter Demineralizer System	Deleted Figure 10.5-2, Fuel Pool Filter Demineralizer System. The following changes have occurred in the Fuel Pool Cooling alignment: the Normal Fuel Pool Cooling system has been permanently removed from service, all fuel pool cooling is performed by the Standby Fuel Pool Cooling system. Demineralizers which are part of the installed FPC system have been permanently removed from service, pool clarity is maintained by filtration and demineralizer units installed in the pool. Consequently, the information regarding system specifications provided by Figure 10.5-2 is no longer applicable.				
10.6	STATION SERVICE WATER SYSTEM				3.3.2	SERVICE WATER SYSTEM	
	10.6.1	Power Generation Objective	Modify Section 10.6.1, Station Service Water System, Power Generation Objective to remove reference to operating and accident conditions, and reflect that the Station Service Water System now supplies only cooling necessary to ensure the safe storage and handling of irradiated fuel.			3.3.2.1	Objective
	10.6.2	Safety Objective	Delete Section 10.6.2, Safety Objective. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the the Station SWS is no longer required to provide cooling water to systems and equipment required to operate under accident conditions.				
	10.6.3	Power Generation Design Basis	Modify Section 10.6.3, Station Service Water System, Power Generation Design Basis to remove reference to operating and accident conditions, and reflect that the Station Service Water System now supplies only cooling necessary to ensure the safe storage and handling of irradiated fuel.			3.3.2.2	Design Basis
	10.6.4	Safety Design Basis	Delete Section 10.6.2, Safety Design Bases. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the the Station SWS is no longer required to provide cooling water to systems and equipment required to operate under accident conditions. The Station SWS Safety Design Bases are no longer applicable.				
	10.6.5	Description	Modify Section 10.6.5, Station Service Water System, Description to remove reference to operating and accident conditions, and reflect that the Station Service Water System now supplies only cooling necessary to ensure the safe storage and handling of irradiated fuel, remove reference to certain automatic features no longer applicable, reflect that Station SWS discharge is aligned to the cooling tower deep basin, and note that Station SWS piping which supplies spent fuel pool cooling is seismic class 1.			3.3.2.3	Description
	10.6.6	Safety Evaluation	Modify Section 10.6.6, Safety Evaluation to delete the term safety from the section title and reword the section to demonstrate that the objectives appropriate to the defueled condtion are met, Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the Station SWS is no longer required to provide cooling water to systems and equipment required to operate under accident conditions. A Station SWS Safety Evaluation is no longer required.			3.3.2.4	Evaluation
	10.6.7	Inspection and Testing	Modify Section 10.6.7, Station Service Water System, Inspection and Testing, to remove reference to operating and accident conditions, and reflect that since the Station Service Water System is in constant operation, special testing and inspections are not required.			3.3.2.5	Inspection and Testing
	Figure 10.6-1a	G-191159, Sh1 Flow Diagram Service Water (Part 1)	No Change		Figure 3.3.2-1	G-191159, Sh1 Flow Diagram Service Water (Part 1)	
	Figure 10.6-1b	G-191159, Sh2 Flow Diagram Service Water (Part 2)	No Change		Figure 3.3.2-2	G-191159, Sh2 Flow Diagram Service Water (Part 2)	
10.7	RESIDUAL HEAT REMOVAL (RHR) SERVICE WATER SYSTEM		The RHR Service Water (RHRSW) System provided a dynamic heat sink for the Residual Heat Removal (RHR) System during normal operation.				
	10.7.1	Power Generation Objective	Delete section 10.7.1, Residual Heat Removal Service Water System, Power Generation Objectives. The RHR Service Water (RHRSW) System provided a dynamic heat sink for the Residual Heat Removal (RHR) System during normal operation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since removal of decay and residual heat is no longer required, operation of the RHR system and RHR SW systems is no longer required. Therefore, the RHR SW power generation objectives are no longer applicable. Residual Heat Removal Service Water System information is obsolete.				

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	10.7.2	Safety Objective	Delete section 10.7.2, Residual Heat Removal Service Water System, Safety Objective. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since restoration and maintenance of RPV water level following a LOCA, Suppression Pool Cooling following a LOCA, and Containment Cooling following a LOCA are no longer required the functions of RHR SW are also no longer required. Therefore, the RHR SW safety objective is no longer applicable. Residual Heat Removal SW System information is obsolete.	
	10.7.3	Power Generation Design Bases	Delete section10.7.3 Residual Heat Removal Service Water System, Power Generation Design Bases The RHR Service Water (RHRSW) System provided a dynamic heat sink for the Residual Heat Removal (RHR) System during normal operation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since removal of decay and residual heat is no longer required, operation of the RHR system and RHR SW systems is no longer required. Therefore, the RHR SW power generation design bases are no longer applicable. Residual Heat Removal Service Water System information is obsolete.	
	10.7.4	Safety Design Bases	Delete section 10.7.4, Residual Heat Removal Service Water System, Safety Design Bases. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since restoration and maintenance of RPV water level following a LOCA, Suppression Pool Cooling following a LOCA, and Containment Cooling following a LOCA are no longer required, the functions of RHR SW are also no longer required. Therefore, the RHR SW safety objective is no longer applicable. Residual Heat Removal SW System information is obsolete.	
	10.7.5	Description	Delete section 10.7.5, Residual Heat Removal Service Water System, Description. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since removal of decay and residual heat, restoration and maintenance of RPV water level following a LOCA, Suppression Pool Cooling following a LOCA, and Containment Cooling following a LOCA are no longer required, the functions of RHR SW are also no longer required. Therefore, the information provided in the RHR SW description section is no longer applicable. Residual Heat Removal SW System information is obsolete.	
	10.7.6	Safety Evaluation	Delete section 10.7.6, Residual Heat Removal Service Water System, Safety Evaluation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since removal of decay and residual heat, restoration and maintenance of RPV water level following a LOCA, Suppression Pool Cooling following a LOCA, and Containment Cooling following a LOCA are no longer required, the functions of RHR SW are also no longer required. Therefore, a RHR SW System safety evaluation is no longer required. Residual Heat Removal SW System information is obsolete.	
	10.7.7	Inspection and Testing	Delete section 10.7.7, Residual Heat Removal Service Water System, Inspection and Testing. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since removal of decay and residual heat, restoration and maintenance of RPV water level following a LOCA, Suppression Pool Cooling following a LOCA, and Containment Cooling following a LOCA are no longer required, the functions of RHR SW are also no longer required. Therefore, Inspection and Testing of the RHR SW system is no longer required. Residual Heat Removal SW System information is obsolete.	
10.8	ALTERNATE COOLING SYSTEM		The Alternate Cooling System (ACS) provided an alternate means of heat removal in the event that the service water pumps become inoperable.	
	10.8.1	Safety Objective	Delete Section 10.8.1, Alternate Cooling System Safety Objective. The Alternate Cooling System (ACS) provided an alternate means of heat removal in the event that the service water pumps become inoperable. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, decay and sensible heat removal from the primary system is no longer required. A backup system for heat removal from the Fuel Pool is no longer required, other strategies are in place to maintain cooling and inventory in the spent fuel pool. Consequently, the Alternate Cooling System safety objective is no longer applicable. Alternate Cooling System information is obsolete.	
	10.8.2	Safety Design Bases	Delete Section 10.8.2, Alternate Cooling System Safety Design Bases. The Alternate Cooling System (ACS) provided an alternate means of heat removal in the event that the service water pumps become inoperable. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, decay and sensible heat removal from the primary system is no longer required. A backup system for heat removal from the Fuel Pool is no longer required, other strategies are in place to maintain cooling and inventory in the spent fuel pool. Consequently, the Alternate Cooling System safety design bases are no longer applicable. Alternate Cooling System information is obsolete.	
	10.8.3	Description	Delete Section 10.8.3, Alternate Cooling System Description. The Alternate Cooling System (ACS) provided an alternate means of heat removal in the event that the service water pumps become inoperable. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, decay and sensible heat removal from the primary system is no longer required. A backup system for heat removal from the Fuel Pool is no longer required, other strategies are in place to maintain cooling and inventory in the spent fuel pool. Consequently, the information provided in the Alternate Cooling System description is no longer applicable. Alternate Cooling System information is obsolete.	

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	10.8.4	Safety Evaluation	Delete Section 10.8.4, Alternate Cooling System Safety Evaluation. The Alternate Cooling System (ACS) provided an alternate means of heat removal in the event that the service water pumps become inoperable. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, decay and sensible heat removal from the primary system is no longer required. A backup system for heat removal from the Fuel Pool is no longer required, other strategies are in place to maintain cooling and inventory in the spent fuel pool. Consequently, a safety evaluation of the Alternate Cooling System description is no longer required. Alternate Cooling System information is obsolete.	
	10.8.5	Inspection and Testing	Delete Section 10.8.5, Alternate Cooling System Inspection and Testing. The Alternate Cooling System (ACS) provided an alternate means of heat removal in the event that the service water pumps become inoperable. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, decay and sensible heat removal from the primary system is no longer required. A backup system for heat removal from the Fuel Pool is no longer required, other strategies are in place to maintain cooling and inventory in the spent fuel pool. Consequently, inspection and testing of the Alternate Cooling System is no longer required. Alternate Cooling System information is obsolete.	
	10.8.6	References	Delete Section 10.8.6, References. The Alternate Cooling System (ACS) provided an alternate means of heat removal in the event that the service water pumps become inoperable. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, decay and sensible heat removal from the primary system is no longer required. A backup system for heat removal from the Fuel Pool is no longer required, other strategies are in place to maintain cooling and inventory in the spent fuel pool. Consequently, Calculation VYC-1279J, "SW Flow Analysis-Deep Basin Back Flow Due to Seismic Line Breaks in Turbine Building & Minimum EDG Flowrates After Common Discharge Header Break" is no longer applicable. Alternate Cooling System information is obsolete.	
	Table 10.8.1	RHRSW Pump Operating Modes for the Alternate Cooling System	Delete Table 10.8.1, RHRSW Pump Operating Modes for the Alternate Cooling System. The Alternate Cooling System (ACS) provided an alternate means of heat removal in the event that the service water pumps become inoperable. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, decay and sensible heat removal from the primary system is no longer required. A backup system for heat removal from the Fuel Pool is no longer required, other strategies are in place to maintain cooling and inventory in the spent fuel pool. Consequently, Table 10.8.1 is no longer applicable. Alternate Cooling System information is obsolete.	
	Figure 10.8-1	Alternate Cooling System	Delete Figure 10.8.1, Alternate Cooling System. The Alternate Cooling System (ACS) provided an alternate means of heat removal in the event that the service water pumps become inoperable. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, decay and sensible heat removal from the primary system is no longer required. A backup system for heat removal from the Fuel Pool is no longer required, other strategies are in place to maintain cooling and inventory in the spent fuel pool. Consequently, the information provided on Figure 10.8-1 concerning the Alternate Cooling System is no longer applicable. Alternate Cooling System information is obsolete.	
10.9	REACTOR BUILDING CLOSED COOLING WATER SYSTEM		The Reactor Building Closed Cooling Water (RBCCW) System provided inhibited demineralized water to cool reactor auxiliary equipment.	
	10.9.1	Safety Objective	Delete Section 10.9.1, RBCCW Safety Objective. The Reactor Building Closed Cooling Water (RBCCW) System provided inhibited demineralized water to cool reactor auxiliary equipment. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Reactor auxiliary systems operation will no longer be required. Consequently, the functions provided by the RBCCW system are no longer required. The RBCCW Safety Objective is no longer applicable. RBCCW information is obsolete.	
	10.0.2	Power Generation Objective	Delete Section 10.9.2, RBCCW Power Generation Objective. The Reactor Building Closed Cooling Water (RBCCW) System provided inhibited demineralized water to cool reactor auxiliary equipment. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Reactor auxiliary systems operation will no longer be required. Consequently, the functions provided by the RBCCW system are no longer required. The RBCCW Power Generation Objective is no longer applicable. RBCCW information is obsolete.	
	10.9.3	Safety Design Bases	Delete Section 10.9.3, RBCCW Safety Design Bases. The Reactor Building Closed Cooling Water (RBCCW) System provided inhibited demineralized water to cool reactor auxiliary equipment. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Reactor auxiliary systems operation will no longer be required. Consequently, the functions provided by the RBCCW system are no longer required. The RBCCW Safety Design Bases are no longer applicable. RBCCW information is obsolete.	

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	10.9.4	Power Generation Design Bases	Delete Section 10.9.4, RBCCW Power Generation Design Bases. The Reactor Building Closed Cooling Water (RBCCW) System provided inhibited demineralized water to cool reactor auxiliary equipment. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Reactor auxiliary systems operation will no longer be required. Consequently, the functions provided by the RBCCW system are no longer required. The RBCCW Power Generation Design Bases are no longer applicable. RBCCW information is obsolete.	
	10.9.5	Description	Delete Section 10.9.5, RBCCW Description. The Reactor Building Closed Cooling Water (RBCCW) System provided inhibited demineralized water to cool reactor auxiliary equipment. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Reactor auxiliary systems operation will no longer be required. Consequently, the functions provided by the RBCCW system are no longer required. The information provided in the RBCCW Description Section is no longer applicable. RBCCW information is obsolete.	
	10.9.6	Safety Evaluation	Delete Section 10.9.6, RBCCW Safety Evaluation. The Reactor Building Closed Cooling Water (RBCCW) System provided inhibited demineralized water to cool reactor auxiliary equipment. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Reactor auxiliary systems operation will no longer be required. Consequently, the functions provided by the RBCCW system are no longer required. An RBCCW Safety Evaluation is no longer required. RBCCW information is obsolete.	
	10.9.7	Inspection and Testing	Delete Section 10.9.7, RBCCW Inspection and Testing. The Reactor Building Closed Cooling Water (RBCCW) System provided inhibited demineralized water to cool reactor auxiliary equipment. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Reactor auxiliary systems operation will no longer be required. Consequently, the functions provided by the RBCCW system are no longer required. RBCCW Inspection and Testing is no longer required. RBCCW information is obsolete.	
	Figure 10.9-1	G-191159, Sh3 RCW Cooling Water System	Delete Figure 10.9-1, RCW Cooling Water System. The Reactor Building Closed Cooling Water (RBCCW) System provided inhibited demineralized water to cool reactor auxiliary equipment. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Reactor auxiliary systems operation will no longer be required. Consequently, the functions provided by the RBCCW system are no longer required. The information provided regarding the RBCCW Cooling Water System on figure 10.9-1 is no longer applicable. RBCCW information is obsolete.	
10.10	TURBINE BUILDING CLOSED COOLING WATER SYSTEM		The Turbine Building Closed Cooling Water (TBCCW) System provided inhibited demineralized cooling water to auxiliary equipment in the Turbine Building requiring cooling water.	
	10.10.1	Power Generation Objective	Delete Section 10.10.1, TBCCW Power Generation Objective. The Turbine Building Closed Cooling Water (TBCCW) System provided inhibited demineralized cooling water to auxiliary equipment in the Turbine Building requiring cooling water. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Auxiliary systems operation in the turbine building will no longer be required. Consequently, the functions provided by the TBCCW system are no longer required. The TBCCW Power Generation Objective is no longer applicable. TBCCW information is obsolete.	
	10.10.2	Power Generation Design Bases	Delete Section 10.10.2, TBCCW Power Generation Design Bases The Turbine Building Closed Cooling Water (TBCCW) System provided inhibited demineralized cooling water to auxiliary equipment in the Turbine Building requiring cooling water. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Auxiliary systems operation in the turbine building will no longer be required. Consequently, the functions provided by the TBCCW system are no longer required. The TBCCW Power Generation Design Bases are no longer applicable. TBCCW information is obsolete.	
	10.10.3	Description	Delete Section 10.10.3, TBCCW Description. The Turbine Building Closed Cooling Water (TBCCW) System provided inhibited demineralized cooling water to auxiliary equipment in the Turbine Building requiring cooling water. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Auxiliary systems operation in the turbine building will no longer be required. Consequently, the functions provided by the TBCCW system are no longer required. The information provided in the TBCCW Description Section is no longer applicable. TBCCW information is obsolete.	

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	10.10.4	Testing and Inspection	Delete Section 10.10.4, TBCCW Inspection and Testing. The Turbine Building Closed Cooling Water (TBCCW) System provided inhibited demineralized cooling water to auxiliary equipment in the Turbine Building requiring cooling water. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Auxiliary systems operation in the turbine building will no longer be required. Consequently, the functions provided by the TBCCW system are no longer required. Testing and Inspection of the TBCCW System is no longer required. TBCCW information is obsolete.				
	Figure 10.10-1	G-191159, Sh 4 Turbine Building Closed Cooling Water System	Delete Figure 10.10-1,Turbine Building Closed Cooling Water System. The Turbine Building Closed Cooling Water (TBCCW) System provided inhibited demineralized cooling water to auxiliary equipment in the Turbine Building requiring cooling water. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Auxiliary systems operation in the turbine building will no longer be required. Consequently, the functions provided by the TBCCW system are no longer required. The information provided on Figure 10.10-1 is no longer applicable. TBCCW information is obsolete.				
10.11	Fire Protection System				3.3.4	FIRE PROTECTION SYSTEM	
	10.11.1	Power Generation Objective	Deleted "Power Generation" from the section title. Since the classification process described in UFSAR section 1.4.1 is no longer used, the classification basis is no longer required and is no applicable to a permanently defueled station.			3.3.4.1	Objective
	10.11.2	Power Generation Design Basis	Deleted "Power Generation" from the section title. Since the classification process described in UFSAR section 1.4.1 is no longer used, the classification basis is no longer required and is no applicable to a permanently defueled station.			3.3.4.2	Design Basis
	10.11.3	Description	Deleted reference to the Appendix "R" Program and the safe shutdown capability analysis since the program no longer applies post certification of permanent defueling. Added the words "The fire protection program for the permanently defueled state has been developed based on the applicable requirements of 10CFR50.48 and BTP APCSB 9.5 1, Appendix A. The Fire Hazards Analysis (FHA) documents existing plant configurations and defines the resources available for the prevention and limitation of damage from fire (Reference 1). In addition to plans and physical configurations for fire protection, fire detection, fire suppression and limitation of fire damage, the FHA also provides an overall description of the fire protection program." to reflect the facility state post-defueling. Removed the words "The following documents form the basis of the Fire Protection Program and are incorporated by reference into the DSAR: • Fire Protection Program • Fire Hazard Analysis to remove the IBR since it is no longer required.			3.3.4.3	Description
	10.11.4	Inspection and Testing	No changes			3.3.4.4	Inspection and Testing
	10.11.5	References	Delete section 10.11.5, References. Only reference in this section, reference to the Safe Shutdown Capability Analysis, was deleted since the analysis no longer applies post certification of permanent defueling. Re-added VY Fire Hazards Analysis as a reference.			3.3.4.5	References
	Figure 10.11-1A	G-191163, Sh1 Fire Protection System	No Changes	Figure 3.3.4-1	G-191163, Sh1 Fire Protection System		
	Figure 10.11-1b	G-191163, Sh2 Fire Protection System (Outer Loop)	No Changes	Figure 3.3.4-2	G-191163, Sh2 Fire Protection System (Outer Loop)		
10.12	STATION HEATING, VENTILATING, AND AIR CONDITIONING SYSTEMS				3.3.5	Heating, Ventilating and Air Conditioning Systems	
	10.12.1	Power Generation Objective	Modify Section 10.12.1 to eliminate reference to the term "Station" and replace with "facility" where appropriate Delete the term "Power Generation" from the section title. Since the term "Power Generation" is based in the classification process described in section 1.4.1 which is no longer used, the term is no longer required and is not applicable to a permanently defueled station.			3.3.5.1	Objective
	10.12.2	Power Generation Design Basis	Modify Section 10.12.2 to eliminate reference to purging the primary containment. Delete the term "Power Generation" from the section title. Since the term "Power Generation" is based in the classification process described in section 1.4.1 which is no longer used, the term is no longer required and is not applicable to a permanently defueled station. Remove reference to purging primary containment. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Primary Containment System are no longer required and the safety design bases are no longer applicable. Primary Containment System information is obsolete.			3.3.5.2	Design Basis
	10.12.3	Description				3.3.5.3	Description
		10.12.3.1 Reactor Building	RB HVAC information moved into this section from Section 5.3				3.3.5.3.1 Reactor Building

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		10.12.3.2	Turbine Building	Modify Section 10.12.3.2 to delete reference to the modular facility located on the turbine deck with a ventilation system independent of the turbine building ventilation system. The facility was a staging area for personnel engaged in main turbine and generator work during refueling outages. Since refueling outages, including turbine and generator overhauls, will no longer be conducted, the functions provided by the modular facility are no longer required. Therefore these functions are obsolete and may be deleted.					3.3.5.3.2	Turbine Building
		10.12.3.3	Main Control Room	Modify Section 10.12.3.3 to delete reference to manual operation of control room ventilation dampers during a LOCA, including reference to manual operation of dampers within 3 hours after the event. Following certification of permanent defueling, the licensee is no longer authorized to emplace or retain fuel in the reactor vessel. Therefore, power operations can no longer occur and a LOCA is not possible. Additionally, the three hour requirement originated in an Appendix R scenario. The 3 hour limit intended to ensure control room temperature remained below 120F to prevent challenging equipment operating limits. Post defueling, Appendix R is no longer applicable.					3.3.5.3.3	Main Control Room
		10.12.3.4	Service Building	No Changes					3.3.5.3.4	Service Building
		10.12.3.5	Radwaste Building	No Changes					3.3.5.3.5	Radwaste Building
		10.12.3.6	Station Heating Boiler System	No Changes					3.3.5.3.6	Heating Boiler System
		10.12.3.7	Advanced Off-Gas Building HVAC	No Changes					3.3.5.3.7	Advanced Off-Gas Building HVAC
	10.12.4	Inspection and Testing		No Changes				3.3.5.4	Inspection and Testing	
	Figure 10.12-1	G-191237, Sh1 HVAC - Flow Diagram Turbine, Service, and Control Room Buildings		No Changes		Figure 3.3.5-1	G-191237, Sh1 HVAC - Flow Diagram Turbine, Service, and Control Room Buildings			
	Figure 10.12-2	G-191237, Sh2 HVAC - Flow Diagram Turbine, Service, and Control Room Buildings		No Changes		Figure 3.3.5-2	G-191237, Sh2 HVAC - Flow Diagram Turbine, Service, and Control Room Buildings			
	Figure 10.12-3	G-191236 HVAC - Flow Diagram Radwaste Building		No Changes		Figure 3.3.5-3	G-191236 HVAC - Flow Diagram Radwaste Building			
	Figure 10.12-4	G-191254 Station Heating Boiler System		No Changes		Figure 3.3.5-4	G-191254 Station Heating Boiler System			
				This figure moved in from section 5.3, Secondary Containment		Figure 3.3.5-5	Reactor Building Heating Ventilation and Air- Conditioning G-191238			
10.13	STATION MAKEUP WATER SYSTEM						3.3.8.1	MAKEUP WATER SYSTEM		
	10.13.1	Power Generation Objective		Modify Section 10.13.1, Station Makeup Water System, Power Generation Objective, to remove reference to operating and accident conditions and reflect that the Station Makeup Wate System now supplies only makeup water necessary to ensure the safe storage and handling of irradiated fuel. Delete the term "Power Generation" from the section title. Since the term "Power Generation" is based in the classification process described in section 1.4.1 which is no longer used, the term is no longer required and is not applicable to a permanently defueled station.				3.3.8.1.1	Objective	
	10.13.2	Power Generation Design Basis		Modify Section 10.13.2, Station Makeup Water System, Power Generation Design Basis, to remove reference to operating and accident conditions and reflect that the Station Makeup Wate System now supplies only makeup water necessary to ensure the safe storage and handling of irradiated fuel. Delete the term "Power Generation" from the section title. Since the term "Power Generation" is based in the classification process described in section 1.4.1 which is no longer used, the term is no longer required and is not applicable to a permanently defueled station.				3.3.8.1.2	Design Basis	
	10.13.3	Description		Modified Section 10.13.3, Description, to remove reference to "potable" water to accommodate changes due to EC 45 613, tracked by FCR 27-003				3.3.8.1.3	Description	
		10.13.3.1	General	Editorial Changes only.					3.3.8.1.3.1	General
		10.13.3.2	Pretreatment System	No changes					3.3.8.1.3.2	Pretreatment System
		10.13.3.3	Demineralizer System	No changes					3.3.8.1.3.3	Demineralizer System
	10.13.4	Inspection and Testing		No changes				3.3.8.1.4	Inspection and Testing	
	Figure 10.13-1	G-191161 Make-Up Water Treatment System		No changes		Figure 3.3.8-1	G-191161 Make-Up Water Treatment System			
10.14	STATION INSTRUMENT AND SERVICE AIR SYSTEMS					3.3.6	INSTRUMENT AND SERVICE AIR SYSTEMS			
	10.14.1	Power Generation Objective		Modify Section 10.14.1, Station Instrument and Service Air Systems, Power Generation Objective, to remove reference to operating and accident conditions and reflect that the Statioı Instrument and Service Air Systems now supply only pressurized air necessary to ensure the safe storage and handling of irradiated fuel. Delete the term "Power Generation" from the section title. Since the term "Power Generation" is based in the classification process described in section 1.4.1 which is no longer used, the term is no longer required and is not applicable to a permanently defueled station.			3.3.6.1	Objective		

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	10.14.2	Power Generation Design Basis		Modify Section 10.14.2, Station Instrument and Service Air Systems, Power Generation Design Basis to remove reference to operating and accident conditions and reflect that the Station Instrument and Service Air Systems now supply only pressurized air necessary to ensure the safe storage and handling of irradiated fuel. Delete the term "Power Generation" from the section title. Since the term "Power Generation" is based in the classification process described in section 1.4.1 which is no longer used, the term is no longer required and is not applicable to a permanently defueled station.			3.3.6.2	Design Basis	
	10.14.3	Description		Modify Section 10.14.3, Station Instrument and Service Air Systems, Description to remove reference to operating and accident conditions, the Compressed nitrogen system and reflect that the Station Instrument and Service Air Systems now supply only pressurized air necessary to ensure the safe storage and handling of irradiated fuel. The Compressed nitrogen System supplied compressed nitrogen for pneumatic instruments and controls located in the primary containment when containment was inerted. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Since primary containment is no longer required and containment inerting will no longer occur, compressed nitrogen for pneumatics in the drywell is no longer required.			3.3.6.3	Description	
	10.14.4	Inspection and Testing		Editorial changes only			3.3.6.4	Inspection and Testing	
	Figure 10.14-1	G-191160, Sh1 Flow Diagram Instrument Air System		No changes		Figure 3.3.6-1	G-191160, Sh1 Flow Diagram Instrument Air System		
	Figure 10.14-2	G-191160, Sh2 Service and Instrument Air System (Reactor		No changes		Figure 3.3.6-2	G-191160, Sh2 Service and Instrument Air System (Reactor		
	Figure 10.14-3	G-191160, Sh3 Instrument Air System Flow Diagram		No changes		Figure 3.3.6-3	G-191160, Sh3 Instrument Air System Flow Diagram		
	Figure 10.14-4	G-191160, Sh4 Instrument Air System Flow Diagram		No changes		Figure 3.3.6-4	G-191160, Sh4 Instrument Air System Flow Diagram		
	Figure 10.14-5	G-191160, Sh5 Flow Diagram Service Air System		No changes		Figure 3.3.6-5	G-191160, Sh5 Flow Diagram Service Air System		
	Figure 10.14-6	G-191160, Sh6 Flow Diagram Service Air System		No changes		Figure 3.3.6-6	G-191160, Sh6 Flow Diagram Service Air System		
	Figure 10.14-7	G-191160, Sh7 Flow Diagram Diesel Generator Starting Air System		No changes		Figure 3.3.6-7	G-191160, Sh7 Flow Diagram Diesel Generator Starting Air System		
	Figure 10.14-8	G-191160, Sh8 Flow Diagram Service Air System		No changes		Figure 3.3.6-8	G-191160, Sh8 Flow Diagram Service Air System		
10.15	STATION POTABLE AND SANITARY WATER SYSTEM						3.3.8.2	POTABLE AND SANITARY WATER SYSTEM	
	10.15.1	Power Generation Objective		Modify Section 10.15.1, Station Potable and Sanitary Water System, Power Generation Objective, to remove reference to operating and accident conditions Delete the term "Power Generation" from the section title. Since the term "Power Generation" is based in the classification process described in section 1.4.1 which is no longer used, the term is no longer required and is not applicable to a permanently defueled station.				3.3.8.2.1	Objective
	10.15.2	Power Generation Design Basis		Modify Section 10.15.2, Station Potable and Sanitary Water System, Power Generation Design Basis to remove reference to operating and accident conditions Delete the term "Power Generation" from the section title. Since the term "Power Generation" is based in the classification process described in section 1.4.1 which is no longer used, the term is no longer required and is not applicable to a permanently defueled station.				3.3.8.2.2	Design Basis
	10.15.3	Description		Modified Section 10.15.3, Description, to remove reference to "potable" water and change number of wells from four to two and remove overly detailed information, to accommodate change due to EC 45 613, tracked by FCR 27-003.				3.3.8.2.3	Description
10.16	STATION EQUIPMENT AND FLOOR DRAINAGE SYSTEMS					4.5.1	EQUIPMENT AND FLOOR DRAINAGE SYSTEMS		
	10.16.1	Power Generation Objective		Modify Section 10.16.1, Station Equipment and Floor Drainage Systems, Power Generation Objective, to remove reference to operating and accident conditions and reflect that the primary station function is now the safe storage and handling of irradiated fuel.			4.5.1.1	Objective	
	10.16.2	Power Generation Design Basis		Modify Section 10.16.2, Station Equipment and Floor Drainage Systems, Power Generation Design Basis to remove reference to operating and accident conditions and reflect that the primary station function is now the safe storage and handling of irradiated fuel.			4.5.1.2	Design Basis	
	10.16.3	Description					4.5.1.3	Description	
		10.16.3.1	General	No Changes				4.5.1.3.1	General
		10.16.3.2	Radioactive Equipment Drainage Systems	Modify Section 10.16.3.2, Radioactive Equipment Drainage Systems, to remove reference to operating and accident conditions and reflect that the primary station function is now the safe storage and handling of irradiated fuel.				4.5.1.3.2	Radioactive Equipment Drainage Systems
		10.16.3.3	Radioactive Floor Drainage Systems	Modify Section 10.16.3.3, Radioactive Floor Drainage Systems, to remove reference to operating and accident conditions and reflect that the primary station function is now the safe storage and handling of irradiated fuel.				4.5.1.3.3	Radioactive Floor Drainage Systems
		10.16.3.4	Radioactive Liquid Chemical Drainage Systems	Modify Section 10.16.3.4, Radioactive Liquid Chemical Drainage Systems, to remove reference to operating and accident conditions and reflect that the primary station function is now the safe storage and handling of irradiated fuel.				4.5.1.3.4	Radioactive Liquid Chemical Drainage Systems

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10.16.3.5	Oil Drainage Systems	Modify Section 10.16.3.5, Oil Drainage Systems, to remove reference to operating and accident conditions and reflect that the primary station function is now the safe storage and handling o irradiated fuel. Editorial modification added to remove the potential impression that an oil-water mixture may be drained to the environment.	4.5.1.3.5	Oil Drainage Systems
10.16.3.6	Nonradioactive Water Drainage Systems	Modify Section 10.16.3.6, Nonradioactive Water Drainage Systems, to remove reference to operating and accident conditions and reflect that the primary station function is now the safe storage and handling of irradiated fuel.	4.5.1.3.6	Nonradioactive Water Drainage Systems
10.16.3.7	Sanitary Drainage Systems	Modify Section 10.16.3.7, Sanitary Drainage Systems, to remove reference to operating and accident conditions and reflect that the primary station function is now the safe storage and handling of irradiated fuel.	4.5.1.3.7	Sanitary Drainage Systems
10.16.4	Inspection and Testing	Modify section 10.16.4, Station Equipment and Floor Drainage Systems, Inspection and Testing to reflect that the facility Equipment and Floor Drainage Systems are in constant operation, special testing and inspection is not required. Delete existing testing and inspection requirements since those are construction and installation requirments and are no longer applicable.	4.5.1.4	Inspection and Testing
10.17	STATION PROCESS SAMPLING SYSTEMS		3.3.7	PROCESS SAMPLING SYSTEMS
10.17.1	Power Generation Objective	Delete "Power Generation" from section title Modify section to reflect defueled state	3.3.7.1	Objective
10.17.2	Power Generation Design Basis	Delete "Power Generation" from section title Modify section to reflect defueled state	3.3.7.2	Design Basis
10.17.3	Description		3.3.7.3	Description
10.17.3.1	General	Modified to reflect defueled state Deleted pointer to Process Rad Monitoring, not required	3.3.7.3.1	General
10.17.3.2	Turbine Building Sampling Panel	Delete this section , not required post defueling, all sample point from abandoned systems		
10.17.3.3	Reactor Building Sampling Panel	Delete this section , not required post defueling, all sample point from abandoned systems		
10.17.3.4	Radwaste Building Sampling Panel	No Changes	3.3.7.3.2	Radwaste Building Sampling Panel
10.17.3.5	Gas Sampling and Monitoring	Editorial change to remove reference to a UFSAR section which no longer exists, otherwise, no changes	3.3.7.3.3	Gas Sampling and Monitoring
Table 10.17.1	Fluid Samples	Modified Table 10.17.1, Fluid Samples to remove fluid sample points no longer used post defueling. The Reactor water, cleanup, main steam line, Standby Liquid Control, Feedwater, closed cooling loop, condensate and Circulating water systems are no longer required following permanent defueling. Therefore, sampling of those systems is no longer required. The information deleted from Table 10.17.1 is obsolete. Remaining information in the table has been relocated into the text body of this section		
Table 10.17.2	Gas Samples	Modified Table 10.17.2, Gas Samples to remove gaseous sample points no longer used post defueling. The Air Ejector off gas, Off Gas Filter samples and Standby Gas Treatment Systems are no longer required following permanent defueling. Therefore, sampling of those systems is no longer required. The information deleted from Table 10.17.2 is obsolete. Remaining information in the table has been relocated into the text body of this section		
Figure 10.17-1	G-191164 Station Process Sampling System	No changes	Figure 3.3.7-1	G-191164 Station Process Sampling System
Figure 10.17-2	G-191165 Station Process Sampling System	No changes	Figure 3.3.7-2	G-191165 Station Process Sampling System
10.18	STATION COMMUNICATIONS SYSTEM		3.3.10	COMMUNICATIONS SYSTEM
10.18.1	Power Generation Objective	Modify Section 10.18.1, Station Communications System, Power Generation Objective, to remove reference to operating and accident conditions and reflect that the primary station function i now the safe storage and handling of irradiated fuel.	3.3.10.1	Objective
10.18.2	Power Generation Design Basis	Modify Section 10.18.2, Station Communications System, Power Generation Design Basis to remove reference to operating and accident conditions and reflect that the primary station functio is now the safe storage and handling of irradiated fuel.	3.3.10.2	Design Basis
10.18.3	Description	Modify Section 10.18.3, Station Communications System,Description to remove reference to operating and accident conditions and reflect that the primary station function is now the saf storage and handling of irradiated fuel.	3.3.10.3	Description
10.18.4	Inspection and Testing	No Changes	3.3.10-4	Inspection and Testing
10.19	STATION LIGHTING SYSTEMS		3.3.9	LIGHTING SYSTEMS
10.19.1	Power Generation Objective	Modify Section 10.19.1, Station Lighting System, Power Generation Objective, to remove reference to operating and accident conditions and reflect that the primary station function is now th safe storage and handling of irradiated fuel.	3.3.9.1	Objective
10.19.2	Power Generation Design Basis	Modify Section 10.19.2, Station Lighting System, Power Generation Design Basis to remove reference to operating and accident conditions and reflect that the primary station function is nov the safe storage and handling of irradiated fuel.	3.3.9.2	Design Basis
10.19.3	Description	Modify Section 10.19.3, Station Lighting System, Description to remove reference to operating and accident conditions and reflect that the primary station function is now the safe storage an handling of irradiated fuel.	3.3.9.3	Description
10.19.4	Inspection and Testing	No changes	3.3.9.4	Inspection and Testing

