Northern States Power Company

414 Nicollet Mall
Minneapolis, Minnesota 55401
Telephone (612) 330-5500

June 29, 1984

Director
Office of Nuclear Reactor Regulation
U S Nuclear Regulatory Commission
Washington, DC 20555

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
Docket Nos. 50-282 License Nos. DPR-42
50-306 DPR-60

Submittal of Revision No. 2 to the Updated Safety Analysis Report (USAR)

Purusant to 10CFR.50.71(e) we are submitting 13 copies of Revision No. 2 to the Updated Safety Analysis Report (USAR) for the Prairie Island Generating Plant. This revision updates the information in the USAR for the period from January 1, 1983 through December 31, 1983.

Exhibit A contains a description and a summary of the safety evaluation for changes, tests and experiments made under the provisions of 10CFR 50.59 during this period.

Exhibit B contains the USAR page changes and instructions for entering the pages. Included in Exhibit B is Revision 9 to the Northern States Power Company Operational Quality Assurance Plan in compliance with 10CFR 50.54(a). Changes included in Revision 9 to the plan are described in Exhibit A (Item 18, page 10) of this letter.

David Musolf
Manager - Nuclear Support Services

DMM/TAP/bd

c: Regional Administrator-III, NRC
NRR Project Manager (w/o enclosure)
Resident Inspector, NRC
G Charnoff

Attachments

A053
1/13

EXHIBIT A

PRAIRIE ISLAND NUCLEAR GENERATING PLANT
ANNUAL REPORT OF CHANGES, TEST AND EXPERIMENTS

January 1, 1983 to December 31, 1983

The following sections include a brief description and a summary of the safety evaluation for those changes, tests and experiments which were carried out without prior NRC approval, pursuant to the requirements of 10CFR 50.59(b).

1. ITEMS WHICH CONTRIBUTE TO HYDROGEN GENERATION INSIDE CONTAINMENT (SE #97)

Description of Change
Portions of galvanized ductwork inside containment which were not necessary were removed so as not to contribute to hydrogen generation.

Summary of Safety Evaluation
Galvanized ductwork was removed from containment resulting in less hydrogen generaton. Hydrogen build-up is thus, slower.

2. ANALYSIS OF ADEQUACY OF STATION ELECTRICAL DISTRIBUTION SYSTEM VOLTAGES (SE# 114)

Description of Change
An "Analysis of Adequacy of Station Electric Distribution System Voltages" was performed and submitted to the NRC as an attachment to a letter dated July 17, 1981. This had been implemented by Reference in USAR Section 8. A re-analysis was performed which considered normal grid conditions with abnormal 1R and 2R transformer configurations and worst case plant loads.

Summary of Safety Evaluation
This analysis does not alter prior acceptance criteria for safeguards equipment electrical support. It establishes bounding conditions that are within the capabilities of the NSP system that assure support in any forseeable event.

3. CIRCULATING WATER INTAKE AND DISCHARGE MODIFICATIONS (78Y073)

Description of Change
This design change provided for the modification of the Circulating Water Intake and Discharge.

Summary of Safety Evaluation
This design change is non-safety related, QA Type III. Portions of the USAR describing the circulating water system are affected and have been updated in Revision 2. Interfaces between plant systems and the circulating water intake and discharge modifications were reviewed and no safety concerns were identified.

4. REACTOR COOLANT VENT SYSTEM (80Y117)

Description of Change
The purpose of this design change is to install the capability of remotely venting noncondensable gases from the reactor coolant system high points, in particular the reactor vessel head and the pressurzier steam space. Noncondensable gases in the reactor coolant system can degrade the core cooling effectiveness of the ECCS. The design bases for the reactor coolant gas vent system was specified in NUREG-0578 (short term lessons learned) and the appropriate sections of the NRC letter to all operating power plants of October 30, 1979.

Summary of Safety Evaluation
The Reactor Coolant Vent System is designed to safely and reliably vent noncondensible gases from either the reactor vessel head or the pressurizer steam space in order to maintain an adequate mode of core cooling. The system design has taken into account those design considerations as contained in the detailed design change/safety evaluation package, which will guarantee that the installation and operation of the system will meet the criteria as defined by NUREG-0578 and related followup letters from the NRC.

5. INCORE THERMO COUPLE MODIFICATIONS (80Y120)

Description of Change
This design change addresses the installation of the inside of containment penetration adaptors and mineral insulated, metal sheathed thermocouple extension cable from the refueling pool-side disconnect to the penetration. This design change modified the incore thermocouple system by: 1) Replacing the Thermo-Electric type K head connector and mate with qualified mating connector. 2) Replace the thermocouple extension wire in containment with a qualified mineral insulated-metal sheathed cable which extended to the containment penetration. 3) The reference temperature junction boxes were replaced and their location changed to outside of containment. 4) Thermocouple extension wire was added from the containment penetrations to the reference junction boxes and copper field wiring was run from the reference junction boxes to the incore panel and computer.

Summary of Safety Evaluation
The objectives of the design change was to upgrade the Incore Thermocouple System to a qualified 1E system. This allows qualified input of Core Exit Thermocouples to both the subcooled margin monitors and the Tech Support Center B&W Recall Computer. This upgrade does not eliminate any of the present functions of the thermocouple system but only upgrades their reliability.

6. HOT SHUTDOWN PANEL MODIFICATIONS (80Y140)

Description of Change
This design change fulfills part of an NSP commitment to the NRC to isolate the indication of selected parameters from the effects of a control room/relay room fire. The indicator portion of the Hot Shutdown Panels was reconfigured to display more parameters using dual channel indicators. When this design change was completed, each Hot Shutdown panel displayed the following protected outputs: Steam Generator Level (2/unit), RC Loop Pressure (2/unit), RC Loop Temperature Th (2/unit), RC Loop Temperature Tc (2/unit), and Pressurizer Level - Cold Cal. Each Hot shutdown Panel will continue to display the following nonisolated signals: Steam Header Pressure (2/unit) and Pressurizer Pressure.

Summary of Safety Evaluation
This design change involved fabrication of subpanels, installation of new instrumentation, bracing the existing Hot Shutdown Panels, cutting away portions of the Hot Shutdown Panel fronts, and mounting the new subpanels in the Hot Shutdown Panel enclosures. The display of plant parameters at the Hot Shutdown Panel will not be degraded by this design change. The indicators selected for this QA III function are of the same type and manufacturer as those selected for some QA I applications at Prairie Island.

