Commonwealth Edison Company
Byron Generating Station
4450 North German Church Road
Byron, IL 61010-9794
Tel 815-234-5441

ComEd

March 31, 2000

LTR: BYRON 2000-0016
File: 2.07.0611

United States Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. STN 50-454 and STN 50-455

Subject: 10 CFR 50.59 Annual Report

Pursuant to the requirements of 10 CFR 50.59(b)(2), Byron Station is providing the required annual report for Facility Operating License Nos. NPF-37 and NPF-66. This report is being provided for the 1999 calendar year and consists of descriptions and safety evaluation summaries for changes to the facility or procedures as described in the Updated Final Safety Analysis Report (UFSAR), and tests or experiments not described in the UFSAR. Also included as part of this report are changes made to features of the fire protection program not previously presented to the NRC Staff.

Please direct any questions regarding this submittal to Brad Adams, Regulatory Assurance Manager, at (815) 234-5441 extension 2280.

Richard P. Lopriore for
William Levis
Site Vice President
Byron Station

WL/BA/kh

Attachment

cc: Regional Administrator – Region III
NRC Senior Resident Inspector – Byron Station

IE47

bcc: NRC Project Manager – NRR – Byron Station (w/o attachments)
Nicholas Reynolds – Winston & Strawn (w/o attachments)
Office of Nuclear Facility Safety – IDNS
Site Vice President – Byron Station (w/o attachments)
Vice President, Regulatory Services (w/o attachments)
Regulatory Assurance Manager – Byron Station (w/o attachments)
Director, Licensing and Compliance – Braidwood and Byron Stations (w/o attachments)
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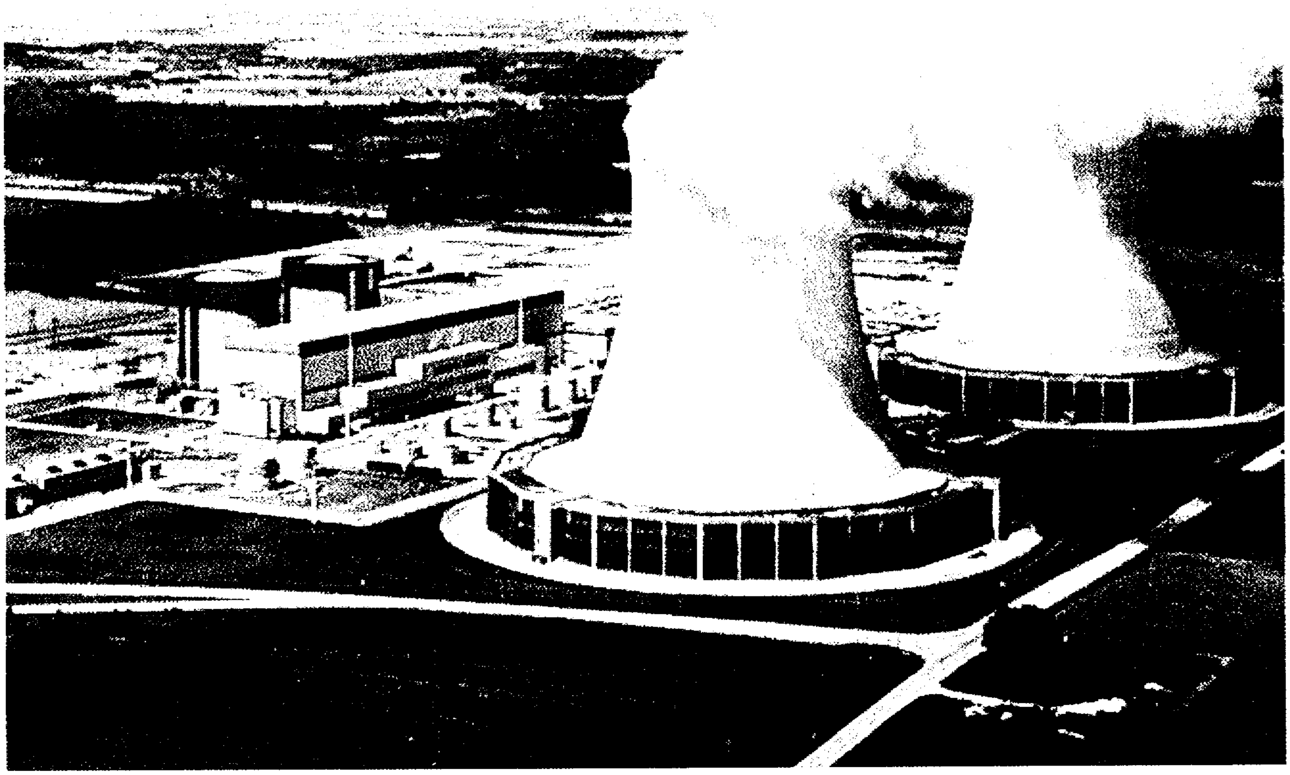
10 CFR 50.59 Summary Report



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Byron Nuclear Power Station

10 CFR 50.59 Summary Report

1999

NRC Docket Nos. 50-454 and 50-455

License Nos. NPF-37 and NPF-66

DESIGN CHANGE PACKAGE – DCP

1.	DCP 9500230	6G-99-0006
2.	DCP 9600185 (Unit 1) & 9600186 (Unit 2)	6G-98-0174
3.	DCP 9600222	6G-98-0195
4.	DCP 9600228	6G-97-0243
5.	DCP 9600377	6G-98-0011
6.	DCP 9600383	6G-98-0105
7.	DCP 9600413 & 9600414	6G-98-0070
8.	DCP 9600419	6G-98-0095
9.	DCP 9600420	6H-98-0170
10.	DCP 9700022 & 9700023	6G-99-0097
11.	DCP 9700114	6G-98-0186
12.	DCP 9700174 & 9700175	6G-98-0104
13.	DCP 9700177 (Unit 1) & 9700178 (Unit 2)	6G-98-0042
14.	DCP 9700228 & 9700230	6G-98-0189
15.	DCP 9700231	6G-98-0308
16.	DCP 9700233	6G-98-0230
17.	DCP 9700313	6G-97-0194
18.	DCP 9700319 & 9700320	6G-98-0122
19.	DCP 9700358 & 9700359	6G-98-0202
20.	DCP 9700386	6G-98-0129
21.	DCP 9700388 (Unit 1) & 9700389 (Unit 2)	6G-98-0158
22.	DCP 9700401	6G-97-0258
23.	DCP 9700402	6G-98-0108
24.	DCP 9700431	6G-98-0178
25.	DCP 9700506 (Unit 1) & 9700507 (Unit 2)	6G-98-0052
26.	DCP 9700511 & 9700512	6H-98-0346
27.	DCP 9700513 & 9700514	6H-98-0370
28.	DCP 9700513 & 9700514	6H-99-0385
29.	DCP 9700517 & 9700518	6H-98-0347
30.	DCP 9700519 & 9700520	6H-98-0369
31.	DCP 9700521 & 9700522	6H-98-0368
32.	DCP 9700521 & 9700522	6H-99-0386
33.	DCP 9700559	6G-98-0263
34.	DCP 9700560	6G-98-0047
35.	DCP 9700568 & 9700569	6G-98-0088
36.	DCP 9700712	6G-98-0062
37.	DCP 9700729 & 9700730	6G-99-0175
38.	DCP 9700780	6G-98-0293
39.	DCP 9700790 & 9700791	6G-98-0006
40.	DCP 9700811	6G-98-0307

DESIGN CHANGES – DCP (CONT'D.)

41.	DCP 9800019	6G-98-0098
42.	DCP 9800050	6H-98-0366
43.	DCP 9800057	6H-98-0365
44.	DCP 9800096	6G-98-0131
45.	DCP 9800122 & 9800123	6G-99-0176
46.	DCP 9800140	6G-99-0046
47.	DCP 9800235	6G-99-0004
48.	DCP 9800239	6G-98-0132
49.	DCP 9800255 & 9800256	6G-98-0275
50.	DCP 9800315	6G-99-0022
51.	DCP 9800347	6H-99-0221
52.	DCP 9800368 & 9800369	6G-99-0081
53.	DCP 9800406	6G-98-0255
54.	DCP 9800417	6G-99-0030
55.	DCP 9800482, 9800483, 9800484, 9800485, 9800494, 9800495, 9800496, 9800497	6G-99-0161
56.	DCP 9800498	6G-99-0012
57.	DCP 9800510	6G-98-0282
58.	DCP 9800532	6G-99-0011
59.	DCP 9800539 & 9800540	6G-99-0144
60.	DCP 9800560 & 9800561	6G-98-0264
61.	DCP 9800567	6H-99-0393
62.	DCP 9800578 & 9800579	6G-99-0156
63.	DCP 9800599	6G-99-0018
64.	DCP 9800610	6G-99-0010
65.	DCP 9800615	6G-98-0299
66.	DCP 9800624 & 9800625	6G-99-0039
67.	DCP 9900018 & 9900021	6G-99-0095
68.	DCP 9900022 & 9900023	6G-98-0156
69	DCP 9900026	6G-99-0050
70.	DCP 9900083	6G-99-0042
71.	DCP 9900101	6G-99-0067
72.	DCP 9900102	6G-99-0068
73.	DCP 9900161 & 9900162	6G-99-0098
74.	DCP 9900207	6G-99-0110
75.	DCP 9900230	6G-99-0149
76.	DCP 9900258 & 9900259	6H-99-0310
77.	DCP 9900289 & 9900290	6G-99-0240
78.	DCP 9900339	6G-99-0197

DRAFT REVISION PACKAGE – DRP (UFSAR CHANGE)

1.	DRP 7-142	6G-98-0032
2.	DRP 7-166	6G-98-0106
3.	DRP 7-167	6G-98-0215
4.	DRP 7-177	6G-98-0277
5.	DRP 7-180	6G-98-0283
6.	DRP 7-185	6G-98-0071
7.	DRP 7-190	6G-98-0074
8.	DRP 7-229	6G-98-0136
9.	DRP 7-241	6G-98-0151
10.	DRP 7-253	6H-98-0196
11.	DRP 7-258	6G-98-0177
12.	DRP 7-272	6H-99-0184
13.	DRP 8-014	6G-98-0304
14.	DRP 8-019	6G-98-0296
15.	DRP 8-032	6G-99-0028
16.	DRP 8-033	6G-99-0040
17.	DRP 8-040	6H-99-0320
18.	DRP 8-043	6G-99-0086
19.	DRP 8-051	6H-99-0371
20.	DRP 8-055	6G-99-0103
21	DRP 8-056	6G-99-0101
22	DRP 8-057	6G-99-0106
23.	DRP 8-060	6G-99-0119
24.	DRP 8-062	6G-99-0121
25	DRP 8-064	6G-99-0124
26.	DRP 8-065	6G-99-0206
27	DRP 8-066	6G-99-0205
28.	DRP 8-076	6G-99-0128
29	DRP 8-077	6G-99-0186
30.	DRP 8-081	6G-99-0135
31.	DRP 8-082	6G-99-0192
32.	DRP 8-083	6G-99-0236
33.	DRP 8-086	6G-99-0188

FIRE PROTECTION REPORT DRAFT REVISION PACKAGE - FDRP

1.	FDRP 18-051	6G-98-0274
2.	FDRP 18-065	6G-98-0173
3.	FDRP 18-070	6G-98-0252
4.	FDRP 18-072	6G-98-0254
5.	FDRP 19-002	6G-98-0271
6.	FDRP 19-006	6G-99-0035
7.	FDRP 19-007	6G-99-0044
8.	FDRP 19-009	6G-97-0045
9.	FDRP 19-014	6G-99-0146
10.	FDRP 19-017	6G-99-0190
11.	FDRP 19-018	6G-99-0170
12.	FDRP 19-019	6G-99-0191
13.	FDRP 19-022	6G-99-0195
14.	FDRP 19-023	6G-99-0200
15.	FDRP 19-024	6G-99-0181

DOCUMENT CHANGE REQUEST – DCR

1.	DCP 9800633	6G-99-0084
2.	DCR 970104 & 990049	6G-99-0076
3.	DCR 980148	6G-99-0210
4.	DCR 980169 & 980170	6H-98-0095
5.	DCR 980181	6G-98-0214
6.	DCR 980237	6G-98-0245
7.	DCR 990098	6G-99-0168
8.	DCR 990149	6G-99-0162
9.	DCR 990188	6G-99-0204
10.	DCR 990213	6H-99-0391
11.	DCR 990263	6G-99-0209

TEMPORARY ALTERATIONS – TA OR TMOD

1.	DCP 9900351	6G-99-0134
2.	TEMP ALT 98-2-031	6H-98-0091
3.	TEMP ALT 98-2-051	6G-98-0229
4.	TEMP ALT 98-2-033	6H-98-0339
5.	TEMP ALT 98-2-067	6G-98-0273
6.	TMOD 98-0-070	6G-98-0303
7.	TEMP ALT 99-1-002	6G-99-0005
8.	TMOD 99-1-010	6G-99-0029
9.	TMOD 99-2-018	6G-99-0083

TEMPORARY ALTERATIONS – TA or TMOD (Cont'd.)

10.	T-MOD	99-1-020	6G-99-0091
11.	T-MOD	99-0-024	6G-99-0102
12.	T-MOD	99-2-025	6G-99-0107
13.	T-MOD	99-1-029	6G-99-0148
14.	T-MOD	99-2-040	6G-99-0196

PROCEDURE CHANGES

1.	BAP 370-3	Rev. 23	6H-99-0139
2.	BAP 1100-3	Rev. 14	6G-99-0153
3.	BAR 0-39-A3 BAR 0-37-A3	Rev. 0 Rev. 0	6H-99-0121
4.	BCP 300-15	Rev. 11	6H-99-0030
5.	BOP AP-80	Rev. 3	6G-99-0080
6.	BOP CV-14	Rev. 14	6H-99-0107
7.	BOP CV-14	Rev. 14	6G-99-0019
8.	BOP CV-14	Rev. 14	6G-99-0033
9.	BOP DG-M1 BOP DG-M1A BOP DG-M1B BOP DG-M2 BOP DG-M2A BOP DG-M2B BAR 1/2PL07J-1-D6 BAR 1/2PL08J-1-D6	Rev. 14 Rev. 6 Rev. 6 Rev. 10 Rev. 3 Rev. 3 Rev. 2 Rev. 2	6H-99-0230
10.	BOP RC-0	Rev. 11	6H-99-0040
11.	BOP RC-8A	Rev. 12	6G-98-0027
12.	BOP RC-9 BOP RC-9A1	Rev. 9 Rev. 0	6G-98-0097
13.	BOP TR-9	Rev. 2	6G-98-0009
14.	BOP VV-13 BAR 1PM09J-A8 BAR 1/2PM09J-A7 BAR 033-C7	Rev. 2 Rev. 2 Rev. 1 Rev. 7	6H-99-0397
15.	BRP 5820-18 BRP 5820-18T1	Rev. 0 Rev. 0	6G-98-0258
16.	0BOL 6.8	Rev. 2	6G-99-0051
17.	1BOL 7.b LCOAR	Rev. 2	6H-99-0281
18.	2BOL 7.b LCOAR	Rev. 1	6H-99-0282
19.	0BOS 0.1-0	Rev. 19	6G-98-0222
20.	1BOS 3.1.1-20	Rev. 18	6H-99-0013
21.	1BOSR 3.1.5-1	Rev. 1	6H-99-0018
22.	1BOSR 3.2.7-605A/B	Rev. 3	6H-99-0177

PROCEDURE CHANGES

23.	1/2BOSR 3.2.9-4	Rev. 2	6G-99-0026
24.	1/2BOSR 6.3.3-1 1/2BOSR 6.3.4-1	Rev. 1 Rev. 1	6H-99-0063
25.	BVP 800-11	Rev. 4	6H-99-0370
26.	1BVS 2.3.5-1	Rev. 12	6G-98-0086
27.	1BVS 4.6.2.2-1	Rev. 12	6G-98-0056
28.	0BVSR DC-3A 0BVSR DC-3B	Rev. 0 Rev. 0	6G-99-0094
29.	SAAD 1990086 (Process Computer Procedure Change)		6G-99-0100
30.	1BVSR 5.5.8.SD.3-1	Rev. 2	6G-99-0053
31.	1/2BVSR AF-3	Rev. 2	6H-99-0336
32.	1BVSR XPT-14 2BVSR XPT-14 1BVSR 3.1.11-1 2BVSR 3.1.11-1	Rev. 2 Rev. 2 Rev. 2 Rev. 2	6H-99-0266
33.	Surveillance Procedures		6G-98-0057