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	Table 10.19.1	Areas Requiring Standby Lighting	Modify Table 10.19.1, Areas Requiring Standby Lighting, to remove reference to operating and accident conditions and reflect that the primary station function is now the safe storage and handling of irradiated fuel. Remaining information in the table has been relocated into the text body in this section.	
10.20	POST-ACCIDENT SAMPLING SYSTEM		The post accident sampling system provided representative samples of reactor coolant for analysis which would be indicative of the extent and development of core damage.	
	10.20.1	Safety Objective	Delete section 10.20.1, Post Accident Sampling System Safety Objective. The post accident sampling system provided representative samples of reactor coolant for analysis which would be indicative of the extent and development of core damage. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions provided by the Post Accident Sampling System are no longer required. The Post Accident Sampling System Safety Objective is no longer applicable. Post Accident Sampling System information is obsolete.	
	10.20.2	Safety Design Basis	Delete section 10.20.2, Post Accident Sampling System Safety Design Basis. The post accident sampling system provided representative samples of reactor coolant for analysis which would be indicative of the extent and development of core damage. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions provided by the Post Accident Sampling System are no longer required. The Post Accident Sampling System Safety Design Basis is no longer applicable. Post Accident Sampling System information is obsolete.	
	10.20.3	Description	Delete section 10.20.3, Post Accident Sampling System Description. The post accident sampling system provided representative samples of reactor coolant for analysis which would be indicative of the extent and development of core damage. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions provided by the Post Accident Sampling System are no longer required. The information provided in the Post Accident Sampling System Description section is no longer applicable. Post Accident Sampling System information is obsolete.	
	10.20.4	Safety Evaluation	Delete section 10.20.4, Post Accident Sampling System Safety Evaluation. The post accident sampling system provided representative samples of reactor coolant for analysis which would be indicative of the extent and development of core damage. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions provided by the Post Accident Sampling System are no longer required. A Post Accident Sampling System Safety Evaluation no longer required. Post Accident Sampling System information is obsolete.	
	10.20.5	Inspection and Testing	Delete section 10.20.5, Post Accident Sampling System Inspection and Testing. The post accident sampling system provided representative samples of reactor coolant for analysis which would be indicative of the extent and development of core damage. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions provided by the Post Accident Sampling System are no longer required. Inspection and Testing of the Post Accident Sampling System is no longer required. Post Accident Sampling System information is obsolete.	
	Figure 10.20-1	G-191165 Fundamental Flow Diagram Post Accident Sampling System	Delete Figure 10.20-1, Fundamental Flow Diagram Post Accident Sampling System. The post accident sampling system provided representative samples of reactor coolant for analysis which would be indicative of the extent and development of core damage. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions provided by the Post Accident Sampling System are no longer required. The information provided on figure 10.20-1 is no longer applicable. Post Accident Sampling System information is obsolete.	
10.21	ADVANCED OFF-GAS CLOSED COOLING WATER SYSTEM		The Advanced Off-Gas Closed Cooling Water System (AOGCCW) provided inhibited glycol-water cooling to auxiliary equipment in the advanced off-gas building.	
	10.21.1	Power Generation Objective	Delete section 10.21.1, Advanced Off-Gas Cooling Water System Power Generation Objective. The Advanced Off-Gas Closed Cooling Water System (AOGCCW) provided inhibited glycol-water cooling to auxiliary equipment in the advanced off-gas building. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions provided by the Advanced Off-Gas Closed Cooling Water System are no longer required.The Advanced Off-Gas Closed Cooling Water System Power Generation Objective is no longer applicable. Advanced Off-Gas Closed Cooling Water System information is obsolete.	

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	10.21.2	Power Generation Design Basis	Delete section 10.21.2, Advanced Off-Gas Cooling Water System Power Generation Design Basis. The Advanced Off-Gas Closed Cooling Water System (AOGCCW) provided inhibited glycol-water cooling to auxiliary equipment in the advanced off-gas building. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions provided by the Advanced Off-Gas Closed Cooling Water System are no longer required.The Advanced Off-Gas Closed Cooling Water System Power Generation Design Basis is no longer applicable. Advanced Off-Gas Closed Cooling Water System information is obsolete.	
	10.21.3	Description	Delete section 10.21.3, Advanced Off-Gas Cooling Water System Description. The Advanced Off-Gas Closed Cooling Water System (AOGCCW) provided inhibited glycol-water cooling to auxiliary equipment in the advanced off-gas building. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions provided by the Advanced Off-Gas Closed Cooling Water System are no longer required.The information provided in the Advanced Off-Gas Closed Cooling Water System Description is no longer applicable. Advanced Off-Gas Closed Cooling Water System information is obsolete.	
	10.21.4	Testing and Inspection	Delete section 10.21.4, Advanced Off-Gas Cooling Water System Inspection and Testing. The Advanced Off-Gas Closed Cooling Water System (AOGCCW) provided inhibited glycol-water cooling to auxiliary equipment in the advanced off-gas building. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions provided by the Advanced Off-Gas Closed Cooling Water System are no longer required. Inspection and testing of the Advanced Off-Gas Closed Cooling Water System is no longer required. Advanced Off-Gas Closed Cooling Water System information is obsolete.	
	Figure 10.21-1	G-191159, Sh6 AOGCW Flow Diagram	Delete Figure 10.21-1, AOGCW Flow Diagram. The Advanced Off-Gas Closed Cooling Water System (AOGCCW) provided inhibited glycol-water cooling to auxiliary equipment in the advanced off-gas building. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions provided by the Advanced Off-Gas Closed Cooling Water System are no longer required.The information provided on Figure 10.21-1 is no longer applicable. Advanced Off-Gas Closed Cooling Water System information is obsolete.	
11.0	**Station Power Conversion Systems**			
11.1	SUMMARY DESCRIPTION		Delete Section 11.1, Summary Description of the Station Power Conversion Systems. The station power conversion systems were those systems which were required to convert the energy of nuclear steam to electrical energy. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and electrical power cannot be generated. Consequently, the functions and information provided by the Station Power Conversion Systems are no longer required. Station Power Conversion Systems information is obsolete.	
11.2	MAIN TURBINE - GENERATOR		The Main Turbine Generator System converted the thermodynamic energy of the steam into electrical energy.	
	11.2.1	Power Generation Objective	Delete Section 11.2.1, Main Turbine - Generator Power Generation Objective. The Main Turbine Generator System converted the thermodynamic energy of the steam into electrical energy. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and electrical power cannot be generated. Consequently, the functions and information provided by the Main Turbine - Generator are no longer required. The Main Turbine - Generator Power Generation Objective is no longer applicable. Main Turbine - Generator information is obsolete.	
	11.2.2	Power Generation Design Basis	Delete Section 11.2.2, Main Turbine - Generator Power Generation Design Basis The Main Turbine Generator System converted the thermodynamic energy of the steam into electrical energy. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and electrical power cannot be generated. Consequently, the functions and information provided by the Main Turbine - Generator are no longer required. The Main Turbine - Generator Power Generation Design Basis is no longer applicable. Main Turbine - Generator information i obsolete.	
	11.2.3	Description	Delete Section 11.2.3, Main Turbine - Generator, Description. The Main Turbine Generator System converted the thermodynamic energy of the steam into electrical energy. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and electrical power cannot be generated. Consequently, the functions and information provided by the Main Turbine - Generator are no longer required. The information provided in the Main Turbine - Generator description section is no longer applicable. Main Turbine - Generator information is obsolete.	
	11.2.4	Testing and Inspection	Delete Section 11.2.4, Main Turbine - Generator,Inspection and Testing. The Main Turbine Generator System converted the thermodynamic energy of the steam into electrical energy. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and electrical power cannot be generated. Consequently, the functions and information provided by the Main Turbine - Generator are no longer required. Inspection and Testing of the Main Turbine - Generator is no longer required. Main Turbine - Generator information is obsolete.	

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	Figure 11.2-1	G-191156 Turbine Generator, Main Steam and Extraction Steam Systems	Delete Figure 11.2-1, Turbine Generator, Main Steam and Extraction Steam Systems. The Main Turbine Generator System converted the thermodynamic energy of the steam into electrical energy. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and electrical power cannot be generated. Consequently, the functions and information provided by the Main Turbine - Generator are no longer required. The information provided on Figure 11.2-1 is no longer applicable. Main Turbine - Generator information is obsolete.	
11.3	MAIN CONDENSER SYSTEM		The Main Condenser System provided a heat sink for steam condensation during normal operation or steam bypass valve operation. The main condenser also provided for the deaeration of condensate, provided NPSH for the condensate pumps, and served as a makeup point for water used in the steam cycle.	
	11.3.1	Power Generation Objective	Delete Section 11.3.1, Main Condenser System, Power Generation Objective. The Main Condenser System provided a heat sink for steam condensation during normal operation or steam bypass valve operation. The main condenser also provided for the deaeration of condensate, provided NPSH for the condensate pumps, and served as a makeup point for water used in the steam cycle. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and, as a result, steam condensation is not required, condensate deaeration is not required, condensate pump NPSH is not required and no steam cycle makeup is required. The functions and information provided by the Main Condenser System are no longer required. The Main Condenser System Power Generation Objective is no longer applicable. Main Condenser information is obsolete.	
	11.3.2	Power Generation Design Bases	Delete Section 11.3.2, Main Condenser System, Power Generation Design Basis The Main Condenser System provided a heat sink for steam condensation during normal operation or steam bypass valve operation. The main condenser also provided for the deaeration of condensate, provided NPSH for the condensate pumps, and served as a makeup point for water used in the steam cycle. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and, as a result, steam condensation is not required, condensate deaeration is not required, condensate pump NPSH is not required and no steam cycle makeup is required. The functions and information provided by the Main Condenser System are no longer required. The Main Condenser System Power Generation Design Bases are no longer applicable. Main Condenser information is obsolete.	
	11.3.3	Description	Delete Section 11.3.3, Main Condenser System, Power Generation Design Basis The Main Condenser System provided a heat sink for steam condensation during normal operation or steam bypass valve operation. The main condenser also provided for the deaeration of condensate, provided NPSH for the condensate pumps, and served as a makeup point for water used in the steam cycle. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and, as a result, steam condensation is not required, condensate deaeration is not required, condensate pump NPSH is not required and no steam cycle makeup wast is required. The functions and information provided by the Main Condenser System are no longer required. The Main Condenser System Power Generation Design Bases are no longer applicable. Main Condenser information is obsolete.	
	11.3.4	Testing and Inspection	Delete Section 11.3.4, Main Condenser System, Power Generation Design Basis The Main Condenser System provided a heat sink for steam condensation during normal operation or steam bypass valve operation. The main condenser also provided for the deaeration of condensate, provided NPSH for the condensate pumps, and served as a makeup point for water used in the steam cycle. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and, as a result, steam condensation is not required, condensate deaeration is not required, condensate pump NPSH is not required and no steam cycle makeup wast is required. The functions and information provided by the Main Condenser System are no longer required. Inspection and Testing of the Main Condenser System is no longer required. Main Condenser information is obsolete.	
11.4	MAIN CONDENSER GAS REMOVAL AND TURBINE SEALING SYSTEMS		The main condenser gas removal system evacuated gases from the main turbine and main condenser during startup and maintained the system free of noncondensible gases during operation. The turbine sealing system acted in conjunction with the gas removal system to prevent air from entering into the system during startup and operation and to prevent steam from leaking to th atmosphere.	
	11.4.1	Power Generation Objective	Delete Section 11.4.1, Main Condenser Gas Removal and Turbine Sealing Systems, Power Generation Objective. The main condenser gas removal system evacuated gases from the main turbine and main condenser during startup and maintained the system free of noncondensible gases during operation. The turbine sealing system acted in conjunction with the gas removal system to prevent air from entering into the system during startup and operation and to prevent steam from leaking to th atmosphere. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and, as a result, main condenser gas removal is not required, and, since the main turbine wil no longer operate, turbine sealing is not required. The functions and information provided by the Main Condenser Gas Removal and Turbine Sealing Systems are no longer required. The Main Condenser Gas Removal and Turbine Sealing Systems Power Generation Objective is no longer applicable. Main Condenser Gas Removal and Turbine Sealing Systems information is obsolete.	

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	11.4.2	Power Generation Design Bases		Delete Section 11.4.2, Main Condenser Gas Removal and Turbine Sealing Systems, Power Generation Objective. The main condenser gas removal system evacuated gases from the main turbine and main condenser during startup and maintained the system free of noncondensible gases during operation. The turbine sealing system acted in conjunction with the gas removal system to prevent air from entering into the system during startup and operation and to prevent steam from leaking to the atmosphere. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and, as a result, main condenser gas removal is not required, and, since the main turbine will no longer operate, turbine sealing is not required. The functions and information provided by the Main Condenser Gas Removal and Turbine Sealing Systems are no longer required. The Main Condenser Gas Removal and Turbine Sealing Systems Power Generation Design Bases are no longer applicable. Main Condenser Gas Removal and Turbine Sealing Systems information is obsolete.	
	11.4.3	Description			
		11.4.3.1	General	Delete Section 11.4.3.1, Main Condenser Gas Removal and Turbine Sealing Systems, General Description. The main condenser gas removal system evacuated gases from the main turbine and main condenser during startup and maintained the system free of noncondensible gases during operation. The turbine sealing system acted in conjunction with the gas removal system to prevent air from entering into the system during startup and operation and to prevent steam from leaking to the atmosphere. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and, as a result, main condenser gas removal is not required, and, since the main turbine will no longer operate, turbine sealing is not required. The functions and information described in the Main Condenser Gas Removal and Turbine Sealing Systems General Description Section are no longerapplicable. Main Condenser Gas Removal and Turbine Sealing Systems information is obsolete.	
		11.4.3.2	Condenser Hogging Pump	Delete Section 11.4.3.2, Main Condenser Gas Removal and Turbine Sealing Systems, Condenser Hogging Pump. The main condenser gas removal system evacuated gases from the main turbine and main condenser during startup and maintained the system free of noncondensible gases during operation. A mechanical ac motor driven remote manually operated V Belt rotary type pump removed non-condensible gasses during startup. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and, as a result, main condenser gas removal is not required. The functions and information described in the Condenser Hogging Pump Section are no longerapplicable. Main Condenser Gas Removal and Turbine Sealing Systems information is obsolete.	
		11.4.3.3	Steam Jet Air Ejectors	Delete Section 11.4.3.3, Main Condenser Gas Removal and Turbine Sealing Systems, Steam Jet Air Ejectors. The main condenser gas removal system evacuated gases from the main turbine and main condenser during startup and maintained the system free of noncondensible gases during operation. During operation, steam jet air ejectors removed gases and vapors from the main condensers using main steam, reduced in pressure, as the driving medium.. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and, as a result, main condenser gas removal is not required. The functions and information described in the Steam Jet Air Ejectors Section are no longerapplicable. Main Condenser Gas Removal and Turbine Sealing Systems information is obsolete.	
		11.4.3.4	Steam Packing Exhauster Unit	Delete Section 11.4.3.4, Main Condenser Gas Removal and Turbine Sealing Systems, Steam Packing Exhauster Unit. The main condenser gas removal system evacuated gases from the main turbine and main condenser during startup and maintained the system free of noncondensible gases during operation. During operation, one steam packing exhauster unit collected and condensed sealing steam and discharged air leakage to the offgas system. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and, as a result, main condenser gas removal is not required. The functions and information described in the Steam Packing Exhauster Unit Section are no longerapplicable. Main Condenser Gas Removal and Turbine Sealing Systems information is obsolete.	
		11.4.3.5	Steam Seal Regulator	Delete Section 11.4.3.5, Main Condenser Gas Removal and Turbine Sealing Systems, Steam Seal Regulator. The main condenser gas removal system evacuated gases from the main turbine and main condenser during startup and maintained the system free of noncondensible gases during operation. During operation, the steam seal regulator maintained the same preselected steam pressure in the packing header regardless of the steam turbine load.. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and, as a result, main condenser gas removal is not required. The functions and information described in the Steam Seal Regulator Section are no longerapplicable. Main Condenser Gas Removal and Turbine Sealing Systems information is obsolete.	

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	11.4.4	Inspection and Testing		Delete Section 11.4.4, Main Condenser Gas Removal and Turbine Sealing Systems, Inspection and Testing. The main condenser gas removal system evacuated gases from the main turbine and main condenser during startup and maintained the system free of noncondensible gases during operation. The turbine sealing system acted in conjunction with the gas removal system to prevent air from entering into the system during startup and operation and to prevent steam from leaking to the atmosphere. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and, as a result, main condenser gas removal is not required, and, since the main turbine will no longer operate, turbine sealing is not required. Inspection and Testing of the Main Condenser Gas Removal and Turbine Sealing Systems General Description Section is not required. Main Condenser Gas Removal and Turbine Sealing Systems information is obsolete.	
	Figure 11.4-1	5920-12598 Turbine Steam Seal System		Delete Figure 11.4-1,Turbine Steam Seal System. The main condenser gas removal system evacuated gases from the main turbine and main condenser during startup and maintained the system free of noncondensible gases during operation. The turbine sealing system acted in conjunction with the gas removal system to prevent air from entering into the system during startup and operation and to prevent steam from leaking to the atmosphere. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and, as a result, main condenser gas removal is not required, and, since the main turbine will no longer operate, turbine sealing is not required. The information provided on Figure 11.4-1 is no longer applicable. Main Condenser Gas Removal and Turbine Sealing Systems information is obsolete.	
11.5	STEAM SYSTEMS			The steam systems conducted steam from the reactor vessel through the primary containment to the steam turbine at a controlled pressure during normal operation, bypassed steam directly to the main condenser to control pressure during reactor vessel heatup and as the turbine was broughton line and when reactor steam generation exceeded the turbine requirements, and during reactor cooldown. Steam systems also provided turbine extraction steam for condensate and reactor feedwater heating.	
	11.5.1	Power Generation Objective		Delete Section 11.5.1, Steam Systems, Power Generation Objective. The steam systems conducted steam from the reactor vessel through the primary containment to the steam turbine at a controlled pressure during normal operation, bypassed steam directly to the main condenser to control pressure during reactor vessel heatup and as the turbine was broughton line and when reactor steam generation exceeded the turbine requirements, and during reactor cooldown. Steam systems also provided turbine extraction steam for condensate and reactor feedwater heating. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and, as a result, steam systems are not required. The Steam Systems Power Generation Objective is no longer applicable. Steam Systems information is obsolete.	
	11.5.2	Power Generation Design Bases		Delete Section 11.5.2, Steam Systems, Power Generation Design Bases The steam systems conducted steam from the reactor vessel through the primary containment to the steam turbine at a controlled pressure during normal operation, bypassed steam directly to the main condenser to control pressure during reactor vessel heatup and as the turbine was broughton line and when reactor steam generation exceeded the turbine requirements, and during reactor cooldown. Steam systems also provided turbine extraction steam for condensate and reactor feedwater heating. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and, as a result, steam systems are not required. The Steam Systems Power Generation Design Bases are no longer applicable. Steam Systems information is obsolete.	
	11.5.3	Safety Design Bases		Delete Section 11.5.3, Steam Systems, Safety Design Bases The steam systems conducted steam from the reactor vessel through the primary containment to the steam turbine at a controlled pressure during normal operation, bypassed steam directly to the main condenser to control pressure during reactor vessel heatup and as the turbine was broughton line and when reactor steam generation exceeded the turbine requirements, and during reactor cooldown. Steam systems also provided turbine extraction steam for condensate and reactor feedwater heating. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and, as a result, steam systems are not required. The Steam Systems Safety Design Bases are no longer applicable. Steam Systems information is obsolete.	
	11.5.4	Description			
		11.5.4.1	Main Steam System	Delete Section 11.5.4.1, Main Steam System Description The main steam system conducted steam from the reactor vessel through the primary containment to the steam turbine. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and, as a result, the main steam system is no longer required. The description of the Main Steam System is no longer applicable. Steam Systems information is obsolete.	
		11.5.4.2	Main Turbine Bypass System	Delete Section 11.5.4.2, Main Steam Turbine Bypass System Description The Main Turbine Bypass system opened whenever the permitted admission of steam into the turbine was less than the amount of steam generated by the reactor. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and, as a result, the main turbine bypass system is no longer required. The description of the main turbine bypass system is no longer applicable. Steam Systems information is obsolete.	

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		11.5.4.3	Extraction Steam	Delete Section 11.5.4.3, Extraction Steam System Description. Extraction Steam provided steam for feedwater pre-heating. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and, as a result, the Extraction Steam system is no longer required. The description of the Extraction Steam System is no longer applicable. Steam Systems information is obsolete.	
	11.5.5	Safety Evaluation		Delete Section 11.5.5, Steam Systems, Safety Evaluation The steam systems conducted steam from the reactor vessel through the primary containment to the steam turbine at a controlled pressure during normal operation, bypassed steam directly to the main condenser to control pressure during reactor vessel heatup and as the turbine was broughton line and when reactor steam generation exceeded the turbine requirements, and during reactor cooldown. Steam systems also provided turbine extraction steam for condensate and reactor feedwater heating. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and, as a result, steam systems are not required. The Steam Systems Safety Evaluation is are no longer applicable. Steam Systems information is obsolete.	
	11.5.6	Tests and Inspections		Delete Section 11.5.6 Steam Systems, Tests and Inspections. The steam systems conducted steam from the reactor vessel through the primary containment to the steam turbine at a controlled pressure during normal operation, bypassed steam directly to the main condenser to control pressure during reactor vessel heatup and as the turbine was broughton line and when reactor steam generation exceeded the turbine requirements, and during reactor cooldown. Steam systems also provided turbine extraction steam for condensate and reactor feedwater heating. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and, as a result, steam systems are not required. Tests and Inspection of the Steam Systems are no longer required. Steam Systems information is obsolete.	
	Figure 11.5-1	5920-569 Primary Steam Piping Arrangement		Delete Figure 11.5-1,Primary Steam Piping Arrangement. The steam systems conducted steam from the reactor vessel through the primary containment to the steam turbine at a controlled pressure during normal operation, bypassed steam directly to the main condenser to control pressure during reactor vessel heatup and as the turbine was broughton line and when reactor steam generation exceeded the turbine requirements, and during reactor cooldown. Steam systems also provided turbine extraction steam for condensate and reactor feedwater heating. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and, as a result, steam systems are not required. The information provided on Figure 11.5-1 is no longer applicable. Steam Systems information is obsolete.	
11.6	CIRCULATING WATER SYSTEMS			The Circulating Water System provided a continuous supply of cooling water pumped from and returned to the Connecticut River or by recirculation flow pumped through cooling towers for steam condensation and heat removal in the condensers.	
	11.6.1	Power Generation Objective		Delete Section 11.6.1, Circulating Water Systems, Power Generation Objective. The Circulating Water System provided a continuous supply of cooling water pumped from and returned to the Connecticut River or by recirculation flow pumped through cooling towers for steam condensation and heat removal in the condensers. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and, as a result, Circulating Water Systems are not required. The Circulating Water Systems Power Generation Objective is no longer applicable. Circulating Water Systems information is obsolete.	
	11.6.2	Power Generation Design Bases		Delete Section 11.6.2, Circulating Water Systems, Power Generation Design Bases. The Circulating Water System provided a continuous supply of cooling water pumped from and returned to the Connecticut River or by recirculation flow pumped through cooling towers for steam condensation and heat removal in the condensers. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and, as a result, Circulating Water Systems are not required. The Circulating Water Systems Power Generation Design Bases are no longer applicable. Circulating Water Systems information is obsolete.	
	11.6.3	Description		Delete Section 11.6.3, Circulating Water Systems, Description. The Circulating Water System provided a continuous supply of cooling water pumped from and returned to the Connecticut River or by recirculation flow pumped through cooling towers for steam condensation and heat removal in the condensers. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and, as a result, Circulating Water Systems are not required. The functions and information provided by the Circulating Water System are no longer required. Information provided in the Circulating Water System Desription section is no longer applicable. Main Condenser information is obsolete.	

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	11.6.4	Inspection and Testing		Delete Section 11.6.4, Circulating Water Systems, Inspection and Testing. The Circulating Water System provided a continuous supply of cooling water pumped from and returned to the Connecticut River or by recirculation flow pumped through cooling towers for steam condensation and heat removal in the condensers. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and, as a result, Circulating Water Systems are not required. Inspection and Testing of the Circulating Water System is no longer required. Information provided in the Circulating Water System Inspection and Testing section is no longer applicable. Main Condenser information is obsolete.	
	Figure 11.6-1	G-191166 Circulating Water Systems		Delete Figure 11.6-1, Circulating Water Systems. The Circulating Water System provided a continuous supply of cooling water pumped from and returned to the Connecticut River or by recirculation flow pumped through cooling towers for steam condensation and heat removal in the condensers. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and, as a result, Circulating Water Systems are not required. The functions and information provided by the Circulating Water System are no longer required. Information provided on Figure 11.6-1 is no longer applicable.Circulating Water information is obsolete.	
11.7	CONDENSATE DEMINERALIZER SYSTEM			The condensate demineralizer system maintained the required purity of feedwater supplied to the reactor.	
	11.7.1	Power Generation Objective		Delete Section 11.7.1, Condensate Demineralizer System, Power Generation Objective. The condensate demineralizer system maintained the required purity of feedwater supplied to the reactor. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and Condensate and Feedwater systems are not required. Consequently, the condensate demineralizer system is no longer required. The Condensate Demineralizer System Power Generation Objective is no longer applicable. Information provided in the Condensate Demineralizer Power Generation Objective section is no longer applicable. Condensate Demineralizer System information is obsolete.	
	11.7.2	Power Generation Design Bases		Delete Section 11.7.2, Condensate Demineralizer System, Power Generation Design Bases. The condensate demineralizer system maintained the required purity of feedwater supplied to the reactor. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and Condensate and Feedwater systems are not required. Consequently, the condensate demineralizer system is no longer required. The Condensate Demineralizer System Power Generation Design Bases are no longer applicable. Information provided in the Condensate Demineralizer Power Generation Design Bases section is no longer applicable. Condensate Demineralizer System information is obsolete.	
	11.7.3	Description			
		11.7.3.1	General	Delete Section 11.7.3.1, Condensate Demineralizer System, General Description. The condensate demineralizer system maintained the required purity of feedwater supplied to the reactor. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and Condensate and Feedwater systems are not required. Consequently, the functions and information provided by the Condensate Demineralizer System are no longer required. Information provided in the Condensate Demineralizer General Description section is no longer applicable. Condensate Demineralizer System information is obsolete.	
		11.7.3.2	Equipment	Delete Section 11.7.3.2, Condensate Demineralizer System, Equipment Description. The condensate demineralizer system maintained the required purity of feedwater supplied to the reactor. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and Condensate and Feedwater systems are not required. Consequently, the functions and information provided by the Condensate Demineralizer System are no longer required. Information provided in the Condensate Demineralizer Equipment Description section is no longer applicable. Condensate Demineralizer System information is obsolete.	
		11.7.3.3	Control and Instrumentation	Delete Section 11.7.3.3, Condensate Demineralizer System, Control and Instrumentation. The condensate demineralizer system maintained the required purity of feedwater supplied to the reactor. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and Condensate and Feedwater systems are not required. Consequently, the functions and information provided by the Condensate Demineralizer System are no longer required. Information provided in the Condensate Demineralizer Control and Instrumentation description section is no longer applicable. Condensate Demineralizer System information is obsolete.	

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		11.7.3.4	Specifications	Delete Section 11.7.3.4, Condensate Demineralizer System, Specifications. The condensate demineralizer system maintained the required purity of feedwater supplied to the reactor. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and Condensate and Feedwater systems are not required. Consequently, the functions and information provided by the Condensate Demineralizer System are no longer required. Information provided in the Condensate Demineralizer Specifications section is no longer applicable. Condensate Demineralizer System information is obsolete.	
	11.7.4	Inspection and Testing		Delete Section 11.7.4, Condensate Demineralizer System, Inspection and Testing. The condensate demineralizer system maintained the required purity of feedwater supplied to the reactor. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and Condensate and Feedwater systems are not required. Consequently, the functions and information provided by the Condensate Demineralizer System are no longer required. Inspection and Testing of the Condensate Demineralizer System is no longer required. Condensate Demineralizer System information is obsolete.	
	Figure 11.7-1	G-191274 Condensate Demineralizer System		Delete Figure 11.7-1, Condensate Demineralizer System. The condensate demineralizer system maintained the required purity of feedwater supplied to the reactor. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and Condensate and Feedwater systems are not required. Consequently, the functions and information provided by the Condensate Demineralizer System are no longer required. Information provided on Figure 11.7-1 is no longer applicable. Condensate Demineralizer System information is obsolete.	
11.8	CONDENSATE AND REACTOR FEEDWATER SYSTEMS			The Condensate and Reactor Feedwater Systems provided demineralized water to the reactor vessel at a rate sufficient to maintain adequate reactor vessel water level.	
	11.8.1	Power Generation Objective		Delete Section 11.8.1, Condensate and Reactor Feedwater Systems, Power Generation Objective. The Condensate and Reactor Feedwater Systems provided demineralized water to the reactor vessel at a rate sufficient to maintain adequate reactor vessel water level. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and Condensate and Feedwater systems are not required.The Condensate and Reactor Feedwater Systems Power Generation Objective is no longer applicable. Information provided in the Condensate and Reactor Feedwater Systems Power Generation Objective section is no longer applicable.Condensate and Reactor Feedwater Systems information is obsolete.	
	11.8.2	Power Generation Design Bases		Delete Section 11.8.2, Condensate and Reactor Feedwater Systems, Power Generation Objective. The Condensate and Reactor Feedwater Systems provided demineralized water to the reactor vessel at a rate sufficient to maintain adequate reactor vessel water level. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and Condensate and Feedwater systems are not required.The Condensate and Reactor Feedwater Systems Power Generation Design Bases are no longer applicable. Information provided in the Condensate and Reactor Feedwater Systems Power Generation Design Bases section is no longer applicable.Condensate and Reactor Feedwater Systems information is obsolete.	
	11.8.3	Description			
		11.8.3.1	Condensate and Reactor Feedwater Pumps	Delete Section 11.8.3.1, Condensate and Reactor Feedwater Pumps. The Condensate and Reactor Feedwater Systems provided demineralized water to the reactor vessel at a rate sufficient to maintain adequate reactor vessel water level. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and Condensate and Feedwater systems are not required.The functions and information provided by the Condensate and Reactor Feedwater Pumps are no longer required. Information provided in the Condensate and Reactor Feedwater Pumps section is no longer applicable.Condensate and Reactor Feedwater Systems information is obsolete.	
		11.8.3.2	Reactor Feedwater Heaters	Delete Section 11.8.3.2, Reactor Feedwater Heaters. The Condensate and Reactor Feedwater Systems provided demineralized water to the reactor vessel at a rate sufficient to maintain adequate reactor vessel water level. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and Condensate and Feedwater systems are not required.The functions and information provided by the Reactor Feedwater Heaters are no longer required. Information provided in the Reactor Feedwater Heaters section is no longer applicable.Condensate and Reactor Feedwater Systems information is obsolete.	