7. TOXIC GAS MONITORS (80Y154)

Description of Change
This project provided for the installation of gas analyzers to monitor for chlorine, ammonia, formaldehyde, and hydrogen chloride and upon actuation place the Control, Relay, and Computer Rooms ventilation system into the recirculation mode through the PAC filter and close the outside air intake dampers. This is in compliance with Regulatory Guides 1.78 and 1.95 covering operator exposure to toxic gases.

Summary of Safety Evaluation
This design change provides for the control system design modification, procurement, and installation of gas analyzers for the Control, Relay, and Computer Room Ventilation System. The regulatory requirements pertaining to Chlorine Detection are contained in the Regulatory Guide 1.95. The Control, Relay and Computer Room Ventilation System consists of two trains of ventilation equipment. When chlorine is detected, the outside air dampers and steam exclusion dampers are closed, and Particulate, Absolute, and Charcoal (PAC) filter dampers are opened. Control Room isolation is accomplished with redundant dampers, Train A and Train B, in each external air path. All equipment necessary for the operation of the chlorine detectors and system actuation are Quality Assurance Type 1 and Design Class 1. The system is designed in accordance with IEEE 344-1975. It is concluded by this evaluation that this change provides additional protection for the control room operation.

8. AUXILIARY FEEDWATER PUMP MODIFICATIONS (80Y240, Part J)

Description of Change
This design change is to insure that the #21 Auxiliary Feedwater Pump (AFW) is operable in the event of a relay room or control room fire. Certain relays were relocated and circuits modified for Appendix R requirements. These modifications included the relocation of a relay to prevent a breaker trip from a fire-induced short, the fusing of conductors to the coils for two relays at the bus cubicle, the installation of a remote transfer switch to insure total electrical isolation by local operation, and where relays cannot be guaranteed to return to normal positions after loss of off-site power and a relay room fire, bypasses while in local operation or remote transfer switches are being installed. In addition, the lube oil pressure relay will be eliminated and a timer to the Aux. Lub Oil Pump installed which will lubricate the #21 AFW for 15 minutes every 24 hours. The same modification will apply to the Aux Lube Oil Pump for #12 AFW.

Summary of Safety Evaluation
This design change does not alter the AFW system mechanically. The time for the Aux. Lube Oil Pumps for #12 and #21 AFW pumps meets lube oil requirements.

Taking relay TD3/B26 out of local operation will not affect the FSAR section on load sequencing. Load restoration will remain the same for remote operation. While the AFW pump is in local operation, activation of relay LRY/B26 would still trip the pump since the remote switch in series with this contact will be from the diesel generator controls. For Appendix R requirements, the remote diesel generator controls, and the load rejection and restoration schemes for "A" Train, are assumed to be unreliable after a relay room fire.

9. INTERNAL HYDROGEN RECOMBINERS (82Y255)

Description of Change
This design change provides for the installation of two redundant internal hydrogen recombiners in each unit 1 & 2 containment. Components of this system include redundant Westinghouse type recombiners inside each containment, redundant power supply and control panels in the Auxiliary Building, new electrical penetration assemblies for power feeds, temperature control instrumentation including thermocouples and reference junction boxes and control wiring and power cables from new MCC to power supplies and recombiners.

Summary of Safety Evaluation
All equipment necessary for the operation of the hydrogen recombiners are QA Type I and Design Class 1. The system is qualified in accordance with IEEE-323, 1974 to the plant environmental specifications as detailed in our response to IE Bulletin 79-01B and IEEE-344, 1975. The recombiners will be powered from new safeguards MCCs located in Train A and B safeguards chiller rooms.

10. REACTOR COOLANT PUMP DISCHARGE CROSS-CONNECT TO THE CHEMICAL AND VOLUME CONTROL SYSTEM (82L702)

Description of Change

This design change installed a permanent cross-connect from the reactor coolant drain pump discharge to the chemical and volume control system. The cross-connect is used for cleaning up the RCS and refueling pool during refueling shutdowns, when the RCS is depressurized and in cold shutdown. The reactor drain tank pumps are needed to provide head for flow through the CVCS for cleaning.

Summary of Safety Evaluatin

All new piping installation and fabrication were in accordance with specifications. Piping was seismically analyzed. Proper quality assurance documentation and qualifications was provided. Chances of operational and installation errors are significantly reduced by this design change due to the elimination of a temporary jumper installation for every refueling outage and a simplified operating procedure for the permanent jumper.

11. MANIPULATOR CRANE "INCHING" AND SPLIT SCREEN UNDERWATER TV INDEXING SYSTEM (83Y410)

Description of Change

This design change is to upgrade the unit 1 and 2 manipulator cranes to include the "inching" crane movement system and the Split Screen Underwater TV Indexing systems. This will give a finer, more precise indexing system for fuel handling in addition to a viewing system for inserting fuel into the core.

Summary of Safety Evaluation

This design change was implemented to aid the operator in the placement of fuel assemblies into the reactor core. Both systems were specifically designed for use on the Prairie Island Manipulator Cranes. The inching modification allows the operator to make very precise bridge and trolley movements while centering the fuel assembly over its intended core location. An "inching" permissive must be switched on prior to the use of the joy stick control. This permissive locks out all the other bridge and trolley controls and transfers the control to inching joy stick. The inching modification is being supplemented by an underwater Split-Screen TV system. This gives the operator a 360° view of the fuel assembly as it is entering or leaving the core. The operator can use the inching control and the TV system to center the fuel assembly to ensure that it can be lowered into the core without contacting the surrounding assemblies. Overall, with the use of these two systems, the movement of fuel will be performed in a more controlled, precise and safe manner than was done in the past.

12. UNINTERRUPTABLE POWER SUPPLIES (83Y415)

Description of Change
This design change replaces the existing four (4) 75 KVA inverters with four (4) Uninterruptable Power Supplies which each include a 5.0 KVA inverter, automatic transfer switch, and a manual bypass switch. The new power supplies allows for automatic transfer of power to an alternate AC source in case of inverter failure. Also the manual bypass switch allows for inverter maintenance under power. When any one of the inverters is bypassed it is annunciated separately in the Control Room.