SPECIAL PROCESS PROCEDURES – SPP

1.	SPP 97-138	6G-98-0079
2.	SPP 97-141 & 97-142	6G-98-0002
3.	SPP 98-007	6G-98-0016
4.	SPP 98-017	6G-98-0060
5.	SPP 98-023	6G-98-0043
6.	SPP 98-046	6G-98-0115
7.	SPP 98-073	6H-99-0214
8.	SPP 98-078	6H-98-0166
9.	SPP 99-005	6G-99-0036
10.	SPP 99-008	6G-99-0032 6H-99-0356
11.	SPP 99-015	6H-99-0125
12.	SPP 99-021	6G-99-0064
13.	SPP 99-023	6G-99-0066
14.	SPP 99-026	6G-99-0130
15.	SPP 99-029	6G-99-0104
16	SPP 99-034	6H-99-0332

MISCELLANEOUS

1.	Byron Unit 1 Cycle 10 (BY1C10) Core Reload Design	6G-99-0060
2.	Byron Unit 1 Cycle 10 (BY1C10) Lead Test Assemblies	6G-99-0061
3.	Byron Unit 1 Cycle 10 (BY1C10) Core Reload Design	6H-99-0173
4.	Byron Unit 2 Cycle 8 (BY2C8) Core Loading Pattern	6G-98-0083
5.	Byron Unit 2 Cycle 9 (BY2C9) Core Reload Design	6G-99-0127
6.	Byron Unit 2 Cycle 8A (BY2C8) Mid-load Reload Design	6G-99-0055
7.	DCP 9900233	6G-99-0154
8.	DCP 9900233	6G-99-0198
9.	Temp. Procedure 1BOSR 3.1.5-1, Rev. 1	6H-99-0083
10.	Various Procedures: BOP SX-3 Rev. 13 0BOSR 7.9.6- 1 Rev. 2 0BOSR 7.9.6-2 Rev. 2 0BOL 7.9 Rev. 2 BAR 0VH01J-1-B5 Rev. 1 BAR 0-38-C4 Rev. 4 BAR 1-20-A5 Rev. 2 BAR 1-20-A11 Rev. 1 BAR 1-20-A12 Rev. 6 BAR 1-20-B5 Rev. 2 BAR 1-20-B11 Rev. 1 BAR 1-20-B12 Rev. 1 BAR 1-20-C12 Rev. 1 BAR 1-20-D5 Rev. 1 BAR 1-20-D11 Rev. 1 BAR 1-20-D12 Rev. 1 BAR 2-20-A5 Rev. 2 BAR 2-20-A11 Rev. 4 BAR 2-20-A12 Rev. 3 BAR 2-20-B5 Rev. 2 BAR 2-20-B11 Rev. 2 BAR 2-20-B12 Rev. 4 BAR 2-20-C12 Rev. 3 BAR 2-20-D5 Rev. 1 BAR 2-20-D11 Rev. 1 BOP 199-EA OS Rev. 16	6H-99-0396

ENGINEERING REQUESTS – ER

1.	ER 9711123	6H-98-0081
2.	ER 9808072	6G-98-0291
3.	ER 9808116	6G-99-0072
4.	ER 9808888	6G-98-0302
5.	ER 9809025	6G-98-0301
6.	ER 9809338	6G-99-0009

WORK REQUESTS – WR

1.	WR 990000334	6G-99-0002
2.	WR 990074856	6G-99-0132

OUT OF SERVICE – OOS

1.	OOS 940012511 & 940012434	6G-98-0113
2.	OOS 940012923 & 940012924	6G-98-0232
3.	OOS 980001044	6G-98-0233
4.	OOS 980001045 & 980001631	6G-98-0234
5.	OOS 980002260	6G-98-0243
6.	OOS 980003840	6G-98-0217
7.	OOS 980012235 & 980012239	6H-99-0241

OTHER

1.	Abnormal Valve Lineup 2SX173	6G-99-0211
2.	Beacon Implementation	6G-98-0250
3.	Byron Tech Review – BYTR 98-017	6G-98-0093
4.	Byron Tech Review – BYTR 98-018	6G-98-0094
5.	Byron Tech Review – BYTR 98-018	6G-98-0130
6.	Byron 2ND Interval Inservice Testing Plant For Pumps and Valves	6H-98-0039
7.	DCP 9800089, 9800090, 9800316, 9800317	6H-98-0131
8.	DCP 9500393	6G-98-0044
9.	Improved Technical Specification (ITS) Bases Change to B3.9.2	6G-99-0052
10.	ODCM Chapter 12, Rev. 1.6	6H-99-0420
11.	ODCM, Rev. 2 , April 1999	6G-99-0077
12.	On-site Review – OSR 97-157	6G-98-0119
13.	PCCIR 6-99-005	6H-99-0010
14.	PCCIR 6-99-011	6H-99-0093
15.	PCCIR 6-99-015	6H-99-0102
16.	PCCIR 6-99-022	6H-99-0122
17.	PCCIR 6-99-024	6H-99-0171

OTHER

18.	Various Procedures 2BISR 3.1.7-001 Rev. 2 2BISR 3.1.7-002 Rev. 2 2BISR 3.1.7-003 Rev. 2 2BISR 3.1.7-004 Rev. 2 2BOA ELEC-1 Rev. 54C 2BOA PRI-5 Rev. 57C 2BOA PRI-6 Rev. 56B 2BOA SEC-4 Rev. 52B BAR 2-18-E16 Rev. 5 BOP RY-5 Rev. 8 BAP 560-1 Rev. 17	6H-99-0154 THRU 6H-99-0164
19.	SAAD 1999-0181	6H-99-0375
20.	Setpoint Scaling Change Request – SSCR 98-029	6G-98-0198
21.	Setpoint Scaling Change Request – SSCR 98-032 & BAR RM11-1-2PR27J, Rev. 3	6H-98-0174
22.	Setpoint Scaling Change Request – SSCR 99-030	6H-99-0311
23.	Tech Eval – BOM-1998-3667-001	6G-99-0089
24.	Technical Specification Bases 3.6.3	6G-99-0015
25.	Technical Specification Amendment 6	6G-99-0016
26.	Westinghouse SECL-99-035	6H-99-0150
27.	WR 980005470	6H-98-0116

TECHNICAL REQUIREMENTS MANUAL – TRM

1.	TRM Change 99-04	6G-99-0031
2.	TRM Section 3.7.b	6G-99-0057
3.	TRM Appendix 0	6G-99-0111
4.	TRM 3.1.k	6G-99-0122
5.	Technical Specification Bases 3.9.4	6G-99-0133
6.	Technical Specification Amendment 106	6H-99-0021

6G-99-0006
DCP 9500230

DESCRIPTION:

The purpose of this Design Change was to modify the Unit 1 Main Steam Power Operated Relief Valve (MS PORV) actuators as follows:

1. Replaced and relocated the open, close, accumulator pneumatic, and accumulator hydraulic pressure switches.
2. Added new Isolation Valves in the sensing lines to the pressure switches.
3. Added a new terminal strip for the pressure switches.
4. Added a new mounting plate and supports for the new pressure switches.
5. Relocated and added a new isolation valve for the hydraulic fluid fill line.
6. Added a new opening and access cover in the fiberglass actuator cover.
7. Increased the open pressure switch, close pressure switch, open relief valve, and close relief valve setpoints.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the probability of an accident is unchanged by the design changes. The modification did not significantly change the actuator weight; thus the MS piping stresses were not adversely affected. The function and operation of the MS PORVs pressure switches were not changed. The replacement pressure switches were recommended by the valve actuator OEM and are designed to have less setpoint drift and are less susceptible to vibration. The improved pressure switch reliability should improve MS PORV reliability. The ability of the MS PORVs to function to mitigate the consequences of an accident was not affected. The Steam Generator Tube Rupture (SGTR) accident analysis assumes an MS PORV fails open on the ruptured SG (off-site dose case) or fails closed (margin to overfill case). The results of this analysis were unchanged by the modification. Therefore, the consequences of an SGTR were not increased. The actuator changes have been seismically and environmentally qualified.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because this design change was installed to improve pressure switch reliability, MS PORV performance, and to improve access to actuator components during calibration activities. The function and operation of the MS PORVs was not changed. No new failure modes were created.

6G-99-0006
DCP 9500230
(Cont'd.)

The new pressure switches have the same failure modes as the replaced pressure switches. The new fill valve has the same failure mode as the replaced valve (failure of the hydraulic fluid pressure boundary). The new valve is a diaphragm valve while the replaced valve is a Schrader valve. The diaphragm valve provides better shutoff and minimizes the chance of a hydraulic fluid leak.

Plant procedures provide assurance that the new manual valves in the tubing to the pressure sensors are properly aligned for operation. Additionally, the pressure switch Isolation Valves are located inside the actuator cover; therefore, it is unlikely that a pressure switch isolation valve could be mispositioned during plant operation.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which Technical Specifications are based.

6G-98-0174, Rev.1
DCP 9600185 (UNIT 1) & 9600186 (U2)

DESCRIPTION:

The purpose of this Design Change was to add a time delay to the auto-close circuit of the RH system miniflow recirculation valves. The relay was installed in the RH miniflow valves Motor Control Center. Flow transients during RH pump starts have caused inadvertent closure of the recirculation valves. The change will prevent inadvertent closure of the recirculation valves.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because this change improves the reliability of the RH pumps by adding assurance that the RH miniflow valves will perform their safety function as intended for safe operation and as analyzed in the UFSAR. The function of the RH miniflow valves will not be changed. Since the function of the RH miniflow valve has not been changed the consequences of the accident are unchanged. The consequences of an accident or offsite dose will not be increased because the function of the RH miniflow valves has not changed as described in the UFSAR.

   This change will not create any new RHR System failure modes nor increase the probability of a malfunction of equipment important to safety during normal or accident operating conditions. The change or a failure of the change does not alter the design, function, or method of performing the function for another SSC that is described in the UFSAR. Addition of the time delay relay, which is qualified for its intended function in the auto close circuit benefits the overall RH system by providing protection for the RHR pumps. The improvement outweighs the potential increases in the probability of a malfunction of equipment important to safety. The change improves the reliability of the RHR pumps; therefore, the probability of a failure of equipment important to safety is decreased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because this change will not impact the RHR System functions so as to create the possibility of an accident or malfunction other than those described in the UFSAR. Failure of the time delay relay will affect the recirculation capacity only, which is not a new type of event as described in the UFSAR. This change merely improves the operability of the RH miniflow to perform its function. The functions of the present RHR System are maintained by the change. Therefore, there is no impact on the UFSAR or its analysis.

6G-98-0174, Rev.1
DCP 9600185 (Unit 1) & 9600186 (Unit 2)
(Cont'd.)

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the changes to add a time delay to the auto-close circuit of the RH system miniflow recirculation valves will not affect the function, operation or margin of safety for any SSCs required by the Tech. Specs. The modification does not involve changes to any parameters upon which the Technical Specifications are based.

6G-98-0195
DCP 9600222

DESCRIPTION:

The purpose of this Design Change was to add welded plate flanges into piping to seal steam and liquid inputs to the Liquid Radwaste Evaporators. A small jumper pipe was added to allow isolation of only the evaporators while retaining the function of other liquid waste treatment equipment. Manually operated Isolation Valves in the auxiliary steam condensate return, were designated as normally closed by procedure. All liquid waste inputs to the evaporators continue to have a flow path to the radwaste mixed bed demineralizer inlet header. This change was installed because the evaporators are no longer used and were becoming degraded.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because failure of any of the affected components is bounded by the Radioactive Liquid Waste System Leak or Failure described in UFSAR Section 15.7.2. This activity does not affect the spent resin storage tank or the boron recycle holdup tanks. The chances of a leak were reduced by the addition of the flanges to provide more reliable isolation points. Also, the high-energy auxiliary steam line was isolated prior to entering the Auxiliary Building reducing the possibility of a HELB. Therefore, the probability of an accident or equipment malfunction is not increased. The affected components are not required to mitigate the consequences of an accident. The amount of radioactivity potentially released is not increased. Therefore, there is no increase in the consequences of the accident or of an equipment malfunction.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because this activity adds no new system interactions or functions. It does remove the auxiliary steam supply from the radwaste evaporator package to reduce the chances for leakage. The piping configurations have been fully qualified to Design Basis requirements. Therefore, no new accidents or malfunctions are created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which Technical Specifications are based.

6G-97-0243
DCP 9600228

DESCRIPTION:

The purpose of this Design Change was to install new vibration monitoring equipment and associated alarms for the VA System supply and exhaust fans and motors. The current vibration monitoring of the VA supply and exhaust fans is done by using hand held equipment. The new Vibration Monitoring (VMS) System will allow operators to survey the equipment condition during normal operation of the VA supply and exhaust fans and motors without entering the plenum.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the addition of a Vibration Monitoring (VMS) System does not affect normal operation or design functions of the VA System supply or exhaust fans and motors. The changes do not adversely affect any other equipment or systems used to mitigate any accidents. This change will not introduce any new failure modes. Modes of failure of the VA fans and their consequences are not adversely changed. Therefore, the probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the vibration sensors will be mounted to the bearing pillow blocks and motor frame housings. The optical speed sensor would be seismically mounted on a bracket secured to the motor. Loss of the vibration-monitoring panel will not affect normal operation or design basis functions of the VA System supply or exhaust fans and motors. This change did not introduce any new failure modes or adversely affect the function or operation of the VA System fans and motors. No other components or systems will be affected by this design change. Therefore, no other accident or malfunction of a type different from those evaluated in the SAR can be created by this design change.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which Tech. Specs. are based.

6G-98-0011
DCP 9600377

DESCRIPTION:

The purpose of this Design Change was to replace two skids as part of the Condensate Polishing System. The first skid is the acid injection skid which consists of an acid day tank, two full capacity acid metering pumps and various supporting valves and instruments. The skids were be completely replaced. The changes between the old and the new skids include:

- Increasing the tank size from 200 gallons to 350 gallons,
- Upgrading the metering pumps to a higher capacity variable speed model,
- Upgrading the piping material from polypropylene to alloy 20 pipe, and
- Upgrading the control system.

The second skid is the caustic injection skid that contains the same basic equipment as the acid skid. The changes are similar to those identified for the acid tank with one exception. Heat trace on the original caustic tank and associated piping is being eliminated by diluting the caustic to a 20-30% solution. A dilution line is being added for this purpose.

Both existing skids have degraded over time due to leaks and internal corrosion to the extent that they have become unreliable.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the acid and caustic skids are used in the regeneration of resin for the condensate cleanup system. The skids have no functions that will lead to an accident or affect the consequences of an accident or malfunction of equipment important to safety. The skids are not relied upon to function to mitigate an accident. Therefore, there is no increase in the probability of occurrence or consequences of an accident.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the acid and caustic skids are used in the regeneration of resin for the condensate cleanup system. There are no new failure modes associated with this change. The only existing failure mode for the skids that could be more severe is leakage of the acid. The increase of the acid tank size would result in an additional 150 gallons of acid to the condensate polishing room. There is no safety-related equipment in the room and there are no systems, which must operate to mitigate an accident located in this room. Therefore, there is no possibility for a new accident or malfunction of equipment important to safety.

6G-98-0011
DCP 9600377
(Cont'd.)

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which Technical Specifications are based. Therefore, the margin of safety defined by the basis of any Technical Specification was not impacted.

6G-98-0105, Rev. 2
DCP 9600383

DESCRIPTION:

The purpose of this Design Change was to install new temporary line stop and supports, new permanent fittings for the temporary line stop, and provides for Isolation Valves to be installed in the WS supply and return headers for Unit 0 and Unit 1 Service Air Compressors (SACs).

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the systems cooled by the WS are not affected by the activity and will continue to function as designed. The WS system has no interface with or impact on equipment important to safety and no interfaces are created by this design change.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the new Isolation Valves will function to isolate flow to and from the Unit 0 and Unit 1 SACS. All piping components and valves are qualified for the design pressure. Therefore, there will be no accident or malfunction of a type different from those evaluated in UFSAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which Tech. Specs. are based.

6G-98-0070
DCP 9600413 & 9600414

DESCRIPTION:

The purpose of this Design Change was to provide installation details to add an interposing time delay relay in the CC Pump Low Discharge Pressure auto start circuitry for all CC pumps. During normal operations, one CC pump is placed in standby to auto start on low discharge pressure, less than 85 psig, in the CC Pump Discharge Header. This auto start feature is provided to automatically start the standby pump in the event the operating pump fails.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because this change does not impact the ability or capability of the Component Cooling Water System (CC) from providing adequate cooling water to various plant components of either unit during normal operation, plant shutdown or after an accident. The CC pumps and affected low discharge pressure instruments do not influence any event responsible for any accident. The automatic start of the standby CC Pump on low discharge pressure is not an ESF function. Therefore, the probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because this change does not adversely impact the operation or function of any plant SSC important to safety, including the CC system, reactor coolant pumps, or containment isolation features. CC system parameters during normal and accident conditions remain unchanged as a result of this change. Therefore, the possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which Tech. Specs. are based.