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		11.8.3.3	Condensate Demineralizer System	Delete Section 11.8.3.3, Condensate Demineralizer System. The Condensate and Reactor Feedwater Systems provided demineralized water to the reactor vessel at a rate sufficient to maintain adequate reactor vessel water level. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and Condensate and Feedwater systems are not required.The functions and information provided by the Condensate Demineralizer System are no longer required. Information provided in the Condensate Demineralizer System section is no longer applicable.Condensate and Reactor Feedwater Systems information is obsolete.	
		11.8.3.4	Steam Jet Air Ejector (SJAE) Intercondensers	Delete Section 11.8.3.4, Steam Jet Air Ejector (SJAE) Intercondensers. The Condensate and Reactor Feedwater Systems provided demineralized water to the reactor vessel at a rate sufficient to maintain adequate reactor vessel water level. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and Condensate and Feedwater systems are not required.The functions and information provided by the Steam Jet Air Ejector (SJAE) Intercondensersare no longer required. Information provided in the Steam Jet Air Ejector (SJAE) Intercondensers section is no longer applicable. Condensate and Reactor Feedwater Systems information is obsolete.	
		11.8.3.5	Steam Packing Exhauster	Delete Section 11.8.3.5, Steam Packing Exhauster. The Condensate and Reactor Feedwater Systems provided demineralized water to the reactor vessel at a rate sufficient to maintain adequate reactor vessel water level. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and Condensate and Feedwater systems are not required.The functions and information provided by the Steam Packing Exhauster are no longer required. Information provided in the Steam Jet Air Ejector (SJAE) Intercondensers section is no longer applicable. Condensate and Reactor Feedwater Systems information is obsolete.	
		11.8.3.6	Feedwater Control Station	Delete Section 11.8.3.6, Feedwater Control Station. The Condensate and Reactor Feedwater Systems provided demineralized water to the reactor vessel at a rate sufficient to maintain adequate reactor vessel water level. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and Condensate and Feedwater systems are not required.The functions and information provided by the Feedwater Control Station are no longer required. Information provided in the Feedwater Control Station section is no longer applicable. Condensate and Reactor Feedwater Systems information is obsolete.	
		11.8.3.7	Condensate and Feedwater System Minimum Flow Bypasses	Delete Section 11.8.3.7, Condensate and Feedwater System Minimum Flow Bypasses. The Condensate and Reactor Feedwater Systems provided demineralized water to the reactor vessel at a rate sufficient to maintain adequate reactor vessel water level. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and Condensate and Feedwater systems are not required.The functions and information provided by the Condensate and Feedwater System Minimum Flow Bypasses are no longer required. Information provided in the Condensate and Feedwater System Minimum Flow Bypasses section is no longer applicable. Condensate and Reactor Feedwater Systems information is obsolete.	
		11.8.3.8	Keep Fill Pressurizing Line	Delete Section 11.8.3.8, Condensate and Feedwater System Keep Fill Pressurizing Line. The Condensate and Reactor Feedwater Systems provided demineralized water to the reactor vessel at a rate sufficient to maintain adequate reactor vessel water level. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and Condensate and Feedwater systems are not required.The functions and information provided by the Keep Fill Pressurizing Line are no longer required. Information provided in the Keep Fill Pressurizing Line section is no longer applicable. Condensate and Reactor Feedwater Systems information is obsolete.	
		11.8.3.9	Condensate Makeup and Reject Systems	Delete Section 11.8.3.9, Condensate and Feedwater System Condensate Makeup and Reject Systems. The Condensate and Reactor Feedwater Systems provided demineralized water to the reactor vessel at a rate sufficient to maintain adequate reactor vessel water level. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and Condensate and Feedwater systems are not required.The functions and information provided by the Condensate Makeup and Reject Systems are no longer required. Information provided in the Condensate Makeup and Reject Systems section is no longer applicable. Condensate and Reactor Feedwater Systems information is obsolete.	

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		11.8.3.10	Condensate and Feedwater Piping	Delete Section 11.8.3.10, Condensate and Feedwater Piping The Condensate and Reactor Feedwater Systems provided demineralized water to the reactor vessel at a rate sufficient to maintain adequate reactor vessel water level. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and Condensate and Feedwater systems are not required.The functions and information provided by the Condensate and Feedwater Piping are no longer required. Information provided in the Condensate and Feedwater Piping section is no longer applicable. Condensate and Reactor Feedwater Systems information is obsolete.	
	11.8.4	Inspection and Testing		Delete Section 11.8.4, Condensate and Reactor Feedwater Systems, Power Generation Objective. The Condensate and Reactor Feedwater Systems provided demineralized water to the reactor vessel at a rate sufficient to maintain adequate reactor vessel water level. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and Condensate and Feedwater systems are not required. Inspection and Testing of the Condensate and Reactor Feedwater Systems is no longer required. Information provided in the Condensate and Reactor Feedwater Systems Inspection and Testing section is no longer applicable. Condensate and Reactor Feedwater Systems information is obsolete.	
	Figure 11.8-1	G-191157, Sh1 Condensate Feedwater and Gas Removal Systems		Delete Figure 11.8-1, Condensate Feedwater and Gas Removal Systems. The Condensate and Reactor Feedwater Systems provided demineralized water to the reactor vessel at a rate sufficient to maintain adequate reactor vessel water level. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and Condensate and Feedwater systems are not required.The Condensate and Reactor Feedwater Systems Power Generation Objective is no longer applicable. Information provided on Figure 11.8-1 is no longer applicable. Condensate and Reactor Feedwater Systems information is obsolete.	
11.9	STATION COOLING TOWER WATER SYSTEM			The Station Cooling Tower Water System provided an alternate means of cooling heated condenser circulating water discharge. The cooling towers transferred heat from the circulating water to the atmosphere and return the cooled circulating water to the river or recirculated it through the Circulating Water System.	
	11.9.1	Power Generation Objective		Delete Section 11.9.1, Station Cooling Tower Water System, Power Generation Objective. The Station Cooling Tower Water System provided an alternate means of cooling heated condenser circulating water discharge. The cooling towers transferred heat from the circulating water to the atmosphere and return the cooled circulating water to the river or recirculated it through the Circulating Water System. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and, as a result, Circulating Water Systems are not required. Consequently, the Station Cooling Tower Water System is no longer required. The Station Cooling Tower Water System Power Generation Objective is no longer applicable. Station Cooling Tower Water System information is obsolete.	
	11.9.2	Power Generation Design Bases		Delete Section 11.9.2, Station Cooling Tower Water System, Power Generation Design Bases. The Station Cooling Tower Water System provided an alternate means of cooling heated condenser circulating water discharge. The cooling towers transferred heat from the circulating water to the atmosphere and return the cooled circulating water to the river or recirculated it through the Circulating Water System. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and, as a result, Circulating Water Systems are not required. Consequently, the Station Cooling Tower Water System is no longer required. The Station Cooling Tower Water System Power Generation Design Bases are no longer applicable. Station Cooling Tower Water System information is obsolete.	
	11.9.3	Description		Delete Section 11.9.3, Station Cooling Tower Water System Description. The Station Cooling Tower Water System provided an alternate means of cooling heated condenser circulating water discharge. The cooling towers transferred heat from the circulating water to the atmosphere and return the cooled circulating water to the river or recirculated it through the Circulating Water System. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and, as a result, Circulating Water Systems are not required. Consequently, the Station Cooling Tower Water System is no longer required. Information provided in the Station Cooling Tower Water System description section is no longer applicable. Station Cooling Tower Water System information is obsolete.	
	11.9.4	Inspection and Testing		Delete Section 11.9.4, Station Cooling Tower Water System Inspection and Testing. The Station Cooling Tower Water System provided an alternate means of cooling heated condenser circulating water discharge. The cooling towers transferred heat from the circulating water to the atmosphere and return the cooled circulating water to the river or recirculated it through the Circulating Water System. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, nuclear steam cannot be produced and, as a result, Circulating Water Systems are not required. Consequently, the Station Cooling Tower Water System is no longer required. Inspection and testing of the Station Cooling Tower Water System is no longer required. Station Cooling Tower Water System information is obsolete.	

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12.0	**Station Structures**						3.2	Facility Structures			
12.1	SUMMARY DESCRIPTION					Modify section 12.1, Station Structures, Summary Description, to delete reference to primary containment. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions of the Primary Containment System are no longer required. Primary Containment System information is obsolete.			1.3.1.2	SUMMARY DESCRIPTION	
12.2	PRINCIPAL STATION STRUCTURES AND FOUNDATIONS					Heading deleted, redundant					
	12.2.1	Loading Considerations for Structures, Foundations, Equipment and Systems				Section 12.2.1, Loading Considerations for Structures, Foundations, Equipment and Systems, delete internal reference, not required, excessive detail Section 12.2.1, Loading Considerations for Structures, Foundations, Equipment and Systems, delete internal reference, not required, excessive detail. Section 12.2.1, Loading Considerations for Structures, Foundations, Equipment and Systems, delete internal reference, not required, excessive detail, section 5.2 has been deleted. Section 12.2.1, Loading Considerations for Structures, Foundations, Equipment and Systems, Delete information regarding turbine building dampers, HELB no longer possible, steam tunnel pressurization due to steam leak no longer possible, damper function no longer required.		3.1.3	Loading Considerations for Structures, Foundations, Equipment and Systems		
		12.2.1.1	Seismic Classification			Section 12.2.1.1, Seismic Classification, delete reference to a main steam line break outside the drywell. Following certification of permanent defueling, a main steam line break outside the drywell is not possible.			3.1.3.1	Seismic Classification	
			12.2.1.1.1	Class I Structures		Section 12.2.1.1.1, Class I Structures, Delete Alternate cooling cell and cooling tower cell from list of class I structures. The functions provided by those structures are no longer required following permanent defueling, those structures no longer meet the definition of class I structures, therefore, there is no requirement to maintain those structures as Class I.				3.1.3.1.1	Class I Structures
			12.2.1.1.2	Class I Equipment		Section 12.2.1.1.2, Class I Equipment, Delete the below listed equipment from the class I equipment list. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Consequently, the functions provided by the below systems are no longer required and that equipment no longer meets the definition of class I equipment. • Control Rods and Drive System • Control Rod Drive Housing Supports • Core Shroud • Core Supports • Recirculation Piping, including valves, pumps and suspension system • Primary Containment Isolation Dampers • Reactor Building Isolation Dampers • All Piping Connections from the Reactor Primary Vessel up to and including the First Isolation Valve External to the Drywell • Primary Containment isolation valves • Reactor Building Isolation Dampers • Reactor Core High Pressure Coolant Injection System • Reactor Core Isolation Cooling System • Standby Liquid Control System • Reactor Core Spray Cooling System • Reactor Building Closed Cooling Water System, alternate cooling and primary containment isolation portions only • Reactor Core Residual Heat Removal System and its associated Service Water System • Alternate Cooling System • Station Standby Gas Treatment System • Station Battery System • Emergency Busses and Other Electrical Gear and Power to Safety Equipment • Instrumentation and control systems: ADS backup nitrogen supply, reactor water level, standby liquid control systems, CRD I&C Scram portions, core standby cooling systems, primary containment isolation systems, control rod position indication systems, reactor protection systems, nuclear instrumentation systems • Condensate Storage Tank and Condensate Transfer System (lines tied to Engineering Safety Systems)				3.1.3.1.2	Class I Equipment
			12.2.1.1.3	Class II Structures		Section 12.2.1.1.3, Class II Structures, Delete the below listed structures from the class II structure list. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Consequently, the functions provided by the below structures are no longer required and that equipment no longer meets the definition of class II structures. • AOG Building • Cooling tower				3.1.3.1.3	Class II Structures

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UFSAR Section					FSAR Conversion to DSAR Change Summary	DSAR Section				
			12.2.1.1.4	Class II Equipment	Section 12.2.1.1.4, Class II Equipment, Delete the below listed equipment from the class II equipment list. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Consequently, the functions provided by the below systems are no longer required and that equipment no longer meets the definition of class II equipment. • Advanced Off-Gas System • Turbine Generator System • Main Condenser System • Reactor Feedwater and Condensate Systems • Reactor Clean-Up Demineralizer System • Turbine System Moisture Separators • Condensate Demineralizer System • Steam Separators and Driers				3.1.3.1.4	Class II Equipment
		12.2.1.2	Sesmic Design					3.1.3.2	Sesmic Design	
			12.2.1.2.1	Class I Structures	Modify section 12.2.1.2.1 to delete reference to the alternate cooling cell. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Consequently, the functions provided by the alternate cooling cell are no longer required. Information regarding the alternate cooling cell is obsolete and may be deleted.				3.1.3.2.1	Class I Structures
			12.2.1.2.2	Class II Structures	No changes				3.1.3.2.2	Class II Structures
			12.2.1.2.3	Equipment Seismic Design	Section 12.2.1.2.3, Equipment Seismic Design, delete the statement regarding snubber operability for the reactor coolant and other safety related systems. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Consequently, the functions provided by the Reactor Coolant system are no longer required and assurance of structural integrity of the Reactor coolant system during an earthquake is not required. Following certification of permanent defueling, only the fuel pool structure, liner, racks and fuel are considered safety related. Those components and structures do not require functional snubbers to meet requirements during an earthquake.				3.1.3.2.3	Equipment Seismic Design
	12.2.2	Reactor Building					3.2.1	Reactor Building		
		12.2.2.1	Function		Section 12.2.2.1, Function, Modify the statement regarding the SSCs enclosed by the reactor building. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Consequently the functions provided by the primary reactor system, primary containment, reactor auxiliary and cooling systems and reactor well pool are no longer required. Following certification of permanent defueling, the reactor building functions only to house and support the spent fuel pool			3.2.1.1	Function	
		12.2.2.2	Description		Section 12.2.2.2, Description, • add the word "structure" following primary containment to reflect the fact that following permanent defueling there is no primary containment function, only the concrete structure. • Delete reference to the system and support facilities for the reactor, including Reactor Water Cleanup, Supplemental Cooling Systems, Reactor Control Rod Drive Hydraulic Systems, as well a refueling functions. The functions provided by those SSCs are not required following permanent defueling. Section 12.2.2.2, Description: • Delete all reference to corrosion of the lower part of the steel drywell liner. Following permanent defueling, the function provided by the steel liner is no longer required. Therefore, corrosion of the steel liner is not a concern. Section 12.2.2.2, Description: • Delete reference to HELB in the reactor building. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Consequently a high energy line break is no longer possible in the reactor building.			3.2.1.2	Description	
		12.2.2.3	Seismic Analysis		No Changes			3.2.1.3	Seismic Analysis	
	12.2.3	Turbine Building					3.2.2	Turbine Building		

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UFSAR Section		FSAR Conversion to DSAR Change Summary	DSAR Section	
12.2.3.1	Function	Section 12.2.3.1, Turbine Building Function: • Paragraph modified to reflect facility condition following cessation of commercial operations and certification of permanent defueling. The functions provided by the turbine generator, condensate, feedwater and water treatment systems are no longer required. The turbine building now provides miscellaneous space for auxiliary equipment.	3.2.2.1	Function
12.2.3.2	Description	Section 12.2.3.2, Turbine Building Description: • Delete portion of turbine building description stating that the TB housed power generation and related equipment. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, power generation will no longer occur. Section 12.2.3.2, Turbine Building Description: • Delete portion of turbine building description stating that blowout panels vent the TB during certain high energy line breaks. . Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, high energy line breaks in the turbine building can no longer occur.	3.2.2.2	Description
12.2.3.3	Seismic Analysis	No Changes	3.2.2.3	Seismic Analysis
12.2.4	Plant Stack		3.2.3	Plant Stack
12.2.4.1	Description	Section 12.2.4.1, Plant Stack Description: • Grammatical changes only.	3.2.3.1	Description
12.2.4.2	Seismic Analysis	No Changes	3.2.3.2	Seismic Analysis
12.2.5	Control Room Building		3.2.4	Control Room Building
12.2.5.1	Description	Section 12.2.5.1, Control Room Building, Description: • Removed reference to station operation, since station operation will not occur following certification of permanent defueling.	3.2.4.1	Description
12.2.5.2	Seismic Analysis	No Changes	3.2.4.2	Seismic Analysis
12.2.6	Circulating Water Intake and Discharge Structures		3.2.5	Circulating Water Intake and Discharge Structures
12.2.6.1	General	12.2.6.1, Circulating Water Intake and Discharge Structures, General. • Delete this section. The general description of the circulating water system is not required. . Following certification of permanent defueling the licensee is no longer authorized to emplace o retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, the functions provided by the circulating water system are no longer required and are obsolete. The intake and discharge-aerating structures are addressed in subsequent sections and are not required in this section.		
12.2.6.2	Intake Structure		3.2.5.1	Intake Structure
12.2.6.2.1	Description	12.2.6.2.1, Intake Structure, Description. • Section reworded to remove excess detail • Deleted references to the functions provided by the circulating water system. Following certification of permanent defueling power operations can no longer occur. Consequently, the functions provided by the circulating water system are no longer required and may be deleted. Circulating Water System information is obsolete.	3.2.5.1.1	Description
12.2.6.2.2	Seismic Analysis	No Changes	3.2.5.1.2	Seismic Analysis
12.2.6.3	Discharge and Aerating Structure	Section 12.2.6.3, Discharge and Aerating Structure. • Section reworded to remove excess detail. • Deleted reference to the functions provided by the circulating water system • Deleted reference to the pumps located in the discharge structure which supply water to the cooling towers. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, the functions provided by the cooling towers are no longer required. Cooling tower related information is obsolete and may be deleted.	3.2.5.2	Discharge and Aerating Structure
12.2.6.4	Cooling Tower Recirculating Water System			
12.2.6.4.1	Description	Section 12.2.6.4.1, Cooling Tower Recirculation Water System, Description. Delete this section. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Consequently, the functions provided by the cooling towers are no longer required. Cooling tower related information is obsolete and may be deleted.		
12.2.6.4.2	Seismic Analysis	Section 12.2.6.4.2, Cooling Tower Recirculation Water System, Seismic Analysis. Delete this section. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Consequently, the functions provided by the cooling towers are no longer required. Since the functions provided by the cooling towers are no longer required, a seismic analysis of the cooling towers is also not required. Cooling tower related information is obsolete and may be deleted.		

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UFSAR Section					DSAR Section			
		12.2.6.5	Cooling Tower Deep Basin	Section 12.2.6.5, Cooling Tower Deep Basin. This section has been reworded to reflect that the function of the deep basin is no longer to provide alternate cooling cell makeup in the event of a loss of the Vernon Dam. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the makeup function of the Deep Basin is no longer required.		3.2.6	Cooling Tower Deep Basin	
	12.2.7	Interim Spent Fuel Storage Installation		No Changes		3.2.7	Interim Spent Fuel Storage Installation	
		12.2.7.1	Description	No Changes			3.2.7.1	Description
		12.2.7.2	Seismic Analysis	No Changes			3.2.7.2	Seismic Analysis
	12.2.8	References		No Changes		3.1.4 3.2.8	References	
	Table 12.2.1	Allowable Stresses for Class I Structures		No Changes		Table 3.1.1	Allowable Stresses for Class I Structures	
	Table 12.2.2	Safety Margins for Several Critical Portions of Major Class I Structures		No Changes		Table 3.1.2	Safety Margins for Several Critical Portions of Major Class I Structures	
	Figure 12.2-1	G-191148 Reactor Building Plan EL. 213'-9		No Changes		Figure 3.2-1	G-191148 Reactor Building Plan EL. 213'-9	
	Figure 12.2-2	G-191148 Reactor Building Plan EL. 252'-6		No Changes		Figure 3.2-1	G-191148 Reactor Building Plan EL. 252'-6	
	Figure 12.2-3	G-191149 Reactor Building Plan EL. 280'		No Changes		Figure 3.2-2	G-191149 Reactor Building Plan EL. 280'	
	Figure 12.2-4	G-191149 Reactor Building Plan EL. 303'		No Changes		Figure 3.2-2	G-191149 Reactor Building Plan EL. 303'	
	Figure 12.2-5	G-191149 Reactor Building Plan EL. 318'-8		No Changes		Figure 3.2-2	G-191149 Reactor Building Plan EL. 318'-8	
	Figure 12.2-6	G-191149 Reactor Building Plan EL. 345'-2		No Changes		Figure 3.2-2	G-191149 Reactor Building Plan EL. 345'-2	
	Figure 12.2-7	G-191150 Reactor Building Section A-A		No Changes		Figure 3.2-3	G-191150 Reactor Building Section A-A	
	Figure 12.2-8	G-191150 Reactor Building Section B-B		No Changes		Figure 3.2-3	G-191150 Reactor Building Section B-B	
	Figure 12.2-9	G-191143 Turbine Building Basement Plan		No Changes		Figure 3.2-4	G-191143 Turbine Building Basement Plan	
	Figure 12.2-10	G-191144 Turbine Building Ground Floor Plan		No Changes		Figure 3.2-5	G-191144 Turbine Building Ground Floor Plan	
	Figure 12.2-11	G-191145 Turbine Building Operating Floor Plan		No Changes		Figure 3.2-6	G-191145 Turbine Building Operating Floor Plan	
	Figure 12.2-12	G-191146 Turbine Building Section A-A		No Changes		Figure 3.2-7	G-191146 Turbine Building Section A-A	
	Figure 12.2-13	G-191146 Turbine Building Section B-B		No Changes		Figure 3.2-7	G-191146 Turbine Building Section B-B	
	Figure 12.2-14	G-191147 Turbine Building Section C-C		No Changes		Figure 3.2-8	G-191147 Turbine Building Section C-C	
	Figure 12.2-15	G-191147 Turbine Building Section D-D		No Changes		Figure 3.2-8	G-191147 Turbine Building Section D-D	
	Figure 12.2-16	Main Stack Geometry		No Changes		Figure 3.2-18	Main Stack Geometry	
				Drawing added since it is referenced in the text body.		Figure 3.2-9	G-191142 PLOT PLAN	
	Figure 12.2-17	G-191592 Control Room Building Plan View (Typical)		No Changes		Figure 3.2-10	G-191592 Control Room Building Plan View (Typical)	
	Figure 12.2-18	G-191595 Control Room Elevation (Eastern Wall)		No Changes		Figure 3.2-11	G-191595 Control Room Elevation (Eastern Wall)	
	Figure 12.2-19	G-191595 Control Room Building Elevation (West Wall)		No Changes		Figure 3.2-11	G-191595 Control Room Building Elevation (West Wall)	

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UFSAR Section			FSAR Conversion to DSAR Change Summary	DSAR Section		
	Figure 12.2-20	G-191595 Control Room Building Elevation (North Wall)	No Changes		Figure 3.2-11	G-191595 Control Room Building Elevation (North Wall)
	Figure 12.2-21	G-191595 Control Room Building Elevation (South Wall)	No Changes		Figure 3.2-11	G-191595 Control Room Building Elevation (South Wall)
	Figure 12.2-22	G-191451 Intake Structure Masonry	No Changes		Figure 3.2-12	G-191451 Intake Structure Masonry
	Figure 12.2-23	G-191452 Intake Structure Masonry	No Changes		Figure 3.2-13	G-191452 Intake Structure Masonry
	Figure 12.2-24	G-191453 Intake Structure Masonry	No Changes		Figure 3.2-14	G-191453 Intake Structure Masonry
	Figure 12.2-25	G-191463 Discharge Structure Masonry	No Changes		Figure 3.2-15	G-191463 Discharge Structure Masonry
	Figure 12.2-26	G-191461, sh 1 Discharge Structure Masonry	No Changes		Figure 3.2-16	G-191461, sh 1 Discharge Structure Masonry
	Figure 12.2-27	G-200347 Aerating Structure Masonry and Reinforcing	No Changes		Figure 3.2-17	G-200347 Aerating Structure Masonry and Reinforcing
	Figure 12.2-28	G-200349 Circulating Water System Partial Plan and Profile	Delete Figure 12.2-28. The general description of the circulating water system is not required. . Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, the functions provided by the circulating water system are no longer required and are obsolete. The information shown on this figure is no longer applicable and is obsolete.			
	Figure 12.2-29	G-191448 Circulating Water System General Plan and Profile	Delete Figure 12.2-29. The general description of the circulating water system is not required. . Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, the functions provided by the circulating water system are no longer required and are obsolete. The information shown on this figure is no longer applicable and is obsolete.			
	Figure 12.2-30	G-200350 Circulating Water System Cooling Tower Piping Sections and Details	Delete Figure 12.2-30. The general description of the circulating water system is not required. . Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Therefore, the functions provided by the circulating water system are no longer required and are obsolete. The information shown on this figure is no longer applicable and is obsolete.			
	Figure 12.2-31	5920-3324 Cooling Tower Typical Plan	Delete Figure 12.2-31. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Consequently, the functions provided by the cooling towers are no longer required. Cooling tower related information is obsolete and may be deleted.			
	Figure 12.2-32	5920-3326 Cooling Tower Typical Section and Fan Deck Plan	Delete Figure 12.2-32. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Consequently, the functions provided by the cooling towers are no longer required. Cooling tower related information is obsolete and may be deleted.			
	Figure 12.2-33	G-200357 Cooling Tower No. 2 Basin Plan View	Delete Figure 12.2-33. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Consequently, the functions provided by the cooling towers are no longer required. Cooling tower related information is obsolete and may be deleted.			
	Figure 12.2-34	G-200357 Cooling Tower No. 2 Basin Elevation View	Delete Figure 12.2-34. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Consequently, the functions provided by the cooling towers are no longer required. Cooling tower related information is obsolete and may be deleted.			
	Figure 12.2-35	G-200353 Cooling Tower No. 1 Basin Plan	Delete Figure 12.2-35. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Consequently, the functions provided by the cooling towers are no longer required. Cooling tower related information is obsolete and may be deleted.			
	Figure 12.2-36	Cooling Tower No. 1 Basin Elevation View	Delete Figure 12.2-36. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur. Consequently, the functions provided by the cooling towers are no longer required. Cooling tower related information is obsolete and may be deleted.			
	Figure 12.2-37	G-191529 Reactor Building Reactor Vessel Pedestal Mat- M+R	No Changes		Figure 3.1-1	G-191529 Reactor Building Reactor Vessel Pedestal Mat- M+R
	Figure 12.2-38	G-191483 Reactor Building Foundation Mat Plan-M+R	No Changes		Figure 3.1-2	G-191483 Reactor Building Foundation Mat Plan-M+R
	Figure 12.2-39	5920-13400 ISFSI Concrete Storage Pad – Location Site Plan	Delete Figure 12.2-39. The figure is not referenced in UFSAR Section 12.2. This information is adequately addressed in the ISFSI SAR.			
	Figure 12.2-40	5920-13402 ISFSI Concrete Storage Pad – Plan, Sections, and Details	Delete Figure 12.2-40. The figure is not referenced in UFSAR Section 12.2. This information is adequately addressed in the ISFSI SAR.			
12.3	RADIATION SHIELDING			4.2	Radiation Shielding	

VY UFSAR					VY DSAR			
UFSAR Section				FSAR Conversion to DSAR Change Summary	DSAR Section			
12.3.1	Power Generation Objective			Modify Section 12.3.1, Radiation Shielding, Power Generation Objective, to delete the term "Power Generation" from the section title. Since the term "Power Generation" is based in the classification process described in section 1.4.1 which is no longer used, the term is no longer required and is not applicable to a permanently defueled station. Combined the power generation objective and safety objective into one section stating the objective of Radiation Shielding in the permanently defueled state.	4.2.1	Objective		
12.3.2	Safety Objective			Delete Section 12.3.2, Radiation Shielding, Safety Objective. Since the term "Safety" is based in the classification process described in section 1.4.1 which is no longer used, the term is no longer required and is not applicable to a permanently defueled station. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2).Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the safety objection of radiation				
12.3.3	Power Generation Design Basis			Delete Section 12.3.3, Radiation Shielding, Power Generation Design Basis. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2).Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related				
12.3.4	Safety Design Basis			Modify Section 12.3.4, Radiation Shielding, Safety Design Basis, to delete the term "Safety" from the section title. Since the term "Safety" is based in the classification process described in UFSAR section 1.4.1 which is no longer used, the term is no longer required and is not applicable to a permanently defueled station. Modify Section 12.3.4, Radiation Shielding, Safety Design Basis, to delete listing of “typical” areas. This information is excessive detail which is not required and is no longer applicable in the defueled state. Modify Section 12.3.4, Radiation Shielding, Safety Design Basis, to delete listing of “typical” areas within the controlled boundary areas. This information is excessive detail which is not require and may no longer be applicable. Modify Section 12.3.4, Radiation Shielding, Safety Design Basis, to delete listing of “typical” zone IV areas within the controlled boundary areas. This information is excessive detail which is not required and may no longer be applicable. Modify Section 12.3.4, Radiation Shielding, Safety Design Basis, to delete listing of areas which will be equipped with local monitoring devices. This information is excessive detail which is not required and may no longer be applicable. Delete internal reference to section 7. Internal reference not required.	4.2.2	Design Basis		
12.3.5	Description				4.2.3	Description		
	12.3.5.1	Materials Description		Modify Section 12.3.5.1, materials description to remove reference to the shielding safety design basis. Information is no longer applicable.		4.2.3.1	Materials Description	
	12.3.5.2	Reactor Building		Modify Section 12.3.5.2, Reactor Building, to remove the discussion regarding sacrificial shielding provided in the drywell. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2).Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the information regarding the functions provided by the sacrificial shielding described in this paragraph is no longer applicable and is obsolete. Modify Section 12.3.5.2, Reactor Building, to remove the discussion regarding sacrificial shielding provided in the drywell. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2).Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the information regarding the functions provided by the sacrificial shielding described in this paragraph is no longer applicable and is obsolete. Modify Section 12.3.5.2, Reactor Building, to remove the discussion regarding biological shield provided in the drywell. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2).Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the information regarding the functions provided by the biological shield described in this paragraph is no longer applicable and is obsolete.		4.2.3.2	Reactor Building	
	12.3.5.3	Turbine Building		Delete Section 12.3.5.3, Radiation Shielding, Turbine Building. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2).Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, generation of Nitrogen-16 gamma radiation is not a concern and does not contribute to the need for shielding in the turbine building. Therefore, the areas listed do not require shielding as the result of power operations. Residual radiation as the result of remaining radioactive material will be shielded on a case basis as required for personnel protection.				
	12.3.5.4	Main Control Room and Technical Support Center (TSC)		Modify Section 12.3.5.4, Main Control Room and Technical Support Center (TSC) to remove the statement regarding additional shielding to the floor above the TSC to limit direct gamma dose to personnel in the TSC during a design basis LOCA. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2).Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, a design basis LOCA is no longer possible and reference to the DB LOCA may be deleted. Modify Section 12.3.5.4, Main Control Room and Technical Support Center (TSC) to remove the statement regarding TEDE during the 30 days following a design basis accident. Following permanent defueling a design basis accident is no longer possible. Modify Section 12.3.5.4 to delete reference to operation at 100% power since VY has ceased power operations.		4.2.3.3	Main Control Room and Technical Support Center (TSC)	
	12.3.5.5	Residual Heat Removal System		Delete Section 12.3.5.5, Radiation Shielding, Residual Heat Removal System. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2).Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions provided by the Residual Heat Removal System are no longer required, the system will not be placed in operation. Residual radiation as the result of remaining radioactive material will be shielded on a case basis as required for personnel protection.				