Summary of Safety Evaluation
All variations between the old inverters and the new Uninterruptable Power Supplies have been evaluated and addressed in the detailed design description and safety evaluation. Included were an evaluation of the 5.0 KVA rating vs 7.5 KVA (the maximum loading is 4.0 (KVA), the addition of an automatic static transfer switch which is designed to transfer the instrument bus load from the inverter, if it fails, to the AC source through a bypass breaker (this transfer switch adds to the reliability of the 120 VAC source being supplied to the instrument bus), addition of a manual bypass switch allowing for inverter repair or maintenance without need to rely on panel 217 to supply the load, seismic considerations and electrical variations.

With the addition of one more alternate source of power to the instrument power buses there will be increase in the reliability of the buses and therefore an increase in the margin of plant safety.

13. INSTALLATION OF A MANUAL ISOLATION VALVE DOWNSTREAM OF CV31420 (83L742)

Description of Change
This design change is for the installation of a manual isolation valve downstream of CV31420. The valve is normally open and was installed in the line so CV31420 may be isolated from RCS pressure and repaired under power.

Summary of Safety Evaluation
The installation of a manual valve in the charging line to the RCS does not create a possibility for an accident or malfunction of a different type than evaluated previously. The valve utilized has been used in many places in the RCS and is compatible with the piping specification. The line has been stress analyzed with the additional valve installed, and the load increases appear to be insignificant. The stress levels are within acceptable limits.

14. CAUSTIC ADDITION SYSTEM (83L745)

Description of Change
This design change performs modification to the Caustic Addition System which assures that the pH limit to which the electrical equipment in containment is qualified will not be exceeded at any time. This design change is a result of an evaluation which was performed in accordance with IE Bulletin 79-01b on the qualification of safety related electrical equipment when exposed to various harsh environmental conditions. The evaluation showed that some accident scenarios resulted in a containment spray pH greater than 13 during the injection phase. The pH would not decrease to 10.5 until the initiation of the recirculation mode. Safety related electrical equipment has been evaluated to withstand a chemical environment consisting of 2100 ppm boric acid with the pH adjusted to a maximun of 10.5 through the addition of sodium hydroxide.

Summary of Safety Evaluation
This design change resulted in the containment spray injection pH varying from 10.45 to 9.1 depending upon pumping combinations (RHR, SIP and CSPs) which is acceptable. Recirculation phase pH could reach a minimum value of 8.2 which is less than the criteria in the Standard Review Plan but still consistent with the Prairie Island FSAR. A pH less than 8.5 but greater than 7 is considered acceptable because; a) the tank volumes and concentrations for this pH value were taken in the worst cast situation, not the normal condition, b) the normal tank levels and solution concentrations would give a sump pH (Recircwater phase pH) of 8.4. c) the changes in pH at this level do not affect stress corrosion cracking at Prairie Island d) at the pH of 8.2 the partition factor is above 2000, consequently, most if not all, of the iodine absorbed during the spray should remain in solution. e) the FSAR analysis takes credit for a reduction of iodine concentration by a factor of only 100 (DE=100). The FSAR also shows that 10CFR 100 guidelines are NOT exceeded if the Caustic Addition System were not installed, f) the recirculation phase water pH can be monitored by sampling to ensure the pH levels are greater than 7. Additives can be used during the recirculation phase to maintain the pH at an acceptable level.

Given the preceeding considerations, the modifications meet the limiting Conditions of Operation, the FSAR and 10CFR 100.

Piping and structural analyses were performed and were still within the acceptable limits.

15. RADIOACTIVE WASTE LIQUID TREATMENT SYSTEM (83L754)

Description of Change
A CHEM-Nuclear 24 inch diameter pressure demineralizer vessel containing activated carbon was installed as a filter media in lieu of the ADT Evaporator in the Radioactive Waste Liquid Treatment System. The filter unit is located in the old Waste Concentrates

Tank Pump Room of the RAD Waste Building. The feed line to the 5 GPM ADT Evaporator was modified to allow flow to the new filter unit or to the evaporator. A line was also added to go from the filter unit back into the 5 GPM Evaporator Room and tie into the distillate discharge line to the Condensate Receiver Tanks. This allowed flow from the ADT Collection Tanks through the ADT Filters to the Activated Charcoal filter and on to the ADT Condensate Receiver Tanks. The waste liquid may then be treated by the ADT Ion Exchangers prior to release from the ADT Monitor Tanks.

Summary of Safety Evaluation
All piping was QA Type III and the tie-ins were made into existing QA Type III lines as defined by plant data files. No new penetrations were made in any structual, fire barrier, or Special Ventilation Boundary walls. All work was performed under normal plant QA III Installation Procedures. The USAR has been updated to reflect the change in the description of the Waste Liquid Treatment System.

16. UNIT 2, CYCLE 8, CORE RELOAD (83L762)

Description of Change
This design change addresses the Core Reload for Unit 2 Cycle 8. During the Prairie Island Unit 2 Cycle 7/8 refueling outage, 40 spent Exxon STANDARD region 7(G) fuel assemblies and 1 Westinghouse region 6(F) assembly was replaced with 20 fresh Exxon TOPROD region 10(J) assemblies, 20 Exxon TOPROD region 9 (I) assemblies that were discharged from Unit 1 at the end of cycle 7 after one cycle of operation, and 1 Westinghouse region 2(B) assembly. The remainder of the core will consist of 40 once burned Exxon TOPROD assemblies region 9(I), 36 twice burned Exxon STANDARD assemblies region 8(H), and 4 twice burned Exxon lead TOPROD assemblies region 8L(H). Three of the assemblies to be placed in the core for Unit 2 Cycle 8 are the Unit 1 assemblies which were reconstituted in April, 1983. The failed pins (1 per assembly) were replaced with pins of similar burnups and power histories. Cycle 8 is projected to last 11,700 MWD/MTU (262 EFPD) including a coast down of 600 MWD/MTU. To achieve this cycle length the loading pattern is a low leakage pattern with 64, 4 w/o gadolinia bearing, fuel pins distributed evenly among 16 of the fresh fuel assemblies.

Summary of Safety Evaluation
The analyses performed in the design and licensing of Unit 2 Cycle 8 operation were done by NSP's Nuclear Analysis Department (NAD) and are summarized in the "Prairie Island Unit 2, Cycle 8 Final Reload Design Report (Reload Safety Evaluation)" Sept., 1983 and the "Prairie Island Unit 2, Cycle 8, Startup and Operations Report" Sept., 1983. The analyses indicate that the core will operate within Technical Specification limits with respect to shutdown reactivity worth, temperature coefficient restrictions, and hot channel peaking factor $F_{\Delta H}$.