6G-98-0095
DCP 9600419

DESCRIPTION:

The purpose of this Design Change was to remove the staircase between elevations 426' and 377' in the Byron Unit 1 Containment Building and the installation of several handrails and miscellaneous construction details around the stairwell. The activity also installed a jib crane inside the Containment Building at elevation 436'8" (attached to the Steam Generator enclosure wall) to serve the opening after the subject staircase was removed. Finally, this activity installed a new steel framing beam at elevation 399' and 400' of the Unit 1 Containment Building near column line R1. The jib crane was required to move small tools in containment without conflicting with use of the overhead crane. The handrails and details were installed to meet personnel safety code requirements. The beam was added to increase the platform area for safety reasons.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because all equipment and structures that was removed and installed to and from containment was done in conjunction with approved design calculations and analysis which evaluated the impact on the plant safety analysis. The structural impact of the changes was determined to be acceptable per the current design basis. Therefore, the changes did not increase the probability or consequences of any accidents in the Safety Analysis Report.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because calculations were completed for this activity which demonstrated that no other equipment or structures were affected by the change. Therefore, there was no possibility of new accidents or malfunctions of equipment important to safety than previously evaluated in the Safety Analysis Report.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this activity did not adversely impact any Technical Specification systems or equipment. Therefore, plant operation remained within the basis of plant Technical Specifications.

6H-98-0170
DCP 9600420

DESCRIPTION:

The purpose of this Design Change was to modify several components in the Unit 2 Containment Building as follows:

a. removal of the west staircase,

b. installation of a jib crane,

c. addition of a steel beam at elevation 398' at Coordinate 10.5", and

d. installation of removable handrails around stairwell C-7

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the remaining 2 staircases meet the requirements of NFPA Codes (See NDIT No. BB-EXT-1359).

   a. The jib crane will not interfere with installed equipment and has been analyzed for adverse structural impact (See Calculation 18.6.2.5-BYR98-077).

   b. The steel beam is a passive structural element intended to provide adequate floor support. The beam will not interfere with installed equipment and has been analyzed for adverse structural impact (See Calculation 23.1.23-BYR98-064).

   c. The handrails are installed during outages only and provide a standard safety precaution for large openings.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because Calculation 18.6.2.5-BYR98-077 and 23.1.23-BYR98-064 were performed to show that no other SSCs were adversely impacted.

3. The margin of safety, as defined in the bases for any Technical Specification, is not reduced because the change does not affect Technical Specification Bases parameters.

6G-99-0097
DCP 9700022 & 9700023

DESCRIPTION:

The purpose of this Design Change was to extend existing RH vent lines to the Auxiliary Building Equipment (WE) and Floor Drain (WF) Systems or to the Boric Acid Processing System (Valve Leakoff Collection). The existing vent valves affected by this change are 1/2RH011A/B, 1/2RH026A/B and 1RH027. In some cases, a second safety related isolation valve is added to the vent line to provide a more accessible isolation point for use by station operators.

Valves 1/2RH026A/B and 1RH027 are considered primary containment boundary valves as identified in 1/2BOSR 6.3.3-1. Valves 1/2RH026B and 1RH027 will be normally open after installation of the DCPs. The containment boundary function will be satisfied by valves 1/2RH032 and 1RH031 respectively.

Sight glasses are added to the vent lines for use by operators during the venting process.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the extension of vent lines on the RH system has no impact on RCS flow rate or on heat removed by the secondary system. A failure of a vent valve or of the piping would result in a pressure boundary leak, which would represent a failure of a small line carrying reactor coolant outside of containment. The probability of this event is not increased because the new vent piping and valves meet the system design requirements and are no more likely to fail than the existing vent lines and valves. Inputs to the leak detection sump during venting may increase the cycling frequency of the sump level switch. The switch is designed to cycle and is calibrated on a periodic basis so there is no increase in the probability of malfunction.

   Consequences of accidents and equipment malfunctions are not changed since the vent valve changes will have no impact on the function of the RH system. The ability of the system to deliver low head safety injection, recirculate water from the containment sump and remove heat from the RCS is not changed. Valves and piping are required to maintain system pressure boundary and in some cases containment integrity. The consequences of failure of these components are unchanged. The function of the leak detection sump continues to be to sense leakage. The consequences of a malfunction of this component is also unchanged.

6G-99-0097
DCP 9700022 & 9700023
(Cont'd.)

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the modifications extend existing vent valve lines. Vent valves, piping, drain lines, and the leak detection sump are all used in a manner consistent with their design. The design of the vents meets the original requirements. Several of the vent lines will direct flow to the Auxiliary Building leak detection sumps. The ability of these sumps to detect leakage is not changed because the vent flow is directed upstream of the weir in the same compartment as the level switch. Valve leakage will be detected by an increase in the sump level. Several of the vents are directed to valve leakoff collection lines. The leakoff system directs borated, potentially contaminated water to a collection tank. The water vented from the RH system is no different than the water leaking through the valve packing. Therefore, use of the collection system to receive the vented water is appropriate. All fluids vented from the RH system, leaked through RH or SI valve packing or drained through the Auxiliary Building equipment and floor drain systems are eventually processed through the liquid Radwaste System. Venting fluid using the valve leak-off drain lines or the leak detection sump drain does not create the possibility of a new malfunction.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the DCPs do not change the frequency of venting nor decrease the effectiveness of the venting operation. Therefore, ECCS piping will remain operable as required in the Technical Specifications.

6G-98-0186, Rev. 1
DCP 9700114

DESCRIPTION:

The purpose of this Design Change was to reroute non-contaminated Auxiliary Building (AB) "A" Chiller Floor Drains to the Turbine Building Fire and Oil Sump. This activity will allow the non-contaminated water to be drained from the AB Chillers and associated piping during maintenance activities.

Additionally, the activity installed a new AB Equipment Drain Header (i.e., 0WE172A) to enable non-contaminated water, such as service water from heat exchangers, to be drained to Essential Service Water (SX) Pump Room Sump "A". From the Sump "A", the water can be pumped to the Turbine Building Floor Drain Tanks, and eventually to the Turbine Building Fire and Oil Sump.

The reason for the activities is to reduce the volume of non-contaminated AB drain water processed as liquid radwaste, thereby reducing the high cost of processing radwaste.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the new drain lines do not increase the probability of an accident (i.e., the accident of concern is AB Flooding) because no new sources of water are added. Additionally, the activities are seismically qualified and supported, so they will not impact safety-related equipment.

   The consequences of offsite dose are not increased because systems involved are transporting only non-contaminated water from the AB.

   The consequences of the activity on the AB flooding analysis is addressed as follows. The two AB Chilled Water Cooling Coil areas on Elev. 451' that are being rerouted to the Turbine Building Fire and Oil sump each contain four floor drains. Only three of the four drains are considered in the AB Flooding Analysis. Converting one of the four drains' piping to 4" from the current 2" diameter will not increase the calculated area flood levels. The AB Chiller area on Elev. 463' has open floor grating and an open stairway down to Elev. 451'. Therefore, converting the floor drain piping size will not increase the calculated floor flood level nor affect the consequences of the AB flood analyses.

6G-98-0186, Rev. 1
DCP 9700114
(Cont'd.)

The new drain header is routed to the SX Sump "A". The new header will not increase the consequences of the AB flooding analysis because the drain connections are capped when not in use. This will prevent drainage of flood water into the SX pump rooms.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the new drain lines function in the same manner as the existing drain lines. The lines have loop seals to prevent air from circulating between adjacent areas similar to the existing drain lines. The new pipes are low energy lines and will transport only non-contaminated water. Active components are not installed as part of the change. Therefore, a different type of accident or malfunction is not created by this change.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because no Technical Specifications were impacted by the changes.

6G-98-0104
DCP 9700174 & 9700175

DESCRIPTION:

The purpose of this Design Change was to provide details for the installation of stem protectors on the 1/2SI8811A/B valves. The stem protectors consist of a length of pipe or conduit threaded into the top of the valve actuator. A pipe/conduit cap is installed on the end of the protector. The Limitorque actuator is pre-threaded and requires no modification. The stem protector length is such that it does not interfere with stem travel. Vent holes are provided to allow air to escape when the stem moves. The reason for the installation is to prevent foreign material from entering the valve through the gap where the stem passes through the actuator.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the stem protector has no impact on the probability of a loss of coolant accident. There is no scenario where the presence of the stem protector could lead to a leak from the Reactor Coolant Pressure Boundary. The SI8811 valves open to provide a source of water from the ECCS sumps to the suction of the RH pumps once the RWST level falls to the switchover set point following a LOCA. This allows the core to remain covered and heat to be removed. The addition of the stem protectors will not prevent the SI8811 valves from opening and performing their design function and does not add new failure modes. The actions assumed in the accident analysis will occur and there will be no increase in consequences. The design of the protector will ensure there is no interference with the valve stem or the actuator internals. The protector will prevent foreign materials from entering the actuator through the gap around the stem. Therefore, there will be no increase in the probability of equipment malfunction. Therefore the consequences of a malfunction of these components will not change. The SI8811 valves will continue to isolate the ECCS sumps from the ECCS system when the sumps are not required to provided a suction source to the RH pumps. The valves will continue to have the capability to open during LOCA situations when the sumps must be aligned to the RH pumps suction.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the new stem protector will not change the function of the Safety Injection System or Valves 1/2SI8811A/B. New interfaces with other systems are not created. There are no new failure modes and existing failure modes remain the same. Therefore, there will be no accident or malfunction of a type different from those evaluated in the UFSAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which Technical Specifications are based.

6G-98-0042
DCP 9700177 (Unit 1) & 9700178 (Unit 2)

DESCRIPTION:

The purpose of this Design Change was to install a relief valve manifold, upstream of each 'N' 4-way hydraulic valve to provide pressure relief of the MSIV hydraulic accumulators. The relief valve assembly will automatically port the hydraulic fluid from the accumulator back to the hydraulic reservoir.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the assembly has been seismically and environmentally qualified and does not adversely affect MSIV operation or function. The modification does not affect the main steam line piping or any other accident initiating components or systems. The closure time of the valve will still be within the limits set forth by Tech. Specs. Therefore, the probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the possibility of the new relief valve sticking open is a possible malfunction. However, the manifold has an orifice installed to ensure that if the relief valve malfunctions, the flow will be limited to less then the makeup capability of the hydraulic pump. This will ensure sufficient hydraulic pressure is present in the accumulator to close the MSIV in the required time. Therefore, the modification does not affect the ability of the MSIVs to perform their design function. Therefore, the change does not create the possibility of an accident or malfunction of a different type from those previously evaluated.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which Tech. Specs. are based.

6G-98-0189
DCP 9700228 & 9700230

DESCRIPTION:

The purpose of this Design Change was to remove the internals from Check Valves 1SX116A, 1SX116B, 2SX116A, and 2SX116B, replace the existing carbon steel Check Valves 1SX194 and 2SX194 with stainless steel check valves, and replace snubber 2SX62010S with a strut.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the changes did not, in themselves, factor into any initiating event for UFSAR accidents. The SX System has not been rerouted and is able to provide sufficient cooling water to components as designed. The SX line from the SX pump cubicle coolers is Category 1 pipe. The stainless steel replacement valves are designed to perform the same function as the original carbon steel valves. Support 2SX62010S seismically supports line 2SXB1AA-3. The Essential Service Water (SX) and Auxiliary Feedwater (AF) systems will respond as designed for each accident and for applicable events which may cause the accident.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the replacement stainless steel check valves have the same failure modes as the existing check valves. The replacement strut meets the design requirements to support line 2SXB1AA-3. The removal of the SX cubicle cooler return line check valve internals will not result in system backflow because no reverse flow path exists due to the system lineup and/or closure of the SX pump discharge check valve. Therefore, the SX and AF Systems will not be operated in a different manner such that the possibility of a new or different kind of accident or malfunction will be created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which Technical Specifications are based.

6G-98-0308
DCP 9700231

DESCRIPTION:

The purpose of this Design Change was to install a Service Air access point on the north end and an Instrument Air access point on the south end of the Circulating Water Pump House (CWPH). Submersible pumps are installed and hard piped from the Circulating Water Flume to a rinse basin (sink) at each of the two Sulfuric Acid Tanks. The sulfuric acid from the tanks is used to control the pH of the Circulating Water system.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the added service and Instrument Air drop points provide air to pressurize the acid tank on the delivery truck to facilitate the acid transfer. The air is supplied from the same air header as the temporary supply lines it replaced. The acid lines and the basic process used to transfer the acid from the delivery truck to the storage tank are not changed. A blow-off (relief) valve is installed on the delivery truck's acid tank to prevent its overpressurization. Water from the flume is still used to flush the residual sulfuric acid from the transfer hoses and to rinse equipment in the existing rinse basin, which remains piped to send the diluted acid to the flume. Equipment important to safety is not affected.

   A sulfuric acid spill would still require a Hazardous Material and Chemical Spill Response as described in Byron Station Procedures and in accordance with CERCLA/SUPERFUND Regulations.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the air line tie-ins and submersible pumps perform functions to facilitate transferring sulfuric acid from delivery trucks to tanks on site in order to control the pH of the water in the Circulating Water system at the flume. They do not increase the possibility of an accident or create any new accidents with respect to the UFSAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which Technical Specifications are based.

6G-98-0230
DCP 9700233

DESCRIPTION:

The purpose of this Design Change was to relocate orifice flanges for Fire Pump 0FP03PA, to a point just upstream of where line 0FP257C discharges into the Circulating Water Pump House (CWPH) Forebay. For both station fire pumps (0FP03PA and 0FP03PB), some of the elbows and piping for the recirculation line are increased in diameter from 6" to 8" and the recirculation line for pump 0FP03PB is straightened to remove two elbows. Orifice plates, with full pipe diameter holes, are inserted between each of the two orifice flanges to act as spacers between the flanges and maintain a pressure boundary.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the recirculation piping is designed in accordance with current piping design criteria to allow recirculation of water back to the Circulating Water Pump House Intake. The fire protection pumps and their ability to supply water to the fire sprinklers, hose stations, and foam systems, to mitigate fires are not affected. The Safety Category 1 Essential Service Water System's ability to supply fire protection water to the Containment Building, for fire mitigation is not affected. No flammable materials or ignition sources are added by this change.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the Fire Protection System will continue to be able to supply water to the plant in order to provide fire protection water. The Essential Service Water System will still be able to supply fire protection water to the Containment Building, as designed, and is not affected by this design change. The fail close recalculation flow control valve ensures the water will be diverted to areas for fire control when needed. The failure mode of the resized lines are unchanged.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which Technical Specifications are based.

6G-97-0194
DCP 9700313

DESCRIPTION:

The purpose of this Design Change was to add an isolation valve between each CO_2 odorizer and the main CO_2 header. Vent valves were also added for each of the three odorizers. The Isolation Valves are locked open during normal system alignment. The isolation valve, for each of the three odorizers, allows the odorizers to be isolated from the pressurized main header during puff testing of the CO_2 system. The vent valve, for each odorizer, is used to depressurize the CO_2 header following the puff testing.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the CO_2 system supplies fire extinguishing CO_2 to various safety related equipment rooms in the event of a fire in that room. The design change does not change the ability of the CO_2 system to deliver the CO_2 to the rooms, when needed, during normal plant alignment. The CO_2 system does not adversely affect plant operation or involve any Design Basis Accidents, including equipment that is important to safety.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the CO_2 system will continue to supply the wintergreen scent from the odorizer upon normal actuation of the suppression system. The odorizer Isolation Valves are locked open during normal system operation to prevent accidental isolation of the odorizers and to ensure odorizer actuation on emergency CO_2 actuation. During puff testing, the odorizer will be isolated to prevent an undesired actuation during the testing. The vent line allows the CO_2 header to be depressurized following puff testing, so the odorizer will not see system pressure and actuate. No new valves are added in the main CO_2 flow path that could potentially block the flow of CO_2 from the CO_2 storage tank to a protected room. The CO_2 Main Generator purge is not affected.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which Tech. Specs. are based.

6G-98-0122
DCP 9700319 (Unit 1) & DCP 9700320 (Unit 2)

DESCRIPTION:

The purpose of this Design Change was to remove the Boron Concentration Measuring System (BCMS) from the Chemical & Volume Control System (CVCS) system by disconnecting and plugging or capping the measuring system inlet and outlet lines. The BCMS System is not very accurate and, therefore, was only usable on an advisory basis. In the case of Byron station, it is not used at all. Boron Concentration samples are taken manually by the Chemistry Department, which are more accurate than the BCMS.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the CVCS will still be able to perform required safety functions, such as controlling RCS Boron Concentration and providing Safety Injection. The charging pumps will remain operational and the CV piping system pressure boundary will be intact following the installation of the tube plugs. Integrity of the primary system pressure boundary is ensured by installation of qualified tube plugs. Therefore, the probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because grab samples are obtained to determine actual Boron Concentration in the CVCS when needed. The disconnected lines will be plugged (and capped) in accordance with existing piping design tables to ensure system pressure boundary integrity. The CV system will still be able to regulate the concentration of boron in the RCS to control reactivity changes. The CV charging pumps will serve as high head safety injection pumps in the Emergency Core Cooling System. The remaining pressurized and abandoned lines are seismically qualified and supported by analysis. Revision of the EWCS design and operating pressures for lines 1CVG3C and 2CVG3C to agree with new pressures the lines are exposed to does not change the integrity of the lines.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which Tech. Specs. are based.