VY UFSAR					VY DSAR			
UFSAR Section				FSAR Conversion to DSAR Change Summary	DSAR Section			
		12.3.5.6	Demineralizer System	Delete Section 12.3.5.6, Radiation Shielding, Demineralizer System. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2).Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions provided by the Demineralizer System are no longer required, the system will not be placed in operation. Residual radiation as the result of remaining radioactive material will be shielded on a case basis as required for personnel protection.				
		12.3.5.7	Advanced Off Gas System	Delete Section 12.3.5.5, Radiation Shielding, Advanced Off-Gas System. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2).Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions provided by the Advanced Off-Gas System are no longer required, the system will not be placed in operation. Residual radiation as the result of remaining radioactive material will be shielded on a case basis as required for personnel protection.				
	12.3.6	Inspection and Testing		Delete Section 12.3.6, Radiation Shielding, Inspection and Testing. This section provides historical information regarding shielding inspections conducted during the construction phase. This information is historical and may be deleted. Additionally, this section commits to radiation surveys conducted upon initiation of reactor operation. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2).Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the reactor operations will not be conducted. The information provided regarding radiation surveys conducted during power operations is no longer applicable, and is obsolete.				
	12.3.7	Surveillance and Testing		Modify Section 12.3.7, Radiation Shielding, Surveillance and Testing, to delete reference to normal station operations and response of safeguards components to accident or mechanical failure. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2).Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, normal station operations are no longer possible, and safeguard components will no longer be operated.		4.2.4	Surveillance and Testing	
	12.3.8	References		Delete Section 12.3.8, References. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2).Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the references listed in section 12.3.8 are no longer applicable and are obsolete				
	Table 12.3.1	General Occupancy Requirements and Corresponding Radiation Levels		Delete table 12.3.1, this information duplicates and is more appropriately contained in facility procedures				
	Table 12.3.2	Shielding Description at Various Areas		Delete Table 12.3.2 Design and expected radiation levels in various buildings. Radiation levels provided on table 12.3.2 which are bounding for operation at 100% power are no longer accurate or applicable since VY has ceased power operations.				
13.0	**Conduct of Operations**				5	Conduct of Operations		
13.1	SUMMARY DESCRIPTIONS			Deleted section 13.1, Conduct of Operations, Summary Description, to delete the history of ownership of the VYNPS. This information is not required in this section and is redundant to section 1.1 Deleted section 13.1, Conduct of Operations, Summary Description, The reporting relationship between the plant manager and the Site VP is redundant to Technical Specifications Delete Section 13.1. The discussion on procedures is redundant to Technical Specifications Section 6.4 Deleted reference to the pre-operational, startup and power test programs, no longer applicable following permanent defueling. Deleted reference to Emergency Operating Procedures; no longer required following permanent defueling. The remaining information in this section is either obsolete or redundant to other chapter 13 sections.				
	13.1.1	References		Deleted Section 13.1.1 References. Only reference no longer cited in body of text.				
13.2	ORGANIZATION AND RESPONSIBILITY			Modified Section 13.2, Conduct of Operations, Organization and Responsibility, to remove reference to "operating" organization and staff since the station will no longer be operating followin certification of permanent defueling. Modified the QA program manual to include "Vermont Yankee" in the title to reflect QA program manual separation from corporate.	5.1	ORGANIZATION AND RESPONSIBILITY		
13.3	TRAINING				5.2	TRAINING		
	13.3.1	Program Description (General)		Modified Section 13.3.1, Training Program Description, General to reflect the station change from operation to the safe storage and handling of irradiated fuel following certification o permanent defueling, Deleted reference to INPO, since VY is no longer associated with the national academy for nuclear training and. as such. INPO does not accredit training programs for nor operating nuclear sites.		5.2.1	Program Description (General)	
	13.3.2	General Employee Training		No changes		5.2.2	General Employee Training	
		13.3.2.1	Access to Plant	No changes			5.2.2.1	Access to Plant
	13.3.3	Fire Brigade Training		Modify section 13.3.3, Conduct of Operations, Fire Brigade Training to remove reference to Appendix R, since appendix R does not apply to permanently defueled stations.		5.2.3	Fire Brigade Training	
	13.3.4	Operations Training		Modify Section 13.3.4, Conduct of Operations, Operations Training, to remove referene to operations positions which do not exist following permanent defueling, delete reerence to the National Academy for Nuclear Training, and delete reference to the VY plant specific simulator, since the simulator has been removed from service.		5.2.4	Operations Training	

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	13.3.5	Craft, Technician, and Technical Staff Training		Modify section 13.3.5, Craft, Technician and Technical Staff Training, to remove reference to program accreditation by the national academy for nuclear training, since the INPO does not offer accreditation for non-operating nuclear sites training programs.		5.2.5	Craft, Technician, and Technical Staff Training	
	13.3.6	Plant Certification Program		Delete section 13.3.6, Conduct of Operations, Plant Certification Program. Since, following certification of permanent defueling, the licensee is no longer authorized to emplace or retain fuel i the reactor vessel, power operations can no longer occur. Since the Plant Certification Program provided selected staff with advanced knowledge of plant operation, the progran is no longer applicable.				
	13.3.7	Training Records		Modified section 13.3.7, Conduct of Operations, Training Records to reflect the fact that appropriate training records will be maintained in accordance with VY records retention policies.		5.2.6	Training Records	
	13.3.8	Training Program Approval and Evaluation		Modified Section 13.3.8 to remove referenct to Entergy Fleet Training Procedures and reflect the appropriate station organizational responsibilities post defueling.		5.2.7	Training Program Approval and Evaluation	
	13.3.9	Responsibility				5.2.8	Responsibility	
		13.3.9.1	Superintendent, Operations Training	Modified Section 13.3.9.1, Responsibility, Superintendent Operations Training, to change the title to Superintendent, Training, to reflect organizational and responsibility changes pos defueling. Deleted references to training programs no longer applicable, included applicable responsibilities formerly the role of the Superintendent, Nuclear Training, since those positions were combined as part of the post defueling organization.			5.2.8.1	Superintendent,Training
		13.3.9.2	Superintendent, Nuclear Training	Delete Secyion 13.3.9.2, Responsibility, Superintendent Nuclear Training. The applicable responsibilities of the Superintendent, Nuclear Training were transitioned to the Superintendent Training position, and the Superintendent Nuclear Training postion was eliminated.				
13.4	Radiation Protection				4.4	Radiation Protection		
	13.4.1	Health Physics		No change		4.4.1	Health Physics	
		13.4.1.1	Personnel Monitoring Systems	No change			4.4.1.1	Personnel Monitoring Systems
		13.4.1.2	Personnel Protective Equipment	No change			4.4.1.2	Personnel Protective Equipment
		13.4.1.3	Change Area and Shower Facilities	No change			4.4.1.3	Change Area and Shower Facilities
		13.4.1.4	Access Control	No change			4.4.1.4	Access Control
		13.4.1.5	Laboratory Facilities	No change			4.4.1.5	Laboratory Facilities
		13.4.1.6	Health Physics Instrumentation	No change			4.4.1.6	Health Physics Instrumentation
		13.4.1.7	Bioassay Program	No change			4.4.1.7	Bioassay Program
	13.4.2	Radioactive Materials Safety Program		No change		4.4.2	Radioactive Materials Safety Program	
		13.4.2.1	Facilities and Equipment	No change			4.4.2.1	Facilities and Equipment
		13.4.2.2	Personnel and Procedures	No change			4.4.2.2	Personnel and Procedures
		13.4.2.3	Required Materials	Modified section 13.4.2.3, Radioactive Materials Safety Program, Required Materials to reflect plant conditions following certification of permanent defueling,			4.4.2.3	Required Materials
13.5	STARTUP AND POWER TEST PROGRAM							
	13.5.1	General Objectives		Delete Section 13.5.1, Startup and Power Test Program, General Objectives. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the information provided in the Startup and Power Test Program General Objectives section is no longer applicable.				
	13.5.2	Initial Fuel Loading and Tests at Atmospheric Pressure		Delete Section 13.5.2 Startup and Power Test Program, Initial Fuel Loading and Tests at Atmospheric Pressure. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the information provided in the Startup and Power Test Program, Initial Fuel Loading and Tests at Atmospheric Pressure section is no longer applicable.				
	13.5.3	Initial Core Heatup from Ambient to Rated Temperature and Pressure		Delete Section 13.5.3, Startup and Power Test Program, Initial Core Heatup from Ambient to Rated Temperature and Pressure. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the information provided in the Startup and Power Test Program, Initial Core Heatup from Ambient to Rated Temperature and Pressure section is no longer applicable.				
	13.5.4	Initial Core Tests From Rated Temperature to 100% Power		Delete Section 13.5.4, Startup and Power Test Program, Initial Core Tests From Rated Temperature to 100% Power. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the information provided in the Startup and Power Test Program,Initial Core Tests From Rated Temperature to 100% Power section is no longer applicable.				

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	13.5.5	Reload Core Startup Testing		Delete Section 13.5.5, Startup and Power Test Program, Reload Core Startup Testing. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, the information provided in the Startup and Power Test Program,Reload Core Startup Testing section is no longer applicable.			
13.6	EMERGENCY PLAN			No Changes	5.3	EMERGENCY PLAN	
13.7	STATION RECORDS			Delete Section 13.7. The information contained in this section is redundant to the information contained in the VY QAPM. Since the UFSAR/DSAR commit to implementation of the VY QAPM, there is no need to repeat that information in the UFSAR/DSAR.			
	13.7.1	Standards		Delete this section, See 13.7 above			
	13.7.2	Definitions		Delete this section, See 13.7 above			
	13.7.3	Retention Time		Delete this section, See 13.7 above			
	13.7.4	Completed Records		Delete this section, See 13.7 above			
13.8	REVIEW AND AUDIT OF OPERATIONS				5.5	REVIEW AND AUDIT OF OPERATIONS	
	13.8.1	General		No Change		5.5.1	General
	13.8.2	On Site Safety Review Committee		No Change		5.5.2	On Site Safety Review Committee
	13.8.3	Safety Review Committee		No Change		5.5.3	Independent Safety Review
13.9	REFUELING OPERATIONS			Delete Section 13.9, Refueling Operations. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, refueling operations will no longer be conducted. The information provided in the refueling operations section is no longer applicable and is obsolete.			
	13.9.1	Regularly Scheduled Refueling Operations		Delete Section 13.9, Refueling Operations. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, refueling operations will no longer be conducted. The information provided in the refueling operations section is no longer applicable and is obsolete.			
	13.9.2	Authority		Delete Section 13.9, Refueling Operations. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, refueling operations will no longer be conducted. The information provided in the refueling operations section is no longer applicable and is obsolete.			
	13.9.3	Refueling Procedures		Delete Section 13.9, Refueling Operations. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Therefore, refueling operations will no longer be conducted. The information provided in the refueling operations section is no longer applicable and is obsolete.			
13.10	Technical Requirements Manual			Discussion modified to reflect changes to TS and the TRM as the result of certification of permanent defueling. Deleted the words: *The TRM is incorporated by reference into the DSAR, is maintained in accordance with Vermont Yankee administrative processes.* Since there is no regulatory requirement to maintain a TRM at the site, or to submit TRM changes to the NRC on a periodic basis, the TRM is not required to be incorporated by reference into the DSAR.	5.6	Technical Requirements Manual	
14.0	Station Safety Analysis				6.0	Safety Analysis	
14.1	Station Safety Analysis			Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.			
	14.1.1	OBJECTIVE		Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.			
14.2	SAFETY DESIGN LIMITS FOR ABNORMAL OPERATIONAL TRANSIENTS			Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.			
14.3	SAFETY DESIGN LIMITS FOR ACCIDENTS			Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.			
14.4	APPROACH TO SAFETY ANALYSES			Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.			
	14.4.1	General		Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.			
	14.4.2	Abnormal Operational Transients		Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.			
	14.4.3	Accidents		Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.			
	14.4.4	Barrier Damage Evaluations		Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.			
		14.4.4.1	Fuel Damage	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.			

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		14.4.4.2	Nuclear System Process Barrier Damage	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.4.4.3	Containment Damage	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
	14.4.5	References		Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
14.5	ANALYSIS OF ABNORMAL OPERATIONAL TRANSIENTS			Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
	14.5.1	Events Resulting in a Nuclear System Pressure Increase		Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.5.1.1	Generator Trip (Turbine Control Valve Fast Closure)	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.5.1.2	Turbine Trip (Turbine Stop Valve Closure)	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.5.1.3	Main Steam Line Isolation Valve Closure	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
	14.5.2	Events Resulting in a Reactor Vessel Water Temperature Decrease		Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.5.2.1	Loss of a Feedwater Heater	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.5.2.2	Shutdown Cooling (RHRS) Malfunction Decreasing	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.5.2.3	Inadvertent Pump Start	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
	14.5.3	Events Resulting in a Positive Reactivity Insertion		Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.5.3.1	Continuous Rod Withdrawal During Power Range Operation	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.5.3.2	Continuous Rod Withdrawal During Reactor Startup	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.5.3.3	Control Rod Removal Error During Refueling	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.5.3.4	Fuel Assembly Insertion Error During Refueling	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
	14.5.4	Events Resulting in a Reactor Vessel Coolant Inventory Decrease		Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.5.4.1	Pressure Regulator Failure	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.5.4.2	Inadvertent Opening of a Safety Relief Valve or Safety Valve	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.5.4.3	Loss of Feedwater Flow	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.5.4.4	Loss of Auxiliary Power	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
	14.5.5	Events Resulting in a Core Coolant Flow Decrease		Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.5.5.1	Recirculation Flow Control Failure Decreasing Flow	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.5.5.2	Trip of One Recirculation Pump	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.5.5.3	Trip of Two Recirculation Pumps	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.5.5.4	Recirculation Pump Seizure	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
	14.5.6	Events Resulting in a Core Coolant Flow Increase		Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.5.6.1	Recirculation Flow Controller Failure Increasing Flow	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.5.6.2	Startup of Idle Recirculation Pump	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
	14.5.7	Event Resulting in a Core Coolant Temperature Increase		Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.5.7.1	Loss of RHR Service Water Flow	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
	14.5.8	Event Resulting in Excess of Coolant Inventory		Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	

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		14.5.8.1	Feedwater Controller Failure Maximum Demand	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
	14.5.9	Loss of Habitability of the Main Control Room		Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.5.9.1	Criteria for Loss of Habitability of the Control Room	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.5.9.2	Conditions	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.5.9.3	Assumptions	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.5.9.4	Evaluation - Achievement of Hot Shutdown Condition	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.5.9.5	Evaluation - Achievement of the Cold Shutdown Condition	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
	14.5.10	References		Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
14.6	ANALYSIS OF DESIGN BASIS ACCIDENTS			Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
				Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
	14.6.1	Introduction		Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
	14.6.2	Control Rod Drop Accident		Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.6.2.1	Identification of Causes	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.6.2.2	Starting Conditions and Assumptions	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.6.2.3	Accident Description	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.6.2.4	Method of Analysis	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.6.2.5	Results and Consequences	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
	14.6.3	Loss of Coolant Accident (LOCA)		Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.6.3.1	Peak Containment Pressure (Reference 52)	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.6.3.2	Long-Term Primary Containment Response	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.6.3.3	Over-Pressure Required for CS and RHR Pump NPSH	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.6.3.4	Metal Water Reaction Effects on the Primary Containment	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.6.3.5	Radiological Consequences	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
	14.6.4	Refueling Accident		Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.6.4.1	Identification of Causes	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.6.4.2	Methods, Assumptions, and Conditions	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.6.4.3	Results and Consequences	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.6.4.4	Fission Product Release to Environs	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.6.4.5	Radiological Effects	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
	14.6.5	Main Steam Line Break Accident		Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.6.5.1	Nuclear System Transient Effects	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.6.5.2	Deleted	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
		14.6.5.3	Radiological Effects	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	
	14.6.6	References		Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.	

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14.7	CONCLUSIONS	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.			
14.8	ANALYTICAL METHODS	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.			
14.8.1	Nuclear Excursion Analysis	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.			
14.8.2	Reactor Vessel Depressurization Analysis	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.			
14.8.3	Reactor Heatup Analysis	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.			
14.8.4	Containment Response Analysis	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.			
14.8.5	Analytical Methods for Evaluating Radiological Effects	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.			
14.8.6	References	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.			
14.9	RADIOLOGICAL CONSEQUENCES FOR NON DESIGN BASIS ACCIDENTS WITH A TID–14844 SOURCE TERM	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.			
14.9.1	Evaluation of Station Systems Using TID 14844 Source Term	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.			
14.9.1.1	Source Term Assumptions	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.			
14.9.1.2	Standby Gas Treatment System	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.			
14.9.1.3	ECCS Components	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.			
14.9.1.4	Electrical Penetrations	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.			
14.9.1.5	Control Room Habitability Performed in Response to NUREG-0737	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.			
14.9.1.6	Materials Within the Containment	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.			
14.9.1.7	Conclusion	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.			
14.9.2	Radiological Consequences of Accidents with a TID-1488 Source Term	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.			
14.9.3	References	Deleted, See FCR 27-006 and FCR 27-020 for justification and explanation of changes.			
14.1	Introduction	See FCR 27-006 and FCR 27-020 for justification and explanation of changes. The VY accident and transient analyses were replaced with new analyses of accidents and events which are applicable in the permanently defueled state with irradiated fuel stored in the spent fuel pool. The new analysis address a fuel handling accident over the spent fuel pool and a drop of a high integrity radioactive material container.		6.1	Introduction
14.2	Acceptance Criteria	See FCR 27-006 and FCR 27-020 for justification and explanation of changes. The VY accident and transient analyses were replaced with new analyses of accidents and events which are applicable in the permanently defueled state with irradiated fuel stored in the spent fuel pool. The new analysis address a fuel handling accident over the spent fuel pool and a drop of a high integrity radioactive material container.		6.2	Acceptance Criteria
14.2.1	DBA Acceptance Criteria	See FCR 27-006 and FCR 27-020 for justification and explanation of changes. The VY accident and transient analyses were replaced with new analyses of accidents and events which are applicable in the permanently defueled state with irradiated fuel stored in the spent fuel pool. The new analysis address a fuel handling accident over the spent fuel pool and a drop of a high integrity radioactive material container.		6.2.1	DBA Acceptance Criteria
14.2.2	Site Event Acceptance Criteria	See FCR 27-006 and FCR 27-020 for justification and explanation of changes. The VY accident and transient analyses were replaced with new analyses of accidents and events which are applicable in the permanently defueled state with irradiated fuel stored in the spent fuel pool. The new analysis address a fuel handling accident over the spent fuel pool and a drop of a high integrity radioactive material container.		6.2.2	Site Event Acceptance Criteria
14.3	Accidents Evaluated	See FCR 27-006 and FCR 27-020 for justification and explanation of changes. The VY accident and transient analyses were replaced with new analyses of accidents and events which are applicable in the permanently defueled state with irradiated fuel stored in the spent fuel pool. The new analysis address a fuel handling accident over the spent fuel pool and a drop of a high integrity radioactive material container.		6.3	Accidents Evaluated
14.3.1	Fuel Handling Accident	See FCR 27-006 and FCR 27-020 for justification and explanation of changes. The VY accident and transient analyses were replaced with new analyses of accidents and events which are applicable in the permanently defueled state with irradiated fuel stored in the spent fuel pool. The new analysis address a fuel handling accident over the spent fuel pool and a drop of a high integrity radioactive material container.		6.3.1	Fuel Handling Accident
14.3.1.1	Analytical Methodology	See FCR 27-006 and FCR 27-020 for justification and explanation of changes. The VY accident and transient analyses were replaced with new analyses of accidents and events which are applicable in the permanently defueled state with irradiated fuel stored in the spent fuel pool. The new analysis address a fuel handling accident over the spent fuel pool and a drop of a high integrity radioactive material container.		6.3.1.1	Analytical Methodology
14.3.1.2	Assembly Drop in SFP with Open Containment Scenario	See FCR 27-006 and FCR 27-020 for justification and explanation of changes. The VY accident and transient analyses were replaced with new analyses of accidents and events which are applicable in the permanently defueled state with irradiated fuel stored in the spent fuel pool. The new analysis address a fuel handling accident over the spent fuel pool and a drop of a high integrity radioactive material container.		6.3.1.2	Assembly Drop in SFP with Open Containment Scenario

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		14.3.1.3	Assembly Drop in SFP with Closed Containment Scenario	See FCR 27-006 and FCR 27-020 for justification and explanation of changes. The VY accident and transient analyses were replaced with new analyses of accidents and events which are applicable in the permanently defueled state with irradiated fuel stored in the spent fuel pool. The new analysis address a fuel handling accident over the spent fuel pool and a drop of a high integrity radioactive material container.			6.3.1.3	Assembly Drop in SFP with Closed Containment Scenario
		14.3.1.4	Software	See FCR 27-006 and FCR 27-020 for justification and explanation of changes. The VY accident and transient analyses were replaced with new analyses of accidents and events which are applicable in the permanently defueled state with irradiated fuel stored in the spent fuel pool. The new analysis address a fuel handling accident over the spent fuel pool and a drop of a high integrity radioactive material container.			6.3.1.4	Software
		14.3.1.5	Assumptions	See FCR 27-006 and FCR 27-020 for justification and explanation of changes. The VY accident and transient analyses were replaced with new analyses of accidents and events which are applicable in the permanently defueled state with irradiated fuel stored in the spent fuel pool. The new analysis address a fuel handling accident over the spent fuel pool and a drop of a high integrity radioactive material container.			6.3.1.5	Assumptions
		14.3.1.6	Inputs	See FCR 27-006 and FCR 27-020 for justification and explanation of changes. The VY accident and transient analyses were replaced with new analyses of accidents and events which are applicable in the permanently defueled state with irradiated fuel stored in the spent fuel pool. The new analysis address a fuel handling accident over the spent fuel pool and a drop of a high integrity radioactive material container.			6.3.1.6	Inputs
		14.3.1.7	Impact of Water Depth on Iodine Decontamination Factor	See FCR 27-006 and FCR 27-020 for justification and explanation of changes. The VY accident and transient analyses were replaced with new analyses of accidents and events which are applicable in the permanently defueled state with irradiated fuel stored in the spent fuel pool. The new analysis address a fuel handling accident over the spent fuel pool and a drop of a high integrity radioactive material container.			6.3.1.7	Impact of Water Depth on Iodine Decontamination Factor
		14.3.1.8	Fuel Damage from Assembly Drop onto SFP Fuel Racks	See FCR 27-006 and FCR 27-020 for justification and explanation of changes. The VY accident and transient analyses were replaced with new analyses of accidents and events which are applicable in the permanently defueled state with irradiated fuel stored in the spent fuel pool. The new analysis address a fuel handling accident over the spent fuel pool and a drop of a high integrity radioactive material container.			6.3.1.8	Fuel Damage from Assembly Drop onto SFP Fuel Racks
		14.3.1.9	Radiological Consequences/Results	See FCR 27-006 and FCR 27-020 for justification and explanation of changes. The VY accident and transient analyses were replaced with new analyses of accidents and events which are applicable in the permanently defueled state with irradiated fuel stored in the spent fuel pool. The new analysis address a fuel handling accident over the spent fuel pool and a drop of a high integrity radioactive material container.			6.3.1.9	Radiological Consequences/Results
14.4		Site Events Evaluated		See FCR 27-006 and FCR 27-020 for justification and explanation of changes. The VY accident and transient analyses were replaced with new analyses of accidents and events which are applicable in the permanently defueled state with irradiated fuel stored in the spent fuel pool. The new analysis address a fuel handling accident over the spent fuel pool and a drop of a high integrity radioactive material container.	6.4		Site Events Evaluated	
	14.4.1		High Integrity Container (HIC) Drop Ev	See FCR 27-006 and FCR 27-020 for justification and explanation of changes. The VY accident and transient analyses were replaced with new analyses of accidents and events which are applicable in the permanently defueled state with irradiated fuel stored in the spent fuel pool. The new analysis address a fuel handling accident over the spent fuel pool and a drop of a high integrity radioactive material container.		6.4.1		High Integrity Container (HIC) Drop Event
		14.4.1.1	Analytical Methodology	See FCR 27-006 and FCR 27-020 for justification and explanation of changes. The VY accident and transient analyses were replaced with new analyses of accidents and events which are applicable in the permanently defueled state with irradiated fuel stored in the spent fuel pool. The new analysis address a fuel handling accident over the spent fuel pool and a drop of a high integrity radioactive material container.			6.4.1.1	Analytical Methodology
		14.4.1.2	Assumptions	See FCR 27-006 and FCR 27-020 for justification and explanation of changes. The VY accident and transient analyses were replaced with new analyses of accidents and events which are applicable in the permanently defueled state with irradiated fuel stored in the spent fuel pool. The new analysis address a fuel handling accident over the spent fuel pool and a drop of a high integrity radioactive material container.			6.4.1.2	Assumptions
		14.4.1.3	Inputs	See FCR 27-006 and FCR 27-020 for justification and explanation of changes. The VY accident and transient analyses were replaced with new analyses of accidents and events which are applicable in the permanently defueled state with irradiated fuel stored in the spent fuel pool. The new analysis address a fuel handling accident over the spent fuel pool and a drop of a high integrity radioactive material container.			6.4.1.3	Inputs
		14.4.1.4	Radiological Consequences/Results	See FCR 27-006 and FCR 27-020 for justification and explanation of changes. The VY accident and transient analyses were replaced with new analyses of accidents and events which are applicable in the permanently defueled state with irradiated fuel stored in the spent fuel pool. The new analysis address a fuel handling accident over the spent fuel pool and a drop of a high integrity radioactive material container.			6.4.1.4	Radiological Consequences/Results
14.5	References			See FCR 27-006 and FCR 27-020 for justification and explanation of changes. The VY accident and transient analyses were replaced with new analyses of accidents and events which are applicable in the permanently defueled state with irradiated fuel stored in the spent fuel pool. The new analysis address a fuel handling accident over the spent fuel pool and a drop of a high integrity radioactive material container. Reference 9 changed from NUMARC 93-01 to NUMARC 91-06, Guidelines for Industry Actions to Assess Shutdown Management, in the References Section. 91-06 is the correct reference and agrees with the text body.	6.5	References		
14.6	Appendices			See FCR 27-006 and FCR 27-020 for justification and explanation of changes. The VY accident and transient analyses were replaced with new analyses of accidents and events which are applicable in the permanently defueled state with irradiated fuel stored in the spent fuel pool. The new analysis address a fuel handling accident over the spent fuel pool and a drop of a high integrity radioactive material container. Section 6.6 – Appendix A, Methodology – Step 2, deleted the sentence: *"Given the complexities of the fuel bundle system and its response impact event, it is not possible to judge the level of conservatism in this set of assumptions, however, the high levels of conservatism in other areas of this calculation should more that make up for any non-conservatism introduced here."* This statement is an engineering judgement estimation of conservatism which is not quantifiable. The statement is not required since adequate conservatism already exists in the calculation. Therefore, since the statement is not quantifiabe and is not required, is has been deleted.	6.6	Appendices		
	Table 14.3.1	FHA Scenarios Analyzed		See FCR 27-006 and FCR 27-020 for justification and explanation of changes. The VY accident and transient analyses were replaced with new analyses of accidents and events which are applicable in the permanently defueled state with irradiated fuel stored in the spent fuel pool. The new analysis address a fuel handling accident over the spent fuel pool and a drop of a high integrity radioactive material container.		Table 6.3.1	FHA Scenarios Analyzed	

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	Table 14.3.2	Input Conditions for FHA	See FCR 27-006 and FCR 27-020 for justification and explanation of changes. The VY accident and transient analyses were replaced with new analyses of accidents and events which are applicable in the permanently defueled state with irradiated fuel stored in the spent fuel pool. The new analysis address a fuel handling accident over the spent fuel pool and a drop of a high integrity radioactive material container.		Table 6.3.2	Input Conditions for FHA
	Table 14.3.3	Undecayed Core Inventory for Radionuclides Important in the Radiological Evaluation of DBAs	See FCR 27-006 and FCR 27-020 for justification and explanation of changes. The VY accident and transient analyses were replaced with new analyses of accidents and events which are applicable in the permanently defueled state with irradiated fuel stored in the spent fuel pool. The new analysis address a fuel handling accident over the spent fuel pool and a drop of a high integrity radioactive material container.		Table 6.3.3	Undecayed Core Inventory for Radionuclides Important in the Radiological Evaluation of DBAs
	Table 14.3.4	Undecayed Gap Activity Available for Release from Fuel Assembly Drop in the SFP	See FCR 27-006 and FCR 27-020 for justification and explanation of changes. The VY accident and transient analyses were replaced with new analyses of accidents and events which are applicable in the permanently defueled state with irradiated fuel stored in the spent fuel pool. The new analysis address a fuel handling accident over the spent fuel pool and a drop of a high integrity radioactive material container.		Table 6.3.4	Undecayed Gap Activity Available for Release from Fuel Assembly Drop in the SFP
	Table 14.3.5	Typical Iodine Decontamination Factors and Iodine Speciation vs Water Depth above Dropped Assembly	See FCR 27-006 and FCR 27-020 for justification and explanation of changes. The VY accident and transient analyses were replaced with new analyses of accidents and events which are applicable in the permanently defueled state with irradiated fuel stored in the spent fuel pool. The new analysis address a fuel handling accident over the spent fuel pool and a drop of a high integrity radioactive material container.		Table 6.3.5	Typical Iodine Decontamination Factors and Iodine Speciation vs Water Depth above Dropped Assembly
	Table 14.3.6	Atmospheric Dispersion Factors for the Postulated FHA	See FCR 27-006 and FCR 27-020 for justification and explanation of changes. The VY accident and transient analyses were replaced with new analyses of accidents and events which are applicable in the permanently defueled state with irradiated fuel stored in the spent fuel pool. The new analysis address a fuel handling accident over the spent fuel pool and a drop of a high integrity radioactive material container.		Table 6.3.6	Atmospheric Dispersion Factors for the Postulated FHA
	Table 14.3.7	EAB TEDE Dose vs Long Decay Time	See FCR 27-006 and FCR 27-020 for justification and explanation of changes. The VY accident and transient analyses were replaced with new analyses of accidents and events which are applicable in the permanently defueled state with irradiated fuel stored in the spent fuel pool. The new analysis address a fuel handling accident over the spent fuel pool and a drop of a high integrity radioactive material container.		Table 6.3.7	EAB TEDE Dose vs Long Decay Time
	Figure 14.3-1	VY FHA – EAB TEDE Dose vs Decay Time	See FCR 27-006 and FCR 27-020 for justification and explanation of changes. The VY accident and transient analyses were replaced with new analyses of accidents and events which are applicable in the permanently defueled state with irradiated fuel stored in the spent fuel pool. The new analysis address a fuel handling accident over the spent fuel pool and a drop of a high integrity radioactive material container.		Figure 6.3-1	VY FHA – EAB TEDE Dose vs Decay Time
	Figure 14.3-2	VY FHA – MCR TEDE Dose vs Decay Time	See FCR 27-006 and FCR 27-020 for justification and explanation of changes. The VY accident and transient analyses were replaced with new analyses of accidents and events which are applicable in the permanently defueled state with irradiated fuel stored in the spent fuel pool. The new analysis address a fuel handling accident over the spent fuel pool and a drop of a high integrity radioactive material container.		Figure 6.3-2	VY FHA – MCR TEDE Dose vs Decay Time
15.0	Aging Management Program			7.0	Aging Management Program	
15.1	SUPPLEMENT FOR RENEWED OPERATING LICENSE		Modify section 15.1, Supplement for Renewed Operating License, to delete reference to the evaluation of time-limited aging analyses for the period of exended operation. On January 12, 2015, Entergy Nuclear Operations (ENO) certified to the Nuclear Regulatory Commission (NRC) that a determination to permanently cease operation at the Vermont Yankee Nuclear Power Station (VYNPS) was made on December 29, 2014 which was the date on which operation ceased at VYNPS. ENO also certified that the fuel has been permanently removed from the VYNPS reactor vessel and placed in the spent fuel pool. ENO acknowledged that, following docketing, the VYNPS license no longer authorized operation of the reactor or emplacement or retention of fuel into the reactor vessel. Consequently , the period of extended operation has ceased and the evaluation of time-limited aging analysis is no longer required.	7.1	Supplement for Renewed Operating License	
15.2	AGING MANAGEMENT PROGRAMS AND ACTIVITIES		On January 12, 2015, Entergy Nuclear Operations (ENO) certified to the Nuclear Regulatory Commission (NRC) that a determination to permanently cease operation at the Vermont Yankee Nuclear Power Station (VYNPS) was made on December 29, 2014 which was the date on which operation ceased at VYNPS. ENO also certified that the fuel has been permanently removed from the VYNPS reactor vessel and placed in the spent fuel pool. ENO acknowledged that, following docketing, the VYNPS license no longer authorized operation of the reactor or emplacement or retention of fuel into the reactor vessel. Cosequently, the following changes were made to Section 15.2 of the VYNPS UFSAR: 1) Eliminate discussions of License Renewal aging management programs, analyses, and commitments that are no longer required to be maintained to support the current condition of the plant; 2) Modify the scopes of programs and commitments to reflect the permanently shutdown and defueled condition; and 3) Make administrative changes including the renumbering and restructuring of the Chapter and its subsections. Refer to FCR 27-014. Specific changes to Section 15.2, Aging Management Programs and Activities include: • change "period of extended operation" to "period of wet fuel storage", since, as described in change summary for section 15.1, the period of extended operation ceased following cessation of operations. • Change "Entergy Quality Assurance Program" to "Quality Assurance Program" per FCR 27-13	7.2	AGING MANAGEMENT PROGRAMS AND ACTIVITIES	
	15.2.1	Buried Piping Inspection Program	No Changes		7.2.1	Buried Piping Inspection Program
	15.2.2	BWR CRD Return Line Nozzle Program	Delete Section 15.2.2, BWR CRD Return Line Nozzle Program, program is not required following certification of permanent defueling. See Section 15.2 Summary of Changes.			
	15.2.3	BWR Feedwater Nozzle Program	Delete Section 15.2.3, BWR Feedwater Nozzle Program, program is not required following certification of permanent defueling. See Section 15.2 Summary of Changes.			
	15.2.4	BWR Penetrations Program	Deleete Section 15.2.4, BWR Penetrations Program, program is not required following certification of permanent defueling. See Section 15.2 Summary of Changes.			
	15.2.5	BWR Stress Corrosion Cracking Program	Delete Section 15.2.5, BWR Stress Corrosion Cracking Program, program is not required following certification of permanent defueling. See Section 15.2 Summary of Changes.			

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15.2.6	BRW Vessel ID Attachment Welds Program	Delete Section 15.2.6, BRW Vessel ID Attachment Welds Program, program is not required following certification of permanent defueling. See Section 15.2 Summary of Changes.			
15.2.7	BWR Vessel Internals Program	Delete Section 15.2.7, BWR Vessel Internals Program, program is not required following certification of permanent defueling. See Section 15.2 Summary of Changes.			
15.2.8	Containment Leak Rate Program	Delete Section 15.2.8, Containment Leak Rate Program, program is not required following certification of permanent defueling. See Section 15.2 Summary of Changes.			
15.2.9	Diesel Fuel Monitoring Program	No Changes		7.2.2	Diesel Fuel Monitoring Program
15.2.10	Environmental Qualification (EQ) of Electric Components Program	Delete Section 15.2.10, Environmental Qualification (EQ) of Electric Components Program, program is not required following certification of permanent defueling. See Section 15.2 Summary o Changes and FCR 27-010.			
15.2.11	Fatigue Monitoring Program	Delete Section 15.2.11 , Fatigue Monitoring Program			
15.2.12	Fire Protection Program	No Changes		7.2.3	Fire Protection Program
15.2.13	Fire Water System Program	No Changes		7.2.4	Fire Water System Program
15.2.14	Flow-Accelerated Corrosion Program	Delete Section 15.2.14 , Flow-Accelerated Corrosiong Program, program is not required following certification of permanent defueling. See Section 15.2 Summary of Changes.			
15.2.15	Heat Exchanger Monitoring Program	Delete Section 15.2.15 , Heat Exchanger Monitoring Program, program is not required following certification of permanent defueling. See Section 15.2 Summary of Changes.			
15.2.16	Inservice Inspection - Containment Inservice Inspection (CII) Program	Delete Section 15.2.16 , Inservice Inspection - Containment Inservice Inspection (CII) Program, program is not required following certification of permanent defueling. See Section 15.2 Summary of Changes.			
15.2.17	Inservice Inspection – Inservice Inspection (ISI) Program	Delete Section 15.2.17 , Inservice Inspection – Inservice Inspection (ISI) Program, program is not required following certification of permanent defueling. See Section 15.2 Summary of Changes			
15.2.18	Instrument Air Quality Program	No Changes		7.2.5	Instrument Air Quality Program
15.2.19	Non-EQ Inaccessible Medium-Voltage Cable Program	Modify Section 15.2.19, Non-EQ Inaccessible Medium-Voltage Cable Program, to remove requirement related to first test being performed prior to the period of extended operation with the exception of the 4.16 kV cables between the UATs and Bus 1 and 2.		7.2.6	Non-EQ Inaccessible Medium-Voltage Cable Program
15.2.20	Non-EQ Instrumentation Circuits Test Review Program	Delete Section 15.2.20, Non-EQ Instrumentation Circuits Test Review Program, program is not required following certification of permanent defueling. See Section 15.2 Summary of Changes.			
15.2.21	Non-EQ Insulated Cables and Connections Program	Delete Section 15.2.21, Non-EQ Insulated Cables and Connections Program, program is not required following certification of permanent defueling. See Section 15.2 Summary of Changes.			
15.2.22	Oil Analysis Program	Modify Section 15.2.22, Oil Analysis Program, to change the word "plant" to "facility".		7.2.7	Oil Analysis Program
15.2.23	One-Time Inspection Program	Delete Section 15.2.23,One-time Inspection Program, program is not required following certification of permanent defueling. See Section 15.2 Summary of Changes.			
15.2.24	Periodic Surveillance and Preventive Maintenance Program	Modify Section 15.2.24, Periodic Surveillance and Preventive Maintenance Program, to • change the word "plant" to "facility". • remove the HPCI gland seal exhauster fan housing and piping, hydrogen analyzer pre-cooler, housings in the RHR corner room, EECS (stet) corner room recirculation units and circulating water from the inspection list.		7.2.8	Periodic Surveillance and Preventive Maintenance Program
15.2.25	Reactor Head Closure Studs Program	Delete Section 15.2.25, Reactor Head Closure Studs Program, program is not required following certification of permanent defueling. See Section 15.2 Summary of Changes.			
15.2.26	Reactor Vessel Surveillance Program	Delete Section 15.2.26, Reactor Vessel Surveillance Program, program is not required following certification of permanent defueling. See Section 15.2 Summary of Changes.			
15.2.27	Selective Leaching Program	Delete Section 15.2.27, Selective Leaching Program, program is not required following certification of permanent defueling. See Section 15.2 Summary of Changes.			
15.2.28	Service Water Integrity Program	Modify Section 15.2.28, Service Water Integrity Program, to: · delete reference to GL 89-13 · change "period of extended operation" to "period of wet fuel storage", since, as described in change summary for section 15.1, the period of extended operation ceased following cessation o operations. · Insert the word "opportunistic" to describe component inspections · Insert the words "and non-safety related" to further describe the heat exchangers to be inspected as a part of this program		7.2.9	Service Water Integrity Program
15.2.29	Structures Monitoring – Masonry Wall Program	Modify Section 15.2.29,Structures Monitoring – Masonry Wall Program, to: · change "period of extended operation" to "period of wet fuel storage", since, as described in change summary for section 15.1, the period of extended operation ceased following cessation o operations.		7.2.10	Structures Monitoring – Masonry Wall Program
15.2.30	Structures Monitoring – Structures Monitoring Program	No Changes		7.2.11	Structures Monitoring – Structures Monitoring Program
15.2.31	Structures Monitoring – Vernon Dam FERC Program	Delete Section 15.2.31, Structures Monitoring – Vernon Dam FERC Program, program is not required following certification of permanent defueling. See Section 15.2 Summary of Changes.			