During cycle 7 the primary coolant activity indicated the presence of at least one failed fuel pin. The decision was made to reconstitute the failed assembly(ies) during the shuffle. NSP's Nuclear Analysis Department has analyzed the effects of replacing the failed pin(s) with an inert zirc rod(s) and has concluded that there would be no unresolved safety questions as long as guidelines for the reconstitution provided by NAD to the plant were followed.

No additional analysis was required for the three fuel assemblies (I-23, 37 and 40) that were reconstituted in April, 1983 since the replacement pins were from Fuel Assembly H-90 and had similar burnups.

In summary, sufficient analyses have been performed to show that the Unit 2 Cycle 8 reload core can be safely loaded into Unit 2 Reactor.

17. UNIT 1, CYCLE 9, CORE RELOAD

Description of Change

This design change addresses the Core Reload for Unit 1 Cycle 9. During the Prairie Island Unit 2 Cycle 8/9 refueling outage, 40 Exxon region 8(H) assemblies and 1 Westinghouse region 2(B) assembly will be replaced with 20 fresh Exxon TOPROD region 11(L) assemblies, 20 once burned Exxon TOPROD region 9(I) from Unit 1 Cycle 7 and 1 twice burned Westinghouse region 2(B) assembly. The remainder of the core will consist of 40 once burned Exxon TOPROD region 10(J) assemblies and 40 once burned Exxon TOPROD region 10A(K) assemblies. Due to outage schedule considerations, the low leakage core design was not used and all new assemblies will be loaded on the outside of the core. This loading pattern allows for a cycle 9 hot full power lifetime of 11, 400 MWD/MTU (approximately 305 EFPDs) assuming an EOL exposure of cycle 8 of 12, 738 MWD/MTU.

Summary of Safety Evaluation

The "Prairie Island Unit 1, Cycle 9, Final Reload Design Report (Reload Safety Evaluation)", October, 1983 validated a safe fuel loading and subsequent operation using the specified loading pattern provided the end of Cycle 8 exposure was 12,738 ± 500 MWD/MTU. The transient analysis and reactor physics sections were completed by NSP's Nuclear Analysis Department (NAD) while the LOCA/ECCS analysis was completed by Exxon Nuclear using data supplied by NSP through the Reload Safety Evaluation. However, Unit 1 was shut down early at an exposure of 11,940 MWD/MTU which is not within the bounds of the Reload Safety Evaluation. The Reload Safety Evaluation was re-analyzed by NSP and verified to be valid. The updated data was also supplied to Exxon and the LOCA/ECCS analysis was also verified to be valid considering this new EOC exposure. The Unit 1 Cycle 9 physics characteristics have been analyzed by NSP's NAD and are contained in "Prairie Island Unit 1 Cycle 9, Startup and Operations Report" December, 1983.

18. CHANGE TO THE OPERATIONAL QUALITY ASSURANCE PLAN APPENDIX 13B

Revision 9 to the NSP Operation Quality Assurance Plan was internally reviewed and approved on April 12, 1984. This revision does not reduce the commitments in NSP's Operational Quality Assurance Program and does not adversely impact the safe operation of the nuclear plants. A summary of significant changes in Revision 9 is presented below. Specific changes with the reason for the change and the basis for concluding no reduction in commitments per 10CFR 50.54(a)(3)(iii), are presented in Appendix D to the plan. The Operational Quality Assurance Plan, Revision 9, is included in Section 13 to the USAR.

1. Change Security and Radiation Environmental Monitoring

Description

Exceptions to sections in ANSI N18.7-1976 concerning security and radiation envrionmental monitoring have been deleted. Management positions in these two areas have been added to the organizational description. These changes place security and radiation environmental monitoring under the quality assurance program.

2. Change Inspections and Test Control

Description

Exceptions to sections in ANSI N18.7-1976 concerning inspections and test control have been deleted. NSP's alternate wording within the plan remains essentially unchanged. NSP's program implements the N18.7 sections and the plan sections.

3. Change Vendor Evaluation and Verification and Vendor Inspections

Description

The differences between vendor evaluation and verification and vendor inspection have been clarified.

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

Revision 2 to the Updated Safety Analysis Report

The attached instructions should be followed when making this revision to the Updated Safety Analysis Report. If you have any questions concerning this revision call:

Terry Pickens (612) 330-5671

Volume No.	Tab	Remove Page No.	Insert Page No.	Rev. No.
I	Document Control	a	a	2
		b	b	2
		c	c	2
		d	d	2
		e	e	2
		f	f	2
		g	g	2
		h	h	2
		i	i	2
		j	j	1
		k	k	2
		l	l	2
		m	m	2
		n	n	2
		o	o	2
		p	p	2
		q	q	2
		r	r	0
		s	s	2
		t	t	2
		u	u	2
		v	v	2
		w	w	2
		x	x	2
		y	y	2
	Table of Contents	iii	iii	1
		iv	iv	2
		v	v	2
	2. Plant Site and Environs	2-v	2-v	2
		2-vi	2-vi	0
		2.9-4	2.9-4	2
		2.9-5	2.9-5	0
		TBL 2.9-5	TBL 2.9-5	2

Volume No.	Tab	Remove Page No.	Insert Page No.	Rev. No.
I	3. Reactor	3-vii	3-vii	2
		3-viii	3-viii	2
		3.1-1	3.1-1	0
		3.1-2	3.1-2	2
		3.1-7	3.1-7	2
		3.1-8	3.1-8	0
		3.2-1	3.2-1	2
		3.2-2	3.2-2	2
		3.2-3	3.2-3	0
		3.2-4	3.2-4	2
		3.2-5	3.2-5	2
		3.3-1	3.3-1	2
		3.3-4	3.3-4	2
		3.3-5	3.3-5	2
		3.3-6	3.3-6	2
		3.4-1	3.4-1	0
		3.6-4	3.6-4	2
		3.6-5	3.6-5	0
		3.8-2	3.8-2	2
		3.8-3	3.8-3	0
		3.8-4	3.8-4	2
		TBL 3.1-1	TBL 3.1-1	0
		TBL 3.2-3	TBL 3.2-3	0
		TBL 3.2-4	TBL 3.2-4	2
		TBL 3.2-5	TBL 3.2-5	2
		TBL 3.2-6	TBL 3.2-6	1
		TBL 3.2-7	TBL 3.2-7	1
		TBL 3.3-1	TBL 3.3-1	2
		TBL 3.3-2	TBL 3.3-2	2
		TBL 3.3-3	TBL 3.3-3	2
		TBL 3.3-4	TBL 3.3-4	2
		TBL 3.4-1	TBL 3.4-1	0

