Table 3.1.4-1 (page 1 of 1)
Control Rod Scram Times

------------------------------------NOTES------------------------------------

1. OPERABLE control rods with scram times not within the limits of this Table are considered "slow."

2. Enter applicable Conditions and Required Actions of LCO 3.1.3, "Control Rod OPERABILITY," for control rods with scram times > 7 seconds to 90% insertion. These control rods are inoperable, in accordance with SR 3.1.3.4, and are not considered "slow."

PERCENT INSERTION	SCRAM TIMES[a][b] (seconds) when REACTOR STEAM DOME PRESSURE ≥ 800 psig
5	0.48
20	0.89
50	1.98
90	3.44

(a) Maximum scram time from fully withdrawn position based on de-energization of scram pilot valve solenoids at time zero.

(b) Scram times as a function of reactor steam dome pressure when < 800 psig are within established limits.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.6	Verify each SLC subsystem manual valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position, or can be aligned to the correct position.	31 days
SR 3.1.7.7	Verify each pump develops a flow rate ≥ 40 gpm at a discharge pressure ≥ 1275 psig.	In accordance with the Inservice Testing Program
SR 3.1.7.8	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	24 months on a STAGGERED TEST BASIS
SR 3.1.7.9	Verify all heat traced piping between storage tank and pump suction is unblocked.	24 months AND Once within 24 hours after piping temperature is restored within the limits of Figure 3.1.7-2
SR 3.1.7.10	Verify sodium pentaborate enrichment is ≥ 30.0 atom percent B-10.	Prior to addition to SLC tank

4.0 DESIGN FEATURES

4.1 Site Location

4.1.1 Site and Exclusion Area Boundaries

The site area boundary follows the Illinois River to the north, the Kankakee River to the east, a country road from Divine extended eastward to the Kankakee River on the south, and the Elgin, Joliet, and Eastern Railway right-of-way on the west. The exclusion area boundary shall be an 800 meter radius from the centerline of the reactor vessels.

4.1.2 Low Population Zone

The low population zone shall be a five mile radius from the centerline of the reactor vessels.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 724 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. The assemblies may contain water rods or other assembly bypass channels. Limited substitutions of Zircaloy, ZIRLO, or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with NRC staff approved codes and methods and have been shown by tests or analyses to comply with all safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 Control Rod Assemblies

The reactor core shall contain 177 cruciform shaped control rod assemblies. The control material shall be boron carbide and hafnium metal as approved by the NRC.

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

9. ANF-89-98(P)(A), Generic Mechanical Design Criteria for BWR Fuel Designs.

10. ANF-91-048(P)(A), Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model.

11. Commonwealth Edison Company Topical Report NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods."

12. EMF-85-74(P), RODEX2A (BWR) Fuel Rod Thermal Mechanical Evaluation Model.

13. NEDE-224011-P-A, "General Electric Standard Application for Reactor Fuel (GESTAR)."

14. NEDC-32981P, "GEXL96 Correlation for ATRIUM 9B Fuel," September 2000.

15. NEDO-32465-A, "BWR Owners' Group Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," August 1996.

16. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel."

17. WCAP-16081-P-A, "10x10 SVEA Fuel Critical Power Experiments and CPR Correlation: SVEA-96 Optima2."

18. WCAP-15682-P-A, "Westinghouse BWR ECCS Evaluation Model: Supplement 2 to Code Description, Qualification and Application."

19. WCAP-16078-P-A, "Westinghouse BWR ECCS Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2 Fuel."

20. WCAP-15836-P-A, "Fuel Rod Design Methods for Boiling Water Reactors - Supplement 1."

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

21. WCAP-15942-P-A, "Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors, Supplement 1 to CENPD-287-P-A."

22. CENPD-390-P-A, "The Advanced PHOENIX and POLCA Codes for Nuclear Design of Boiling Water Reactors."

The COLR will contain the complete identification for each of the TS referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements).