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	15.2.32	System Walkdown Program		Modify Section 15.2.32, System Walkdown Program, to: · change "period of extended operation" to "period of wet fuel storage", since, as described in change summary for section 15.1, the period of extended operation ceased following cessation o operations.		7.2.12	System Walkdown Program
	15.2.33	Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program		Delete Section 15.2.33, Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program, program is not required following certification of permanent defueling. See Section 15.2 Summary of Changes.			
	15.2.34	Water Chemistry Control – Auxiliary Systems Program		Modify Section 15.2.34, Water Chemistry Control – Auxiliary Systems Program, to: · delete reference to the Stator Cooling Water System · change the word "plant" to "facility". · delete reference to the One-Time Inspection Program, since that program has been deleted.		7.2.13	Water Chemistry Control – Auxiliary Systems Program
	15.2.35	Water Chemistry Control – BWR Program		Modify Section 15.2.35, Water Chemistry Control –BWR Program, to: · delete reference to the EPRI Report 1008192 BWRVIP-130 · insert reference to BWR Water Chemistry Guidlines, 2008 Revision, BWRVIP-190 · delete reference to primary water chemistry and hydrogen water chemistry · delete reference to the One-Time Inspection Program, since that program has been deleted.		7.2.14	Water Chemistry Control – BWR Program
	15.2.36	Water Chemistry Control – Closed Cooling Program		Modify Section 15.2.36, Water Chemistry Control –Closed Cooling Program, to: · delete reference to the TBCCW system, AOG Closed Cooling Water system, and AOG Refrigerant skid water system. · delete reference to the One-Time Inspection Program, since that program has been deleted.		7.2.15	Water Chemistry Control – Closed Cooling Program
	15.2.37	Bolting Integrity Program		No Changes		7.2.16	Bolting Integrity Program
	15.2.38	Metal Enclosed Bus Inspection Program		No Changes		7.2.17	Metal Enclosed Bus Inspection Program
	15.2.39	Bolted Cable Connections Program		No Changes		7.2.18	Bolted Cable Connections Program
	15.2.40	Neutron Absorber Monitoring Program		No Changes		7.2.19	Neutron Absorber Monitoring Program
	15.2.41	Protective Coating Monitoring and Maintenance Program		Delete Section 15.2.41, Protective Coating Monitoring and Maintenance Program, program is not required following certification of permanent defueling. See Section 15.2 Summary o Changes.			
15.3	EVALUATION OF TIME-LIMITED AGING ANALYSES			Delete section 15.3, Evaluation of Timel-Limited Aging Analyses. On January 12, 2015, Entergy Nuclear Operations (ENO) certified to the Nuclear Regulatory Commission (NRC) that a determination to permanently cease operation at the Vermont Yankee Nuclear Power Station (VYNPS) was made on December 29, 2014 which was the date on which operation ceased at VYNPS. ENO also certified that the fuel has been permanently removed from the VYNPS reactor vessel and placed in the spent fuel pool. ENO acknowledged that, following docketing, the VYNPS license no longer authorized operation of the reactor or emplacement or retention of fuel into the reactor vessel. Consequently , the period of extended operation has ceased and the evaluation of time-limited aging analysis is no longer required.			
	15.3.1	Reactor Vessel Neutron Embrittlement		Delete section 15.3.1, Reactor Vessel Neutron Embrittlement, information is not required following certification of permanent defueling. See Section 15.3 Summary of Changes.			
		15.3.1.1	Reactor Vessel Fluence	Delete section 15.3.1.1, Reactor Vessel Fluence, information is not required following certification of permanent defueling. See Section 15.3 Summary of Changes.			
		15.3.1.2	Pressure-Temperature Limits	Delete section 15.3.1.2, Pressure-Temperature Limits, information is not required following certification of permanent defueling. See Section 15.3 Summary of Changes.			
		15.3.1.3	Charpy Upper-Shelf Energy	Delete section 15.3.1.3, Charpy Upper-Shelf Energy, information is not required following certification of permanent defueling. See Section 15.3 Summary of Changes.			
		15.3.1.4	Adjusted Reference Temperature	Delete section 15.3.1.4, Adjusted Reference Temperature, information is not required following certification of permanent defueling. See Section 15.3 Summary of Changes.			
		15.3.1.5	Reactor Vessel Circumferential Weld Inspection Relief	Delete section 15.3.1.5, Reactor Vessel Circumferential Weld Inspection Relief, information is not required following certification of permanent defueling. See Section 15.3 Summary of Changes.			
		15.3.1.6	Reactor Vessel Axial Weld Failure Probability	Delete section 15.3.1.6, Reactor Vessel Axial Weld Failure Probability			
	15.3.2	Metal Fatigue		Delete section 15.3.2, Metal Fatigue, information is not required following certification of permanent defueling. See Section 15.3 Summary of Changes.			
		15.3.2.1	Class 1 Metal Fatigue	Delete section 15.3.2.1, Class 1 Metal Fatigue, information is not required following certification of permanent defueling. See Section 15.3 Summary of Changes.			
		15.3.2.2	Non-Class 1 Metal Fatigue	Delete section 15.3.2.2, Non-Class 1 Metal Fatigue, information is not required following certification of permanent defueling. See Section 15.3 Summary of Changes.			
		15.3.2.3	Environmental Effects on Fatigue	Delete section 15.3.2.3, Environmental Effects on Fatigue, information is not required following certification of permanent defueling. See Section 15.3 Summary of Changes.			
	15.3.3	Environmental Qualification of Electrical Components		Delete section 15.3.3, Environmental Qualification of Electrical Components, information is not required following certification of permanent defueling. See Section 15.3 Summary of Changes.			
	15.3.4	Fatigue of Primary Containment, Attached Piping, and Components		Delete section 15.3.4, Fatigue of Primary Containment, Attached Piping, and Components, information is not required following certification of permanent defueling. See Section 15.3 Summary of Changes.			

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	15.3.5	Core Plate Rim Hold-Down Bolt Loss of Preload	Delete section 15.3.5, Core Plate Rim Hold-Down Bolt Loss of Preload, information is not required following certification of permanent defueling. See Section 15.3 Summary of Changes.			
	15.3.6	Lower Plenum Fatigue Analysis	Delete section 15.3.6, Lower Plenum Fatigue Analysis			
15.4	REFERENCES		Modify Section 15.4, Referenes, to remove those references no longer cited in the body of the text.	15.3	REFERENCES	
15.5	LIST OF LICENSE RENEWAL COMMITMENTS		Text modified to add " commitments remaining applicable following permanent cessation of operations and certification of permanent defueling. Modify Section 15.4, References, to remove those references no longer cited in the body of the text.	15.4	LIST OF LICENSE RENEWAL COMMITMENTS	
	1	Guidance for performing examinations of buried piping	No Changes		1	Guidance for performing examinations of buried piping
	2	Inspect Fifteen (15) percent of the top guide locations	Commitment Eliminated by FCR 27/014			
	3	Enhance The Diesel Fuel Monitoring Program UT Tank bottom every 10 yrs	No Changes		3	Enhance The Diesel Fuel Monitoring Program UT Tank bottom every 10 yrs
	4	Enhance The Diesel Fuel Monitoring Program modify tank bottom acceptance criteria	No Changes		4	Enhance The Diesel Fuel Monitoring Program modify tank bottom acceptance criteria
	5	Modify Fatigue Monitoring Program	Commitment Eliminated by FCR 27/014			
	6	Use manual cycle counting to track and compare accumulated cycles	Commitment Eliminated by FCR 27/014			
	7	Establish allowable number of effective transients	Commitment Eliminated by FCR 27/014			
	8	Enhance procedures to specify that fire damper frames in fire barriers will be inspected for corrosion	No Changes		8	Enhance procedures to specify that fire damper frames in fire barriers will be inspected for corrosion
	9	Enhance procedures to state that the diesel engine sub-systems (including the fuel supply line) will be observed while the pump is running	No Changes		9	Enhance procedures to state that the diesel engine sub-systems (including the fuel supply line) will be observed while the pump is running
	10	Enhance Fire Water System Progam Procedures to specify that a representative sample of sprinklers which have been in service for 50 years will be tested.	No Changes		10	Enhance Fire Water System Progam Procedures to specify that a representative sample of sprinklers which have been in service for 50 years will be tested.
	11	Enhance the Fire Water System Program to specify that wall thickness evaluations of fire protection piping will be performed on system components using non-intrusive techniques to identify evidence of loss of material due to corrosion.	No Changes		11	Enhance the Fire Water System Program to specify that wall thickness evaluations of fire protection piping will be performed on system components using non-intrusive techniques to identify evidence of loss of material due to corrosion.
	12	Implement Heat Exchanger Monitoring Program	Commitment Eliminated by FCR 27/014			
	13	Implement the Non-EQ Inaccessible Medium-Voltage Cable Program	No Changes		13	Implement the Non-EQ Inaccessible Medium-Voltage Cable Program
	14	Implement the Non-EQ Instrumentation Circuits Test Review Program	Commitment Eliminated by FCR 27/014			
	15	Implement the Non-EQ Insulated Cables and Connections Program	Commitment Eliminated by FCR 27/014			
	16	Implement the One-Time Inspection Program	The One-Time Inspection Program was performed; thus, the commitment was met and closed by FCR 27/014.			

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17	Enhance the Periodic Surveillance and Preventive Maintenance Program to assure that the effects of aging will be managed	No Changes	17	Enhance the Periodic Surveillance and Preventive Maintenance Program to assure that the effects of aging will be managed
18	Enhance the Reactor Vessel Surveillance Program to proceduralize the data analysis, acceptance criteria, and corrective actions described in the program description	Commitment Eliminated by FCR 27/014		
19	Implement the Selective Leaching Program	The Selective Leaching Program was performed; thus, the commitment was met and closed by FCR 27/014.		
20	Enhance the Structures Monitoring Program to specify that process facility crane rails and girders, condensate storage tank (CST) enclosure, CO2 tank enclosure, nitrogen tank enclosure and restraining wall, CST pipe trench, diesel generator cable trench, fuel oil pump house, service water pipe trench, man-way seals and gaskets, and hatch seals and gaskets are included in the program.	No Changes	20	Enhance the Structures Monitoring Program to specify that process facility crane rails and girders, condensate storage tank (CST) enclosure, CO2 tank enclosure, nitrogen tank enclosure and restraining wall, CST pipe trench, diesel generator cable trench, fuel oil pump house, service water pipe trench, man-way seals and gaskets, and hatch seals and gaskets are included in the program.
21	Add guidance for performing structural examinations of wood to identify loss of material, cracking, and change in material properties to the Structures Monitoring Program.	Commitment Eliminated by FCR 27/014		
22	Enhance Guidance for performing structural examinations of elastomers to identify cracking and change in material properties in the Structures Monitoring Program procedure	No Changes	22	Enhance Guidance for performing structural examinations of elastomers to identify cracking and change in material properties in the Structures Monitoring Program procedure
23	Add Guidance for performing structural examinations of PVC cooling tower fill to identify cracking and change in material properties to the Structures Monitoring Program procedure.	Commitment Eliminated by FCR 27/014		
24	System walkdown guidance documents will be enhanced to perform periodic system engineer inspections of systems in scope and subject to aging management review .	No Changes	24	System walkdown guidance documents will be enhanced to perform periodic system engineer inspections of systems in scope and subject to aging management review .
25	Implement the Thermal Aging and Neutron Irradiation Embrittlement of Cast Austenitic Stainless Steel (CASS) Program	Commitment Eliminated by FCR 27/014		
26	Procedures will be enhanced to flush the John Deere Diesel Generator cooling water system and replace the coolant and coolant conditioner every three years.	No Changes	26	Procedures will be enhanced to flush the John Deere Diesel Generator cooling water system and replace the coolant and coolant conditioner every three years.

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27	At least 2 years prior to entering the period of extended operation, for the locations identified in NUREG/CR-6260 for BWRs of the VY vintage, VY will refine our current fatigue analyses to include the effects of reactor water environment and verify that the cumulative usage factors (CUFs) are less than 1.	Commitment Eliminated by FCR 27/014			
28	Revise program procedures to indicate that the Instrument Air Program will maintain instrument air quality in accordance with ISA S7.3	No Changes		28	Revise program procedures to indicate that the Instrument Air Program will maintain instrument air quality in accordance with ISA S7.3
29	VYNPS will perform one of the following: 1. Install core plate wedges, or, 2. Complete a plant-specific analysis to determine acceptance criteria for continued inspection of core plate hold down bolting	The plant-specific analysis to determine the acceptance criteria for the continued inspection of core plate hold down bolting was performed. Thus, the commitment was met and closed by FCR 27/014.			
30	Revise System Walkdown Program to specify CO2 system inspections every 6 months.	No Changes		30	Revise System Walkdown Program to specify CO2 system inspections every 6 months.
31	Revise Fire Water System Program to specify annual fire hydrant gasket inspections and flow tests.	No Changes		31	Revise Fire Water System Program to specify annual fire hydrant gasket inspections and flow tests.
32	Implement the Metal Enclosed Bus Program	No Changes		32	Implement the Metal Enclosed Bus Program
33	Include within the Structures Monitoring Program provisions that will ensure an engineering evaluation is made on a periodic basis (at least once every five years) of groundwater samples to assess aggressiveness of groundwater to concrete.	No Changes		33	Include within the Structures Monitoring Program provisions that will ensure an engineering evaluation is made on a periodic basis (at least once every five years) of groundwater samples to assess aggressiveness of groundwater to concrete.
34	Implement the Bolting Integrity Program.	No Changes		34	Implement the Bolting Integrity Program.
35	Provide within the System Walkdown Training Program a process to document biennial refresher training of Engineers to demonstrate inclusion of the methodology for aging management of plant equipment	No Changes		35	Provide within the System Walkdown Training Program a process to document biennial refresher training of Engineers to demonstrate inclusion of the methodology for aging management of plant equipment
36	If technology to inspect the hidden jet pump thermal sleeve and core spray thermal sleeve welds has not been developed and approved by the NRC at least two years prior to the period of extended operation, VYNPS will initiate plant-specific action to resolve this issue.	Commitment Eliminated by FCR 27/014			
37	This Commitment superseded by steam dryer license condition 3.S.	Commitment Superseded. Placeholder deleted by FCR 27/014.			

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38	The BWRVIP-116 report which was approved by the Staff will be implemented at VYNPS with the conditions documented in Sections 3 and 4 of the Staff's final SE dated March 1, 2006, for the BWRVIP-116 report.	Commitment Eliminated by FCR 27/014			
39	If the VYNPS standby capsule is removed form the reactor vessel without the intent to test it, the capsule will be stored in a manner which maintains it in a condition which would permit its future use, including during the period of extended operation, if necessary.	Commitment Eliminated by FCR 27/014			
40	This Commitment has been deleted and replaced with Commitment 43.	Commitment Deleted. Placeholder deleted by FCR 27/014.			
41	This Commitment has been deleted and replaced with Commitment 43.	Commitment Deleted. Placeholder deleted by FCR 27/014.			
42	Implement the Bolted Cable Connections Program.	No Changes		42	Implement the Bolted Cable Connections Program.
43	Establish and implement a program that will require testing of the two 13.8 kV cables from the two Vernon Hydro Station 13.8 kV switchgear buses to the 13.8 kV / 69 kV step up transformers	Commitment Eliminated by FCR 26/011			
44	This Commitment has been deleted and replaced with Commitment 54.	Commitment Superseded. Placeholder deleted by FCR 27/014.			
45	Enhance the Service Water Integrity Program to require a periodic visual inspection of the RHRSW pump motor cooling coil internal surface for loss of material.	Commitment Eliminated by FCR 27/014			
46	Enhance the Diesel Fuel Monitoring Program to specify that fuel oil in the fire pump diesel storage (day) tank and the John Deere diesel storage tank will be analyzed according to ASTM D975 and for particulates per ASTM D2276.	No Changes		46	Enhance the Diesel Fuel Monitoring Program to specify that fuel oil in the fire pump diesel storage (day) tank and the John Deere diesel storage tank will be analyzed according to ASTM D975 and for particulates per ASTM D2276.
47	Enhance the Diesel Fuel Monitoring Program to specify that fuel oil in the common portable fuel oil storage tank will be analyzed according to ASTM D975, per ASTM D2276 for particulates, and per ASTM D2709 for water and sediment.	No Changes		47	Enhance the Diesel Fuel Monitoring Program to specify that fuel oil in the common portable fuel oil storage tank will be analyzed according to ASTM D975, per ASTM D2276 for particulates, and per ASTM D2709 for water and sediment.
48	Perform an internal inspection of the underground Service Water piping before entering the period of extended operation.	The internal inspection of the underground Service Water piping was performed; thus, the commitment was met and closed. Placeholder eliminated by FCR 27/014.			
49	Revise station procedures to specify fire hydrant hose testing, inspection, and replacement, if necessary, in accordance with NFPA code specifications for fire hydrant hoses	No Changes		49	Revise station procedures to specify fire hydrant hose testing, inspection, and replacement, if necessary, in accordance with NFPA code specifications for fire hydrant hoses

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	50	During the period of extended operation, review the Vernon Dam owner FERC required report(s) at a minimum of every five years to confirm that the Vernon Dam owner is performing the required FERC inspections.	Commitment Eliminated by FCR 27/014		
	51	Perform an evaluation of operating experience at extended power uprate (EPU) levels prior to the period of extended operation to ensure that operating experience at EPU levels is properly addressed by the aging management programs.	The evaluation of operating experience at extended power uprate levels was performed; thus, the commitment was met and closed. Placeholder eliminated by FCR 27/014.		
	52	Implement the Neutron Absorber Monitoring Program	No Changes	52	Implement the Neutron Absorber Monitoring Program
	53	During the period of extended operation, VYNPS will perform periodic volumetric examinations of small-bore Class 1 socket and butt welds.	Commitment Eliminated by FCR 27/014		
	54	Prior to the PEO, VYNPS will inspect portions of the standby gas treatment system buried piping. During the PEO, inspections of two carbon steel piping segments in the standby gas treatment system and four carbon steel piping segments in the service water system will be performed every 10 years	No Changes	54	Prior to the PEO, VYNPS will inspect portions of the standby gas treatment system buried piping. During the PEO, inspections of two carbon steel piping segments in the standby gas treatment system and four carbon steel piping segments in the service water system will be performed every 10 years
	55	Enhance safety-related coatings programs and procedures to be consistent with the recommendations of NUREG-1801, Section XI.S8, Protective Coating Monitoring and Maintenance Program.	Commitment Eliminated by FCR 27/014		
Appendices					
A	SEISMIC ANALYSES				
	A.1	Summary Description	Editorial changes. Deletions to remove excess detail.	A.1	Summary Description
	A.2	SPECIFIC DESIGN ANALYSIS FOR THE DRYWELL, SUPPRESSION CHAMBER AND REACTOR BUILDING	No Changes. This is an independent report by H. J. Sexton Engineering	A.2	SPECIFIC DESIGN ANALYSIS FOR THE DRYWELL, SUPPRESSION CHAMBER AND REACTOR BUILDING
	A.3	Turbine Building	Editorial changes only. Turbine Building analysis should remain at least while the Diesel Generators are relied upon as a standby power source.	A.3	Turbine Building
	Figure A.3-1	Turbine Building Bent	No Changes	Figure A.3-1	Turbine Building Bent
	Figure A.3-2	Turbine Building Bent	No Changes	Figure A.3-2	Turbine Building Bent
	A.4	Plant Stack	No Changes	A.4	Plant Stack
	Figure A.4-1	Ventilation Stack Mathematical Model	No Changes	Figure A.4-1	Ventilation Stack Mathematical Model
	Figure A.4-2	Ventilation Stack Maximum Shear	No Changes	Figure A.4-2	Ventilation Stack Maximum Shear
	Figure A.4-3	Ventilation Stack Maximum Movement	No Changes	Figure A.4-3	Ventilation Stack Maximum Movement

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	Figure A.4-4	Ventilation Stack Maximum Deflection Diagram		No Changes	Figure A.4-4		Ventilation Stack Maximum Deflection Diagram
	A.5	Earthquake Analysis of the Control Building		No changes, this is an independent report by EBASCO Services	A.5		Earthquake Analysis of the Control Building
	A.6	Intake Structure		No changes. Seismic analysis remains while portion of the Intake Structure remains Cat 1.	A.6		Intake Structure
	Figure A.6-1	PLAN AT EL 230.00'		No Changes	Figure A.6-1		PLAN AT EL 230.00'
	Figure A.6-2	SECTIONS		No Changes	Figure A.6-2		SECTIONS
	Figure A.6-3	INTAKE STRUCTURE SERVICE WATER BAY AREA MATHEMATICAL MODELS (N-S)		No Changes	Figure A.6-3		INTAKE STRUCTURE SERVICE WATER BAY AREA MATHEMATICAL MODELS (N-S)
	Figure A.6-4	INTAKE STRUCTURE - SERVICE WATER BAY AREA -MAXIMUM ACCELERATIONS (N-S)		No Changes	Figure A.6-4		INTAKE STRUCTURE - SERVICE WATER BAY AREA -MAXIMUM ACCELERATIONS (N-S)
	Figure A.6-5	Service Water Bay Area, Maximum Displacements (N-S)		No Changes	Figure A.6-5		Service Water Bay Area, Maximum Displacements (N-S)
	Figure A.6-6	Intake Structure – Service Water Bay Area, Maximum Shears (N-S)		No Changes	Figure A.6-6		Intake Structure – Service Water Bay Area, Maximum Shears (N-S)
	Figure A.6-7	Intake Structure – Service Water Bay Area, Maximum Moments (N-S)		No Changes	Figure A.6-7		Intake Structure – Service Water Bay Area, Maximum Moments (N-S)
	A.7	Alternate Cooling Cell		Delete Appendix A, Section A.7 Alternate Cooling Cell Seismic Analysis. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the functions provided by the Alternate Cooling Cell are no longer required. The information provided in the Alternate Cooling Cell Seismic Analysis is no longer applicable, and is considere obsolete.			
	A.8	VERMONT YANKEE NUCLEAR POWER STATION ADDITIONAL INFORMATION CONCERNING SEISMIC ANALYSIS OF CLASS I PIPING		Delete Appendix A, Section A.8, VERMONT YANKEE NUCLEAR POWER STATION ADDITIONAL INFORMATION CONCERNING SEISMIC ANALYSIS OF CLASS I PIPING. The one-time analysis documented in A.8 has been superseded by any number of analyses performed since that time as further discussed in, primarily, App A.9.			
	A.9	DESCRIPTION, SCOPE, AND DESIGN METHODOLOGY USED FOR THE REANALYSIS OF SEISMIC CLASS I PIPING SUBSEQUENT TO INITIAL OPERATION			A.9		DESCRIPTION, SCOPE, AND DESIGN METHODOLOGY USED FOR THE REANALYSIS OF SEISMIC CLASS I PIPING SUBSEQUENT TO INITIAL OPERATION
		A.9.1	Introduction	No Changes		A.9.1	Introduction
		A.9.2	Description of Analysis Methodology	Modify Section A.9.2 to: ·incorporate editorial changes ·Delete section A.9.2.8, Method 7, since it applied only to the reactor recirculation system, and the functions provided by that system are not required post certification of permanent defueling.		A.9.2	Description of Analysis Methodology

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		A.9.3	Scope of Seismic Class I Piping Systems Modified and Methodology Used	Modified Section A.9.3 to delete reference to the following SSCs, since the functions provided by those SSCs are no longer required following certification of permanent defueling: -RBCCW RHR SBGT Feedwater HPCI RCIC SBLC CS Clean-Up Water Demineralizer System Main Steam Drains Containment Inerting CRD Instrument air to AdS, MSIVs Sampling System Misc Small bore piping Containment atmosphere Control		A.9.3	Scope of Seismic Class I Piping Systems Modified and Methodology Used
		A.9.4	Description of Floor Amplified Response Spectra Used for Seismic Reanalysis of Piping	No Changes		A.9.4	Description of Floor Amplified Response Spectra Used for Seismic Reanalysis of Piping
		A.9.5	Summary and Conclusions	No Changes		A.9.5	Summary and Conclusions
	A.10	PRIMARY STRUCTURE SEISMIC ANALYSIS		No Changes	A.10	PRIMARY STRUCTURE SEISMIC ANALYSIS	
	Table A.10.2-1a	Summary of Nodal Hydrodynamic Masses for Horizontal Direction		No Changes		Table A.10.2-1a	Summary of Nodal Hydrodynamic Masses for Horizontal Direction
	Table A.10.2-2a	Horizontal Model Nodal Masses/Elevations		No Changes		Table A.10.2-2a	Horizontal Model Nodal Masses/Elevations
	Table A.10.2-2b	Horizontal Model Nodal Masses/Elevations		No Changes		Table A.10.2-2b	Horizontal Model Nodal Masses/Elevations
	Table A.10.2-2c	Horizontal Model Nodal Masses/Elevations		No Changes		Table A.10.2-2c	Horizontal Model Nodal Masses/Elevations
	Table A.10.2-2d	Horizontal Model Nodal Masses/Elevations		No Changes		Table A.10.2-2d	Horizontal Model Nodal Masses/Elevations
	Table A.10.2-3a	Vertical Model Nodal Masses/Elevations		No Changes		Table A.10.2-3a	Vertical Model Nodal Masses/Elevations
	Table A.10.2-3b	Vertical Model Nodal Masses/Elevations		No Changes		Table A.10.2-3b	Vertical Model Nodal Masses/Elevations
	Table A.10.2-4a	Horizontal Model Beam Element Properties North South Direction		No Changes		Table A.10.2-4a	Horizontal Model Beam Element Properties North South Direction
	Table A.10.2-4b	Horizontal Model Beam Element Properties North South Direction		No Changes		Table A.10.2-4b	Horizontal Model Beam Element Properties North South Direction
	Table A.10.2-4c	Horizontal Model Beam Element Properties North South Direction		No Changes		Table A.10.2-4c	Horizontal Model Beam Element Properties North South Direction
	Table A.10.2-5a	Horizontal Model Beam Element Properties East West Direction		No Changes		Table A.10.2-5a	Horizontal Model Beam Element Properties East West Direction
	Table A.10.2-5b	Horizontal Model Beam Element Properties East West Direction		No Changes		Table A.10.2-5b	Horizontal Model Beam Element Properties East West Direction
	Table A.10.2-5c	Horizontal Model Beam Element Properties East West Direction		No Changes		Table A.10.2-5c	Horizontal Model Beam Element Properties East West Direction
	Table A.10.2-5d	Horizontal Model Beam Element Properties East West Direction		No Changes		Table A.10.2-5d	Horizontal Model Beam Element Properties East West Direction
	Table A.10.2-6a	Vertical Model Beam Element Properties		No Changes		Table A.10.2-6a	Vertical Model Beam Element Properties

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	Table A.10.2-6b	Vertical Model Beam Element Properties	No Changes		Table A.10.2-6b	Vertical Model Beam Element Properties
	Table A.10.2-6c	Vertical Model Beam Element Properties	No Changes		Table A.10.2-6c	Vertical Model Beam Element Properties
	Table A.10.2-7a	Spring Element Stiffnesses Horizontal Model	No Changes		Table A.10.2-7a	Spring Element Stiffnesses Horizontal Model
	Table A.10.2-8a	Spring Element Stiffnesses Vertical Model	No Changes		Table A.10.2-8a	Spring Element Stiffnesses Vertical Model
	Table A.10.2-9a	Natural Frequencies of the N-S Model	No Changes		Table A.10.2-9a	Natural Frequencies of the N-S Model
	Table A.10.2-10a	Natural Frequencies of the E-W Model	No Changes		Table A.10.2-10a	Natural Frequencies of the E-W Model
	Table A.10.2-11a	Natural Frequencies of the Vertical Model	No Changes		Table A.10.2-11a	Natural Frequencies of the Vertical Model
	Figure A.10.2-1	Reactor Building Complex Longitudinal Section	No Changes		Figure A.10.2-1	Reactor Building Complex Longitudinal Section
	Figure A.10.2-2	Reactor Building Complex Seismic Horizontal Model	No Changes		Figure A.10.2-2	Reactor Building Complex Seismic Horizontal Model
	Figure A.10.2-3	Reactor Building Complex Seismic Vertical Model	No Changes		Figure A.10.2-3	Reactor Building Complex Seismic Vertical Model
	Figure A.10.2-4	Stabilizer System	No Changes		Figure A.10.2-4	Stabilizer System
	Figure A.10.2-5	Drywell Geometric Figure	No Changes		Figure A.10.2-5	Drywell Geometric Figure
	Figure A.10.3-1	Acceleration Time History (0.14G), H1 Motion, DT = .01 Sec. NPT = 2049	No Changes		Figure A.10.3-1	Acceleration Time History (0.14G), H1 Motion, DT = .01 Sec. NPT = 2049
	Figure A.10.3-2	A.10.3 4 GESSAR II vs. NRC ARS	No Changes		Figure A.10.3-2	A.10.3 4 GESSAR II vs. NRC ARS
	Figure A.10.3-3	Acceleration Time History. (0.14G), V Motion, DT = .01 Sec. NPT = 2201	No Changes		Figure A.10.3-3	Acceleration Time History. (0.14G), V Motion, DT = .01 Sec. NPT = 2201
	Figure A.10.3-4	GESSAR II vs. NRC ARS	No Changes		Figure A.10.3-4	GESSAR II vs. NRC ARS
	Figure A.10.3-5	GESSAR II vs. NRC ARS	No Changes		Figure A.10.3-5	GESSAR II vs. NRC ARS
	Figure A.10.3-6	GESSAR II vs. NRC ARS	No Changes		Figure A.10.3-6	GESSAR II vs. NRC ARS
B	STATION RESEARCH DEVELOPMENT AND FURTHER INFORMATION, REQUIREMENTS AND RESOLUTIONS					
	B.1	SUMMARY DESCRIPTION	Delete Appendix B, Section B.1, Station Research, Development and Further Information, Requirements and Resolutions, Summary Description. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the information provided in Section B.1 is no longer applicable and is considered obsolete.			
	B.2	AREAS SPECIFIED IN THE VERMONT YANKEE AEC-ACRS CONSTRUCTION PERMIT LETTERS	Delete Appendix B, Section B.2, Station Research, Development and Further Information, Requirements and Resolutions, Areas Specified in the Vermont Yankee AEC-ACRS Construction Permit Letters. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the information provided in Section B.2 regarding resolution o the eleven concern items prior to initial operation of the facility is no longer applicable and is considered obsolete.			
	B.3	AREAS SPECIFIED IN THE VERMONT YANKEE AEC ACRS CONSTRUCTION PERMIT SAFETY EVALUATION REPORT (Refer to Table 1.10.3)	Delete Appendix B, Section B.3, Station Research, Development and Further Information, Requirements and Resolutions, Areas Specified in the Vermont Yankee AEC-ACRS Construction Permit Safety Evaluation Report Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the information provided in Section B.3 regarding resolution o items of concern prior to initial operation of the facility is no longer applicable and is considered obsolete.			