6G-98-0202
DCP 9700358 & 9700359

DESCRIPTION:

The purpose of this Design Change was to temporarily disable the essential service water (SX) low flow alarm when the DG is at standby or running below 280 rpm. The alarm was enabled 20 seconds after speed reached 280 rpm.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the DG will auto start on a LOOP as designed. Since this is a condition resulting from an accident, low flow of SX cooling water will not prevent the DG from starting and delivering power to the ESF buses as designed. This change will delay the low flow of cooling water alarm but does not eliminate it and will not trip the DG during accident conditions. Therefore the probability of the accident is not increased. The change will block the cooling water low flow alarm but will not prevent the DG from starting during accident conditions. The DG will provide power to the ESF buses to do its safety function and therefore the consequences of the accident resulting in an offsite release is not increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the change will not create the possibility of an accident or malfunction of a type different from those evaluated in the SAR. The 20 second delay is provided by a spare contact of an existing time delay relay that currently is used to inhibit the lube oil (engine lube oil and turbo lube oil) trips and alarms until the DG has reached 280 RPM . The low flow alarm does not provide any control or trip functions for the DG but is used for an abnormal condition alarm only. Disabling the alarm for 20 seconds after the engine starts will not have an adverse affect on the DG because the 20 second delay is negligible compared to the time the DG can be run without SX cooling water with no adverse effects. Operator action is still required and can still be accomplished prior to a malfunction of the DG.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the alarm does not provide any control function for the DG. The alarm is to alert the operators to an abnormal condition. The 20 sec. delay is added to allow flow to be established thereby eliminating a nuisance alarm. This is well within the thermal limit of the engine to operate without cooling water. The DG and SX cooling water valves will operate in the exact manner they presently do. Any abnormal flow conditions that become evident during the running of the DG will be alarmed after the DG has started running and the 20 second delay has elapsed.

6G-98-0129
DCP 9700386

DESCRIPTION:

The purpose of this Design Change was to connect the spare COM-5 Overcurrent Relays (OCR) in the Reactor Coolant Pump (RCP) 6.9KV switchgear (SWGR) cubicle to the redundant trip coil circuit and set the relays to the same settings as the primary circuit OCR. In addition, the existing CO-11 relay was replaced with the CO-8 OCR in order to improve the RCP motor locked rotor protection. This relay change results in better coordination or time delay relay function and helps avoid inadvertent tripping of the RCP while still providing the necessary protection for a locked rotor condition.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because this activity is strictly an enhancement to the RCP circuit breaker trip circuit for penetration protection. The change does not interface with the RCS fluid system of the physical components associated with the fluid system. Inadvertent actuation of the replaced relays or the added set of second relays does not increase the probability of inadvertent tripping different than the current configuration. This design change does not alter the functions originally designed, rather improves the original design. No additions or changes are made to components, which are credited to mitigate the consequences of an accident (i.e. loss of partial RCS flow). Therefore, there is no increase in the probability or consequence of any accident previously evaluated in the Safety Analysis Report.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report Is not created because the change connects two spare COM-5 relays located in the same cubicle to the second trip coil circuit which has a totally independent redundant power source. Necessary electrical separation between redundant divisions is maintained. The new OCRs are seismically mounted and qualified to the same requirements as the existing ones and the electrical connections are the same for both trip circuits. The failure modes for the change are the same as the current (i.e. partial loss of RCS flow) which has been analyzed and found acceptable in the SAR.

   Replacement of the CO-11 with the CO-8 relay was initiated to provide better RCP motor protection for degraded voltage conditions. The intent is unchanged, however, the overall level of RCP protection with this change is improved. Given the above, there is no possibility of an accident or malfunction of a different type other than those previously evaluated in the Safety Analysis Report.

6G-98-0129
DCP 9700386
(Cont'd.)

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this activity was evaluated against plant Technical Specifications and it was determined that there was no impact on any Technical Specifications.

6G-98-0158
DCP 9700388 (Unit 1) & 9700389 (Unit 2)

DESCRIPTION:

The purpose of this Design Change was to modify the existing ESFAS Phase A Containment Isolation testing circuitry for the CV letdown line containment isolation safety related valves 1/2CV8152 and 1/2CV8160 from "GO" testing (valve stroke closed) to "NO-GO" testing (position of the end device remains unchanged while the actuating safety circuit being tested via control power to assure circuit integrity). The solid state protection system cabinets, safeguards test cabinets, auxiliary safeguard relay cabinet, post accident hydrogen monitor Containment Isolation Valves, electrical penetration and main control board were affected to accomplish the necessary rewiring associated with this design change.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the ESFAS test circuit does not control plant parameters and thus is not the initiator of any accident or transient. The change affected the electrical circuit components in the control cabinets. No changes were made to the mechanical components that may affect the reactor pressure boundary. Only the method of circuit testing and interlocking is being changed from "GO" to "NO-GO". The relocation of valves 1PS230A/B ESFAS actuation from slave relay K607 to K605 did not change the function or operation of the valves as described in the SAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the ESFAS test circuit does not control plant parameters and thus is not the initiator of any accident or transient. Failure effects and inadvertent actuation of these components is no different then the previous circuit. This design change does not affect the operation of, or alter the functions of the valves as originally designed. All original design and performance criteria for the affected valves will be met. The relocation of valves 1PS230A/B ESFAS actuation from slave relay K607 to K605 did not change the function or operation of the valves to isolate the post LOCA hydrogen monitor during accident conditions.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the Technical Specification requirement to test the slave relay is not affected by the change. The method of testing is the same as described in the UFSAR. The relocation of valves 1PS230A/B ESFAS actuation from slave relay K607 to K605 did not change the function or operation of the valves as described in the Tech. Specs.

6G-97-0258
DCP 9700401

DESCRIPTION:

The purpose of this Design Change Package was to rewire the limit and torque switch valve travel interruption contacts per the guidance provided in NRC Notice IEN 92-18 on the Power Operated Relief Valve (PORV) Block Motor Operated (MOVs) (EPNs 1RY8000A/B). The modification arranged the connection to the LS/TS such that the valve circuit will be interrupted by LS/TS preventing mechanical damage to the MOV as a result of a fire in the MCR as postulated in the IEN for Unit 1.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the modification to the valve circuitry is to further enhance the reliability of the operation of the valves during a postulated Main Control Room Fire. The modification does not change any of the precursors assumed for any of the Design Basis Accidents. The probability of the Design Basis Accidents remains unchanged as a result of this modification. The rewiring of the LS/TS contacts does not change the function or operation of the valves and their circuitry.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the rewiring of the LS/TS switches does not create the possibility for a different accident. The rewiring is to increase the reliability of the function of the valve during a postulated Main Control Room Fire accident. The change is to address the recommendations provided by the NRC on IEN 92-18. The modification is to prevent a catastrophic mechanical failure of the valve if the valve spuriously actuates as a result of a Main Control Room fire that creates "smart hot shorts" in the conductors of the Main Control Room cables. The modification is to prevent a malfunction of a different type previously evaluated in the SAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the rewiring of the LS/TS for these valves does not affect any parameters upon which Technical Specifications are based.

6G-98-0108
DCP 9700402

DESCRIPTION:

The purpose of this Design Change Package was to rewire the limit and torque switch valve travel interruption contacts per the guidance provided in NRC Notice IEN 92-18 on the Power Operated Relief Valve (PORV) Block Motor Operated (MOVs) (EPNs 2RY8000A/B). The modification arranged the connection to the LS/TS such that the valve circuit will be interrupted by LS/TS preventing mechanical damage to the MOV as a result of a fire in the MCR as postulated in the IEN for Unit 2.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the modification to the valve circuitry is to further enhance the reliability of the operation of the valves during a postulated Main Control Room Fire. The modification does not change any of the precursors assumed for any of the Design Basis Accidents. The probability of the Design Basis Accidents remains unchanged as a result of this modification. The rewiring of the LS/TS contacts does not change the function or operation of the valves and their circuitry.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the rewiring of the LS/TS switches does not create the possibility for a different accident. The rewiring is to increase the reliability of the function of the valve during a postulated Main Control Room Fire accident. The change is to address the recommendations provided by the NRC on IEN 92-18. The modification is to prevent a catastrophic mechanical failure of the valve if the valve spuriously actuates as a result of a Main Control Room fire that creates "smart hot shorts" in the conductors of the Main Control Room cables. The modification is to prevent a malfunction of a different type previously evaluated in the SAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the rewiring of the LS/TS for these valves does not affect any parameters upon which Technical Specifications are based

6G-98-0178
DCP 9700431

DESCRIPTION:

The purpose of this Design Change was to replace eight rotor assemblies in Auxiliary Building Ventilation (VA) Exhaust fans 0VA02CA, CB, CC, and CD. Each fan receives one new first stage assembly, one new second stage assembly, and one new fan shaft. Each rotor assembly consists of 16 forged aluminum alloy fan blades and a split two piece hub. The original cast aluminum blades, single piece hubs, and shafts are removed from each fan because they are subject to cyclic fatigue failure caused by unfavorable fan inlet conditions. The new forged aluminum blades offer superior fatigue resistance over cast aluminum blades.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because changing VA exhaust fan blades, hubs, and shafts does not impact the probability of an accident or alter initial accident conditions. The VA System operation is not adversely impacted. The change does not impact plant environmental conditions or adversely affect the potential spread of radioactive contamination. System functions will be maintained, accident consequences are not increased as a result of this design change. The probability of a VA exhaust fan malfunction will be reduced because the potential for a fan blade failure is reduced by installing forged aluminum blades. Increasing the reliability of the VA exhaust fans will help to improve the overall reliability of the VA System and, therefore, will not adversely impact other systems or components served by the VA System. The fans remain seismically qualified and fan aerodynamic performance remains unchanged. (The failure consequences of VA exhaust fans with forged aluminum blades split two piece hubs, and new fan shafts is the same as for the original fan design.) Fan failure will continue to have the potential for impacting other Auxiliary Building safety-related equipment or for injuring personnel. Following a Loss of Coolant Accident, radioactive leaks from ECCS equipment is filtered through redundant non-accessible filter plenums, with high efficiency filters and charcoal adsorbers, prior to exhausting to the atmosphere. The operation of the Auxiliary Building Exhaust Filter System and the resultant effect on offsite dose are unchanged by the fan design change.

6G-98-0178
DCP 9700431
(Cont'd.)

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because this Auxiliary Building Ventilation System is unrelated to the sequence of events leading to the initiation of a Design Basis Accident. The temporary reduction of VA System airflow during design change installation will not cause Auxiliary Building maximum temperatures to exceed EQ zone limits or otherwise impact operability of VA supplied equipment. During installation, the VA System will continue to operate in accordance with approved station Operating Procedures. The fan design change does not change the direction of airflow in the Auxiliary Building. Since the VA System's accident mitigation and normal functions remain unchanged, the possibility of an accident or malfunction of a different type from those previously evaluated in the SAR is not created. The new fan blades were evaluated by the manufacturer to ensure that other fan components such as the bearings, motor, and appurtenances will not be adversely affected to create the probability of a different type of malfunction.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which Technical Specifications are based.

6G-98-0052
DCP 9700506 (UNIT 1) & DCP 9700507 (U2)

DESCRIPTION:

The purpose of this Design Change was to add the following components to each diesel generator (DG).

1. Vent valves to each turbocharger lube oil filter,

2. A pneumatic snubber, consisting of a short piece of tubing and an elbow, for the crankcase high pressure shutdown/control switch,

3. Spring washers for the bolts on the transition spool-piece between the turbocharger and the exhaust manifold, and

4. New supply air fittings for the air turning gear motor.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the DGs are active after a Loss of Offsite Power accident and are used to mitigate the accident. The pneumatic snubber and turning gear motor air supply connections are not active when the DGs are emergency started. The turbocharger lube oil filter vent valves will make maintenance easier. The spring washers will increase the DGs reliability during operations. The changes do not affect the function or manner of the DGs performing their functions. The changes enhance DG maintenance and reliability. Therefore, the probability and consequences of an evaluated accident or malfunction of equipment important to safety does not increase.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the activities do not adversely affect plant operation. They provide ease of maintenance and increased reliability of the DGs. No new failure modes are added by the activity. The function of the DGs and manner of performing that function is unaffected by the activity. Therefore, the changes do not adversely impact systems or functions so as to create the possibility of an accident or malfunction of a different type from those evaluated in the SAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which Technical Specifications are based.

6H-98-0346
DCP 9700511 & 9700512

DESCRIPTION:

The purpose of this Design Change was to add a relief valve to prevent piping and valves associated with containment penetrations PC-06 and PC-10 and the Chilled Water system inside containment. This prevents piping associated with these penetrations from being overpressurized during certain faulted condition events. This activity was performed in direct response to commitments made to NRC Generic Letter 96-06.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the relief valves on the chilled water inlet piping are installed to mitigate the potential consequences of a design basis loss of coolant accident or Main Steamline Break accident and would not initiate any new accident. In addition, given the fact that chilled water provides no safety function during accident conditions and that the relief path from the valves is inside containment, there is no potential increase in the probability or consequences of any accidents previously evaluated in the Safety Analysis Report.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the installed relief valves do not create any new failure modes or malfunctions different that previously evaluated in the Safety Analysis Report. The potential impact on internal containment flooding was analyzed and determined to be negligible. In addition, the potential failure of the valves would not represent a new accident since the valves relief inside of containment. The relief valves are installed to prevent a potential loss of containment integrity.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the addition of the relief valves does not adversely impact any Technical Specification systems. Containment integrity will be maintained via this installation and plant operation will remain within the basis of plant Technical Specifications.

6H-98-0370
DCP 9700513 & 9700514

DESCRIPTION:

The purpose of this Design Change was to add relief valves to the Reactor Building Equipment Drain piping associated with containment penetration PC-11 of both units to prevent potential overpressurization of the piping due to thermal expansion of trapped water following certain faulted condition events. This activity was performed in direct response to commitments made to NRC Generic Letter 96-06.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the relief valves installed on the Reactor Building Equipment Drain piping are installed to mitigate the potential consequences of a design basis loss of coolant accident or Main Steamline Break accident by relieving into containment in the event that overpressurization occurs in the piping after containment isolation following the accident. The installation does not initiate any accident or malfunction and given the fact that this system provides no safety function during accident conditions and that the relief path from the valve is inside containment, there is no potential increase in the probability or consequences of any accidents previously evaluated in the Safety Analysis Report.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the installed relief valves do not create any new failure modes or malfunctions different that previously evaluated in the Safety Analysis Report. The potential impact on internal containment flooding was analyzed and determined to be negligible. In addition, the potential failure of the valves would not represent a new accident since the valves relieve inside of containment. The relief valves are installed to prevent a potential loss of containment integrity following the accident.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the addition of the relief valves does not adversely impact any Technical Specification systems. Containment integrity will be maintained via this installation and plant operation will remain within the basis of plant Technical Specifications.

6H-99-0385
DCP 9700513 & 9700514

DESCRIPTION:

The purpose of this Design Change was to add a 3/4" x 1" thermal relief valve on the Reactor Coolant Drain Tank (RCDT) pump discharge piping between the PC-11 inside containment isolation valve 1/2RE1003 and the outside containment isolation valve 1/2RE9170. The discharge of the relief valve is routed back to the RCDT through the 2" accumulator tank drain line 1/2RE07AD-2". The design change is to address potential thermally induced overpressurization of isolated water-filled piping sections in containment as discussed in NRC Generic Letter 96-06.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the added relief valve does not affect the function of the RE system. The relief valve will be subjected to LLRT to verify its seat tightness for containment isolation. Further, the addition of the relief valve does not alter the function, but will increase the reliability of the Containment Isolation Valves/piping during accident condition.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the relief valve does not adversely impact the function of the RE system during normal or accident conditions. Plant operation is not changed and no new failure modes are introduced.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any Technical Specification parameters. Containment integrity continues to be maintained and plant operation will remain within the requirement of Tech. Specs.