Volume No.	Tab	Remove Page No.	Insert Page No.	Rev. No.
I	3. Reactor			
		FIG 3.3-1	FIG 3.3-1	[illegible]
		FIG 3.3-2	FIG 3.3-2	2
		FIG 3.3-3	FIG 3.3-3	2
		FIG 3.3-4	FIG 3.3-4	2
		FIG 3.3-5	FIG 3.3-5	2
		FIG 3.3-6	FIG 3.3-6	2
		-	FIG 3.6-5	2
	4. Reactor Coolant System	4-iii	4-iii	2
		4-iv	4-iv	1
		4-vii	4-vii	2
		4-viii	(BLANK PAGE)	NA
		4.3-8	4.3-8	0
		4.3-9	4.3-9	2
		-	4.3-9a	2
		-	(BLANK PAGE)	NA
		4.4-5	4.4-5	0
		4.4-6	4.4-6	2
		4.4-7	4.4-7	2
		4.5-1	4.5-1	2
		4.5-2	4.5-2	2
		4.5-3	4.5-3	0
		4.5-4	4.5-4	2
		4.6-1	4.6-1	0
		4.8-3	4.8-3	1
		4.8-4	4.8-4	2
		TBL 4.3-13	TBL 4.3-13	2
		TBL 4.3-14	TBL 4.3-14	2
		TBL 4.5-1 (3 of 4)	TBL 4.5-1 (3 of 4)	2

Volume No.	Tab	Remove Page No.	Insert Page No.	Rev. No.
I	4. Reactor Coolant System	TBL 4.5-1 (4 of 4)	TBL 4.5-1 (4 of 4)	2
		TBL 4.7-1 (1 of 3)	TBL 4.7-1 (1 of 3)	2
		TBL 4.7-1 (2 of 3)	TP 4.7-1 (2 of 3)	2
		TBL 4.7-1 (3 of 3)	TBL 4.7-1 (3 of 3)	2
		FIG 4.1-1	FIG 4.1-1	2
		-	FIG 4.1-1a	2
		FIG 4.5-1	-	
II	5. Plant Containment Systems	5-i	5-i	2
		5-ii	5-ii	2
		5-iii	5-iii	2
		5-iv	5-iv	0
		5-v	5-v	0
		5-vi	5-vi	2
		5-vii	5-vii	2
		5-viii	5-viii	0
		5-ix	5-ix	2
		5-x	(BLANK PAGE)	NA
		5.1-1	5.1-1	2
		5.1-2	5.1-2	0
		5.2-2	5.2-2	2
		5.2-3	5.2-3	0
		5.2-4	5.2-4	2
		5.2-5	5.2-5	2
		5.2-6	5.2-6	0
		5.2-7	5.2-7	2
		5.2-8	5.2-8	2
		5.2-9	5.2-9	2

Volume No.	Tab	Remove Page No.	Insert Page No.	Rev. No.
II	5. Plant Containment Systems	-	5.2-9a	2
		-	(BLANK PAGE)	NA
		5.2-12	5.2-12	0
		5.2-13	5.2-13	2
		5.2-16	5.2-16	2
		5.2-17	5.2-17	2
		5.2-18	5.2-18	2
		5.2-18a	5.2-18a	2
		5.2-18b	5.2-18b	1
		5.2-19	5.2-19	2
		5.2-20	5.2-20	2
		5.2-21	5.2-21	0
		5.2-22	5.2-22	2
		5.2-23	5.2-23	2
		5.2-24	5.2-24	2
		5.2-25	5.2-25	2
		5.2-26	5.2-26	2
		5.2-27	5.2-27	2
		5.2-28	5.2-28	2
		5.2-29	5.2-29	2
		5.2-30	5.2-30	2
		5.2-31	5.2-31	2
		5.2-32	5.2-32	2
		5.2-33	5.2-33	2
		5.2-34	5.2-34	2
		5.3-1	5.3-1	2
		5.3-2	5.3-2	2
		5.3-3	5.3-3	0
		5.3-4	5.3-4	2
		5.3-5	5.3-5	2
		5.3-6	5.3-6	2
		5.3-7	5.3-7	0

Volume No.	Tab	Remove Page No.	Insert Page No.	Rev. No.
II	5. Plant Containment Systems	5.3-10	5.3-10	2
		5.3-11	5.3-11	2
		5.3-12	5.3-12	2
		5.3-13	5.3-13	0
		5.4-2	5.4-2	0
		5.4-3	5.4-3	2
		5.4-4	5.4-4	2
		5.4-5	5.4-5	2
		-	5.4-5a	2
		-	5.4-5b	2
		5.4-10	5.4-10	2
		5.4-11	5.4-11	0
		5.4-16	5.4-16	2
		5.4-17	5.4-17	2
		-	5.4-18	2
		-	(BLANK PAGE)	NA
		TBL 5.2-1 (1 of 17)	TBL 5.2-1 (1 of 17)	2
		TBL 5.2-1 (2 OF 17)	TBL 5.2-1 (2 OF 17)	1
		TBL 5.2-1 (5 of 17)	TBL 5.2-1 (5 of 17)	2
		TBL 5.2-1 (6 of 17)	TBL 5.2-1 (6 of 17)	1
		TBL 5.2-1 (7 of 17)	TBL 5.2-1 (7 of 17)	1
		TBL 5.2-1 (8 of 17)	TBL 5.2-1 (8 of 17)	2
		TBL 5.2-1 (13 of 17)	TBL 5.2-1 (13 of 17)	1
		TBL 5.2-1 (14 of 17)	TBL 5.2-1 (14 of 17)	2
		TBL 5.2-1 (15 of 17)	TBL 5.2-1 (15 of 17)	2
		TBL 5.2-1 (16 of 17)	TBL 5.2-1 (16 of 17)	1