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

Table 3.1.4-1 (page 1 of 1)
Control Rod Scram Times

NOTES

1. OPERABLE control rods with scram times not within the limits of this Table are considered "slow."

2. Enter applicable Conditions and Required Actions of LCO 3.1.3, "Control Rod OPERABILITY," for control rods with scram times > 7 seconds to 90% insertion. These control rods are inoperable, in accordance with SR 3.1.3.4, and are not considered "slow."

PERCENT INSERTION	SCRAM TIMES[a][b] (seconds) when REACTOR STEAM DOME PRESSURE ≥ 800 psig
5	0.48
20	0.89
50	1.98
90	3.44

(a) Maximum scram time from fully withdrawn position based on de-energization of scram pilot valve solenoids at time zero.

(b) Scram times as a function of reactor steam dome pressure when < 800 psig are within established limits.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.6	Verify each SLC subsystem manual valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position, or can be aligned to the correct position.	31 days
SR 3.1.7.7	Verify each pump develops a flow rate ≥ 40 gpm at a discharge pressure ≥ 1275 psig.	In accordance with the Inservice Testing Program
SR 3.1.7.8	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	24 months on a STAGGERED TEST BASIS
SR 3.1.7.9	Verify all heat traced piping between storage tank and pump suction is unblocked.	24 months <u>AND</u> Once within 24 hours after piping temperature is restored within the limits of Figure 3.1.7-2
SR 3.1.7.10	Verify sodium pentaborate enrichment is ≥ 30.0 atom percent B-10.	Prior to addition to SLC tank

4.0 DESIGN FEATURES

4.1 Site Location

4.1.1 Site and Exclusion Area

The site consists of approximately 784 acres on the east bank of the Mississippi River opposite the mouth of the Wapsipinicon River, approximately three miles north of the village of Cordova, Rock Island County, Illinois. The exclusion area shall not be less than 380 meters from the centerline of the chimney.

4.1.2 Low Population Zone

The low population zone shall be a three mile radius from the centerline of the chimney.

4.2 Reactor Core

4.2.1 Fuel Assemblies

The reactor shall contain 724 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. The assemblies may contain water rods or other assembly bypass channels. Limited substitutions of Zircaloy or ZIRLO filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with NRC staff approved codes and methods and have been shown by tests or analyses to comply with all safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions.

4.2.2 Control Rod Assemblies

The reactor core shall contain 177 cruciform shaped control rod assemblies. The control material shall be boron carbide and hafnium metal as approved by the NRC.

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

10. Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors/Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors: Methodology for Analysis of Assembly Channel Bowing Effects/NRC Correspondence, ANF-524(P)(A).

11. COTRANSA 2: A Computer Program for Boiling Water Reactor Transient Analyses, ANF-913(P)(A).

12. Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model, ANF-91-048(P)(A).

13. Commonwealth Edison Topical Report NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods."

14. ANFB Critical Power Correlation Application for Coresident Fuel, EMF-1125(P)(A).

15. EMF-85-74(P), RODEX2A(BWR) Fuel Rod Thermal Mechanical Evaluation Model, Supplement 1(P)(A) and Supplement 2 (P)(A), Siemens Power Corporation, February 1998.

16. NEDC-32981P, "GEXL96 Correction for ATRIUM 9B Fuel."

17. NEDO-32465-A, "BWR Owners' Group Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications," August 1996.

18. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel."

19. WCAP-16081-P-A, "10x10 SVEA Fuel Critical Power Experiments and CPR Correlation: SVEA-96 Optima2."

20. WCAP-15682-P-A, "Westinghouse BWR ECCS Evaluation Model: Supplement 2 to Code Description, Qualification and Application."

21. WCAP-16078-P-A, "Westinghouse BWR ECCS Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2 Fuel."

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

22. WCAP-15836-P-A, "Fuel Rod Design Methods for Boiling Water Reactors - Supplement 1."

23. WCAP-15942-P-A, "Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors, Supplement 1 to CENPD-287-P-A."

24. CENPD-390-P-A, "The Advanced PHOENIX and POLCA Codes for Nuclear Design of Boiling Water Reactors."

The COLR will contain the complete identification for each of the TS referenced topical reports used to prepare the COLR (i.e., report number, title, revision, date, and any supplements).

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.