6H-98-0347
DCP 9700517 & 9700518

DESCRIPTION:

The purpose of this Design Change was to add a relief valve to the Fire Protection water supply header piping (1/2FP04C-4") inside of Containment of both units to prevent potential overpressurization of the piping due to thermal expansion of trapped water following certain faulted condition events. This activity was performed in direct response to commitments made to NRC Generic Letter 96-06.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the relief valve installed on the fire protection header is installed to mitigate the potential consequences of a design basis loss of coolant accident or Main Steamline Break accident by relieving into containment in the event that overpressurization occurs in the fire protection supply header after containment isolation following the accident. This installation would not initiate any accident and given the fact that fire protection provides no safety function during accident conditions and that the relief path from the valve is inside containment, there is no potential increase in the probability or consequences of any accidents previously evaluated in the Safety Analysis Report.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the installed relief valve does not create any new failure modes or malfunctions different that previously evaluated in the Safety Analysis Report. The potential impact on internal containment flooding was analyzed and determined to be negligible. In addition, the potential failure of the valve would not represent a new accident since the valve relieves inside of containment. The relief valve is installed to prevent a potential loss of containment integrity following the accident.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the addition of the relief valve does not adversely impact any Technical Specification systems. Containment integrity will be maintained via this installation and plant operation will remain within the basis of plant Technical Specifications.

6H-98-0369
DCP 9700519 & 9700520

DESCRIPTION:

The purpose of this Design Change was to add a 1/2" bypass line around valves 1/2RY8030 in the Primary Water to Containment supply piping. The bypass line includes a spring loaded check valve to prevent potential overpressurization of isolated piping due to thermal expansion of trapped water following certain faulted condition events. The check valve will allow water to flow to the Pressurizer Relief Tanks (PRTs) when the design pressure (200 psig) is obtained. This activity was performed in direct response to commitments made to NRC Generic Letter 96-06.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the bypass line and check valve installed on the RY piping associated with containment penetration PC-44 is installed to mitigate the potential consequences of a design basis loss of coolant accident or Main Steamline Break accident by relieving to the PRT in the event that overpressurization occurs in the piping after containment isolation following the accident. This installation does not initiate any accident and given the fact that this system provides no safety function during accident conditions and that the relief path from the installation is not outside containment, there is no potential increase in the probability or consequences of any accidents previously evaluated in the Safety Analysis Report.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the installed bypass line and spring loaded check valve do not create any new failure modes or malfunctions different that previously evaluated in the Safety Analysis Report. There was no potential impact on any analyzed accidents. In addition, the potential failure of the valve would not represent a new accident or failure mode since the bypass line would relieve to the PRT. This modification was installed to prevent a potential loss of containment integrity following the accident.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the addition of the bypass line and spring loaded check valve does not adversely impact any Technical Specification systems. Containment integrity will be maintained via this installation and plant operation will remain within the basis of plant Technical Specifications.

6H-98-0368
DCP 9700521 & 9700522

DESCRIPTION:

The purpose of this Design Change was to add a ¾" x 1" thermal relief valve on the containment floor drain (RF) sump pump discharge piping between the PC-47 inside containment isolation valve 1/2RF026 and the outside containment isolation valve, 1/2RF027. The discharge of the relief valve is routed to the Reactor Building Floor Drain System. The design change is to address potential thermally induced overpressurization of isolated water-filled piping sections in containment as discussed in NRC Generic Letter 96-06.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the added relief valve does not affect the function of the RF system. The relief valve will be subjected to LLRT to verify its seat tightness for containment isolation. Further, the addition of the relief valve does not alter the function, but will increase the reliability of the Containment Isolation Valves/piping during accident condition.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the relief valve does not adversely impact the function of the RF system during normal or accident conditions. Plant operation is not changed and no new failure modes are introduced.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any Technical Specification parameters. Containment integrity continues to be maintained and plant operation will remain within the requirement of Tech. Specs.

6H-99-0386
DCP 9700521 & 9700522

DESCRIPTION:

The purpose of the Design Change was to add relief valves to the Reactor Coolant Building Floor Drain piping associated with containment penetration PC-47 of both units to prevent potential overpressurization of the piping due to thermal expansion of trapped water following certain faulted condition events. This activity was performed in direct response to commitments made to NRC Generic Letter 96-06.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the relief valves installed on the Reactor Coolant Building Floor Drain piping are installed to mitigate the potential consequences of a design basis loss of coolant accident or Main Steamline Break accident by relieving into containment in the event that overpressurization occurs in the piping after containment isolation following the accident. The installation does not initiate any accident or malfunction. The new relief valve is classified as a containment isolation valve and is therefore subject to Local Leak Rate Testing (LLRT). The configuration ensures that potential containment leakage is maintained below the LLRT limit. There is no potential increase in the probability or consequences of any accidents previously evaluated in the Safety Analysis Report.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the installed relief valves do not create any new failure modes or malfunctions different that previously evaluated in the Safety Analysis Report. The potential impact on the safety analysis was analyzed and determined acceptable. In addition, the potential failure of the valves would not represent a new accident. The relief valves are installed to prevent a potential loss of containment integrity following the accident.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the addition of the relief valves does not adversely impact any Technical Specification systems. Containment integrity will be maintained via this installation and plant operation will remain within the basis of plant Technical Specifications.

6G-98-0263
DCP 9700559

DESCRIPTION:

The purpose of this Design Change was to modify the 2AF005 A-H valves in the AFW system for Byron Unit 2. The valves were modified with replacement trim and actuator components so that the open position of the valves will be limited to provide a maximum combined flow to the faulted Steam Generator of 464 GPM during a postulated Steam Generator Tube Rupture Event and to ensure all other minimum and maximum flow limits are maintained. This maximum flow of 464 GPM is credited in the accident analyses as specified in the B-B SGTR Analysis. The full open position of the valve will be limited by replacing actuator components to reduce the effective full open stroke from 2 inches to approximately 1.5 inches and will be "fine tuned" by adjusting the valve plug stem connection to the valve actuator. In addition, the existing valve trim components will be replaced to improve the valve's flow characteristics and facilitate setting the maximum flow limit for the valve.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because

   STEAM SYSTEM PIPING FAILURE
   The probability of a Steam System Piping Failure is not increased by the modification to the 2AF005A-H valves. The modification involves replacing the valve trim components for improved flow control and limiting the full open position of the valve by adjusting the valve stem. The modification has no impact on the probability of a steam system piping failure. This probability is defined by the main steam system piping and piping components whose integrity is not affected by this modification.

   In the event of a Steam System Piping Failure, the required AFW flows to the intact and faulted Steam Generators will still be met as specified in NFM: Calculation PSA-B-97-18. This change does not affect the parameters listed in UFSAR Section 15.1.5.3, which are used to assess the radiological consequences of a MSLB. The modification to the 2AF005 A-H valves will not result in an increase to the consequences of a Steam System Piping Failure.

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(Cont'd.)

LOSS OF FEEDWATER

The probability of a Loss of Normal Feedwater is not increased by the modification to the 2AF005A-H valves. Section 15.2.7.1 of the UFSAR describes a Loss of Normal Feedwater resulting from either a Main Feedwater System pump failures, valve malfunctions or loss of AC power. This modification involves limiting the maximum flow through the 2AF005 A-H valves by installing new trim components and adjusting the travel of the valve stem. The modification has no impact on the above mentioned initiators of a Loss of Normal Feedwater. In addition, the AFW system is designed to be in a standby status during normal operation of the main feed system. AFW operation during Loss of Normal Feedwater bounds that for Loss of Non-Emergency AC Power to the Plant Auxiliaries and Inadvertent Operation of ECCS During Power Operation.

The modification to the 2AF005 A-H valves did not result in an increase to the consequences of a Loss of Normal Feedwater. The required minimum AFW flows to all four Steam Generators will be maintained as specified in NFM: Calculation PSA-B-97-18. The radiological consequences of the above accidents are bounded by the Main Steam Line Break and this modification has no impact on the potential release paths nor any impact of systems credited for accident mitigation, therefore, there is no potential increase in the consequences of the accidents.

FEEDWATER LINE BREAK ACCIDENT

Section 15.2.8.1 of the UFSAR defines a major feedwater line break as a break in a feedwater line large enough to prevent the addition of sufficient feedwater to the Steam Generators to maintain shell side fluid inventory. The probability of a Feedwater System Pipe Break is a function of the accident initiators. The probability is not increased by the modification to the 2AF005A-H valves. The modification involves replacing the valve trim components for improved flow control and limiting the full open position of the valve by adjusting the valve stem. The modification has no impact on the pressure boundary function of the main and auxiliary feedwater systems and will not affect the Feedwater System Pipe Break accident initiators.

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(Cont'd.)

The modification to the 2AF005A-H valves did not result in an increase to the consequences of a Feedwater System Pipe Break. The required minimum AFW flows to the three unaffected Steam Generators will be maintained for decay heat removal as specified in NFM Calculation PSA-B-97-18. The feedwater line break with the most significant consequences would be one that occurred inside the containment between a Steam Generator and the feedwater check valve. In this case, the contents of the Steam Generator would be released to the Containment. Since no fuel failures are postulated, the radioactivity released would be less than that for the Steam Line Break. This modification has no impact on the potential release paths nor any impact on systems credited for accident mitigation, therefore, there is no potential increase in the consequences of the accidents, and the event is still bounded by the Steam Line Break event.

STEAM GENERATOR TUBE RUPTURE
The probability of a Steam Generator Tube Rupture is not increased by the modification to the 2AF005A-H valves. The modification involves replacing the valve trim components for improved flow control and limiting the full open position of the valve by adjusting the valve stem. The SGTR accident assumes the complete severance of a single Steam Generator tube. The modification will have no impact on Steam Generator tubing.

The modification to the 2AF005A-H valves will not result in an increase to the consequences of a Steam Generator Tube Rupture. These valves are designed to throttle flow but will fail in the open position upon loss of electrical power and/or loss of Instrument Air. This modification will limit the maximum AFW flow to the faulted Steam Generator to provide additional margin in preventing an overfill condition from occurring. This modification will limit AFW flow during this event to within the limits credited in the accident analyses. This modification has no impact on the systems or components used to mitigate the dose consequences of the SGTR, since the minimum required flows for the SGTR offsite dose calculation, as specified in NFM: Calculation PSA-B-97-18, are still met. Therefore, the consequences evaluated for the SGTR event are not increased.

LOSS OF COOLANT ACCIDENT
The probability of a Loss of Coolant Accident is not increased by the modification to the 2AF005A-H valves. The modification involves replacing the valve trim components for improved flow control and limiting the full open position of the valve by adjusting the valve stem. The modification has no impact on the pressure boundary function of the Reactor Coolant System and will not affect the accident initiators for a Loss of Coolant accident.

6G-98-0263
DCP 9700559
(Cont'd.)

The required minimum AFW flows to the Steam Generators will be maintained. Therefore, heat removal from the Steam Generators is not adversely affected. The systems and components assumed to mitigate the radiological consequences are not affected by this modification. Therefore, the consequences of the LOCA are not affected by this modification.

The 2AF005A-H valves will be modified with new trim components and throttled to a maximum flow setting. The valve modification involves the use of safety-related components and will be seismically qualified. These valves are designed to fail in the open position in the event of loss of electrical power and/or loss of Instrument Air. The modification will limit this failed open position in order to satisfy the current maximum AFW flow limits. Modification of the 2AF005A-H valves will not increase the probability of malfunction of components important to safety.

Modification of the 2AF005A-H valves will not affect the valves capability to perform the design function. The valve will continue to fail open on a loss of power or air such that the failure of the supporting systems will not affect the fail position of the valve. The components have been reviewed to ensure they do not affect the seismic qualification of the valve nor impact the piping system adversely. The consequences of the valve failing open have been reduced as a result of limiting the valve stroke to limit the maximum flow. This modification will increase the margin to overfill following a SGTR Event. This modification therefore, will not increase the consequences of a malfunction of equipment important to safety.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because there are no new or different events created as a result of the modification to the 2AF005A-H valves. The modification to the valves will ensure all maximum and minimum AFW flow requirements will be met. The designed failure mode of the valve is unaffected. The replacement trim is comprised of separate discs which are welded together and then captured, compressed and torqued within the valve body so as to preclude the possibility of the trim components and/or the plug components from becoming loose parts so as not to increase the likelihood of failure. No new credible accidents are created as a result of this modification. Based on the above, there is not the possibility of an accident or malfunction of a type different from those evaluated in the SAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which Technical Specifications are based.

6G-98-0047
DCP 9700560

DESCRIPTION:

The purpose of this Design Change was to install a removable screen in front of the rollup door (i.e., Door Number SD196) of the River Screen House (RSH). The new screen will keep wind blown debris and insects from entering the RSH when the rollup door is open. The rollup door is maintained open during warm weather to increase natural ventilation and minimize the amount of time the supply air fans are run, thereby, lowering noise emissions from the RSH.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because during the summer, installation of the screen and opening the rollup door will improve natural circulation within the RSH. The additional natural circulation will lower the temperature inside the RSH and minimize the operation of the supply air fans. For postulated failures of the supply air fans or loss of power during the summer, the open rollup door will improve circulation and the maximum temperature within the RSH will be less than previously evaluated (i.e., 115° F).

   When the outside air temperature drops below 60° F, administrative controls will close the rollup doors and the ventilation system will function as originally designed. The normal temperatures in the RSH will remain between 65° F and 104° F. For a postulated loss of heating event, the minimum calculated RSH temperature should be unchanged because the rollup door will be manually closed when the outside air temperature drops below 60° F. In the unlikely event the rollup door was not closed when the air temperature drops below 60° F, an alarm in the Main Control Room will annunciate at 50° F and operator actions will be taken to close the door and restore the room temperature or start the Essential Service Water (ESW) Pumps prior to the temperature in the RSH dropping below 40° F.

   The RSH structure is not designed to protect the equipment from tornadoes, therefore keeping the door open during the summer does not change the risk of equipment damage due to tornado generated missiles.

   The exterior walls of the RSH and the rollup door do not carry a fire rating and are not credited as a fire barrier. Therefore, the consequences of a postulated fire at the RSH are not changed.

6G-98-0047
DCP 9700560
(Cont'd.)

The screen outer frame will be seismically anchored to the RSH structure. During a seismic event the screen frame will remain in place. The new screen components (i.e., screen panels, screen door, and inner frame) are not seismically qualified. Failure of these components would most likely be buckling or distortion of the screens. However, in the event where the screens totally collapse into the RSH, there are no safety related components located near the door, which could be affected.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because all conceived failure mechanisms are addressed above. Therefore, the possibility of an accident or malfunction not previously evaluated does not exist.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because no Technical Specifications were impacted by the changes.

6G-98-0088, Rev. 1
DCP 9700568 & 9700569

DESCRIPTION:

The purpose of this Design Change was to remove four Process Radiation (PR) skids (1PR51J, 2PR51J, 1PR52J, 2PR52J) from the Containment Building Equipment and Emergency Hatches for both Byron Unit 1 and Byron Unit 2 (two skids per unit). In addition, removed with both PR skids for each unit are six manual Containment Isolation Valves (PR033A,B,C,D, PR002E,F) and two containment isolation Check Valves (PR002G,H). Two outlet test connection Isolation Valves (PR072 and PR073), used as connection points for performing leak rate tests on two of the Containment Isolation Valves, were also removed for the two PR skids per unit. Following removal of the PR equipment, each remaining open pipe end is sealed at one end with a welded pipe cap. A total of eight pipe caps are welded to eight open pipes for the four containment hatches (two per unit). In total, for Units 1 and 2, 12 manual Containment Isolation Valves, 4 containment isolation check valves, 4 outlet test connection Isolation Valves and 4 PR sample skids were removed. The electrical supply cables for the PR skids were pulled back and terminated at a nearby electrical box. The pipe caps were installed to meet the requirement of ASME Section 111, Subsection NE.

In addition to removals of the PR skids, a detail is provided in the Design Change Package to allow temporary removal of a handrail, located at the Equipment Hatch Personnel Door just inside the Unit 1 Containment, to allow for movement of equipment into and out of the containment. The detail allows the handrail to be easily removed and easily reinstalled as needed. The handrail installation is optional and only done if needed. An existing Unit 1 gallery steel support is also trimmed to prevent contact and damage to a cable pan. Notes for repair of the damaged cable pan are also provided.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the removal of the containment isolation check valves, removal of the PR sampling skids, change to the handrails and gallery steel support, and the cable pan repair do not affect or cause an increase in the probability or consequences of an accident or malfunction of equipment important to safety. A failure caused by the activity is not a precursor to any of the postulated accidents listed in question 10. The Containment Isolation Valves, PR skids, containment personnel hatches, and handrail operate separately from the steam (Main Steam) and feedwater system piping, ECCS, CVCS, RCS, nuclear fuel, rod cluster control assemblies, and fuel handling equipment, and have no impact on their operation or probability of an accident or transient.

6G-98-0088, Rev. 1
DCP 9700568 & 9700569
(Cont'd.)