Volume No.	Tab	Remove Page No.	Insert Page No.	Rev. No.
II	5. Plant Containment Systems	TBL 5.3-3 (3 of 3)	TBL 5.3-3 (3 of 3)	0
		TBL 5.3-4	TBL 5.3-4	2
		TBL 5.4-1 (1 of 2)	TBL 5.4-1 (1 of 2)	2
		TBL 5.4-1 (2 of 2)	TBL 5.4-1 (2 of 2)	0
		FIG 5.2-2	FIG 5.2-2	2
		FIG 5.2-4 (1 of 2)	FIG 5.2-4 (1 of 3)	2
		FIG 5.2-4 (2 of 2)	FIG 5.2-4 (2 of 3)	2
		-	FIG 5.2-4 (3 of 3)	2
		FIG 5.2-10	FIG 5.2-10	2
		FIG 5.4-1	FIG 5.4-1	2
	6. Engineered Safeguards	6-i	6-i	0
		6-ii	6-ii	2
		6-iii	6-iii	2
		6-iv	6-iv	0
		6.2-23	6.2-23	0
		6.2-24	6.2-24	2
		6.2-25	6.2-25	0
		6.2-26	6.2-26	2
		-	6.2-26a	2
		-	(BLANK PAGE)	NA
		6.2-27	6.2-27	2
		6.2-28	6.2-28	0
		6.2-29	6.2-29	2
		6.2-30	6.2-30	0

Volume No.	Tab	Remove Page No.	Insert Page No.	Rev. No.
II	6. Engineered Safeguards	6.4-11	6.4-11	2
		6.4-12	6.4-12	0
		FIG 6.3-1	FIG 6.3-1	2
	7. Plant Instrumentation and Control Systems	7.8-6	7.8-6	0
		7.8-7	7.8-7	2
		FIG 7.6-1	FIG 7.6-1	2
III	8. Plant Electrical Systems	8.3-4	8.3-4	2
		8.3-5	8.3-5	1
		8.5-1	8.5-1	0
		8.6-1	8.6-1	2
		8.9-1	8.9-1	2
		8.10-1	8.10-1	2
		FIG 8.2-2	FIG 8.2-2	2
		FIG 8.5-2	FIG 8.5-2	2
	9. Plant Radioactive Waste Control Systems	9.2-2	9.2-2	0
		9.2-3	9.2-3	2
		9.3-4	9.3-4	0
		9.4-1	9.4-1	2
		FIG 9.1-6	FIG 9.1-6	2
		FIG 9.1-8	FIG 9.1-8	2

Volume No.	Tab	Remove Page No.	Insert Page No.	Rev. No.
III	10. Plant Auxiliary Systems	10-i	10-i	2
		10-ii	10-ii	0
		10-iii	10-iii	2
		10-iv	10-iv	2
		10-v	10-v	2
		10-vi	10-vi	0
		10.2-7	10.2-7	2
		10.2-8	10.2-8	0
		10.2-17	10.2-17	2
		10.2-18	10.2-17a	2
		-	(BLANK PAGE)	NA
		-	10.2-18	0
		10.3-2	10.3-2	2
		10.3-3	10.3-2a	2
		-	(BLANK PAGE)	NA
		-	10.3-3	0
		10.3-10	10.3-10	0
		10.3-11	10.3-11	2
		-	10.3-11a	2
		-	(BLANK PAGE)	NA
		10.3-26	10.3-26	0
		10.3-27	10.3-27	2
		-	(BLANK PAGE)	NA
		-	10.4-1	0
		10.6-1	10.6-1	2
		10.6-2	10.6-2	2
		FIG 10.2-5	FIG 10.2-5	2
		FIG 10.2-7	FIG 10.2-7	2
		FIG 10.3-6	FIG 10.3-6	2
IV	11. Plant Power Conversion Systems	11.4-4	11.4-4	0
		11.5-1	11.5-1	2
		11.5-2	11.5-2	2
		11.6-1	11.6-1	0

Volume No.	Tab	Remove Page No.	Insert Page No.	Rev. No.
IV	11. Plant Power Conversion Systems	11.9-2	11.9-2	0
		11.9-3	11.9-3	2
		11.9-4	11.9-4	2
		11.9-5	11.9-5	1
		FIG 11.1-17	FIG 11.1-17	2
		FIG 11.1-18	FIG 11.1-18	2
	12. Plant Structures and Shielding	12-ix	12-ix	2
		12-x	12-x	2
		12-xi	12-xi	0
		12-xii	12-xii	2
		12.2-104	12.2-104	1
		12.2-105	12.2-105	2
		12.2-106	12.2-106	1
		12.2-107	12.2-107	2
		12.2-108	12.2-108	2
		12.2-109	12.2-109	2
		-	12.2-110	2
		-	12.2-111	2
		-	12.2-112	2
		-	(BLANK PAGE)	NA
		12.3-5	12.3-5	1
		12.3-6	12.3-6	2
		12.5-2	12.5-2	1
		12.5-3	12.5-3	2
		-	TBL 12.2-40 (1 of 2)	2
		-	TBL 12.2-40 (2 of 2)	2
		-	TBL 12.2-41	2
		-	(Blank Page)	NA
	13. Plant Operations	13-i	13-i	1
		13-ii	13-ii	2
		13.2-4	13.2-4	2
		13.3-1	13.3-1	2

Volume No.	Tab	Remove Page No.	Insert Page No.	Rev. No.
IV	13. Plant Operations	13.4-1	13.4-1	2
		13.4-2	13.4-2	2
		13.4-3	13.4-3	2
		13.5-1	13.5-1	0
		13.6-1	13.6-1	0
		-	13.7-1	2
		-	13.8-1	2
	13A. Operational QA Plan	- Remove the Operational QA Plan Revision 8 (cover sheet, Table of Contents and 78 pages).		
		- Insert the Operational QA Plan Revision 9 (cover sheet, Table of Contents and 91 pages).		
V	14. Safety and Accident Analysis			
		14-i	14-i	2
		14-ii	14-ii	2
		14-iii	14-iii	2
		14-iv	14-iv	2
		14-v	14-v	2
		14-vi	14-vi	0
		14-vii	14-vii	2
		14-viii	14-viii	2
		14-ix	14-ix	2
		14-x	14-x	2
		14-xi	14-xi	2
		14-xii	14-xii	2
		14.2-1	14.2-1	2
		14.2-2	14.2-2	2
		14.3-1	14.3-1	2
		14.3-2	14.3-2	2
		-	14.3-3	2
		-	(BLANK PAGE)	NA