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	B.4		AREAS SPECIFIED IN OTHER RELATED AEC ACRS CONSTRUCTION AND OPERATING PERMIT LETTERS (Refer to Table 1.10.4)	Delete Appendix B, Section B.4, Station Research, Development and Further Information, Requirements and Resolutions, Areas Specified in Other Related AEC ACRS Construction and Operating Permit Letters . Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the information provided in Section B.4 regarding resolution o items of concern related to operation of the facility is no longer applicable and is considered obsolete.	
	B.5		AREAS SPECIFIED IN OTHER RELATED AEC STAFF CONSTRUCTION OR OPERATING PERMIT SAFETY EVALUATION REPORTS	Delete Appendix B, Section B.5, Station Research, Development and Further Information, Requirements and Resolutions, Areas Specified in Other AEC Staff Construction or Operating Permit Safety Evaluation Reports . Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the information provided in Section B.5 regarding resolution o items of concern related to operation of the facility is no longer applicable and is considered obsolete. Additionally, information regarding Tornado and Missile Protection related to GE Topical Report "Tornado Protection For The Spent Storage Pool," APED 5696, November, 1968 was relocated to section 10.2, Spent Fuel Storage.	
	B.6		SUMMARY CONCLUSIONS	Delete Appendix B, Section B.6, Station Research, Development and Further Information, Requirements and Resolutions, Summary Conclusions . Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, the conclusions provided in Section B.6 regarding resolution of items of concern related to operation of the facility is no longer applicable and are considered obsolete.	
C			STRUCTURAL LOADING CRITERIA	Delete Appendix C, Structural Loading Criteria. The capacity that the station structures, systems, equipment, and components should withstand the forces of the simultaneous occurrence of a design basis accident (DBA) and a maximum hypothetical earthquake (MHE) event was not specified as a station design requirement when the VY construction permit was issued in December 1967. The information in this appendix is therefore an identification and summary of the capability of the as built VYNPS design to conform to the new AEC requirement of April 1968 mentioned above. The material in Appendix C was prepared at the time of initial plant licensing and was based on the plant design at that time. Plant design changes, subsequent to that time, have resulted in situations that may alter the magnitude of stresses reported in Appendix C. Therefore, the information in Appendix C is a one-time snapshot analysis of the VY seismic design to demonstrate that the design was robust enough to meet the requirement that class I SSCs would survive a concurrent DBA and MHE. The information is historical and may be deleted.	
	C.1		Structural Loading Criteria	Delete Appendix C.1, Structural Loading Criteria. The capacity that the station structures, systems, equipment, and components should withstand the forces of the simultaneous occurrence of a design basis accident (DBA) and a maximum hypothetical earthquake (MHE) event was not specified as a station design requirement when the VY construction permit was issued in December 1967. The information in this appendix is therefore an identification and summary of the capability of the as built VYNPS design to conform to the new AEC requirement of April 1968 mentioned above. The material in Appendix C was prepared at the time of initial plant licensing and was based on the plant design at that time. Plant design changes, subsequent to that time, have resulted in situations that may alter the magnitude of stresses reported in Appendix C. Therefore, the information in Appendix C is a one-time snapshot analysis of the VY seismic design to demonstrate that the design was robust enough to meet the requirement that class I SSCs would survive a concurrent DBA and MHE. The information is historical and may be deleted.	
		C.1.1	Summary Description	Delete Appendix C.1.1, Summary Description. The capacity that the station structures, systems, equipment, and components should withstand the forces of the simultaneous occurrence of a design basis accident (DBA) and a maximum hypothetical earthquake (MHE) event was not specified as a station design requirement when the VY construction permit was issued in December 1967. The information in this appendix is therefore an identification and summary of the capability of the as built VYNPS design to conform to the new AEC requirement of April 1968 mentioned above. The material in Appendix C was prepared at the time of initial plant licensing and was based on the plant design at that time. Plant design changes, subsequent to that time, have resulted in situations that may alter the magnitude of stresses reported in Appendix C. Therefore, the information in Appendix C is a one-time snapshot analysis of the VY seismic design to demonstrate that the design was robust enough to meet the requirement that class I SSCs would survive a concurrent DBA and MHE. The information is historical and may be deleted.	
	C.2		LOADING CRITERIA FOR GENERAL ELECTRIC (GE) SUPPLIED CLASS I COMPONENTS	Delete Appendix C.2,Loading Criteria for General Electric (GE) Supplied Class I Components. The capacity that the station structures, systems, equipment, and components should withstand the forces of the simultaneous occurrence of a design basis accident (DBA) and a maximum hypothetical earthquake (MHE) event was not specified as a station design requirement when the VY construction permit was issued in December 1967. The information in this appendix is therefore an identification and summary of the capability of the as built VYNPS design to conform to the new AEC requirement of April 1968 mentioned above. The material in Appendix C was prepared at the time of initial plant licensing and was based on the plant design at that time. Plant design changes, subsequent to that time, have resulted in situations that may alter the magnitude of stresses reported in Appendix C. Therefore, the information in Appendix C is a one-time snapshot analysis of the VY seismic design to demonstrate that the design was robust enough to meet the requirement that class I SSCs would survive a concurrent DBA and MHE. The information is historical and may be deleted.	
		C.2.1	Intent and Scope		
		C.2.2	Loading Conditions and Allowable Limits		
		C.2.3	Governing Loading Conditions and Criteria		
		C.2.4	Reactor Pressure Vessel		
		C.2.5	Reactor Vessel Internals		
		C.2.6	Piping Systems		
		C.2.7	Equipment		

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UFSAR Section				FSAR Conversion to DSAR Change Summary	DSAR Section			
		C.2.8	Instrumentation					
	C.3	Loading Criteria for Class I Piping		Delete Appendix C.3, Loading Criteria for Class I Piping. The capacity that the station structures, systems, equipment, and components should withstand the forces of the simultaneous occurrence of a design basis accident (DBA) and a maximum hypothetical earthquake (MHE) event was not specified as a station design requirement when the VY construction permit was issued in December 1967. The information in this appendix is therefore an identification and summary of the capability of the as built VYNPS design to conform to the new AEC requirement of April 1968 mentioned above. The material in Appendix C was prepared at the time of initial plant licensing and was based on the plant design at that time. Plant design changes, subsequent to that time, have resulted in situations that may alter the magnitude of stresses reported in Appendix C. Therefore, the information in Appendix C is a one-time snapshot analysis of the VY seismic design to demonstrate that the design was robust enough to meet the requirement that class I SSCs would survive a concurrent DBA and MHE. The information is historical and may be deleted.				
	C.4	LOADING CRITERIA FOR EBASCO SUPPLIED CLASS I COMPONENTS		Delete Appendix C.4, Loading Criteria for EBASCO Supplied Class I Components. The capacity that the station structures, systems, equipment, and components should withstand the forces of the simultaneous occurrence of a design basis accident (DBA) and a maximum hypothetical earthquake (MHE) event was not specified as a station design requirement when the VY construction permit was issued in December 1967. The information in this appendix is therefore an identification and summary of the capability of the as built VYNPS design to conform to the new AEC requirement of April 1968 mentioned above. The material in Appendix C was prepared at the time of initial plant licensing and was based on the plant design at that time. Plant design changes, subsequent to that time, have resulted in situations that may alter the magnitude of stresses reported in Appendix C. Therefore, the information in Appendix C is a one-time snapshot analysis of the VY seismic design to demonstrate that the design was robust enough to meet the requirement that class I SSCs would survive a concurrent DBA and MHE. The information is historical and may be deleted.				
D	QA PROGRAM			Delete Appendix D, QA Program. Section 1.9, Quality Assurance Program, establishes the criteria to be applied to systems requiring Quality Assurance which prevent or mitigate the consequences of postulated accidents which could cause undue risk to the health and safety of the public. The Quality Assurance Program Manual (QAPM) implements the requirements of 10CFR50 Appendix B. Appendix D functioned as a placekeeper and is no longer required since the UFSAR is being wholly rewritten to the DSAR to reflect the permanently defueled state.				
	D.0	QA (QA) Program During the Operations Phase		Delete Section D.0, Quality Assurance (QA) Program During the Operations Phase. Section 1.9, Quality Assurance Program, establishes the criteria to be applied to systems requiring Quality Assurance which prevent or mitigate the consequences of postulated accidents which could cause undue risk to the health and safety of the public. The Quality Assurance Program Manual (QAPM) implements the requirements of 10CFR50 Appendix B. Appendix D functioned as a placekeeper and is no longer required since the UFSAR is being wholly rewritten to the DSAR to reflect the permanently defueled state.				
E	STACK RELEASE LIMIT CALCULATIONS			Delete Appendix E, Stack Release Limit Calculations. Appendix E to the Vermont Yankee Final Safety Analysis Report (FSAR) was previously eliminated from the FSAR because it contained information which had been superseded by the implementation of the Vermont Yankee Off Site Dose Calculation Manual (ODCM). Information on release calculational methods can be found in the ODCM. The ODCM also includes reporting requirements for actual public doses resulting from the release calculations. These results are published in the Annual Radioactive Effluent Release Report. This section functioned as a placekeeper and is no longer required since the UFSAR is being wholly rewritten to the DSAR to reflect the permanently defueled state.				
F	CONFORMANCE WITH AEC DESIGN CRITERIA			In a Staff Requirements Memorandum on SECY-92-223, the NRC approved a proposal in which it was recognized that plants with construction permits issued before May 21, 1971 were not licensed to meet the final General Design Criteria. The memo recognized that while compliance with the intent of the final General Design Criteria was important, backfitting of these requirements to older plants would provide little or no safety benefit. The purpose of Appendix F, as originally submitted was to demonstrate that, although VY was not required to comply with the General Design Criteria, the design and construction of the Vermont Yankee Nuclear Power Station had been performed in accordance with the General Design Criteria proposed in July, 1967. The final version of the General Design Criteria was published in February, 1971 as 10CFR50 Appendix A. Differences between the proposed and final versions of the criteria included a consolidation from 70 to 64 criteria and general elaboration of design requirement details. Since, at the time of issuance, the Commission stressed that the final version of the criteria were not new requirements and were promulgated to more clearly articulate the licensing requirements and practices in effect at the time, Appendix F was not updated to the final GDCs. Consequently, the information provided in Appendix F is a late 1960's snapshot of how VY design and construction met the intent of the GDCs. The information is historical, has not been updated, is not required since VY is not required to meet the GDCs. The information provided in Appendix F does not reflect the permanently defueled state. This information is obsolete and may be deleted. Information regarding compliance with the intent of the GDCs determined to be applicable in the defueled condition has been relocated to section 3.1.1. All other information in this section has been deleted.		3.1.1	Conformance with 10 CFR 50 Appendix A, General Design Criteria	

VY UFSAR					VY DSAR
UFSAR Section				FSAR Conversion to DSAR Change Summary	DSAR Section
	F.1	SUMMARY DESCRIPTION		Delete Appendix F.1, Summary Description. In a Staff Requirements Memorandum on SECY-92-223, the NRC approved a proposal in which it was recognized that plants with construction permits issued before May 21, 1971 were not licensed to meet the final General Design Criteria. The memo recognized that while compliance with the intent of the final General Design Criteria was important, backfitting of these requirements to older plants would provide little or no safety benefit. The purpose of Appendix F, as originally submitted was to demonstrate that, although VY was not required to comply with the General Design Criteria, the design and construction of the Vermont Yankee Nuclear Power Station had been performed in accordance with the General Design Criteria proposed in July, 1967. The final version of the General Design Criteria was published in February, 1971 as 10CFR50 Appendix A. Differences between the proposed and final versions of the criteria included a consolidation from 70 to 64 criteria and general elaboration of design requirement details. Since, at the time of issuance, the Commission stressed that the final version of the criteria were not new requirements and were promulgated to more clearly articulate the licensing requirements and practices in effect at the time, Appendix F was not updated to the final GDCs. Consequently, the information provided in Appendix F is a late 1960's snapshot of how VY design and construction met the intent of the GDCs. The information provided in Appendix F.1, Summary Description is historical, has not been updated, and is not required since VY is not required to meet the GDCs. The information provided in Appendix F.1, Summary Description, does not reflect the permanently defueled state. This information is obsolete and may be deleted.	
	F.2	CRITERION CONFORMANCE			
		F.2.1	Group I--Overall Plant Requirements (Criteria 1-5)	Delete Section F.2.1, Group I--Overall Plant Requirements (Criteria 1-5). In a Staff Requirements Memorandum on SECY-92-223, the NRC approved a proposal in which it was recognized that plants with construction permits issued before May 21, 1971 were not licensed to meet the final General Design Criteria. The memo recognized that while compliance with the intent of the final General Design Criteria was important, backfitting of these requirements to older plants would provide little or no safety benefit. The purpose of Appendix F, as originally submitted was to demonstrate that, although VY was not required to comply with the General Design Criteria, the design and construction of the Vermont Yankee Nuclear Power Station had been performed in accordance with the General Design Criteria proposed in July, 1967. The final version of the General Design Criteria was published in February, 1971 as 10CFR50 Appendix A. Differences between the proposed and final versions of the criteria included a consolidation from 70 to 64 criteria and general elaboration of design requirement details. Since, at the time of issuance, the Commission stressed that the final version of the criteria were not new requirements and were promulgated to more clearly articulate the licensing requirements and practices in effect at the time, Appendix F was not updated to the final GDCs. Consequently, the information provided in Appendix F is a late 1960's snapshot of how VY design and construction met the intent of the GDCs. The information provided in Section F.2.1 is historical, has not been updated, and is not required since VY is not required to meet the GDCs. The information provided in Section F.2.1 does not reflect the permanently defueled state. This information is obsolete and may be deleted.	
		F.2.2	Group II-Protection by Multiple Fission Barriers (Criteria 6-10)	Delete Section F.2.2, Group II-Protection by Multiple Fission Barriers (Criteria 6-10). In a Staff Requirements Memorandum on SECY-92-223, the NRC approved a proposal in which it was recognized that plants with construction permits issued before May 21, 1971 were not licensed to meet the final General Design Criteria. The memo recognized that while compliance with the intent of the final General Design Criteria was important, backfitting of these requirements to older plants would provide little or no safety benefit. The purpose of Appendix F, as originally submitted was to demonstrate that, although VY was not required to comply with the General Design Criteria, the design and construction of the Vermont Yankee Nuclear Power Station had been performed in accordance with the General Design Criteria proposed in July, 1967. The final version of the General Design Criteria was published in February, 1971 as 10CFR50 Appendix A. Differences between the proposed and final versions of the criteria included a consolidation from 70 to 64 criteria and general elaboration of design requirement details. Since, at the time of issuance, the Commission stressed that the final version of the criteria were not new requirements and were promulgated to more clearly articulate the licensing requirements and practices in effect at the time, Appendix F was not updated to the final GDCs. Consequently, the information provided in Appendix F is a late 1960's snapshot of how VY design and construction met the intent of the GDCs. The information provided in Section F.2.2 is historical, has not been updated, and is not required since VY is not required to meet the GDCs. The information provided in Section F.2.2 does not reflect the permanently defueled state. This information is obsolete and may be deleted.	

VY UFSAR					VY DSAR
UFSAR Section				FSAR Conversion to DSAR Change Summary	DSAR Section
		F.2.3	Group III-Nuclear and Radiation Controls (Criteria 11-18)	Delete Section F.2.3, Group III-Nuclear and Radiation Controls (Criteria 11-18). In a Staff Requirements Memorandum on SECY-92-223, the NRC approved a proposal in which it was recognized that plants with construction permits issued before May 21, 1971 were not licensed to meet the final General Design Criteria. The memo recognized that while compliance with the intent of the final General Design Criteria was important, backfitting of these requirements to older plants would provide little or no safety benefit. The purpose of Appendix F, as originally submitted was to demonstrate that, although VY was not required to comply with the General Design Criteria, the design and construction of the Vermont Yankee Nuclear Power Station had been performed in accordance with the General Design Criteria proposed in July, 1967. The final version of the General Design Criteria was published in February, 1971 as 10CFR50 Appendix A. Differences between the proposed and final versions of the criteria included a consolidation from 70 to 64 criteria and general elaboration of design requirement details. Since, at the time of issuance, the Commission stressed that the final version of the criteria were not new requirements and were promulgated to more clearly articulate the licensing requirements and practices in effect at the time, Appendix F was not updated to the final GDCs. Consequently, the information provided in Appendix F is a late 1960's snapshot of how VY design and construction met the intent of the GDCs. The information provided in Section F.2.3 is historical, has not been updated, and is not required since VY is not required to meet the GDCs. The information provided in Section F.2.3 does not reflect the permanently defueled state. This information is obsolete and may be deleted.	
		F.2.4	Group IV--Reliability and Testability of Protection Systems (Criteria 19-26)	Delete Section F.2.4, Group IV--Reliability and Testability of Protection Systems (Criteria 19-26). In a Staff Requirements Memorandum on SECY-92-223, the NRC approved a proposal in which it was recognized that plants with construction permits issued before May 21, 1971 were not licensed to meet the final General Design Criteria. The memo recognized that while compliance with the intent of the final General Design Criteria was important, backfitting of these requirements to older plants would provide little or no safety benefit. The purpose of Appendix F, as originally submitted was to demonstrate that, although VY was not required to comply with the General Design Criteria, the design and construction of the Vermont Yankee Nuclear Power Station had been performed in accordance with the General Design Criteria proposed in July, 1967. The final version of the General Design Criteria was published in February, 1971 as 10CFR50 Appendix A. Differences between the proposed and final versions of the criteria included a consolidation from 70 to 64 criteria and general elaboration of design requirement details. Since, at the time of issuance, the Commission stressed that the final version of the criteria were not new requirements and were promulgated to more clearly articulate the licensing requirements and practices in effect at the time, Appendix F was not updated to the final GDCs. Consequently, the information provided in Appendix F is a late 1960's snapshot of how VY design and construction met the intent of the GDCs. The information provided in Section F.2.4 is historical, has not been updated, and is not required since VY is not required to meet the GDCs. The information provided in Section F.2.4 does not reflect the permanently defueled state. This information is obsolete and may be deleted.	
		F.2.5	Group V--Reactivity Control (Criteria 27-32)	Delete Section F.2.5, Group V--Reactivity Control (Criteria 27-32). In a Staff Requirements Memorandum on SECY-92-223, the NRC approved a proposal in which it was recognized that plants with construction permits issued before May 21, 1971 were not licensed to meet the final General Design Criteria. The memo recognized that while compliance with the intent of the final General Design Criteria was important, backfitting of these requirements to older plants would provide little or no safety benefit. The purpose of Appendix F, as originally submitted was to demonstrate that, although VY was not required to comply with the General Design Criteria, the design and construction of the Vermont Yankee Nuclear Power Station had been performed in accordance with the General Design Criteria proposed in July, 1967. The final version of the General Design Criteria was published in February, 1971 as 10CFR50 Appendix A. Differences between the proposed and final versions of the criteria included a consolidation from 70 to 64 criteria and general elaboration of design requirement details. Since, at the time of issuance, the Commission stressed that the final version of the criteria were not new requirements and were promulgated to more clearly articulate the licensing requirements and practices in effect at the time, Appendix F was not updated to the final GDCs. Consequently, the information provided in Appendix F is a late 1960's snapshot of how VY design and construction met the intent of the GDCs. The information provided in Section F.2.5 is historical, has not been updated, and is not required since VY is not required to meet the GDCs. The information provided in Section F.2.5 does not reflect the permanently defueled state. This information is obsolete and may be deleted.	
		F.2.6	Group VI--Reactor Coolant Pressure Boundary (Criteria 33-36)	Delete Section F.2.6, Group VI--Reactor Coolant Pressure Boundary (Criteria 33-36). In a Staff Requirements Memorandum on SECY-92-223, the NRC approved a proposal in which it was recognized that plants with construction permits issued before May 21, 1971 were not licensed to meet the final General Design Criteria. The memo recognized that while compliance with the intent of the final General Design Criteria was important, backfitting of these requirements to older plants would provide little or no safety benefit. The purpose of Appendix F, as originally submitted was to demonstrate that, although VY was not required to comply with the General Design Criteria, the design and construction of the Vermont Yankee Nuclear Power Station had been performed in accordance with the General Design Criteria proposed in July, 1967. The final version of the General Design Criteria was published in February, 1971 as 10CFR50 Appendix A. Differences between the proposed and final versions of the criteria included a consolidation from 70 to 64 criteria and general elaboration of design requirement details. Since, at the time of issuance, the Commission stressed that the final version of the criteria were not new requirements and were promulgated to more clearly articulate the licensing requirements and practices in effect at the time, Appendix F was not updated to the final GDCs. Consequently, the information provided in Appendix F is a late 1960's snapshot of how VY design and construction met the intent of the GDCs. The information provided in Section F.2.6 is historical, has not been updated, and is not required since VY is not required to meet the GDCs. The information provided in Section F.2.6 does not reflect the permanently defueled state. This information is obsolete and may be deleted.	

VY UFSAR					VY DSAR
UFSAR Section				FSAR Conversion to DSAR Change Summary	DSAR Section
		F.2.7	Group VII--Engineered Safety Features (Criteria 37-65)	Delete Section F.2.7, Group VII--Engineered Safety Features (Criteria 37-65). In a Staff Requirements Memorandum on SECY-92-223, the NRC approved a proposal in which it was recognized that plants with construction permits issued before May 21, 1971 were not licensed to meet the final General Design Criteria. The memo recognized that while compliance with the intent of the final General Design Criteria was important, backfitting of these requirements to older plants would provide little or no safety benefit. The purpose of Appendix F, as originally submitted was to demonstrate that, although VY was not required to comply with the General Design Criteria, the design and construction of the Vermont Yankee Nuclear Power Station had been performed in accordance with the General Design Criteria proposed in July, 1967. The final version of the General Design Criteria was published in February, 1971 as 10CFR50 Appendix A. Differences between the proposed and final versions of the criteria included a consolidation from 70 to 64 criteria and general elaboration of design requirement details. Since, at the time of issuance, the Commission stressed that the final version of the criteria were not new requirements and were promulgated to more clearly articulate the licensing requirements and practices in effect at the time, Appendix F was not updated to the final GDCs. Consequently, the information provided in Appendix F is a late 1960's snapshot of how VY design and construction met the intent of the GDCs. The information provided in Section F.2.7 is historical, has not been updated, and is not required since VY is not required to meet the GDCs. The information provided in Section F.2.7 does not reflect the permanently defueled state. This information is obsolete and may be deleted.	
		F.2.8	Group VIII--Fuel and Waste Storage Systems (Criteria 66-69)	Delete Section F.2.8, Group VIII--Fuel and Waste Storage Systems (Criteria 66-69). In a Staff Requirements Memorandum on SECY-92-223, the NRC approved a proposal in which it was recognized that plants with construction permits issued before May 21, 1971 were not licensed to meet the final General Design Criteria. The memo recognized that while compliance with the intent of the final General Design Criteria was important, backfitting of these requirements to older plants would provide little or no safety benefit. The purpose of Appendix F, as originally submitted was to demonstrate that, although VY was not required to comply with the General Design Criteria, the design and construction of the Vermont Yankee Nuclear Power Station had been performed in accordance with the General Design Criteria proposed in July, 1967. The final version of the General Design Criteria was published in February, 1971 as 10CFR50 Appendix A. Differences between the proposed and final versions of the criteria included a consolidation from 70 to 64 criteria and general elaboration of design requirement details. Since, at the time of issuance, the Commission stressed that the final version of the criteria were not new requirements and were promulgated to more clearly articulate the licensing requirements and practices in effect at the time, Appendix F was not updated to the final GDCs. Consequently, the information provided in Appendix F is a late 1960's snapshot of how VY design and construction met the intent of the GDCs. The information provided in Section F.2.8 is historical, has not been updated, and is not required since VY is not required to meet the GDCs. The information provided in Section F.2.8 does not reflect the permanently defueled state. This information is obsolete and may be deleted.	
		F.2.9	Group IX--Plant Effluents (Criterion 70)	Delete Section F.2.9, Group IX--Plant Effluents (Criterion 70). In a Staff Requirements Memorandum on SECY-92-223, the NRC approved a proposal in which it was recognized that plants with construction permits issued before May 21, 1971 were not licensed to meet the final General Design Criteria. The memo recognized that while compliance with the intent of the final General Design Criteria was important, backfitting of these requirements to older plants would provide little or no safety benefit. The purpose of Appendix F, as originally submitted was to demonstrate that, although VY was not required to comply with the General Design Criteria, the design and construction of the Vermont Yankee Nuclear Power Station had been performed in accordance with the General Design Criteria proposed in July, 1967. The final version of the General Design Criteria was published in February, 1971 as 10CFR50 Appendix A. Differences between the proposed and final versions of the criteria included a consolidation from 70 to 64 criteria and general elaboration of design requirement details. Since, at the time of issuance, the Commission stressed that the final version of the criteria were not new requirements and were promulgated to more clearly articulate the licensing requirements and practices in effect at the time, Appendix F was not updated to the final GDCs. Consequently, the information provided in Appendix F is a late 1960's snapshot of how VY design and construction met the intent of the GDCs. The information provided in Section F.2.9 is historical, has not been updated, and is not required since VY is not required to meet the GDCs. The information provided in Section F.2.9 does not reflect the permanently defueled state. This information is obsolete and may be deleted.	
G	STATION METEOROLOGY				
	G.1	INITIAL ON SITE METEOROLOGICAL PROGRAM		The Initial On-Site Meteorological Program collected meteorological data from August, 1967 to July, 1968, for analysis as the basis for selection of a suitable site diffusion model for the original VY Plant Design and Analysis Report. Section G.1 contains a discussion of the results of that monitoring program and the meteorological model used for dose calculations in the original VY Plant Design and Analysis Report. The On Site Meteorological Data Collection Program was upgraded in early 1976 to meet the intent of Revision 0 of Regulatory Guide 1.23. Section G.2 describes the current on site monitoring program and presents wind and stability data summaries for one full year of operation; January 1, 1980 through December 31, 1980. Section G.2 includes a discussion of the data summaries and a comparison is made between data collected by the initial monitoring program (August 1967 July 1968) and data collected by the current monitoring program (January 1980 December 1980). It was concluded that results from both monitoring programs are compatible, and that both programs produced data bases which are representative of site meteorology. Consequently, the information provided in Section G.1 is considered historical, is obsolete and may be deleted from the DSAR.	

VY UFSAR					VY DSAR
UFSAR Section				FSAR Conversion to DSAR Change Summary	DSAR Section
		G.1.1	Introduction	Delete Section G.1.1, Introduction. The Initial On-Site Meteorological Program collected meteorological data from August, 1967 to July, 1968, for analysis as the basis for selection of a suitable site diffusion model for the original VY Plant Design and Analysis Report. Section G.1 contains a discussion of the results of that monitoring program and the meteorological model used for dose calculations in the original VY Plant Design and Analysis Report. The On Site Meteorological Data Collection Program was upgraded in early 1976 to meet the intent of Revision 0 of Regulatory Guide 1.23. Section G.2 describes the current on site monitoring program and presents wind and stability data summaries for one full year of operation; January 1, 1980 through December 31, 1980. Section G.2 includes a discussion of the data summaries and a comparison is made between data collected by the initial monitoring program (August 1967 July 1968) and data collected by the current monitoring program (January 1980 December 1980). It was concluded that results from both monitoring programs are compatible, and that both programs produced data bases which are representative of site meteorology. The information provided in section G.2 supercedes the information in section G.1. Consequently, the information provided in Section G.1.1 is considered historical, is obsolete and may be deleted from the DSAR.	
		G.1.2	Description of the Monitoring Program	Delete Section G.1.2, Description of the Monitoring Program. The Initial On-Site Meteorological Program collected meteorological data from August, 1967 to July, 1968, for analysis as the basis for selection of a suitable site diffusion model for the original VY Plant Design and Analysis Report. Section G.1 contains a discussion of the results of that monitoring program and the meteorological model used for dose calculations in the original VY Plant Design and Analysis Report. The On Site Meteorological Data Collection Program was upgraded in early 1976 to meet the intent of Revision 0 of Regulatory Guide 1.23. Section G.2 describes the current on site monitoring program and presents wind and stability data summaries for one full year of operation; January 1, 1980 through December 31, 1980. Section G.2 includes a discussion of the data summaries and a comparison is made between data collected by the initial monitoring program (August 1967 July 1968) and data collected by the current monitoring program (January 1980 December 1980). It was concluded that results from both monitoring programs are compatible, and that both programs produced data bases which are representative of site meteorology. The information provided in section G.2 supercedes the information in section G.1. Consequently, the information provided in Section G.1.2 is considered historical, is obsolete and may be deleted from the DSAR.	
		G.1.3	Results	Delete Section G.1.3, Results. The Initial On-Site Meteorological Program collected meteorological data from August, 1967 to July, 1968, for analysis as the basis for selection of a suitable site diffusion model for the original VY Plant Design and Analysis Report. Section G.1 contains a discussion of the results of that monitoring program and the meteorological model used for dose calculations in the original VY Plant Design and Analysis Report. The On Site Meteorological Data Collection Program was upgraded in early 1976 to meet the intent of Revision 0 of Regulatory Guide 1.23. Section G.2 describes the current on site monitoring program and presents wind and stability data summaries for one full year of operation; January 1, 1980 through December 31, 1980. Section G.2 includes a discussion of the data summaries and a comparison is made between data collected by the initial monitoring program (August 1967 July 1968) and data collected by the current monitoring program (January 1980 December 1980). It was concluded that results from both monitoring programs are compatible, and that both programs produced data bases which are representative of site meteorology. The information provided in section G.2 supercedes the information in section G.1. Consequently, the information provided in Section G.1.3 is considered historical, is obsolete and may be deleted from the DSAR.	
		G.1.4	Comparison with PDAR Diffusion Model	Delete Section G.1.4, Comparison with PDAR Diffusion Model. The Initial On-Site Meteorological Program collected meteorological data from August, 1967 to July, 1968, for analysis as the basis for selection of a suitable site diffusion model for the original VY Plant Design and Analysis Report. Section G.1 contains a discussion of the results of that monitoring program and the meteorological model used for dose calculations in the original VY Plant Design and Analysis Report. The On Site Meteorological Data Collection Program was upgraded in early 1976 to meet the intent of Revision 0 of Regulatory Guide 1.23. Section G.2 describes the current on site monitoring program and presents wind and stability data summaries for one full year of operation; January 1, 1980 through December 31, 1980. Section G.2 includes a discussion of the data summaries and a comparison is made between data collected by the initial monitoring program (August 1967 July 1968) and data collected by the current monitoring program (January 1980 December 1980). It was concluded that results from both monitoring programs are compatible, and that both programs produced data bases which are representative of site meteorology. The information provided in section G.2 supercedes the information in section G.1. Consequently, the information provided in Section G.1.4 is considered historical, is obsolete and may be deleted from the DSAR.	
		G.1.5	Conclusions	Delete Section G.1.5, Conclusions. The Initial On-Site Meteorological Program collected meteorological data from August, 1967 to July, 1968, for analysis as the basis for selection of a suitable site diffusion model for the original VY Plant Design and Analysis Report. Section G.1 contains a discussion of the results of that monitoring program and the meteorological model used for dose calculations in the original VY Plant Design and Analysis Report. The On Site Meteorological Data Collection Program was upgraded in early 1976 to meet the intent of Revision 0 of Regulatory Guide 1.23. Section G.2 describes the current on site monitoring program and presents wind and stability data summaries for one full year of operation; January 1, 1980 through December 31, 1980. Section G.2 includes a discussion of the data summaries and a comparison is made between data collected by the initial monitoring program (August 1967 July 1968) and data collected by the current monitoring program (January 1980 December 1980). It was concluded that results from both monitoring programs are compatible, and that both programs produced data bases which are representative of site meteorology. The information provided in section G.2 supercedes the information in section G.1. Consequently, the information provided in Section G.1.5 is considered historical, is obsolete and may be deleted from the DSAR.	