Following an accident or transient that releases radioactivity into and/or increases the pressure in the Containment Building as listed on question 10, the welded pipe caps will provide improved containment integrity, compared to the active and passive Containment Isolation Valves they replace, by preventing leakage from the Containment Building at the modified PR piping points by providing a solid barrier in conformance with ASME Section 111, subsection NE. The remaining lines are seismically analyzed. The removed PR51J and 52J skids are not required for accident mitigation or post accident sampling and are non-safety related. The removable handrail and trimmed gallery steel support are seismically supported and do not affect the consequences of an accident or transient. A damaged cable pan that carries non-safety-related cables is repaired and zinc is restored, as needed, without increasing analyzed zinc levels in containment.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because removal of the PR skids will have no affect on post accident sampling of the containment atmosphere because this is done by other detectors such as components for the CASP (Containment Air Sampling Panel). In the event of an accident where radioactive material is released into containment and the Containment Building is pressurized due to an accident, the sealed process radiation lines will provide a containment pressure isolation boundary to prevent the release of radioactive material from the containment through the modified PR piping which is designed to the requirements of ASME Section 111, Ssubsection NE. Replacing the Containment Isolation Valves with welded pipe caps will eliminate the potential for post-accident leakage through the hatch penetrations in the airlock.

   The trimmed gallery support was analyzed and will seismically and structurally support the associated gallery system. With the gallery support trimmed, it will not collide with cable tray pans and damage them when the equipment hatch is removed. Making an existing handrail in containment removable is acceptable because it is seismically installed and will not fall out and damage other equipment, including equipment important to safety. It is stored in a safe configuration when removed. Work on the cable pan repairs existing damage. Therefore, this change does not create the possibility of a different type of accident or malfunction than any previously evaluated.

3. The margin of Safety, as defined in the basis for any Technical Specification, is not reduced because the containment isolation for the penetrations through the containment hatches is provided by welded caps which meets ASME III, Subsection NE criteria. These caps provided better containment isolation than the passive and active valves they replace. The ability to close the airlocks or other Containment Isolation Valves is not affected. The margin of safety, as defined in the Bases for any Technical Specification, is not reduced because the removal of the subject skids and the implementation of the activity does not affect any parameters upon which Technical Specifications are based.

6G-98-0062
DCP 9700712

DESCRIPTION:

The purpose of this Design Change was to install a permanent Instrument Air supply to the Byron Unit 1 Refuel Machine. The old air supply for the Refuel Machine on Unit 1 was a temporary connection to station air. Unit 2's design was walked down, reviewed, and found to be a permanent Instrument Air supply. This exempt change revised Unit 1's installation to be of similar design as Unit 2. The air supply revisions incorporate the vendor's (Raytheon) recommendations as documented on vendor drawing details. The activity will improve the reliability of the FHS pneumatic gripper by supplying it with Instrument Air rather than Station Air.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because a dropped Fuel Assembly accident is defined as the dropping of a spent Fuel Assembly onto the core during refueling which results in the rupture of the cladding of fuel rods. This DCP changed the air source for the gripper on the Refueling Machine. However, the Refueling Machine function, structure and Fuel Assembly gripping and lifting mechanisms will not be altered. This change enhanced the reliability of the gripper. Therefore, the probability of a fuel handling accident in containment or the malfunction of equipment important to safety is not increased. Since these changes will not alter the purge or radiation monitoring systems designed to mitigate the consequences of the accident as analyzed in the UFSAR, the accident consequences will not be increased and the UFSAR analysis remains valid and bounding for the changes. The Refuel Machine and its components are not required to affect, support, or maintain safe shutdown of the reactor.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the pneumatic air supply for the refuel machine is used to actuate the Fuel Assembly gripper. The air cylinders actuating the gripper mechanism are equipped with backup springs, which close the gripper in the event of loss of air to the cylinder and contains a spring-actuated mechanical lock, which prevents the gripper from opening unless the gripper is under a compressive load. Air operated valves are equipped with safety locking rings to prevent inadvertent actuation. The pneumatic air supply does not initiate any accidents, and the changes do not affect the initiators of any accidents. This change does not affect any other equipment or systems used to mitigate any accident. The function of the FHS has not changed. Therefore, there is no possibility of an accident or malfunction of a type different form those evaluated in the SAR.

6G-98-0062
DCP 9700712
(Cont'd.)

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which Technical Specifications are based.

6G-99-0175
DCP 9700729 & 9700730

DESCRIPTION:

The purpose of this Design Change was to replace the ultrasonic flow indicator on the Chemical and Volume Control miniflow line with an orifice plate differential pressure type flow indicator.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the modification did not increase the probability of operator action, operator error, test sequence error or a malfunction or mechanical failure of the CV or other plant systems. The replacement mechanical flow indicator has no electrical connections or manual/automatic control functions associated with it. The change to the flow indicator did not reduce the capability of the CV system to operate and mitigate accidents.

   The miniflow lines are designed to meet ASME Section III Code requirements and analyzed for deadweights, OBE, and SSE Seismic Events. The probability of flow blockage in the CV miniflow line is unchanged.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the change did not adversely affect the operation of the CV or any other system. The possibility of a flow blockage due to debris in the line is not an increased concern because the new orifice hole is larger than the holes in the break down orifices located upstream of the flow measuring orifice. No system controls are affected.

   The miniflow lines are still designed to meet ASME Section III requirements so the possibility of miniflow failure or failure to unchanged portions of CV piping is not changed. The change from an ultrasonic flow meter to a dP flowmeter does not create any new failure modes. The mechanical gauge provides flow data for insurance of CV pump performance.

6G-99-0175
DCP 9700729 & 9700730
(Cont'd.)

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the added piping configuration for the mechanical flow indicator is designed in accordance with the appropriate Asme Code and analyzed for deadweight, OBE, and SSE Seismic Events. The ability of the Charging Pumps to operate and to supply borated water to the RCS, during normal operation, unit shutdown, and following and accident is not affected due to modification of the miniflow line flow indicator because the main Charging Pump flow path and pump operation are not affected. The miniflow indicator provides local indication only and does not cause automatic actuations. The manual seal injection throttle valves are not impacted by the miniflow flow indication modification.

6G-98-0293
DCP 9700780

DESCRIPTION:

The purpose of this Design Change was to add permanent lead shielding to a portion of line 1RH02AA-8" to reduce radiation dose rates in the robotic repair and Auxiliary Building Equipment Drains (WE) sump area by approximately 9 Rem per year. The lead shielding is in the form of coated lead sheets as supplied by Advanced Shielding Technology. The half-round lead is coated by a proprietary material designated "Rad Klad 340". A maximum of 27 pounds of lead per foot was added to the 8" line. The lead sheets are removable to allow access for inspection, maintenance or replacement.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because line 1RH02AA8" and its supports remain qualified for all loading conditions to ensure the RH system pressure boundary will remain intact. In the unlikely event of leakage, the leak detection sumps will function to limit the severity of flooding. Piping failures due to stress corrosion cracking are no more probable because the lead shielding material is suitable for this application and because the shielding will be applied over the existing pipe insulation. The probability of damage to other safety related equipment by loose lead sheets was not increased since the lead was secured to the seismically qualified pipe by stainless steel banding that is loaded well below its load carrying capacity. Therefore, there is no increase in the probability of accidents or malfunctions. There will be no change in consequences of accidents or malfunctions since line 1RH02AA8" will continue to perform its function. RH system performance will not be degraded by the presence of the lead shielding.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because there are no failure modes of a type different from those evaluated in the SAR. The design of the lead shielding installation includes requirements to ensure that the shielding is well supported and will not become loose. The shielding will not lead to piping failures due to the weight increase or through stress corrosion cracking. There are no functional changes to line 1RH02AA8" or the RH system. There are no new interfaces between the RH system and other systems. Additional equipment failures are not expected due to installation of the design change. Therefore, there are no accidents or malfunctions different from those previously evaluated.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which Technical Specifications are based.

6G-98-0006
DCP 9700790 & 9700791

DESCRIPTION:

The purpose of this Design Change was to replace the existing reactor coolant valves (i.e., Kerotest 3/4" Series 1500 Y-type Globe valves) with similar valves made by a different manufacturer (i.e., Anchor/Darling 3/4" - 1878 Globe valves). These valves are atmospheric vents for the Reactor Coolant System and the RCS Loop Bypass lines.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the activity did not increase the probability or the consequences of an accident or malfunction of equipment important to safety. The piping has been analyzed for the slight difference in weight and the center of gravity location for the new valves and found to be acceptable. The replacement valves were to be more reliable than the existing valves. The function of the valves is to maintain RCS pressure in that the valves are part of the primary coolant system boundary. Failure of the 3/4" valves is bounded by current LOCA analyses for breaks of 2", 3" and 4". These valves are normally isolated and blind flanged during power operation.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the activity did not introduce new failure modes or alter the consequences of existing failures. The replacement valves have the same fit and function as the originally supplied components with an insignificant change in form. Pipe stress analyses have been reviewed and the activity has no impact on the current analyses.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not impact any parameters upon which Technical Specifications are based.

6G-98-0307
DCP 9700811

DESCRIPTION:

The purpose of this Design Change was to add a Pressure Indicator and an isolation valve to the Process Sampling tubing downstream from each of the SX to PS Isolation Valves. These valves, 0SX173A and 0SX173B, are on the SX return headers and are accessible in the Turbine Building base mat near the Feedwater Heater Drain Coolers on each unit. The new Pressure Indicators and Isolation Valves were installed in the non-safety PS tubing.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the change did not affect the initiation of any accident evaluated in the UFSAR.

   The affected portion of the PS system does not function to mitigate the consequences of an accident. The change did not affect equipment important to safety and did not create or modify any system interface. The ability of the safety related SX System to perform its safety function was not affected by the design change. Failure of the affected portion of the PS system could not lead to an accident and could not affect any safety related equipment or equipment important to safety.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the new components were designed to the same conditions as the affected portion of the PS System. No accidents were affected by the design change. No equipment important to safety was adversely affected and no system interactions were added or revised.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which Technical Specifications are based.

6G-98-0098
DCP 9800019

DESCRIPTION:

The purpose of this Design Change was to reroute the Demineralized Water (WM) supply to the Fuel Transfer System console 1FH02J. The previous embedded WM supply to the console was capped and abandoned. A new hose connection off of line 1WM86A and a cover plate detail for the floor opening adjacent to the console was provided. WM line 1WM62A is embedded in concrete. During construction activities, the embedded portion of the pipe was damaged. It was decided to reroute the WM line above the concrete. An additional hose connection was installed at the request of maintenance. This new connection will be used during decontamination of the refuel cavity and the fuel transfer canal.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the change rerouted non-safety related WM piping previously embedded in concrete. The affected line provides water for the fuel transfer system control console reservoir. The piping and supports are designed and installed to meet all design requirements. Failure or leakage from this piping is no different then failure or leakage of the previously installed piping. No accident is affected by the WM system in Containment during Modes 5 and 6. During other operating modes the line is isolated. Also this line does not mitigate the consequences of any accident. Therefore, the probability and consequences of an accident or malfunction of equipment do not increase.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the change did not adversely affect plant operations. No new failure modes were added by the activity. The function of 1WM62A is unaffected. Therefore, these changes did not adversely impact systems or functions so as to create the possibility of an accident or malfunction of a type different from those evaluated in the SAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which Technical Specifications are based.

6H-98-0366
DCP 9800050

DESCRIPTION:

The purpose of this Design Change was to install fillet welds to secure the Byron Unit 1 Upper Reactor Head Rod Cluster Control Assembly (RCCA) Thermal Sleeve Guides to the thermal sleeve at any penetration, as necessary. The fillet weld detail is intended as a generic alternate detail that may be used in lieu of the original plug weld design, where necessary.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because this activity does not change the function of the RCS thermal sleeve guide to ensure proper RCCA alignment during reactor head installation. This activity is provided to further secure the guide in place and reduce the potential for the guide to become loose or detached. Control rod function or operation is not affected by this change. No boundary to radioactive release is affected by this change. Therefore, the probability or consequences of a reactivity or power distribution anomaly or any other analyzed accident are not increased by this activity.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the ability of the thermal sleeve guides to ensure proper alignment nor the function of the control rods is adversely affected by this activity. No other safety related equipment is impacted, and therefore, no new failure modes, equipment malfunction, or new accidents are created by this activity.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this activity does not adversely impact any Technical Specification systems. Control rod operability and operation are maintained for all plant modes by this activity. Therefore, plant operation will remain within the basis of plant Technical Specifications.

6H-98-0365
DCP 9800057

DESCRIPTION:

The purpose of this Design Change was to install fillet welds to secure the Byron Unit 2 Upper Reactor Head Rod Cluster Control Assembly (RCCA) Thermal Sleeve Guides to the Thermal Sleeve at any penetration, as necessary. The fillet weld detail is intended as a generic alternate detail that may be used in lieu of the original plug weld design, where necessary.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because this activity does not change the function of the RCS thermal sleeve guide to ensure proper RCCA alignment during reactor head installation. This activity is provided to further secure the guide in place and reduce the potential for the guide to become loose or detached. Control rod function or operation is not affected by this change. No boundary to radioactive release is affected by this change. Therefore, the probability or consequences of a reactivity or power distribution anomaly or any other analyzed accident are not increased by this activity.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the ability of the Thermal Sleeve guides to ensure proper alignment nor the function of the control rods is adversely affected by this activity. No other safety related equipment is impacted, and therefore, no new failure modes, equipment malfunction, or new accidents are created by this activity.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this activity does not adversely impact any Technical Specification systems. Control rod operability and operation are maintained for all plant modes by this activity. Therefore, plant operation will remain within the basis of plant Technical Specifications.

6G-98-0131
DCP 9800096

DESCRIPTION:

The purpose of this Design Change was to install new labyrinth seals on the Auxiliary Building ventilation supply fan bearings. A shaft seal was also installed. Prior to the change, the VA Supply Fans were consuming a large quantity of oil. The labyrinth seals are expected to reduce the oil consumption. The new shaft seal prevents air from flowing through the bearing. Air flowing through the bearing causes a differential pressure inside the fan and causes the bearing to use more oil.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the VA Supply Fans cannot cause leakage in the RCS piping. Therefore, the probability of a LOCA will not increase. No new failure modes are added by the activity. The new labyrinth and shaft seals improve the reliability of the fans and do not increase the probability of malfunction. The supply fans are used to mitigate the consequences of the accident. The changes do not affect the post-accident operation of the supply fans or off-site dose. The supply fan air flow capacity is unaffected. Therefore, consequences of an accident or malfunction do not increase.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because this DCP does adversely affect plant operations. No new fan failure modes are created by the activity. The function of the supply fans and manner of performing that function are unaffected by the. Therefore, these changes do not adversely impact systems or functions so as to create the possibility of an accident or malfunction of a type different from those evaluated in the SAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which Technical Specifications are based.

6G-99-0176
DCP 9800122 & 9800123

DESCRIPTION:

The purpose of this Design Change was to replace the power supplies in the Westinghouse 7300 Process Instrumentation & Control (I & C) Cabinets from a primary/backup configuration to a load sharing configuration. Also to upgrade the power supply input and output signal wiring. The change allows the power supplies to operate at a lower current and designed voltage. It also allows the power supplies to be removed without disconnecting the wiring.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the Process I&C cabinets work in part with the Solid State Protection System (SSPS) to provide safe operation and control of primary and secondary plant systems during steady state and transient plant conditions. These cabinets provide the initiating signals to SSPS in order to mitigate the consequences of accident conditions discussed in UFSAR Chapter 15. These changes do not alter the protection and control system but improves the supply of power to the circuits. This change does not interface with any fluid systems, any change of valves, breach of piping, any work associated with ventilation system and any radiation detection and/or protection system equipment. Since no fuel damage and/or no RCS pressure boundary break/leakage to secondary fluids is postulated, the activity does not increase the consequences of an accident resulting in a release of an Off-Site Dose. The change will not add any new failure mode and will not affect transient or accident conditions. Therefore this modification does not increase the probability of occurrence or the consequence of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report but rather enhances the Process I&C Cabinet power supply performance.

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(Cont'd.)

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because this modification enhances the power supplies (PS) of the I&C cabinets. Minor alterations in the PS configuration does not alter the function, operation or interactions of the Process I&C Protection and Control System as described in the SAR but improves the supply of power to the circuits. Failure of power supplies in both the current shared auctioneered schemes is the same. There are no changes to the circuits. Only the PS transformer tap settings are changed on one power supply to be equal to the other. As in the existing arrangement, if one power supply fails the alternate one is capable of carrying the entire load and no circuits are affected. This change does not alter this scheme but rather improves it by balancing the load. No changes are being made to the existing Process I&C cabinets that were previously analyzed. Therefore the modification did not create the possibility of a different type accident or malfunction than any previously evaluated.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this change has no impact on the Technical Specification 3.3 (Instrumentation), its Bases and the associated surveillance.