Volume No.	Tab	Remove Page No.	Insert Page No.	Rev. No.
V	14. Safety and Accident Analysis			
		14.4-1	14.4-1	2
		14.4-2	14.4-1a	2
		-	14.4-1b	2
		-	14.4-1c	2
		-	14.4-1d	2
		-	14.4-2	2
		14.4-3	14.4-3	2
		14.4-4	14.4-4	2
		14.4-5	14.4-5	0
		14.4-6	14.4-6	2
		14.4-7	14.4-7	2
		14.4-8	14.4-8	2
		14.4-9	14.4-9	0
		14.4-10	14.4-10	2
		14.4-11	14.4-11	2
		14.4-12	14.4-12	0
		14.4-13	14.4-13	2
		14.4-14	14.4-14	2
		14.4-15	14.4-15	2
		14.4-16	14.4-16	2
		14.4-17	14.4-17	2
		14.4-18	14.4-18	2
		14.4-19	14.4-19	2
		14.4-20	14.4-20	2
		14.4-21	14.4-21	2
		14.4-22	14.4-22	2
		14.4-23	14.4-23	0
		14.4-24	14.4-24	2
		14.4-25	14.4-25	2
		14.4-26	14.4-26	2

Volume No.	Tab	Remove Page No.	Insert Page No.	Rev. No.
V	14. Safety and Accident Analysis			
		14.5-1	14.5-1	2
		14.5-2	14.5-2	0
		14.5-5	14.5-5	2
		14.5-6	14.5-6	2
		14.5-7	14.5-7	2
		14.5-8	14.5-8	2
		14.5-9	14.5-9	2
		14.5-10	14.5-10	2
		14.5-11	14.5-11	2
		14.5-12	14.5-12	0
		14.5-21	14.5-21	2
		14.5-22	14.5-22	2
		14.5-23	14.5-23	2
		14.5-24	14.5-24	2
		14.5-25	14.5-25	2
		14.5-26	14.5-26	2
		14.5-27	14.5-27	2
		14.5-28	(BLANK PAGE)	NA
		14.6-1	14.6-1	0
		14.6-2	14.6-2	2
		14.6-3	14.6-3	2
		14.6-4	14.6-4	2
		14.6-5	14.6-5	2
		14.6-6	14.7-1	2
		14.6-7	-	
		14.7-1	-	
		14.9-15	14.9-15	0
		14.9-16	14.9-16	2
		14.10-1	14.10-1	2
		14.10-2	14.10-2	2

Volume No.	Tab	Remove Page No.	Insert Page No.	Rev. No.
V	14. Safety and Accident Analysis			
		TBL 14.3-1 (1 of 2)	TBL 14.3-1	2
		TBL 14.3-1 (2 of 2)	TBL 14.3-2	2
		TBL 14.3-2	TBL 14.3-3	2
		TBL 14.3-3	TBL 14.4-1	2
		TBL 14.3-4	TBL 14.4-2	0
		TBL 14.4-1	TBL 14.4-3	2
		TBL 14.4-2	TBL 14.5-1	2
		TBL 14.5-1 (1 of 2)	TBL 14.5-2	2
		TBL 14.5-1 (2 of 2)	TBL 14.5-3	2
		TBL 14.5-2 (1 OF 2)	TBL 14.5-4	2
		TBL 14.5-2 (2 of 2)	(BLANK PAGE)	NA
		TBL 14.5-3 (1 of 2)	TBL 14.6-1 (1 of 2)	2
		TBL 14.5-3 (2 of 2)	-	
		TBL 14.5-4 (1 of 2)	-	
		TBL 14.5-4 (2 of 2)	-	
		TBL 14.6-1 (1 of 2)	-	
		TBL 14.6-1 (2 of 2)	TBL 14.6-1 (2 of 2)	2
		TBL 14.6-2 (1 of 2)	TBL 14.6-2 (1 of 2)	0
		TBL 14.6-6	TBL 14.6-6	2
		TBL 14.6-7	TBL 14.6-7	2
		FIG 14.3-1	FIG 14.3-1	2
		FIG 14.3-2	FIG 14.3-2	2

Volume No.	Tab	Remove Page No.	Insert Page No.	Rev. No.
V	14. Safety and Accident Analysis			
		FIG 14.4-1	FIG 14.4-1	2
		FIG 14.4-2	FIG 14.4-2	2
		FIG 14.4-3	FIG 14.4-3	2
		FIG 14.4-4	FIG 14.4-4	2
		FIG 14.4-5	FIG 14.4-5	2
		FIG 14.4-6	FIG 14.4-6	2
		FIG 14.4-7	FIG 14.4-7	2
		FIG 14.4-8	FIG 14.4-8	2
		FIG 14.4-9	FIG 14.4-9	2
		FIG 14.4-10	FIG 14.4-10	2
		FIG 14.4-11	FIG 14.4-11	2
		FIG 14.4-12	FIG 14.4-12	2
		FIG 14.4-13	FIG 14.4-13	2
		FIG 14.4-14	FIG 14.4-14	2
		FIG 14.4-15	FIG 14.4-15	2
		FIG 14.4-16	FIG 14.4-16	2
		FIG 14.4-17	FIG 14.4-17	2
		FIG 14.4-18	FIG 14.4-18	2
		FIG 14.4-19	FIG 14.4-19a	2
		-	FIG 14.4-19b	2
		FIG 14.4-20	FIG 14.4-20	2
		FIG 14.4-21	FIG 14.4-21	2
		FIG 14.4-22	FIG 14.4-22	2