VY UFSAR					VY DSAR
UFSAR Section				FSAR Conversion to DSAR Change Summary	DSAR Section
		Table G.1.1	Monthly Wind Summary August 1967	Delete Table G.1.1, Monthly Wind Summary August 1967. The Initial On-Site Meteorological Program collected meteorological data from August, 1967 to July, 1968, for analysis as the basis for selection of a suitable site diffusion model for the original VY Plant Design and Analysis Report. Section G.1 contains a discussion of the results of that monitoring program and the meteorological model used for dose calculations in the original VY Plant Design and Analysis Report. The On Site Meteorological Data Collection Program was upgraded in early 1976 to meet the intent of Revision 0 of Regulatory Guide 1.23. Section G.2 describes the current on site monitoring program and presents wind and stability data summaries for one full year of operation; January 1, 1980 through December 31, 1980. Section G.2 includes a discussion of the data summaries and a comparison is made between data collected by the initial monitoring program (August 1967 July 1968) and data collected by the current monitoring program (January 1980 December 1980). It was concluded that results from both monitoring programs are compatible, and that both programs produced data bases which are representative of site meteorology. The information provided in section G.2 superceded the information in section G.1. Consequently, the information provided in Table G.1.1 is considered historical, is obsolete and may be deleted from the DSAR.	
		Table G.1.2	Monthly Wind Summary September 1967	Delete Table G.1.2, Monthly Wind Summary September 1967. The Initial On-Site Meteorological Program collected meteorological data from August, 1967 to July, 1968, for analysis as the basis for selection of a suitable site diffusion model for the original VY Plant Design and Analysis Report. Section G.1 contains a discussion of the results of that monitoring program and the meteorological model used for dose calculations in the original VY Plant Design and Analysis Report. The On Site Meteorological Data Collection Program was upgraded in early 1976 to meet the intent of Revision 0 of Regulatory Guide 1.23. Section G.2 describes the current on site monitoring program and presents wind and stability data summaries for one full year of operation; January 1, 1980 through December 31, 1980. Section G.2 includes a discussion of the data summaries and a comparison is made between data collected by the initial monitoring program (August 1967 July 1968) and data collected by the current monitoring program (January 1980 December 1980). It was concluded that results from both monitoring programs are compatible, and that both programs produced data bases which are representative of site meteorology. The information provided in section G.2 superceded the information in section G.1. Consequently, the information provided in Table G.1.2 is considered historical, is obsolete and may be deleted from the DSAR.	
		Table G.1.3	Monthly Wind Summary October 1967	Delete Table G.1.3, Monthly Wind Summary October 1967. The Initial On-Site Meteorological Program collected meteorological data from August, 1967 to July, 1968, for analysis as the basis for selection of a suitable site diffusion model for the original VY Plant Design and Analysis Report. Section G.1 contains a discussion of the results of that monitoring program and the meteorological model used for dose calculations in the original VY Plant Design and Analysis Report. The On Site Meteorological Data Collection Program was upgraded in early 1976 to meet the intent of Revision 0 of Regulatory Guide 1.23. Section G.2 describes the current on site monitoring program and presents wind and stability data summaries for one full year of operation; January 1, 1980 through December 31, 1980. Section G.2 includes a discussion of the data summaries and a comparison is made between data collected by the initial monitoring program (August 1967 July 1968) and data collected by the current monitoring program (January 1980 December 1980). It was concluded that results from both monitoring programs are compatible, and that both programs produced data bases which are representative of site meteorology. The information provided in section G.2 superceded the information in section G.1. Consequently, the information provided in Table G.1.3 is considered historical, is obsolete and may be deleted from the DSAR.	
		Table G.1.4	Monthly Wind Summary November 1967	Delete Table G.1.4, Monthly Wind Summary November 1967. The Initial On-Site Meteorological Program collected meteorological data from August, 1967 to July, 1968, for analysis as the basis for selection of a suitable site diffusion model for the original VY Plant Design and Analysis Report. Section G.1 contains a discussion of the results of that monitoring program and the meteorological model used for dose calculations in the original VY Plant Design and Analysis Report. The On Site Meteorological Data Collection Program was upgraded in early 1976 to meet the intent of Revision 0 of Regulatory Guide 1.23. Section G.2 describes the current on site monitoring program and presents wind and stability data summaries for one full year of operation; January 1, 1980 through December 31, 1980. Section G.2 includes a discussion of the data summaries and a comparison is made between data collected by the initial monitoring program (August 1967 July 1968) and data collected by the current monitoring program (January 1980 December 1980). It was concluded that results from both monitoring programs are compatible, and that both programs produced data bases which are representative of site meteorology. The information provided in section G.2 superceded the information in section G.1. Consequently, the information provided in Table G.1.4 is considered historical, is obsolete and may be deleted from the DSAR.	
		Table G.1.5	Monthly Wind Summary December 1967	Delete Table G.1.5, Monthly Wind Summary December 1967. The Initial On-Site Meteorological Program collected meteorological data from August, 1967 to July, 1968, for analysis as the basis for selection of a suitable site diffusion model for the original VY Plant Design and Analysis Report. Section G.1 contains a discussion of the results of that monitoring program and the meteorological model used for dose calculations in the original VY Plant Design and Analysis Report. The On Site Meteorological Data Collection Program was upgraded in early 1976 to meet the intent of Revision 0 of Regulatory Guide 1.23. Section G.2 describes the current on site monitoring program and presents wind and stability data summaries for one full year of operation; January 1, 1980 through December 31, 1980. Section G.2 includes a discussion of the data summaries and a comparison is made between data collected by the initial monitoring program (August 1967 July 1968) and data collected by the current monitoring program (January 1980 December 1980). It was concluded that results from both monitoring programs are compatible, and that both programs produced data bases which are representative of site meteorology. The information provided in section G.2 superceded the information in section G.1. Consequently, the information provided in Table G.1.5 is considered historical, is obsolete and may be deleted from the DSAR.	

VY UFSAR					VY DSAR
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		Table G.1.6	Monthly Wind Summary January 1968	Delete Table G.1.6, Monthly Wind Summary January 1968. The Initial On-Site Meteorological Program collected meteorological data from August, 1967 to July, 1968, for analysis as the basis for selection of a suitable site diffusion model for the original VY Plant Design and Analysis Report. Section G.1 contains a discussion of the results of that monitoring program and the meteorological model used for dose calculations in the original VY Plant Design and Analysis Report. The On Site Meteorological Data Collection Program was upgraded in early 1976 to meet the intent of Revision 0 of Regulatory Guide 1.23. Section G.2 describes the current on site monitoring program and presents wind and stability data summaries for one full year of operation; January 1, 1980 through December 31, 1980. Section G.2 includes a discussion of the data summaries and a comparison is made between data collected by the initial monitoring program (August 1967 July 1968) and data collected by the current monitoring program (January 1980 December 1980). It was concluded that results from both monitoring programs are compatible, and that both programs produced data bases which are representative of site meteorology. The information provided in section G.2 superceded the information in section G.1. Consequently, the information provided in Table G.1.6 is considered historical, is obsolete and may be deleted from the DSAR.	
		Table G.1.7	Monthly Wind Summary February 1968	Delete Table G.1.7, Monthly Wind Summary February 1968. The Initial On-Site Meteorological Program collected meteorological data from August, 1967 to July, 1968, for analysis as the basis for selection of a suitable site diffusion model for the original VY Plant Design and Analysis Report. Section G.1 contains a discussion of the results of that monitoring program and the meteorological model used for dose calculations in the original VY Plant Design and Analysis Report. The On Site Meteorological Data Collection Program was upgraded in early 1976 to meet the intent of Revision 0 of Regulatory Guide 1.23. Section G.2 describes the current on site monitoring program and presents wind and stability data summaries for one full year of operation; January 1, 1980 through December 31, 1980. Section G.2 includes a discussion of the data summaries and a comparison is made between data collected by the initial monitoring program (August 1967 July 1968) and data collected by the current monitoring program (January 1980 December 1980). It was concluded that results from both monitoring programs are compatible, and that both programs produced data bases which are representative of site meteorology. The information provided in section G.2 superceded the information in section G.1. Consequently, the information provided in Table G.1.7 is considered historical, is obsolete and may be deleted from the DSAR.	
		Table G.1.8	Monthly Wind Summary March 1968	Delete Table G.1.8, Monthly Wind Summary March 1968. The Initial On-Site Meteorological Program collected meteorological data from August, 1967 to July, 1968, for analysis as the basis for selection of a suitable site diffusion model for the original VY Plant Design and Analysis Report. Section G.1 contains a discussion of the results of that monitoring program and the meteorological model used for dose calculations in the original VY Plant Design and Analysis Report. The On Site Meteorological Data Collection Program was upgraded in early 1976 to meet the intent of Revision 0 of Regulatory Guide 1.23. Section G.2 describes the current on site monitoring program and presents wind and stability data summaries for one full year of operation; January 1, 1980 through December 31, 1980. Section G.2 includes a discussion of the data summaries and a comparison is made between data collected by the initial monitoring program (August 1967 July 1968) and data collected by the current monitoring program (January 1980 December 1980). It was concluded that results from both monitoring programs are compatible, and that both programs produced data bases which are representative of site meteorology. The information provided in section G.2 superceded the information in section G.1. Consequently, the information provided in Table G.1.8 is considered historical, is obsolete and may be deleted from the DSAR.	
		Table G.1.9	Monthly Wind Summary April 1968	Delete Table G.1.9, Monthly Wind Summary April 1968. The Initial On-Site Meteorological Program collected meteorological data from August, 1967 to July, 1968, for analysis as the basis for selection of a suitable site diffusion model for the original VY Plant Design and Analysis Report. Section G.1 contains a discussion of the results of that monitoring program and the meteorological model used for dose calculations in the original VY Plant Design and Analysis Report. The On Site Meteorological Data Collection Program was upgraded in early 1976 to meet the intent of Revision 0 of Regulatory Guide 1.23. Section G.2 describes the current on site monitoring program and presents wind and stability data summaries for one full year of operation; January 1, 1980 through December 31, 1980. Section G.2 includes a discussion of the data summaries and a comparison is made between data collected by the initial monitoring program (August 1967 July 1968) and data collected by the current monitoring program (January 1980 December 1980). It was concluded that results from both monitoring programs are compatible, and that both programs produced data bases which are representative of site meteorology. The information provided in section G.2 superceded the information in section G.1. Consequently, the information provided in Table G.1.9 is considered historical, is obsolete and may be deleted from the DSAR.	
		Table G.1.10	Monthly Wind Summary May 1968	Delete Table G.1.10, Monthly Wind Summary May 1968. The Initial On-Site Meteorological Program collected meteorological data from August, 1967 to July, 1968, for analysis as the basis for selection of a suitable site diffusion model for the original VY Plant Design and Analysis Report. Section G.1 contains a discussion of the results of that monitoring program and the meteorological model used for dose calculations in the original VY Plant Design and Analysis Report. The On Site Meteorological Data Collection Program was upgraded in early 1976 to meet the intent of Revision 0 of Regulatory Guide 1.23. Section G.2 describes the current on site monitoring program and presents wind and stability data summaries for one full year of operation; January 1, 1980 through December 31, 1980. Section G.2 includes a discussion of the data summaries and a comparison is made between data collected by the initial monitoring program (August 1967 July 1968) and data collected by the current monitoring program (January 1980 December 1980). It was concluded that results from both monitoring programs are compatible, and that both programs produced data bases which are representative of site meteorology. The information provided in section G.2 superceded the information in section G.1. Consequently, the information provided in Table G.1.10 is considered historical, is obsolete and may be deleted from the DSAR.	

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		Table G.1.11	Monthly Wind Summary June 1968	Delete Table G.1.11, Monthly Wind Summary June 1968. The Initial On-Site Meteorological Program collected meteorological data from August, 1967 to July, 1968, for analysis as the basis for selection of a suitable site diffusion model for the original VY Plant Design and Analysis Report. Section G.1 contains a discussion of the results of that monitoring program and the meteorological model used for dose calculations in the original VY Plant Design and Analysis Report. The On Site Meteorological Data Collection Program was upgraded in early 1976 to meet the intent of Revision 0 of Regulatory Guide 1.23. Section G.2 describes the current on site monitoring program and presents wind and stability data summaries for one full year of operation; January 1, 1980 through December 31, 1980. Section G.2 includes a discussion of the data summaries and a comparison is made between data collected by the initial monitoring program (August 1967 July 1968) and data collected by the current monitoring program (January 1980 December 1980). It was concluded that results from both monitoring programs are compatible, and that both programs produced data bases which are representative of site meteorology. The information provided in section G.2 superceded the information in section G.1. Consequently, the information provided in Table G.1.11 is considered historical, is obsolete and may be deleted from the DSAR.	
		Table G.1.12	Monthly Wind Summary July 1968	Delete Table G.1.12, Monthly Wind Summary July 1968. The Initial On-Site Meteorological Program collected meteorological data from August, 1967 to July, 1968, for analysis as the basis for selection of a suitable site diffusion model for the original VY Plant Design and Analysis Report. Section G.1 contains a discussion of the results of that monitoring program and the meteorological model used for dose calculations in the original VY Plant Design and Analysis Report. The On Site Meteorological Data Collection Program was upgraded in early 1976 to meet the intent of Revision 0 of Regulatory Guide 1.23. Section G.2 describes the current on site monitoring program and presents wind and stability data summaries for one full year of operation; January 1, 1980 through December 31, 1980. Section G.2 includes a discussion of the data summaries and a comparison is made between data collected by the initial monitoring program (August 1967 July 1968) and data collected by the current monitoring program (January 1980 December 1980). It was concluded that results from both monitoring programs are compatible, and that both programs produced data bases which are representative of site meteorology. The information provided in section G.2 superceded the information in section G.1. Consequently, the information provided in Table G.1.12 is considered historical, is obsolete and may be deleted from the DSAR.	
		Table G.1.13	Seasonal Wind Summary Summer; August 1967, June July 1968	Delete Table G.1.13, Seasonal Wind Summary Summer; August 1967, June July 1968. The Initial On-Site Meteorological Program collected meteorological data from August, 1967 to July, 1968, for analysis as the basis for selection of a suitable site diffusion model for the original VY Plant Design and Analysis Report. Section G.1 contains a discussion of the results of that monitoring program and the meteorological model used for dose calculations in the original VY Plant Design and Analysis Report. The On Site Meteorological Data Collection Program was upgraded in early 1976 to meet the intent of Revision 0 of Regulatory Guide 1.23. Section G.2 describes the current on site monitoring program and presents wind and stability data summaries for one full year of operation; January 1, 1980 through December 31, 1980. Section G.2 includes a discussion of the data summaries and a comparison is made between data collected by the initial monitoring program (August 1967 July 1968) and data collected by the current monitoring program (January 1980 December 1980). It was concluded that results from both monitoring programs are compatible, and that both programs produced data bases which are representative of site meteorology. The information provided in section G.2 superceded the information in section G.1. Consequently, the information provided in Table G.1.13 is considered historical, is obsolete and may be deleted from the DSAR.	
		Table G.1.14	Seasonal Wind Summary Fall; September November 1967	Delete Table G.1.14, Seasonal Wind Summary Fall; September November 1967. The Initial On-Site Meteorological Program collected meteorological data from August, 1967 to July, 1968, for analysis as the basis for selection of a suitable site diffusion model for the original VY Plant Design and Analysis Report. Section G.1 contains a discussion of the results of that monitoring program and the meteorological model used for dose calculations in the original VY Plant Design and Analysis Report. The On Site Meteorological Data Collection Program was upgraded in early 1976 to meet the intent of Revision 0 of Regulatory Guide 1.23. Section G.2 describes the current on site monitoring program and presents wind and stability data summaries for one full year of operation; January 1, 1980 through December 31, 1980. Section G.2 includes a discussion of the data summaries and a comparison is made between data collected by the initial monitoring program (August 1967 July 1968) and data collected by the current monitoring program (January 1980 December 1980). It was concluded that results from both monitoring programs are compatible, and that both programs produced data bases which are representative of site meteorology. The information provided in section G.2 superceded the information in section G.1. Consequently, the information provided in Table G.1.14 is considered historical, is obsolete and may be deleted from the DSAR.	
		Table G.1.15	Seasonal Wind Summary Winter; December 1967 February 1968	Delete Table G.1.15, Seasonal Wind Summary Winter; December 1967 February 1968. The Initial On-Site Meteorological Program collected meteorological data from August, 1967 to July, 1968, for analysis as the basis for selection of a suitable site diffusion model for the original VY Plant Design and Analysis Report. Section G.1 contains a discussion of the results of that monitoring program and the meteorological model used for dose calculations in the original VY Plant Design and Analysis Report. The On Site Meteorological Data Collection Program was upgraded in early 1976 to meet the intent of Revision 0 of Regulatory Guide 1.23. Section G.2 describes the current on site monitoring program and presents wind and stability data summaries for one full year of operation; January 1, 1980 through December 31, 1980. Section G.2 includes a discussion of the data summaries and a comparison is made between data collected by the initial monitoring program (August 1967 July 1968) and data collected by the current monitoring program (January 1980 December 1980). It was concluded that results from both monitoring programs are compatible, and that both programs produced data bases which are representative of site meteorology. The information provided in section G.2 superceded the information in section G.1. Consequently, the information provided in Table G.1.15 is considered historical, is obsolete and may be deleted from the DSAR.	

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		Table G.1.16	Seasonal Wind Summary Spring; March May 1968	Delete Table G.1.16, Seasonal Wind Summary Spring; March May 1968. The Initial On-Site Meteorological Program collected meteorological data from August, 1967 to July, 1968, for analysis as the basis for selection of a suitable site diffusion model for the original VY Plant Design and Analysis Report. Section G.1 contains a discussion of the results of that monitoring program and the meteorological model used for dose calculations in the original VY Plant Design and Analysis Report. The On Site Meteorological Data Collection Program was upgraded in early 1976 to meet the intent of Revision 0 of Regulatory Guide 1.23. Section G.2 describes the current on site monitoring program and presents wind and stability data summaries for one full year of operation; January 1, 1980 through December 31, 1980. Section G.2 includes a discussion of the data summaries and a comparison is made between data collected by the initial monitoring program (August 1967 July 1968) and data collected by the current monitoring program (January 1980 December 1980). It was concluded that results from both monitoring programs are compatible, and that both programs produced data bases which are representative of site meteorology. The information provided in section G.2 superceded the information in section G.1. Consequently, the information provided in Table G.1.16 is considered historical, is obsolete and may be deleted from the DSAR.	
		Table G.1.17	Annual Wind Summary August 1967 July 1968	Delete Table G.1.17, Annual Wind Summary August 1967 July 1968. The Initial On-Site Meteorological Program collected meteorological data from August, 1967 to July, 1968, for analysis as the basis for selection of a suitable site diffusion model for the original VY Plant Design and Analysis Report. Section G.1 contains a discussion of the results of that monitoring program and the meteorological model used for dose calculations in the original VY Plant Design and Analysis Report. The On Site Meteorological Data Collection Program was upgraded in early 1976 to meet the intent of Revision 0 of Regulatory Guide 1.23. Section G.2 describes the current on site monitoring program and presents wind and stability data summaries for one full year of operation; January 1, 1980 through December 31, 1980. Section G.2 includes a discussion of the data summaries and a comparison is made between data collected by the initial monitoring program (August 1967 July 1968) and data collected by the current monitoring program (January 1980 December 1980). It was concluded that results from both monitoring programs are compatible, and that both programs produced data bases which are representative of site meteorology. The information provided in section G.2 superceded the information in section G.1. Consequently, the information provided in Table G.1.17 is considered historical, is obsolete and may be deleted from the DSAR.	
		Table G.1.18	Wind Direction Persistence Summary in 22.5 Degree Sectors All Classes of Stability Combined	Delete Table G.1.18, Wind Direction Persistence Summary in 22.5 Degree Sectors All Classes of Stability Combined. The Initial On-Site Meteorological Program collected meteorological data from August, 1967 to July, 1968, for analysis as the basis for selection of a suitable site diffusion model for the original VY Plant Design and Analysis Report. Section G.1 contains a discussion of the results of that monitoring program and the meteorological model used for dose calculations in the original VY Plant Design and Analysis Report. The On Site Meteorological Data Collection Program was upgraded in early 1976 to meet the intent of Revision 0 of Regulatory Guide 1.23. Section G.2 describes the current on site monitoring program and presents wind and stability data summaries for one full year of operation; January 1, 1980 through December 31, 1980. Section G.2 includes a discussion of the data summaries and a comparison is made between data collected by the initial monitoring program (August 1967 July 1968) and data collected by the current monitoring program (January 1980 December 1980). It was concluded that results from both monitoring programs are compatible, and that both programs produced data bases which are representative of site meteorology. The information provided in section G.2 superceded the information in section G.1. Consequently, the information provided in Table G.1.18 is considered historical, is obsolete and may be deleted from the DSAR.	
		Figure G.1-1	Annual Wind Rose, All Stability Classes Combined	Delete Figure G.1-1, Annual Wind Rose, All Stability Classes Combined. The Initial On-Site Meteorological Program collected meteorological data from August, 1967 to July, 1968, for analysis as the basis for selection of a suitable site diffusion model for the original VY Plant Design and Analysis Report. Section G.1 contains a discussion of the results of that monitoring program and the meteorological model used for dose calculations in the original VY Plant Design and Analysis Report. The On Site Meteorological Data Collection Program was upgraded in early 1976 to meet the intent of Revision 0 of Regulatory Guide 1.23. Section G.2 describes the current on site monitoring program and presents wind and stability data summaries for one full year of operation; January 1, 1980 through December 31, 1980. Section G.2 includes a discussion of the data summaries and a comparison is made between data collected by the initial monitoring program (August 1967 July 1968) and data collected by the current monitoring program (January 1980 December 1980). It was concluded that results from both monitoring programs are compatible, and that both programs produced data bases which are representative of site meteorology. The information provided in section G.2 superceded the information in section G.1. Consequently, the information provided in Figure G.1-1 is considered historical, is obsolete and may be deleted from the DSAR.	
		Figure G.1-2	Fall Wind Rose, All Stability Classes Combined	Delete Figure G.1-2, Fall Wind Rose, All Stability Classes Combined. The Initial On-Site Meteorological Program collected meteorological data from August, 1967 to July, 1968, for analysis as the basis for selection of a suitable site diffusion model for the original VY Plant Design and Analysis Report. Section G.1 contains a discussion of the results of that monitoring program and the meteorological model used for dose calculations in the original VY Plant Design and Analysis Report. The On Site Meteorological Data Collection Program was upgraded in early 1976 to meet the intent of Revision 0 of Regulatory Guide 1.23. Section G.2 describes the current on site monitoring program and presents wind and stability data summaries for one full year of operation; January 1, 1980 through December 31, 1980. Section G.2 includes a discussion of the data summaries and a comparison is made between data collected by the initial monitoring program (August 1967 July 1968) and data collected by the current monitoring program (January 1980 December 1980). It was concluded that results from both monitoring programs are compatible, and that both programs produced data bases which are representative of site meteorology. The information provided in section G.2 superceded the information in section G.1. Consequently, the information provided in Figure G.1-2 is considered historical, is obsolete and may be deleted from the DSAR.	

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		Figure G.1-3	Winter Wind Rose, All Stability Classes Combined	Delete Figure G.1-3, Winter Wind Rose, All Stability Classes Combined. The Initial On-Site Meteorological Program collected meteorological data from August, 1967 to July, 1968, for analysis as the basis for selection of a suitable site diffusion model for the original VY Plant Design and Analysis Report. Section G.1 contains a discussion of the results of that monitoring program and the meteorological model used for dose calculations in the original VY Plant Design and Analysis Report. The On Site Meteorological Data Collection Program was upgraded in early 1976 to meet the intent of Revision 0 of Regulatory Guide 1.23. Section G.2 describes the current on site monitoring program and presents wind and stability data summaries for one full year of operation; January 1, 1980 through December 31, 1980. Section G.2 includes a discussion of the data summaries and a comparison is made between data collected by the initial monitoring program (August 1967 July 1968) and data collected by the current monitoring program (January 1980 December 1980). It was concluded that results from both monitoring programs are compatible, and that both programs produced data bases which are representative of site meteorology. The information provided in section G.2 superceded the information in section G.1. Consequently, the information provided in Figure G.1-3 is considered historical, is obsolete and may be deleted from the DSAR.	
		Figure G.1-4	Spring Wind Rose, All Stability Classes Combined	Delete Figure G.1-4, Spring Wind Rose, All Stability Classes Combined. The Initial On-Site Meteorological Program collected meteorological data from August, 1967 to July, 1968, for analysis as the basis for selection of a suitable site diffusion model for the original VY Plant Design and Analysis Report. Section G.1 contains a discussion of the results of that monitoring program and the meteorological model used for dose calculations in the original VY Plant Design and Analysis Report. The On Site Meteorological Data Collection Program was upgraded in early 1976 to meet the intent of Revision 0 of Regulatory Guide 1.23. Section G.2 describes the current on site monitoring program and presents wind and stability data summaries for one full year of operation; January 1, 1980 through December 31, 1980. Section G.2 includes a discussion of the data summaries and a comparison is made between data collected by the initial monitoring program (August 1967 July 1968) and data collected by the current monitoring program (January 1980 December 1980). It was concluded that results from both monitoring programs are compatible, and that both programs produced data bases which are representative of site meteorology. The information provided in section G.2 superceded the information in section G.1. Consequently, the information provided in Figure G.1-4 is considered historical, is obsolete and may be deleted from the DSAR.	
		Figure G.1-5	Summer Wind Rose, All Stability Classes Combined	Delete Figure G.1-5, Summer Wind Rose, All Stability Classes Combined. The Initial On-Site Meteorological Program collected meteorological data from August, 1967 to July, 1968, for analysis as the basis for selection of a suitable site diffusion model for the original VY Plant Design and Analysis Report. Section G.1 contains a discussion of the results of that monitoring program and the meteorological model used for dose calculations in the original VY Plant Design and Analysis Report. The On Site Meteorological Data Collection Program was upgraded in early 1976 to meet the intent of Revision 0 of Regulatory Guide 1.23. Section G.2 describes the current on site monitoring program and presents wind and stability data summaries for one full year of operation; January 1, 1980 through December 31, 1980. Section G.2 includes a discussion of the data summaries and a comparison is made between data collected by the initial monitoring program (August 1967 July 1968) and data collected by the current monitoring program (January 1980 December 1980). It was concluded that results from both monitoring programs are compatible, and that both programs produced data bases which are representative of site meteorology. The information provided in section G.2 superceded the information in section G.1. Consequently, the information provided in Figure G.1-5 is considered historical, is obsolete and may be deleted from the DSAR.	
		Figure G.1-6	Annual Wind Rose, Moderately and Slightly Stable Conditions	Delete Figure G.1-6, Annual Wind Rose, Moderately and Slightly Stable Conditions. The Initial On-Site Meteorological Program collected meteorological data from August, 1967 to July, 1968, for analysis as the basis for selection of a suitable site diffusion model for the original VY Plant Design and Analysis Report. Section G.1 contains a discussion of the results of that monitoring program and the meteorological model used for dose calculations in the original VY Plant Design and Analysis Report. The On Site Meteorological Data Collection Program was upgraded in early 1976 to meet the intent of Revision 0 of Regulatory Guide 1.23. Section G.2 describes the current on site monitoring program and presents wind and stability data summaries for one full year of operation; January 1, 1980 through December 31, 1980. Section G.2 includes a discussion of the data summaries and a comparison is made between data collected by the initial monitoring program (August 1967 July 1968) and data collected by the current monitoring program (January 1980 December 1980). It was concluded that results from both monitoring programs are compatible, and that both programs produced data bases which are representative of site meteorology. The information provided in section G.2 superceded the information in section G.1. Consequently, the information provided in Figure G.1-6 is considered historical, is obsolete and may be deleted from the DSAR.	
		Figure G.1-7	Annual Wind Rose with Precipitation Occurrence	Delete Figure G.1-7, Annual Wind Rose with Precipitation Occurrence. The Initial On-Site Meteorological Program collected meteorological data from August, 1967 to July, 1968, for analysis as the basis for selection of a suitable site diffusion model for the original VY Plant Design and Analysis Report. Section G.1 contains a discussion of the results of that monitoring program and the meteorological model used for dose calculations in the original VY Plant Design and Analysis Report. The On Site Meteorological Data Collection Program was upgraded in early 1976 to meet the intent of Revision 0 of Regulatory Guide 1.23. Section G.2 describes the current on site monitoring program and presents wind and stability data summaries for one full year of operation; January 1, 1980 through December 31, 1980. Section G.2 includes a discussion of the data summaries and a comparison is made between data collected by the initial monitoring program (August 1967 July 1968) and data collected by the current monitoring program (January 1980 December 1980). It was concluded that results from both monitoring programs are compatible, and that both programs produced data bases which are representative of site meteorology. The information provided in section G.2 superceded the information in section G.1. Consequently, the information provided in Figure G.1-7 is considered historical, is obsolete and may be deleted from the DSAR.	

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UFSAR Section				FSAR Conversion to DSAR Change Summary	DSAR Section
		Figure G.1-8	Annual Wind Rose with No Precipitation Occurrence	Delete Figure G.1-8, Annual Wind Rose with No Precipitation Occurrence. The Initial On-Site Meteorological Program collected meteorological data from August, 1967 to July, 1968, for analysis as the basis for selection of a suitable site diffusion model for the original VY Plant Design and Analysis Report. Section G.1 contains a discussion of the results of that monitoring program and the meteorological model used for dose calculations in the original VY Plant Design and Analysis Report. The On Site Meteorological Data Collection Program was upgraded in early 1976 to meet the intent of Revision 0 of Regulatory Guide 1.23. Section G.2 describes the current on site monitoring program and presents wind and stability data summaries for one full year of operation; January 1, 1980 through December 31, 1980. Section G.2 includes a discussion of the data summaries and a comparison is made between data collected by the initial monitoring program (August 1967 July 1968) and data collected by the current monitoring program (January 1980 December 1980). It was concluded that results from both monitoring programs are compatible, and that both programs produced data bases which are representative of site meteorology. The information provided in section G.2 superceded the information in section G.1. Consequently, the information provided in Figure G.1-8 is considered historical, is obsolete and may be deleted from the DSAR.	
		Figure G.1-9	Monthly Inversion Frequency and Average Wind Speed During Inversions	Delete Figure G.1-9, Monthly Inversion Frequency and Average Wind Speed During Inversions. The Initial On-Site Meteorological Program collected meteorological data from August, 1967 to July, 1968, for analysis as the basis for selection of a suitable site diffusion model for the original VY Plant Design and Analysis Report. Section G.1 contains a discussion of the results of that monitoring program and the meteorological model used for dose calculations in the original VY Plant Design and Analysis Report. The On Site Meteorological Data Collection Program was upgraded in early 1976 to meet the intent of Revision 0 of Regulatory Guide 1.23. Section G.2 describes the current on site monitoring program and presents wind and stability data summaries for one full year of operation; January 1, 1980 through December 31, 1980. Section G.2 includes a discussion of the data summaries and a comparison is made between data collected by the initial monitoring program (August 1967 July 1968) and data collected by the current monitoring program (January 1980 December 1980). It was concluded that results from both monitoring programs are compatible, and that both programs produced data bases which are representative of site meteorology. The information provided in section G.2 superceded the information in section G.1. Consequently, the information provided in Figure G.1-9 is considered historical, is obsolete and may be deleted from the DSAR.	
		Figure G.1-10	Number of Hours for Each Wind Speed Group for 32 Hour December 1967 Inversion	Delete Figure G.1-10, Number of Hours for Each Wind Speed Group for 32 Hour December 1967 Inversion. The Initial On-Site Meteorological Program collected meteorological data from August, 1967 to July, 1968, for analysis as the basis for selection of a suitable site diffusion model for the original VY Plant Design and Analysis Report. Section G.1 contains a discussion of the results of that monitoring program and the meteorological model used for dose calculations in the original VY Plant Design and Analysis Report. The On Site Meteorological Data Collection Program was upgraded in early 1976 to meet the intent of Revision 0 of Regulatory Guide 1.23. Section G.2 describes the current on site monitoring program and presents wind and stability data summaries for one full year of operation; January 1, 1980 through December 31, 1980. Section G.2 includes a discussion of the data summaries and a comparison is made between data collected by the initial monitoring program (August 1967 July 1968) and data collected by the current monitoring program (January 1980 December 1980). It was concluded that results from both monitoring programs are compatible, and that both programs produced data bases which are representative of site meteorology. The information provided in section G.2 superceded the information in section G.1. Consequently, the information provided in Figure G.1-10 is considered historical, is obsolete and may be deleted from the DSAR.	
		Figure G.1-11	Wind Direction Distribution During 32 Hour December Inversion	Delete Figure G.1-11, Wind Direction Distribution During 32 Hour December Inversion. The Initial On-Site Meteorological Program collected meteorological data from August, 1967 to July, 1968, for analysis as the basis for selection of a suitable site diffusion model for the original VY Plant Design and Analysis Report. Section G.1 contains a discussion of the results of that monitoring program and the meteorological model used for dose calculations in the original VY Plant Design and Analysis Report. The On Site Meteorological Data Collection Program was upgraded in early 1976 to meet the intent of Revision 0 of Regulatory Guide 1.23. Section G.2 describes the current on site monitoring program and presents wind and stability data summaries for one full year of operation; January 1, 1980 through December 31, 1980. Section G.2 includes a discussion of the data summaries and a comparison is made between data collected by the initial monitoring program (August 1967 July 1968) and data collected by the current monitoring program (January 1980 December 1980). It was concluded that results from both monitoring programs are compatible, and that both programs produced data bases which are representative of site meteorology. The information provided in section G.2 superceded the information in section G.1. Consequently, the information provided in Figure G.1-11 is considered historical, is obsolete and may be deleted from the DSAR.	
		Figure G.1-12	Duration of Longest Inversion in Each Month Between August 1967 and July 1968	Delete Figure G.1-12, Duration of Longest Inversion in Each Month Between August 1967 and July 1968. The Initial On-Site Meteorological Program collected meteorological data from August, 1967 to July, 1968, for analysis as the basis for selection of a suitable site diffusion model for the original VY Plant Design and Analysis Report. Section G.1 contains a discussion of the results of that monitoring program and the meteorological model used for dose calculations in the original VY Plant Design and Analysis Report. The On Site Meteorological Data Collection Program was upgraded in early 1976 to meet the intent of Revision 0 of Regulatory Guide 1.23. Section G.2 describes the current on site monitoring program and presents wind and stability data summaries for one full year of operation; January 1, 1980 through December 31, 1980. Section G.2 includes a discussion of the data summaries and a comparison is made between data collected by the initial monitoring program (August 1967 July 1968) and data collected by the current monitoring program (January 1980 December 1980). It was concluded that results from both monitoring programs are compatible, and that both programs produced data bases which are representative of site meteorology. The information provided in section G.2 superceded the information in section G.1. Consequently, the information provided in Figure G.1-12 is considered historical, is obsolete and may be deleted from the DSAR.	