6G-99-0046
DCP 9800140

DESCRIPTION:

The purpose of this Design Change was to replace the existing 23 Joseph Oat Spent Fuel Pool Storage Racks with 24 Holtec International Spent Fuel Pool Storage Racks. Due to design differences, 114 additional storage cells will be gained. The Holtec storage racks shall not provide for failed fuel canisters. Due to the degradation of the neutron poison (Boraflex) in the existing Joseph Oat storage racks, a soluble Boron Concentration of 2000 PPM is required to control criticality of the spent fuel during normal Spent Fuel Pool operations. The neutron poison (Boral) used in the Holtec International storage racks will not require soluble boron to control criticality. The result of this reduction in soluble boron will have no adverse effect on the Spent Fuel Pool. Soluble boron will still be required to control criticality in the unlikely event of a misplaced Fuel Assembly in the Spent Fuel Pool, however the required concentration will be 300 PPM. There is no adverse effect on the operation of the Spent Fuel Pool as a result of maintaining 300 PPM soluble boron.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because

   The probability of accidents were considered for:

   a). Spent Fuel Assembly dropped onto the Spent Fuel Pool floor,

   b). Spent Fuel Assembly dropped between racks,

   c). Spent Fuel Assembly dropped between a rack and the Spent Fuel Pool wall,

   d). Spent Fuel Assembly loaded contrary to placement restrictions,

   e). Spent Fuel Assembly dropped onto a rack,

   f). Spent fuel cask drop,

   g). Change in Spent Fuel Pool water temperature,

   h). Loss of Spent Fuel Pool cooling,

   i). Loss of Spent Fuel Pool water level, and

   j). Water quality of Spent Fuel Pool.

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(Cont'd.)

Spent Fuel Assembly drooped onto the Spent Fuel Pool floor

The probability and consequences of dropping a spent Fuel Assembly onto the Spent Fuel Pool liner is not increased as a result of reracking. Spent fuel handling tools will not change nor will the method / procedure of handling spent Fuel Assemblies. Additionally, the small increase in the number of spent fuel storage cells gained as a result of reracking will not affect the probability of the accident since the number of spent Fuel Assemblies in the Spent Fuel Storage Pool is not an input to the initial conditions of the accident evaluation. The maximum drop distance for a Fuel Assembly will not change as a result of reracking, therefore, the consequences remain unchanged.

Spent Fuel Assembly dropped between racks

The probability and consequences of dropping a spent Fuel Assembly between racks is not increased as a result of reracking due to the similarity of the Holtec International racks and the existing Joseph Oat racks. Also, as indicated above, there is no change to the spent fuel handling tools. This accident was previously evaluated under the safety analysis that supports the Technical Specification LCO 3.7.15 and was shown to have no effect on reactivity. This is considered a bounding analysis and is applicable to the new design since the Holtec International rack layout precludes a reactivity increase due to this fuel handling accident.

Spent Fuel Assembly dropped between a rack and Spent Fuel Pool wall

The probability and consequences of dropping a spent Fuel Assembly between a rack and the Spent Fuel Pool wall is not increased as a result of reracking due to the similarity of the Holtec International racks and the existing Joseph Oat racks. Also, as indicated above, there is no change to the spent fuel handling tools. The worst case scenario consists of a fresh Fuel Assembly, of the highest allowed enrichment, accidentally placed in a cutout area between a rack and the new fuel elevator or tool bracket. The consequences of this event remain within the design basis criticality limit of < 0.95 keff assuming a minimum Boron Concentration of 220 PPM in the Spent Fuel Pool, water (the minimum soluble Boron Concentration, in conjunction with this design change, is 300 PPM for conservatism).

Spent Fuel Assembly loaded contrary placement restrictions

The probability and consequences of placing a spent Fuel Assembly contrary to placement restrictions is not increased as a result of reracking since the existing spent fuel storage racks are already of a two region layout, similar to the Holtec International racks. Additionally, the possibility of a misplaced Fuel Assembly is minimized by the independent verification of the Nuclear Component Transfer List that prescribes the exact location of each Fuel Assembly. The worst case scenario of placing a Fuel Assembly of the highest enrichment (i.e., 5.0 weight percent U-235) into a Region II rack cell was shown to remain within the design basis criticality limit of 0. 95 keff assuming a minimum soluble Boron Concentration of 220 PPM in the Spent Fuel Pool water.

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(Cont'd.)

Spent Fuel Assembly dropped onto a rack

The probability and consequences of dropping a spent Fuel Assembly onto the Holtec International racks is not increased as a result of reracking. Spent fuel handling tools will not change nor will the method / procedure of handling spent Fuel Assemblies. Additionally, the small increase in the number of spent fuel storage cells gained as a result of reracking will not affect the probability of the accident since the number of spent Fuel Assemblies in the Spent Fuel Storage Pool is not an input to the initial conditions of the accident evaluation. The consequences are shown to meet all existing Design Basis requirements as described in the Byron and Braidwood UFSAR. Analysis of the spent fuel drop accidents onto the top of a spent fuel storage rack (shallow drop), and a deep drop into the bottom of a cell resulting in impact at the bottom of the rack cell, were performed to demonstrate that the spent fuel rack retains its structural integrity and capability to safely store spent fuel in adjacent cells. The damage due to a perfectly vertical drop, on the top of a rack, bounds an inclined Fuel Assembly drop because the impact energy is focused on a single cell wall, which results in maximum cell blockage. The radiological consequences of the drop onto the Spent Fuel Pool liner, shallow drop onto the top of the rack and deep drop into the bottom of a rack cell are bounded by the existing UFSAR basis of 314 ruptured rods. The UFSAR design basis dose is shown to be much less than the 10 CFR 100 reference values of 300 Rem to the thyroid and 25 Rem to the whole body.

Spent fuel cask dropped onto a rack

The probability and consequences of a cask drop is not increased a result of reracking. There are no changes to the systems, structures, or components associated with movement of the spent fuel cask. Although the Spent Fuel Handling Building crane trolley stops will be procedurally removed for installation of the Holtec International racks, at no time will there be movement of the cask with these trolley stops temporarily removed. The cask is shown by the Byron and Braidwood UFSAR to be isolated from the Spent Fuel Pool by the combination of guard walls, which are designed to withstand the impact of a cask drop, and both administrative and physical controls. These controls are designed to preclude the Fuel Handling Building crane from travel over the Spent Fuel Pool. There are also trolley stops on the crane bridge, which physically prevents the main hook of the crane from traveling into the spent fuel storage area when handling a spent fuel cask. Spent Fuel Pool rack installation activities and cask handling will not be performed simultaneously, thus minimizing the possibility of improper movement of the cask. This practice is consistent with the Byron and Braidwood UFSAR relative to new fuel operations. There will be no changes to any of the equipment, procedures or operations relative to spent fuel cask handling that are associated with this design change. Installation of the lift rig storage device will be performed under the requirements of NUREG 0612.

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(Cont'd.)

Change in Spent Fuel Pool water temperature
The probability and consequences of a change to the Spent Fuel Pool water temperature is not increased as a result of reracking. A bulk pool temperature analysis has been performed and there are no features of this design change that would prompt a Spent Fuel Pool water temperature increase above what is analyzed.

A new analysis was performed for a Spent Fuel Pool water temperature of 4° C (39° F), which is well below the lowest normal operating temperature of 50° F. Because the temperature coefficient of reactivity in the Spent Fuel Pool is negative, temperatures greater than 4° C will result in a decrease in reactivity.

Loss of Spent Fuel Pool cooling
The probability and consequences of the loss of Spent Fuel Pool cooling is not increased as a result of reracking. The Spent Fuel Pool cooling system has no emergency function during an accident. In the event of the failure of a Spent Fuel Pool pump or loss of cooling to a Spent Fuel Pool heat exchanger, the second cooling train provides 100% backup capability, thus assuring continued cooling of the Spent Fuel pit. The results of the unlikely event of a failure of the return line to the Spent Fuel Pool downstream of the two Spent Fuel Pool heat exchangers would be a rise in pool water temperature followed by an increase in evaporative losses. These losses could be made up indefinitely from the Safety Category 1 Fire Protection System. The Spent Fuel Pool cooling system has no emergency function during an accident.

Loss of Spent Fuel Pool water level
The probability and consequences of the loss of Spent Fuel Pool water level is not increased as a result of reracking. System piping is arranged so that failure of any pipeline cannot drain the Spent Fuel Pool below the water level required for adequate radiation shielding. Reracking the Spent Fuel Pool does impact the current system piping arrangement. With 26 feet of water between the top of the active region of the Spent Fuel Assemblies stored in the high-density racks and the Spent Fuel Pool low level alarm, sufficient shielding remains over the fuel after any postulated Spent Fuel Pool dewatering incident to allow recovery operations to continue.

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(Cont'd.)

Water quality of Spent Fuel Pool
The probability and consequences of a change (decrease) in the water quality of the Spent Fuel Pool is not increased as result of reracking. The Holtec International Spent Fuel Pool Storage Racks are manufactured completely of inorganic materials that have proven use in Spent Fuel Pool environments. The degradation of the organic material Boraflex in the existing racks has been determined to have an adverse affect on the quality of the Spent Fuel Pool water not only with respect to water clarity but also in the unacceptable addition of soluble silica which results from the Boraflex degradation. A change in consequences would be the result of cladding damage of a Spent Fuel Assembly. The Holtec International spent fuel storage racks have been determined, as indicated above, to have no increased affect on the consequences of damage to a Spent Fuel Assembly.

Seismic Events
The probability of a seismic event is unchanged as a result of reracking. Seismic Events are acts of nature and their probability is not affected by the type of racks in the pool The new racks have been designed and analyzed as Seismic Category 1 Structures. Also, the cumulative damage factor under one SSE and 20 OBE events was calculated to be well below the acceptance limits of 1.0 for fatigue failure of the Spent Fuel Pool liner under all loading conditions.

The bounding condition of an isolated rack (either style) has been evaluated for a seismic event. Results of the evaluation show that the rack displacement is less than the minimum normal rack gap. Therefore, it is concluded that the interim rack configurations are kinematically stable. Installation activities will meet NUREG 0612 requirements and staging locations of new racks have been evaluated as acceptable.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the replacement of the existing Byron Spent Fuel Pool was evaluated for the possibility of creating a new or different accident. The following cases were reviewed.

   a. An accidental drop of a rack into the Spent Fuel Pool, and

   b. additional heat load resulting from the additional storage capacity.

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(Cont'd.)

A construction accident resulting in a drop of a rack, aluminum pick, lift rig, or lift rig storage device is an extremely unlikely event. Operability of the cranes will be checked prior to use. Lift equipment and rigging will also be inspected prior to use. Operators of lift equipment and cranes will be trained prior to use. Safe load paths will be followed and Byron Station commitments to the provisions of NUREG-0612 will be implemented by use of written procedures that have been utilized for numerous other similar rack installation projects. The Technical Requirements Manual requires that Fuel Handling Building Crane loads be limited to 2000 pounds when traveling over Fuel Assemblies. A component drop would present limited structural damage to the Spent Fuel Pool slab on grade, due to the slab being founded on rock and soil. Local concrete crushing and possible liner puncture could occur. Failure of the liner would not result in a significant loss of water and no safety related equipment would be affected by the leakage. Make up water is available from three separate sources:

1. Two 500, 000 gallon borated Refueling Water Storage Tanks

2. Two 500,000 gallon non-category 1 Primary Water Tank unborated back up water sources, and

3. The unborated Safety Category 1 Fire Protection System, available for Spent Fuel Pool water make up. A component drop, therefore, does not create the possibility of creating a new or different kind of accident.

The additional heat load resulting from the additional storage capacity of 114 cells (i.e., approximately 4%) has been evaluated for the possibility of creating a new or different kind of accident. The existing Spent Fuel Pool cooling system has been shown to be capable of removing the decay heat generated by the additional spent Fuel Assemblies utilizing the standard Byron Station Operating Procedures.

The fuel pool rack and fuel configurations have been analyzed considering criticality, thermal hydraulic and structural effects. The increase in storage capacity is achieved by the installation of additional racks of similar, but improved design, which are passive components. No new operating schemes or active equipment types will be required to store additional Fuel Assemblies in the fuel pools. The possibility of a different type of accident occurring is not created since the new racks meet or exceed the requirements applicable to the existing racks.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because although the margin of safety defined in the bases for the Technical Specifications is not reduced, as indicated below, it was identified that Technical Specification changes are required. A Technical Specification change request has been submitted (i.e. letter from R. Krich to NRC, dated 3/24/99. This modification will not be installed until receipt of the requested Technical Specification changes.

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(Cont'd.)

Discussion for Technical Specification 3.7.15 "Spent Fuel Pool Boron Concentration
The function of the Spent Fuel Pool is to store Fuel Assemblies in a subcritical and coolable configuration throughout all environmental and abnormal loadings, such as earthquakes, dropped Fuel Assemblies, or loss of Spent Fuel Pool cooling. Because the Joseph Oat storage racks need to be OPERABLE during and until rack change-out is complete, the Technical Specification needs to address both the requirements for use of the Joseph Oat storage racks as well as the Holtec International storage racks.

The Holtec International spent fuel storage racks are designed to meet all applicable requirements for safe storage of spent fuel and are functionally compatible with the Spent Fuel Pool.

The Holtec Licensing Report has analyzed the consequences of this reracking project by area. In each area, (i.e., criticality, seismic, structural, thermal hydraulics, and radiological exposure), design basis margins of safety will be maintained. Installation controls specified in Byron Station commitments to NUREG-0612 preserve the margins of safety with regard to heavy load restrictions. Compliance with the Byron Station design basis limits and procedure adherence will preclude reducing margins of safety.

The margin of safety, for either the Joseph Oat or Holtec International racks, or any combination of the two type of racks, is not reduced as demonstrated by analysis of the seismic, structural, thermal hydraulic, criticality, and radiological aspects of this design change. The Byron Station design basis Spent Fuel Pool maximum bulk temperature acceptance limit of 140° F has been demonstrated to be preserved by analysis. In addition, it has been shown that under the maximum decay heat load scenario of a full core discharge with a 100 hour hold time, following a normal discharge occurring 17 days earlier, the local water temperature was shown to be subcooled. Cladding failure due to extreme thermal stress was shown to be not credible. Criticality calculations show that keff will be maintained within the margin specified above for both the Joseph Oat and the Holtec International racks. The new Holtec International Spent Fuel Pool Storage Racks have been designed in accordance with the Byron Station Design Bases requirements and the NRC OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications, as amended on January 18, 1979.

The change in the temperature of the Spent Fuel Pool water was evaluated for the potential increase in reactivity. The new Holtec rack analysis was performed assuming a Spent Fuel Pool water temperature of 4° C (39° F), which is well below the lowest normal operating temperature of 50° F. Because the reactivity temperature coefficient in the Spent Fuel Pool is negative, temperatures greater than 4° C will result in a decrease in reactivity. As a result, loss of Spent Fuel Pool temperature control has no adverse affect on fuel stored in Holtec International storage racks.

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DCP 9800140
(Cont'd.)

Since all aspects of the design change have been demonstrated to be within the existing Design Bases for Byron Station and the NRC requirements applicable to spent fuel storage, the changes do not involve a reduction in the margin of safety.

Discussion for Technical Specification 3.7.16 "Spent Fuel Pool Storage"
The function of the Spent Fuel Pool is to store Fuel Assemblies in a subcritical and coolable configuration throughout all environmental and abnormal loadings, such as earthquakes, dropped Fuel Assemblies, or loss of Spent Fuel Pool cooling. Because the Joseph Oat storage racks need to be OPERABLE during and until rack change-out is complete, the Technical Specification needs to address both the requirements for use of the Joseph Oat storage racks as well as the Holtec International storage racks.

The Holtec International spent fuel storage racks are designed to meet all applicable requirements for safe storage of spent fuel and are functionally compatible with the Spent Fuel Pool.

The Holtec Licensing Report has analyzed the consequences of this reracking project by area. In each area, (i.e., criticality, seismic, structural, thermal hydraulics, and radiological exposure), design basis margins of safety will be maintained. Installation controls specified in Byron Stations commitments to NUREG-0612 preserve the margins of safety with regard to heavy load restrictions. Compliance with the Byron Station Design Basis Limits and procedure adherence will preclude reducing margins of safety.

The margin of safety, for either the Joseph Oat or Holtec International racks, or any combination of the two type of racks, is not reduced as demonstrated by analysis of the seismic, structural, thermal hydraulic, criticality, and radiological aspects of this design change. The Byron Station Design basis Spent Fuel Pool maximum bulk temperature acceptance limit of 140° F has been demonstrated to be preserved by analysis. Criticality calculations show that keff will be maintained within the margin specified above for both the Joseph Oat and the Holtec International racks. The new Holtec International Spent Fuel Pool Storage Racks have been designed in accordance with the Byron Station Design Bases requirements and the NRC OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications, as amended on January 18, 1979.