Volume No.	Tab	Remove Page No.	Insert Page No.	Rev. No.
V	14. Safety and Accident Analysis			
		FIG 14.4-23	FIG 14.4-23	2
		FIG 14.4-24	FIG 14.4-24	2
		FIG 14.4-25	FIG 14.4-25	2
		FIG 14.4-26	FIG 14.4-26	2
		FIG 14.4-27a	FIG 14.4-27	2
		FIG 14.4-27b	-	
		FIG 14.4-28	FIG 14.4-28	2
		FIG 14.4-29	FIG 14.4-29	2
		FIG 14.4-30	FIG 14.4-30	2
		FIG 14.4-31	FIG 14.4-31	2
		FIG 14.4-32	FIG 14.4-32	2
		FIG 14.4-33	FIG 14.4-33	2
		FIG 14.4-34	FIG 14.4-34	2
		FIG 14.4-35	FIG 14.4-35	2
		FIG 14.4-36	FIG 14.4-36	2
		FIG 14.4-37	FIG 14.4-37	2
		FIG 14.4-38	FIG 14.4-38	2
		FIG 14.4-39	FIG 14.4-39	2
		FIG 14.4-40	FIG 14.4-40	2
		FIG 14.4-41	FIG 14.4-41	2
		FIG 14.4-42	FIG 14.4-42	2
		FIG 14.4-43	FIG 14.4-43	2
		FIG 14.4-44	FIG 14.4-44	2

Volume No.	Tab	Remove Page No.	Insert Page No.	Rev. No.
V	14. Safety and Accident Analysis			
		FIG 14.4-45	FIG 14.4-45	2
		FIG 14.4-46	FIG 14.4-46	2
		FIG 14.4-47	FIG 14.4-47	2
		FIG 14.4-48	FIG 14.4-48a	2
		-	FIG 14.4-48b	2
		-	FIG 14.4-48c	2
		FIG 14.4-49	-	
		FIG 14.4-50	-	
		FIG 14.4-51	-	
		FIG 14.4-52	-	
		FIG 14.4-53	-	
		FIG 14.4-54	-	
		FIG 14.4-55	-	
		FIG 14.4-56	-	
		FIG 14.4-57	-	
		FIG 14.4-58	-	
		FIG 14.4-59	-	
		FIG 14.4-60	-	
		FIG 14.4-61	-	
		FIG 14.4-62	-	
		FIG 14.4-63	-	
		FIG 14.4-64	-	
		FIG 14.4-65	-	

Volume No.	Tab	Remove Page No.	Insert Page No.	Rev. No.
V	14. Safety and Accident Analysis			
		FIG 14.4-66	-	
		FIG 14.4-67	-	
		FIG 14.4-68	-	
		FIG 14.4-69	-	
		FIG 14.4-70	-	
		FIG 14.4-71	-	
		FIG 14.4-72	-	
		FIG 14.4-73	-	
		FIG 14.4-74a	-	
		FIG 14.4-74b	-	
		FIG 14.4-74c	-	
		FIG 14.5-2	FIG 14.5-2	2
		FIG 14.5-3	FIG 14.5-3	2
		FIG 14.5-4	FIG 14.5-4	2
		FIG 14.5-5	FIG 14.5-5	2
		FIG 14.5-6	FIG 14.5-6	2
		FIG 14.5-7	FIG 14.5-7	2
		FIG 14.5-8	FIG 14.5-8	2
		FIG 14.5-9	FIG 14.5-9	2
		FIG 14.5-10	FIG 14.5-10	2
		FIG 14.5-11	FIG 14.5-11	2
		FIG 14.5-12	FIG 14.5-12	2
		FIG 14.5-13	-	

Volume No.	Tab	Remove Page No.	Insert Page No.	Rev. No.
V	14. Safety and Accident Analysis			
		FIG 14.5-14	-	
		FIG 14.5-15	-	
		FIG 14.5-16	-	
		FIG 14.5-17	-	
		FIG 14.5-18	-	
		FIG 14.5-19	-	
		FIG 14.5-20	-	
		FIG 14.5-21	-	
		FIG 14.5-22	-	
		FIG 14.5-23	-	
		FIG 14.5-24	-	
		FIG 14.5-25	-	
		FIG 14.5-26	-	
		FIG 14.5-27	-	
		FIG 14.6-1	FIG 14.6-1	2
		FIG 14.6-2	FIG 14.6-2	2
		FIG 14.6-3	FIG 14.6-3	2
		FIG 14.6-4	FIG 14.6-4	2
		FIG 14.6-5	FIG 14.6-5	2
		FIG 14.6-6	FIG 14.6-6	2
		FIG 14.6-7	FIG 14.6-7	2
		FIG 14.6-8	FIG 14.6-8	2
		FIG 14.6-9	FIG 14.6-9	2

Volume No.	Tab	Remove Page No.	Insert Page No.	Rev. No.
V	14. Safety and Accident Analysis			
		FIG 14.6-10	FIG 14.6-10	2
		FIG 14.6-11	FIG 14.6-11	2
		FIG 14.6-12	FIG 14.6-12	2
		FIG 14.6-13	FIG 14.6-13	2
		FIG 14.6-14	FIG 14.6-14	2
		FIG 14.6-15	FIG 14.6-15	2
		FIG 14.6-16	FIG 14.6-16	2
		FIG 14.6-17	FIG 14.6-17	2
		FIG 14.6-18	FIG 14.6-18	2
		FIG 14.6-19	FIG 14.6-19	2
		-	FIG 14.6-20	2
		FIG 14.6-21	FIG 14.6-21	2
		FIG 14.6-22	FIG 14.6-22	2
		FIG 14.6-23	FIG 14.6-23	2
		FIG 14.6-24	FIG 14.6-24	2
		FIG 14.6-25	FIG 14.6-25	2
		FIG 14.6-26	FIG 14.6-26	2
		FIG 14.6-27	FIG 14.6-27	2
		FIG 14.6-28	FIG 14.6-28	2
		FIG 14.6-29	FIG 14.6-29	2
		FIG 14.6-30	FIG 14.6-30	2
		FIG 14.6-31	FIG 14.6-31	2
		FIG 14.6-32	FIG 14.6-32	2

Volume No.	Tab	Remove Page No.	Insert Page No.	Rev. No.
V	14. Safety and Accident Analysis			
		FIG 14.6-33	-	
		FIG 14.6-34	-	
		FIG 14.6-35	-	
		FIG 14.6-36	-	
		FIG 14.6-37	-	
		FIG 14.6-38	-	
		FIG 14.6-39	-	
		FIG 14.6-40	-	
		FIG 14.6-41	-	
		FIG 14.6-42	-	
		FIG 14.6-43	-	
		- Insert tab titled, "14A. Offsite Dose Calculations for High Burnup Fuel" after FIG 14.9-7.		
	14A. Offsite Dose Calculations for High Burnup Fuel			
		- Insert the Offsite Dose Calculations for High Burnup Fuel (cover sheet and 18 pages).		