VY UFSAR					VY DSAR				
UFSAR Section				FSAR Conversion to DSAR Change Summary	DSAR Section				
		Figure G.1-13	Cumulative Frequency of Occurrence of	Delete Figure G.1-13, Cumulative Frequency of Occurrence of . The Initial On-Site Meteorological Program collected meteorological data from August, 1967 to July, 1968, for analysis as the basis for selection of a suitable site diffusion model for the original VY Plant Design and Analysis Report. Section G.1 contains a discussion of the results of that monitoring program and the meteorological model used for dose calculations in the original VY Plant Design and Analysis Report. The On Site Meteorological Data Collection Program was upgraded in early 1976 to meet the intent of Revision 0 of Regulatory Guide 1.23. Section G.2 describes the current on site monitoring program and presents wind and stability data summaries for one full year of operation; January 1, 1980 through December 31, 1980. Section G.2 includes a discussion of the data summaries and a comparison is made between data collected by the initial monitoring program (August 1967 July 1968) and data collected by the current monitoring program (January 1980 December 1980). It was concluded that results from both monitoring programs are compatible, and that both programs produced data bases which are representative of site meteorology. The information provided in section G.2 superceded the information in section G.1. Consequently, the information provided in Figure G.1-13 is considered historical, is obsolete and may be deleted from the DSAR.					
	G.2	CURRENT ON SITE METEOROLOGICAL PROGRAM		Section G.2 describes the current on site meteorological monitoring program and presents wind and stability data summaries for one full year of operation; January 1, 1980 through December 31, 1980. A discussion of the data summaries is included, and a comparison is made between data collected by the initial monitoring program (August 1967 July 1968) and data collected by the current monitoring program (January 1980 December 1980). Results from both monitoring programs were compatible, and both programs produced data bases which were representative of site meteorology. Section G.2 supports the meteorology section of the SAR. Section G.2 supports meeting the requirements of Reg Guide 1.23, Rev. 0. No changes are required.		G.2	CURRENT ON SITE METEOROLOGICAL PROGRAM		
		G.2.1	Introduction	No Changes			G.2.1	Introduction	
		G.2.2	Description of the Monitoring Program	No Changes			G.2.2	Description of the Monitoring Program	
		G.2.3	Results	No Changes			G.2.3	Results	
		Table G.2.1	Meteorological Data Recovery Rates for 1980	No Changes			Table G.2.1	Meteorological Data Recovery Rates for 1980	
		Table G.2.2	Joint Frequency Distribution of Wind Speed, Wind Direction, and Stability	No Changes			Table G.2.2	Joint Frequency Distribution of Wind Speed, Wind Direction, and	
		Table G.2.3	Joint Frequency Distribution of Wind Speed, Wind Direction, and Stability	No Changes			Table G.2.3	Joint Frequency Distribution of Wind Speed, Wind Direction, and	
		Table G.2.4	Wind Direction Persistence Summary (35 foot level)	No Changes			Table G.2.4	Wind Direction Persistence Summary (35 foot level)	
		Table G.2.5	Wind Direction Persistence Summary (297 foot level)	No Changes			Table G.2.5	Wind Direction Persistence Summary (297 foot level)	
		Table G.2.6	Inversion Persistence Summary (198-33 foot Delta T)	No Changes			Table G.2.6	Inversion Persistence Summary (198-33 foot Delta T)	
		Table G.2.7	Inversion Persistence Summary (295-33 foot Delta T)	No Changes			Table G.2.7	Inversion Persistence Summary (295-33 foot Delta T)	
		Figure G.2-1	Location of Primary and Backup Meteorological Towers	No Changes			Figure G.2-1	Location of Primary and Backup Meteorological Towers	
		Figure G.2-2	Spring Wind Rose (35 foot level) March 1980 May 1980	No Changes			Figure G.2-2	Spring Wind Rose (35 foot level) March 1980 May 1980	
		Figure G.2-3	Summer Wind Rose (35 foot level) June 1980 August 1980	No Changes			Figure G.2-3	Summer Wind Rose (35 foot level) June 1980 August 1980	
		Figure G.2-4	Autumn Wind Rose (35 foot level) September 1980 November 1980	No Changes			Figure G.2-4	Autumn Wind Rose (35 foot level) September 1980 November 1980	
		Figure G.2-5	Winter Wind Rose (35 foot level) January 1980 February 1980; December 1980	No Changes			Figure G.2-5	Winter Wind Rose (35 foot level) January 1980 February 1980; December 1980	
		Figure G.2-6	Annual Wind Rose (35 foot level) January 1980 December 1980	No Changes			Figure G.2-6	Annual Wind Rose (35 foot level) January 1980 December 1980	

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		Figure G.2-7	Spring Wind Rose (297 foot level) March 1980 May 1980	No Changes			Figure G.2-7	Spring Wind Rose (297 foot level) March 1980 May 1980
		Figure G.2-8	Summer Wind Rose (297 foot level) June 1980 August 1980	No Changes			Figure G.2-8	Summer Wind Rose (297 foot level) June 1980 August 1980
		Figure G.2-9	Autumn Wind Rose (297 foot level) September 1980 November 1980	No Changes			Figure G.2-9	Autumn Wind Rose (297 foot level) September 1980 November 1980
		Figure G.2-10	Winter Wind Rose (297 foot level) January 1980 February 1980; December 1980	No Changes			Figure G.2-10	Winter Wind Rose (297 foot level) January 1980 February 1980; December 1980
		Figure G.2-11	Annual Wind Rose (297 foot level) January 1980 December 1980	No Changes			Figure G.2-11	Annual Wind Rose (297 foot level) January 1980 December 1980
H	THERMAL-HYDRAULIC DESIGN							
	H.1	Summary Description		Delete Section H.1, Thermal Hydraulic Design, Summary Description. The material presented in this appendix describes the analytical methodology used in the initial reactor core design. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related transients are no longer possible. Since a reactor core can no longer be loaded into the reactor vessel, reactor core design criteria, thermal margins and thermal limits are no longer applicable and are obsolete. Additionally, since Sections H.2 and H.3 are deleted from the DSAR, the summary of those sections presented in Section H.1 is no longer applicable and is obsolete.				
	H.2	CORE THERMAL DESIGN		Section H.2, Core Thermal Design, reviewed initial reactor core design criteria and, by presentation of analytical data, showed the existence of adequate thermal margins. The thermal operating limits for the Vermont Yankee Nuclear Power Station were evaluated for each cycle of operation. These were presented in the current cycle Core Performance Analysis Report as well as in the Technical Specifications. Section H.2 also supplemented the core thermal design information given in Section 3.7 of the UFSAR.				
		H.2.1	Introduction	Delete Section H.2.1, Introduction. Section H.2, Core Thermal Design, reviewed initial reactor core design criteria and, by presentation of analytical data, showed the existence of adequate core thermal margins. Section H.2 supplemented the core thermal design information given in Section 3.7 of the UFSAR. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related transients are no longer possible. Since a reactor core can no longer be loaded into the reactor vessel, reactor core design criteria, thermal margins and thermal limits are no longer applicable and are obsolete. Additionally, UFSAR Section 3.7 has been deleted.				
		H.2.2	Reactor Core Design Bases Limits	Delete Section H.2.2, Reactor Core Design Bases Limits. Section H.2, Core Thermal Design, reviewed initial reactor core design criteria and, by presentation of analytical data, showed the existence of adequate core thermal margins. Section H.2 supplemented the core thermal design information given in Section 3.7 of the UFSAR. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related transients are no longer possible. Since a reactor core can no longer be loaded into the reactor vessel, reactor core design criteria, thermal margins and thermal limits are no longer applicable and are obsolete. Additionally, UFSAR Section 3.7 has been deleted.				
		H.2.3	Typical Initial Core Performance	Delete Section H.2.3,Typical Initial Core Performance. Section H.2, Core Thermal Design, reviewed initial reactor core design criteria and, by presentation of analytical data, showed the existence of adequate core thermal margins. Section H.2 supplemented the core thermal design information given in Section 3.7 of the UFSAR. Section H.2.3 presented expected initial core performance characteristics. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related transients are no longer possible. Since a reactor core can no longer be loaded into the reactor vessel, reactor core design criteria, thermal margins and thermal limits are no longer applicable and are obsolete. Initial core performance characteristics are historical and are no longer applicable. Additionally, UFSAR Section 3.7 has been deleted.				

VY UFSAR					VY DSAR
UFSAR Section				FSAR Conversion to DSAR Change Summary	DSAR Section
		H.2.4	Thermal Margin	Delete Section H.2.4, Thermal Margins. Section H.2, Core Thermal Design, reviewed initial reactor core design criteria and, by presentation of analytical data, showed the existence of adequate core thermal margins. Section H.2 supplemented the core thermal design information given in Section 3.7 of the UFSAR. Section H.2.4 discussed the thermal margin between the proposed reactor design limits and the and fuel damage limits Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related transients are no longer possible. Since a reactor core can no longer be loaded into the reactor vessel, reactor core design criteria, thermal margins and thermal limits are no longer applicable and are obsolete. Additionally, UFSAR Section 3.7 has been deleted	
		H.2.5	Transient Operation	Delete Section H.2.5, Transient Operation. Section H.2, Core Thermal Design, reviewed initial reactor core design criteria and, by presentation of analytical data, showed the existence of adequate core thermal margins. Section H.2 supplemented the core thermal design information given in Section 3.7 of the UFSAR. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related transients are no longer possible. Since a reactor core can no longer be loaded into the reactor vessel, transient operations can no longer occur, reactor core design criteria, thermal margins and thermal limits are no longer applicable and are obsolete. Additionally, UFSAR Section 3.7 has been deleted.	
		H.2.6	Summary	Delete Section H.2.6, Summary. Section H.2, Core Thermal Design, reviewed initial reactor core design criteria and, by presentation of analytical data, showed the existence of adequate core thermal margins. Section H.2 supplemented the core thermal design information given in Section 3.7 of the UFSAR. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related transients are no longer possible. Since a reactor core can no longer be loaded into the reactor vessel, transient operations can no longer occur, reactor core design criteria, thermal margins and thermal limits are no longer applicable and are obsolete. Additionally, UFSAR Section 3.7 has been deleted.	
		H.2.7	Conclusions	Delete Section H.2.7, Conclusions. Section H.2, Core Thermal Design, reviewed initial reactor core design criteria and, by presentation of analytical data, showed the existence of adequate core thermal margins. Section H.2 supplemented the core thermal design information given in Section 3.7 of the UFSAR. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related transients are no longer possible. Since a reactor core can no longer be loaded into the reactor vessel, transient operations can no longer occur, reactor core design criteria, thermal margins and thermal limits are no longer applicable and are obsolete. The conclusions drawn in this section are no longer applicable. Additionally, UFSAR Section 3.7 has been deleted.	
		H.2.8	REFERENCES	Delete Section H.2.8, References Section H.2, Core Thermal Design, reviewed initial reactor core design criteria and, by presentation of analytical data, showed the existence of adequate core thermal margins. Section H.2 supplemented the core thermal design information given in Section 3.7 of the UFSAR. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related transients are no longer possible. Since a reactor core can no longer be loaded into the reactor vessel, transient operations can no longer occur, reactor core design criteria, thermal margins and thermal limits are no longer applicable and are obsolete. The documents referenced in this section are no longer applicable. Additionally, UFSAR Section 3.7 has been deleted.	
		Table H.2.1	PEAK LINEAR HEAT GENERATION RATE (INITIAL REACTOR CORE DESIGN BASES)	Delete Table H.2.1, Peak Linear Heat Generation Rate (Initial Reactor Core Design Bases) Section H.2, Core Thermal Design, reviewed initial reactor core design criteria and, by presentation of analytical data, showed the existence of adequate core thermal margins. Section H.2 supplemented the core thermal design information given in Section 3.7 of the UFSAR. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related transients are no longer possible. Since a reactor core can no longer be loaded into the reactor vessel, transient operations can no longer occur, reactor core design criteria, thermal margins and thermal limits are no longer applicable and are obsolete. The information provided in Table H.2.1 is historical, is no longer applicable and is obsolete. Additionally, UFSAR Section 3.7 has been deleted.	
		Table H.2.2	RESULTS OF TRANSIENTS (INITIAL REACTOR CORE DESIGN BASES)	Delete Table H.2.2, Results of Transients (Initial Reactor Core Design Bases). Section H.2, Core Thermal Design, reviewed initial reactor core design criteria and, by presentation of analytical data, showed the existence of adequate core thermal margins. Section H.2 supplemented the core thermal design information given in Section 3.7 of the UFSAR. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related transients are no longer possible. Since a reactor core can no longer be loaded into the reactor vessel, transient operations can no longer occur, reactor core design criteria, thermal margins and thermal limits are no longer applicable and are obsolete. The information provided in Table H.2.2 is historical, is no longer applicable and is obsolete. Additionally, UFSAR Section 3.7 has been deleted.	

VY UFSAR					VY DSAR
UFSAR Section				FSAR Conversion to DSAR Change Summary	DSAR Section
		Table H.2.3	LIMIT COMPARISONS (INITIAL REACTOR CORE DESIGN BASES)	Delete Table H.2.3, Limit Comparisons (Initial Reactor Core Design Bases). Section H.2, Core Thermal Design, reviewed initial reactor core design criteria and, by presentation of analytical data, showed the existence of adequate core thermal margins. Section H.2 supplemented the core thermal design information given in Section 3.7 of the UFSAR. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related transients are no longer possible. Since a reactor core can no longer be loaded into the reactor vessel, transient operations can no longer occur, reactor core design criteria, thermal margins and thermal limits are no longer applicable and are obsolete. The information provided in Table H.2.3 is historical, is no longer applicable and is obsolete. Additionally, UFSAR Section 3.7 has been deleted.	
	H.3	EFFECT OF UNCERTAINTIES ON CRITICAL HEAT FLUX MARGIN		Section H.3 summarized the core thermal hydraulic uncertainty analysis and presented additional information regarding the effect and resolution of uncertainties in key areas of the initial reactor core design, e.g., the calculations of flow in a single channel and the calculations of the minimum critical heat flux ratio.	
		H.3.1	Introduction and Summary	Delete Section H.3.1, Introduction and Summary Section H.3, Effects of Uncertainties on Critical Heat Flux Margin , summarized the core thermal hydraulic uncertainty analysis and presented additional information regarding the effect and resolution of uncertainties in key areas of the initial reactor core design, e.g., the calculations of flow in a single channel and the calculations of the minimum critical heat flux ratio. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related transients are no longer possible. Since a reactor core can no longer be loaded into the reactor vessel, reactor core design criteria, thermal margins and thermal limits are no longer applicable and are obsolete. Consequently, information regarding critical heat flux margin uncertainties is historical, no longer applicable and is obsolete.	
		H.3.2	Definitions and Objectives of the Analysis	Delete Section H.3.2, Definitions and Objectives of the Analysis. Section H.3, Effects of Uncertainties on Critical Heat Flux Margin , summarized the core thermal hydraulic uncertainty analysis and presented additional information regarding the effect and resolution of uncertainties in key areas of the initial reactor core design, e.g., the calculations of flow in a single channel and the calculations of the minimum critical heat flux ratio.. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related transients are no longer possible. Since a reactor core can no longer be loaded into the reactor vessel, reactor core design criteria, thermal margins and thermal limits are no longer applicable and are obsolete. Consequently, information regarding critical heat flux margin uncertainties is historical, no longer applicable and is obsolete.	
		H.3.3	Analytical Procedures and Results	Delete Section H.3.3, Analytical Procedures and Results. Section H.3, Effects of Uncertainties on Critical Heat Flux Margin , summarized the core thermal hydraulic uncertainty analysis and presented additional information regarding the effect and resolution of uncertainties in key areas of the initial reactor core design, e.g., the calculations of flow in a single channel and the calculations of the minimum critical heat flux ratio.. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related transients are no longer possible. Since a reactor core can no longer be loaded into the reactor vessel, reactor core design criteria, thermal margins and thermal limits are no longer applicable and are obsolete. Consequently, information regarding critical heat flux margin uncertainties is historical, no longer applicable and is obsolete.	
		H.3.4	Reactor Model	Delete Section H.3.4, Reactor Model. Section H.3, Effects of Uncertainties on Critical Heat Flux Margin , summarized the core thermal hydraulic uncertainty analysis and presented additional information regarding the effect and resolution of uncertainties in key areas of the initial reactor core design, e.g., the calculations of flow in a single channel and the calculations of the minimum critical heat flux ratio.. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related transients are no longer possible. Since a reactor core can no longer be loaded into the reactor vessel, reactor core design criteria, thermal margins and thermal limits are no longer applicable and are obsolete. Consequently, information regarding the reactor model used in critical heat flux margin uncertainties and analysis is historical, no longer applicable and is obsolete.	
		H.3.5	The Calculational Process	Delete Section H.3.5, The Calculational Process. Section H.3, Effects of Uncertainties on Critical Heat Flux Margin , summarized the core thermal hydraulic uncertainty analysis and presented additional information regarding the effect and resolution of uncertainties in key areas of the initial reactor core design, e.g., the calculations of flow in a single channel and the calculations of the minimum critical heat flux ratio.. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related transients are no longer possible. Since a reactor core can no longer be loaded into the reactor vessel, reactor core design criteria, thermal margins and thermal limits are no longer applicable and are obsolete. Consequently, information regarding the calculational process used in critical heat flux margin uncertainties and analysis is historical, no longer applicable and is obsolete.	

VY UFSAR					VY DSAR
UFSAR Section				FSAR Conversion to DSAR Change Summary	DSAR Section
		H.3.6	Results	Delete Section H.3.6, Results. Section H.3, Effects of Uncertainties on Critical Heat Flux Margin , summarized the core thermal hydraulic uncertainty analysis and presented additional information regarding the effect and resolution of uncertainties in key areas of the initial reactor core design, e.g., the calculations of flow in a single channel and the calculations of the minimum critical heat flux ratio.. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related transients are no longer possible. Since a reactor core can no longer be loaded into the reactor vessel, reactor core design criteria, thermal margins and thermal limits are no longer applicable and are obsolete. Consequently, information regarding the results of the analysis used in critical heat flux margin uncertainties are historical, no longer applicable and obsolete.	
		H.3.7	Conclusions	Delete Section H.3.7, Conclusions. Section H.3, Effects of Uncertainties on Critical Heat Flux Margin , summarized the core thermal hydraulic uncertainty analysis and presented additional information regarding the effect and resolution of uncertainties in key areas of the initial reactor core design, e.g., the calculations of flow in a single channel and the calculations of the minimum critical heat flux ratio.. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related transients are no longer possible. Since a reactor core can no longer be loaded into the reactor vessel, reactor core design criteria, thermal margins and thermal limits are no longer applicable and are obsolete. Consequently, conclusions regarding the results of the analysis used in critical heat flux margin uncertainties are historical, no longer applicable and are obsolete.	
		H.3.8	References	Delete Section H.3.7, References Section H.3, Effects of Uncertainties on Critical Heat Flux Margin , summarized the core thermal hydraulic uncertainty analysis and presented additional information regarding the effect and resolution of uncertainties in key areas of the initial reactor core design, e.g., the calculations of flow in a single channel and the calculations of the minimum critical heat flux ratio.. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related transients are no longer possible. Since a reactor core can no longer be loaded into the reactor vessel, reactor core design criteria, thermal margins and thermal limits are no longer applicable and are obsolete. Consequently, documents listed in the references section are no longer applicable and are obsolete.	
		Table H.3.1	Uncertainty Incore Power Determination	Delete Table H.3.1, Uncertainty Incore Power Determination Section H.3, Effects of Uncertainties on Critical Heat Flux Margin , summarized the core thermal hydraulic uncertainty analysis and presented additional information regarding the effect and resolution of uncertainties in key areas of the initial reactor core design, e.g., the calculations of flow in a single channel and the calculations of the minimum critical heat flux ratio.. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related transients are no longer possible. Since a reactor core can no longer be loaded into the reactor vessel, reactor core design criteria, thermal margins and thermal limits are no longer applicable and are obsolete. Consequently,the information provided in Table H.3.1 is no longer applicable and is obsolete.	
		Table H.3.2	Overpower Occurrences	Delete Table H.3.2, Overpower Occurrences. Section H.3, Effects of Uncertainties on Critical Heat Flux Margin , summarized the core thermal hydraulic uncertainty analysis and presented additional information regarding the effect and resolution of uncertainties in key areas of the initial reactor core design, e.g., the calculations of flow in a single channel and the calculations of the minimum critical heat flux ratio.. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related transients are no longer possible. Since a reactor core can no longer be loaded into the reactor vessel, reactor core design criteria, thermal margins and thermal limits are no longer applicable and are obsolete. Consequently,the information provided in Table H.3.2 is no longer applicable and is obsolete.	
		Table H.3.3	Manufacturing Tolerance Affecting Local Power	Delete Table H.3.3, Manufacturing Tolerance Affecting Local Power. Section H.3, Effects of Uncertainties on Critical Heat Flux Margin , summarized the core thermal hydraulic uncertainty analysis and presented additional information regarding the effect and resolution of uncertainties in key areas of the initial reactor core design, e.g., the calculations of flow in a single channel and the calculations of the minimum critical heat flux ratio.. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related transients are no longer possible. Since a reactor core can no longer be loaded into the reactor vessel, reactor core design criteria, thermal margins and thermal limits are no longer applicable and are obsolete. Consequently,the information provided in Table H.3.3 is no longer applicable and is obsolete.	

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		Table H.3.4	Base Operating State	Delete Table H.3.4, Base Operating State. Section H.3, Effects of Uncertainties on Critical Heat Flux Margin , summarized the core thermal hydraulic uncertainty analysis and presented additional information regarding the effect and resolution of uncertainties in key areas of the initial reactor core design, e.g., the calculations of flow in a single channel and the calculations of the minimum critical heat flux ratio.. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related transients are no longer possible. Since a reactor core can no longer be loaded into the reactor vessel, reactor core design criteria, thermal margins and thermal limits are no longer applicable and are obsolete. Consequently,the information provided in Table H.3.4 is no longer applicable and is obsolete.	
I	ESTIMATE OF PROBABILITIES OF PIPE BREAK			The Estimate of Probabilities of a Pipe Break presented in Appendix I, attempted to predict the rate of occurrence of through wall cracks during the life of the plant for each piping system and component.	
	I.1	PROBABILITY OF LEAKING FAILURES		Delete Section I.1, Probability of Leaking Failures Appendix I, Estimate of Probabilities of Pipe Break, attempted to assign a probability to predict the rate of occurrence of through wall cracks during the life of the plant. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related transients and design basis accidents are no longer possible. Prediction of the occurrence of through wall cracks for those SSCs which remain in operation following permanent defueling is not required since a failure of those SSCs which remain in service will not initiate a design basis accident, result in fuel damage or threaten the health and safety of the public. Therefore, the information provided in Section I.1 regarding the probability of leaking failures is no longer applicable. This information is historical and is obsolete.	
	I.2	CRITICAL CRACK SIZE		Delete Section I.2, Critical Crack Size. Appendix I, Estimate of Probabilities of Pipe Break, attempted to assign a probability to predict the rate of occurrence of through wall cracks during the life of the plant. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related transients and design basis accidents are no longer possible. Prediction of the occurrence of through wall cracks for those SSCs which remain in operation following permanent defueling is not required since a failure of those SSCs which remain in service will not initiate a design basis accident, result in fuel damage or threaten the health and safety of the public. Therefore, the information provided in Section I.2 regarding critical crack size is no longer applicable. This information is historical and is obsolete.	
	I.3	LEAKAGE FLOW FROM CIRCUMFERENTIAL CRACKS		Delete Section I.3,Leakage Flow From Circumferential Cracks. Appendix I, Estimate of Probabilities of Pipe Break, attempted to assign a probability to predict the rate of occurrence of through wall cracks during the life of the plant. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related transients and design basis accidents are no longer possible. Prediction of the occurrence of through wall cracks for those SSCs which remain in operation following permanent defueling is not required since a failure of those SSCs which remain in service will not initiate a design basis accident, result in fuel damage or threaten the health and safety of the public. Therefore, the information provided in Section I.3 regarding leakage flow from circumferential cracks is no longer applicable. This information is historical and is obsolete.	
	I.4	PROBABILITY OF LINE BREAK		Delete Section I.4, Probability of a Line Break. Appendix I, Estimate of Probabilities of Pipe Break, attempted to assign a probability to predict the rate of occurrence of through wall cracks during the life of the plant. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related transients and design basis accidents are no longer possible. Prediction of the occurrence of through wall cracks for those SSCs which remain in operation following permanent defueling is not required since a failure of those SSCs which remain in service will not initiate a design basis accident, result in fuel damage or threaten the health and safety of the public. Therefore, the information provided in Section I.4 regarding line break probability is no longer applicable. This information is historical and is obsolete.	
	I.5	References		Delete Section I.5, References. Appendix I, Estimate of Probabilities of Pipe Break, attempted to assign a probability to predict the rate of occurrence of through wall cracks during the life of the plant. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related transients and design basis accidents are no longer possible. Prediction of the occurrence of through wall cracks for those SSCs which remain in operation following permanent defueling is not required since a failure of those SSCs which remain in service will not initiate a design basis accident, result in fuel damage or threaten the health and safety of the public. Therefore, the documents referenced in section I.5 are no longer applicable. This information is historical and is obsolete.	

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	Figure I-1	Summary of Fracture Data for Piping Materials	Delete Figure I-1, Summary of Fracture Date for Piping Materials Appendix I, Estimate of Probabilities of Pipe Break, attempted to assign a probability to predict the rate of occurrence of through wall cracks during the life of the plant. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related transients and design basis accidents are no longer possible. Prediction of the occurrence of through wall cracks for those SSCs which remain in operation following permanent defueling is not required since a failure of those SSCs which remain in service will not initiate a design basis accident, result in fuel damage or threaten the health and safety of the public. Therefore, the summary information provided on Figure I-1 regarding fracture data for piping materials is no longer applicable. This information is historical and is obsolete.	
	Figure I-2	Relationship of Leak Rate to Stress Intensity Factor	Delete Figure I-2, Relationship of Leak Rate to Stress Intensity Factor. Appendix I, Estimate of Probabilities of Pipe Break, attempted to assign a probability to predict the rate of occurrence of through wall cracks during the life of the plant. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related transients and design basis accidents are no longer possible. Prediction of the occurrence of through wall cracks for those SSCs which remain in operation following permanent defueling is not required since a failure of those SSCs which remain in service will not initiate a design basis accident, result in fuel damage or threaten the health and safety of the public. Therefore, the information provided on Figure I-2 regarding the relationship between leak rate and stress intensity factor is no longer applicable. This information is historical and is obsolete.	
	Figure I-3	Probability that a Line Break Results from a Leaking Crack	Delete Figure I-3, Probability that a Line Break Results from a Leaking Crack. Appendix I, Estimate of Probabilities of Pipe Break, attempted to assign a probability to predict the rate of occurrence of through wall cracks during the life of the plant. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related transients and design basis accidents are no longer possible. Prediction of the occurrence of through wall cracks for those SSCs which remain in operation following permanent defueling is not required since a failure of those SSCs which remain in service will not initiate a design basis accident, result in fuel damage or threaten the health and safety of the public. Therefore, the information provided on Figure I-3 showing the probability that a line break will result from a leaking crack is no longer applicable. This information is historical and is obsolete.	
J	STRUCTURAL EVALUATION OF REACTOR RECIRCULATION SYSTEM			
	J.1	Scope and Intent	Delete Section J.1, Structural Evaluation of Reactor Recirculation System, Scope and Intent. The Reactor Recirculation System provided a variable moderator (coolant) flow to the reactor core to adjust reactor power level. The system was designed to assure adequate fuel barrier thermal margin remained following Recirculation Pump System malfunctions and that failure of piping integrity did not compromise the ability of the reactor vessel internals to maintain a refloodable volume. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, fuel barrier thermal margin and core reflood following design basis accidents is no longer applicable. As a result, a structural evaluation of the Reactor Recirculation System is no longer required. Reactor Recirculation System information is obsolete.	
	J.2	Loading Conditions	Delete Section J.2, Structural Evaluation of Reactor Recirculation System, Loading Condtions. The Reactor Recirculation System provided a variable moderator (coolant) flow to the reactor core to adjust reactor power level. The system was designed to assure adequate fuel barrier thermal margin remained following Recirculation Pump System malfunctions and that failure of piping integrity did not compromise the ability of the reactor vessel internals to maintain a refloodable volume. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, fuel barrier thermal margin and core reflood following design basis accidents is no longer applicable. As a result, a structural evaluation of the Reactor Recirculation System is no longer required. Reactor Recirculation System information is obsolete.	
	J.3	Allowable Limits	Delete Section J.3, Structural Evaluation of Reactor Recirculation System, Allowable Limits. The Reactor Recirculation System provided a variable moderator (coolant) flow to the reactor core to adjust reactor power level. The system was designed to assure adequate fuel barrier thermal margin remained following Recirculation Pump System malfunctions and that failure of piping integrity did not compromise the ability of the reactor vessel internals to maintain a refloodable volume. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, fuel barrier thermal margin and core reflood following design basis accidents is no longer applicable. As a result, a structural evaluation of the Reactor Recirculation System is no longer required. Reactor Recirculation System information is obsolete.	

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	J.4	Discussion of Results	Delete Section J.4, Structural Evaluation of Reactor Recirculation System, Discussion of Results. The Reactor Recirculation System provided a variable moderator (coolant) flow to the reactor core to adjust reactor power level. The system was designed to assure adequate fuel barrier thermal margin remained following Recirculation Pump System malfunctions and that failure of piping integrity did not compromise the ability of the reactor vessel internals to maintain a refloodable volume. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, fuel barrier thermal margin and core reflood following design basis accidents is no longer applicable. As a result, a structural evaluation of the Reactor Recirculation System is no longer required. Reactor Recirculation System information is obsolete.	
	J.5	References	Delete Section J.5, Structural Evaluation of Reactor Recirculation System, References. The Reactor Recirculation System provided a variable moderator (coolant) flow to the reactor core to adjust reactor power level. The system was designed to assure adequate fuel barrier thermal margin remained following Recirculation Pump System malfunctions and that failure of piping integrity did not compromise the ability of the reactor vessel internals to maintain a refloodable volume. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, fuel barrier thermal margin and core reflood following design basis accidents is no longer applicable. As a result, a structural evaluation of the Reactor Recirculation System is no longer required. References are no longer applicable. Reactor Recirculation System information is obsolete.	
	TABLE J.3-1	RHR and Recirculation Systems Piping Load Combinations	Delete Table J.3-1, RHR and Recirculation Systems Piping Load Combinations. The Reactor Recirculation System provided a variable moderator (coolant) flow to the reactor core to adjust reactor power level. The system was designed to assure adequate fuel barrier thermal margin remained following Recirculation Pump System malfunctions and that failure of piping integrity did not compromise the ability of the reactor vessel internals to maintain a refloodable volume. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. Consequently, fuel barrier thermal margin and core reflood following design basis accidents is no longer applicable. As a result, a structural evaluation of the Reactor Recirculation System is no longer required. The information provided on table J.3-1 is no longer applicable. Reactor Recirculation System information is obsolete.	
K	CORE SHROUD REPAIR			
	K.1	INTRODUCTION AND SUMMARY	Delete section K.1, Core Shroud Repair, Introduction and Summary. Appendix K summarizes the design of the core shroud repair for the Vermont Yankee Nuclear Power Station. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. As described in the justification for elimination of the Reactor Vessel and Appurtenances (UFSAR Section 4.2) from the DSAR, the functions provided by the Reactor Vessel and internals, including the core shroud, are no longer required. Consequently, the historical information provided in this section regarding core shroud repairs is no longer applicable and is considered obsolete.	
	K.2	BACKGROUND	Delete section K.2, Core Shroud Repair, Background. Appendix K summarizes the design of the core shroud repair for the Vermont Yankee Nuclear Power Station. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. As described in the justification for elimination of the Reactor Vessel and Appurtenances (UFSAR Section 4.2) from the DSAR, the functions provided by the Reactor Vessel and internals, including the core shroud, are no longer required. Consequently, the historical information provided in this section regarding core shroud repairs is no longer applicable and is considered obsolete.	
	K.3	DESCRIPTION OF REPAIR	Delete section K.3, Core Shroud Repair, Description of Repair. Appendix K summarizes the design of the core shroud repair for the Vermont Yankee Nuclear Power Station. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. As described in the justification for elimination of the Reactor Vessel and Appurtenances (UFSAR Section 4.2) from the DSAR, the functions provided by the Reactor Vessel and internals, including the core shroud, are no longer required. Consequently, the historical information provided in this section regarding core shroud repairs is no longer applicable and is considered obsolete.	
	K.4	STRUCTURAL AND DESIGN EVALUATION	Delete section K.4, Core Shroud Repair, Structural and Design Evaluation. Appendix K summarizes the design of the core shroud repair for the Vermont Yankee Nuclear Power Station. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. As described in the justification for elimination of the Reactor Vessel and Appurtenances (UFSAR Section 4.2) from the DSAR, the functions provided by the Reactor Vessel and internals, including the core shroud, are no longer required. Consequently, the historical information provided in this section regarding core shroud repairs is no longer applicable and is considered obsolete.	

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	K.5	SEISMIC ANALYSES	Delete section K.5, Core Shroud Repair, Seismic Analysis. Appendix K summarizes the design of the core shroud repair for the Vermont Yankee Nuclear Power Station. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. As described in the justification for elimination of the Reactor Vessel and Appurtenances (UFSAR Section 4.2) from the DSAR, the functions provided by the Reactor Vessel and internals, including the core shroud, are no longer required. Consequently, the seismic analysis provided in this section regarding core shroud repairs is no longer applicable and is considered obsolete.	
	K.6	SYSTEMS EVALUATION	Delete section K.6, Core Shroud Repair, Systems Evaluation. Appendix K summarizes the design of the core shroud repair for the Vermont Yankee Nuclear Power Station. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. As described in the justification for elimination of the Reactor Vessel and Appurtenances (UFSAR Section 4.2) from the DSAR, the functions provided by the Reactor Vessel and internals, including the core shroud, are no longer required. Consequently, the Systems evaluation provided in this section regarding core shroud repairs is no longer applicable and is considered obsolete.	
	K.7	MATERIALS AND FABRICATION	Delete section K.7, Materials and Fabrication. Appendix K summarizes the design of the core shroud repair for the Vermont Yankee Nuclear Power Station. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. As described in the justification for elimination of the Reactor Vessel and Appurtenances (UFSAR Section 4.2) from the DSAR, the functions provided by the Reactor Vessel and internals, including the core shroud, are no longer required. Consequently, the description of the materials and fabrication process used is no longer applicable and is considered obsolete.	
	K.8	REFERENCES	Delete section K.8, References. Appendix K summarizes the design of the core shroud repair for the Vermont Yankee Nuclear Power Station. Following certification of permanent defueling the licensee is no longer authorized to emplace or retain fuel in the reactor vessel IAW 10CFR50.82(a)(2). Since it is no longer possible to load a nuclear core, power operations can no longer occur and core related design basis accidents are no longer possible. As described in the justification for elimination of the Reactor Vessel and Appurtenances (UFSAR Section 4.2) from the DSAR, the functions provided by the Reactor Vessel and internals, including the core shroud, are no longer required. Consequently, the references listed in Section K.8 are longer applicable and are considered obsolete.	