The change in the temperature of the Spent Fuel Pool water was evaluated for the potential increase in reactivity. The new Holtec rack analysis was performed assuming a Spent Fuel Pool water temperature of 4° C (39° F), which is well below the lowest normal operating temperature of 50° F. Because the reactivity temperature coefficient in the Spent Fuel Pool is negative, temperatures greater than 4° C will result in a decrease in reactivity. As a result, loss of Spent Fuel Pool temperature control has no adverse affect on fuel stored in Holtec International storage racks.

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DCP 9800140
(Cont'd.)

Since all aspects of the design change have been demonstrated to be within the existing Design Bases for Byron Station and the NRC requirements applicable to spent fuel storage, the changes do not involve a reduction in the margin of safety.

Discussion for Tech Spec 4.3.1 "Criticality"

Section 4.0 of Tech Specs identifies the design features of certain components. There is no Bases section associated with this Tech Spec section. As a result, corresponding UFSAR section(s) were reviewed.

Similar to the discussion presented above for Tech Spec Sections 3.7.15 and 3.7.16, the Joseph Oat storage racks need to be OPERABLE during and until rack change-out is complete, the Technical Specification needs to address both the requirements for use of the Joseph Oat storage racks as well as the Holtec International storage racks. The UFSAR shall be revised consistent with proposed Tech Spec changes upon approval by the NRC. DRP 8-036 shall control UFSAR changes necessitated as a result of reracking the Spent Fuel Pool.

Since all aspects of the design change have been demonstrated to be within the existing Design Bases for Byron Station and the NRC requirements applicable to spent fuel storage, the changes do not involve a reduction in the margin of safety.

Discussion for Technical Specification 4.3.3 "Capacity"
Section 4.0 of Technical Specification identifies the Design Features of certain components. There is no Bases Section associated with this Technical Specification Section. As a result, corresponding UFSAR section(s) were reviewed.

For the Holtec International storage racks, a total of 396 storage cells are provided in 4 racks in Region 1 and 2588 storage cells are provided in 20 racks in Region 2 for a total of 2984 storage cells. Increasing the number of available spent fuel storage cells has been determined to have no impact on the margin of safety for Spent Fuel Pool operation. The UFSAR shall be revised consistent with Tech Spec changes upon approval by the NRC. DRP 8-036 shall control UFSAR changes necessitated as a result of reracking the Spent Fuel Pool.

Since all aspects of the design change have been demonstrated to be within the existing Design Bases for Byron Station and the NRC requirements applicable to spent fuel storage, the changes do not involve a reduction in the margin of safety.

6G-99-0004
DCP 9800235

DESCRIPTION:

The purpose of this Design Change was to replace the Automatic Waste Gas O_2 Analyzer and sequencer with new analyzer and sample selector switch. The Waste Gas Oxygen (O_2) Recorder 0ARGW8003 and H_2 recorder 0AR-GW8000 were deleted. The new analyzer, 0AIT-GW8003 provides the high O_2 alarm. Chart recording of O_2 and H_2 concentration of Waste Gas Processing System is deleted by this change. Automatic sequencing of sample points of WGPS were deleted and replaced by a manual selection using selector switch. The local gauges, 0PI-GW006 and 0PI-GW007 in Panel 0GW01J were changed from 1 ½" to 4 ½" for easy reading. The existing Hydrogen analyzer remains in place with its EPN changed from 0AT-GW8000 to 0AIT-GW8000.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the change affects waste gas monitoring functions only and cannot initiate an accident. The monitoring function is not required to mitigate an accident after it has occurred. The new instrumentation meets or exceeds the specifications of the existing instrumentation and the method of using this instrumentation remains unchanged. The consequences of an accident are the same regardless of the specific gas monitoring instrumentation since the initial system conditions (radioactivity) assumed prior to an accident are unchanged. The change is being made in order to improve instrument reliability, which will maintain or decrease the probability of equipment malfunction. In the event that either the existing or modified waste gas monitoring instrumentation failed to function properly, the ability to detect explosive gas concentration would be lost.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the change will not introduce any new failures in the operation of WGPS. The change only performs monitoring functions and will have the same malfunction scenarios. The change maintains the original design of monitoring oxygen levels to provide adequate prevention of explosive gas mixtures in the WGPS. The overall reliability of monitoring oxygen to prevent explosive gas mixture in WGPS is increased by the change. The change does not adversely affect the systems, structures, or components important to safety so as to create a possibility of a malfunction or a possibility of accidents from those evaluated in the SAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which Technical Specifications are based.

6G-98-0132
DCP 9800239

DESCRIPTION:

The purpose of this Design Change was to support the temporary removal of the motor operator for valve 1SX011 to conduct maintenance repair activities. The operator was removed from the valve while the HBC gear operator was left on effectively locking the valve stem in place. The 1SX011 is one of three return header cross tie valves provided for each unit used to direct service water return to the appropriate essential service water cooling tower. The valve operator was removed while the valve was in its fully closed position, thus leaving the valve closed by virtue of the self locking gear set. The activity was controlled by the procedure requirements of the Temporary Alteration Program.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the Technical Specifications require the ability of the unit Crosstie Valves to shut upon demand to provide for two independent SX Systems. Locking the valve in its closed position is consistent with this requirement. The activity did not increase the probability of occurrence of an accident or a different malfunction.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the temporary activity did not create conditions beyond the original design parameters or design basis values for the valve's function. The valve was temporarily left with the HBC gear assembly, which consists of a 60:1 gear ratio, that effectively locked in place the valve to prevent drifting due to hydrodynamically induced torque.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the temporary removal of the valve operator for maintenance repairs did not change or affect any parameters upon which Technical Specifications are based.

6G-98-0275, Rev. 1
DCP 9800255 & 9800256

DESCRIPTION:

The purpose of this Design Change was to replace the Diesel Generator's Cooling Jacket Water pneumatic controllers, 1/2TIC-DG247A, B due to part obsolescence.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the design enhancement maintains the performance of the diesel generator jacket cooling water system by replacing the obsolete controller. Thus no functional changes are made to the DG system and the probability of a malfunction of the equipment important to safety is not increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because this design change does not adversely impact the capability of the system to respond to an accident or transient. No functions or operating characteristics will be altered.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because replacement of obsolete equipment with same functional capabilities does not affect any parameters upon which Technical Specifications are based

6G-99-0022
DCP 9800315

DESCRIPTION:

The purpose of this Design Change was to replace the hot and cold water pressure reducing valves (PRV) (0WM952 and 0WM953) with low leakage valves, adds accessible Isolation Valves (0WM216 and 0WM217), and relocates the Hot Water PRV to an accessible location for maintenance in the future. In addition, a check valve (0WM220) is added to the cold water PRV supply line to prevent back flow of potentially contaminated laundry water into the Demineralized Water (WM) system. The existing Cold Water PRVs is a combination PRV and check valve. The existing Hot Water supply design includes a check valve to prevent reverse flow. The current PRVs are not designed to be low leakage valves. When the laundry activities are shutdown, the existing PRVs leak by causing pressures to build up in the piping downstream of each PRV. Eventually the pressure exceeds the relief valve set pressure and lifts the relief valves, which drains to the laundry drain tank.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the laundry valves are non-safety related and do not affect any equipment important to safety. Failure of the PRVs could only affect the laundry equipment, which is not considered equipment important to safety. All piping has been qualified for normal operating loads and SSE Loads to prevent damage of nearby Category 1 equipment. Therefore, the probability of a malfunction of equipment important to safety has not increased. The Laundry Hot and Cold Water PRVs have no affect on the off site dose nor could any malfunction of these valves create adverse radiological consequences. The WM System is not relied upon to mitigate the consequences of a malfunction of equipment important to safety. Hence, the consequences of a malfunction of equipment important to safety will not increase.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the change does not adversely impact the Demineralized Water System or operation of any other plant systems. There are no new credible malfunctions and the failure modes of the new PRVs are similar to that of the existing PRVs. Therefore, there is no possibility of a malfunction of a different type from those discussed in the UFSAR. These components are part of the non-safety related system and have no impact on equipment important to safety. The demineralized water supply to the laundry system has no potential to cause radiation exposure to the public. Therefore, the possibility of a different type of accident is not created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because no Technical Specifications are affected by this DCP.

6H-99-0221
DCP 9800347

DESCRIPTION:

The purpose of this Design Change was to install permanent lead shielding on line 2CV01 E-3.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the lead shielding is a passive component that does not perform any mechanical movement or interact with CVCS or any other SSCs. The operation and intended function of CVCS or any other SSCs remain unchanged by the placement of sheet lead shielding. Therefore, the placement of lead shielding has no affect on the plant operation.

   The affected portion of CVCS piping, and associated piping supports and structures have been evaluated for the installation of lead shielding. The results of the evaluation in Calculation W000332 show that the installation of lead shielding on selected portions of letdown line on CVCS has no adverse effect on the piping components, supports and structures.

   Since the radiation shielding is a passive component with no known failure mechanisms, and it does not interact with any other SSCs important to safety, the radiation shielding will not cause the malfunction of other SSCs or equipment important to safety.

   The consequences of an accident cannot be increased since the lead shielding does not introduce any conditions, which would increase the radiation release levels in the plant.

   Radiation shielding attenuates the resultant radiation level and doses to as low as reasonably achievable (ALARA). However, since the placement of lead shielding is localized to small portion of CVCS, the effect on the consequences of the accident is negligible.

   Furthermore, radiation shielding does not interact with any SSCs that mitigates the consequence of the accident. Therefore, the consequence of the accident (off-site dose) is not increased by the placement of lead shielding on CVCS.

6H-99-0221
DCP 9800347
(Cont'd.)

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the lead shield has no known failure mechanisms. The lead shield is a passive component that does not perform any mechanical movement or interact with any other SSCs that may lead to the failure of other SSCs. Therefore, the placement of lead shield does not create any new failure modes, nor affect the failure modes of any other SSCs under applicable operating and accident conditions.

   The thermal induced failure of coating material during normal or DBA condition is incredible. Furthermore, amount of coating material is so minor (i.e., 1/32 inch coating thickness) that the capacity of Fire Zone and the amount of combustible material in the Fire Zone is not adversely affected by the addition of this coating material. As per the coating material and sheet lead manufacturer's, the Rad Klad 340 coating material does not contain any halogens that degrades the mechanical properties of stainless steel when in contact with products such as chlorine, bromine or iodine.

   Therefore, the coated sheet lead shielding and the sheet lead does not create nor affect the failure modes of CVCS or any other SSCs under any applicable operating and accident conditions.

   The lead shielding only increases the loads on the building structures, which have been addressed in an engineering evaluation and are found to be acceptable. No new failures are introduced as a result of this additional loads. Therefore the possibility for an accident or malfunction remain the same.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because there are no Technical Specification Requirements for the installation or the placement of lead shielding.

6G-99-0081
DCP 9800368 & 9800369

DESCRIPTION:

The purpose of this Design Change was to add high differential pressure alarm for safety valve room ventilation exhaust fans 1/2VV01CA, CB, CC, and CD.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the change does not involve or interface with any systems, structures, or components important to safety. The activity does not alter the initial conditions used in the UFSAR analysis for any accidents. The components involved do not initiate any accidents or mitigate the consequences of any accident analyzed in the SAR. The function of any system, structure or component important to safety is not altered by this change.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because this design change does not adversely impact UFSAR accident-related systems, structures, and components (SSCs). The change does not introduce any adverse interactions between any SSCs.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because adding the high dP alarm for exhaust fans 1/2VV01CA, CB, CC, and CD does not affect any parameters upon which Technical Specifications are based.

6G-98-0255
DCP 9800406

DESCRIPTION:

The purpose of this Design Change was to replace Steam Generator (SG) access closure bolts with studs, nuts, and washers. SG closures include primary and secondary manways, handholes, access and inspection openings. The activity provided installation of hardware upgrades recommended in Westinghouse Technical Bulletin 87-01, not previously installed. During a recent cycle, one of the inspection openings leaked, possibly because the gasket was unevenly compressed. Westinghouse Technical Bulletin 87-01 recommends hardware upgrades, among other measures, to reduce the potential for leakage on the SG closures.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because failure of these SG closures is bounded by the analyzed accidents of a LOCA, Loss of Feedwater, and a Main Steam Line Break. The SG closures are smaller or the same size as the piping assumed to break in these accidents. The new studs and nuts are designed to the same codes and standards as the existing bolting. No other equipment important to safety is affected. Therefore, the probability of accidents and equipment malfunction is unchanged. The changes to the SG access opening bolting do not affect the capability of the SGs to mitigate any accident analyzed in the SAR. No other system, structure or component used to mitigate the accidents are affected. Therefore, consequences of accidents and malfunctions of equipment important to safety are not increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the activity does not create any new failure modes. The new studs and nuts are designed to the same Asme Code requirements as the existing bolting. No other systems or functions are affected by this change. Therefore, the activity will not adversely impact systems or functions so as to create the possibility of an accident or malfunction of a type different from those evaluated in the SAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which Technical Specifications are based.

6G-99-0030
DCP 9800417

DESCRIPTION:

The purpose of this Design Change was to increase the size of the Technical Support Center (TSC) chiller liquid refrigerant lines. Existing O.D. 5/8" copper lines (0VV25A and 0VV28A), solenoid valves (0VV033A and 0VV033B), liquid indicators, and fittings are replaced with O.D. 7/8" tubing and components. The change also installed a temperature switch to prevent chiller operation during low outside ambient temperature conditions.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because increasing the size of TSC chiller liquid refrigerant lines and adding a compressor temperature switch will not impact the probability or consequences of a decrease in reactor coolant Inventory or any other accident or transient. The TSC chiller is not relied upon to directly mitigate the consequences of any UFSAR accident, transient, or malfunction of equipment important to safety. However, TSC personnel do provide emergency response support, including analysis and diagnosis of abnormal plant conditions and radioactivity releases to the environment. The ability of personnel to perform these functions is not reduced by this change. The TSC chiller liquid lines and compressor temperature switch are not located near safety-related equipment and will not impact safety-related structures.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because increasing the size of TSC chiller liquid lines and adding a compressor temperature switch will not create the possibility of a new type of accident or transient. No new failure modes of the liquid lines are created. Failure of the TSC chiller will not create a new type of accident or transient. The TSC chiller equipment has no impact on equipment important to safety. The TSC chiller change will not create any new interactions with equipment important to safety.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because increasing the size of TSC chiller liquid refrigerant lines and adding a TSC refrigerant compressor temperature switch change does not impact any Technical Specifications.

6G-99-0161
DCP 9800482, 9800483, 9800484, 9800485,
9800494, 9800495, 9800496, 9800497

DESCRIPTION:

The purpose of these Design Changes was to replace the existing Kerotest lift Check Valves, 1/2SI8819A,B,C,D, with Flowserve swing check valves. The activity is expected to improve the reliability of the check valves.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the Flowserve check valves were designed to the same codes and standards as the Kerotest check valves. The piping stress levels remain within code allowables. Therefore, there was no change in the probability of a LOCA. The change had no adverse affect on the operation of Containment Isolation, SI System or ECCS. Therefore, the consequences of the accidents or transients were unchanged. The change had no affect on equipment failures or malfunctions. No new failure modes were created by changing the check valves from a spring lift piston check to a swing check valve. Both types of check valves could fail to open or fail to close and isolate. The probability of a check valve failing to open (stuck closed) is not increased. Potential causes of a stuck closed check valve are corrosion, foreign material and wear. Both the new and old valves are stainless steel. The wear surface materials for both valves are similar. The replacement valve was selected due to improved access for inspection and maintenance. The risk of foreign material is unchanged.

   The probability of a check valve failing to close (stuck open or improper seating) was unchanged. Potential causes of a check valve failing to close are corrosion, foreign material, wear, improper installation/design misapplication. The wear surface materials for both valves are similar. The replacement valve was selected due to improved access for maintenance. The risk of foreign material is unchanged. The swing check valves are designed for and were installed in a horizontal run of pipe with the hinge pin horizontal and the bonnet on top. The flow velocity for the valves is greater than the flow required to hold the disc open firmly against the backstop. Thus disc flutter is not a concern. The minimum dp required to achieve design allowable seat leakage on the swing check valves is 100 psi. The actual check valve dp during operation and during online SI pump testing is well above 100 psi.

6G-99-0161
DCP 9800482, 9800483, 9800484, 9800485
9800494, 9800495, 9800496, 9800497
(Cont'd.)

The change did not adversely affect equipment required to mitigate a possible check valve failure or any other failure. ECCS flow is balanced to provide adequate flow with a single train failure. The check valve change had a small impact on the ECCS throttle valve position. ECCS flow was rebalanced/verified after installation of the new check valves.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the Flowserve check valves provide the same isolation to separate the RCS from the lower pressure SI piping as the Kerotest check valves. The allowable RCS PIV leakage is unchanged. Thus the probability of overpressurization of low pressure piping or components and the possibility of an unanalyzed LOCA outside of containment are unchanged. No new failure modes have been created. The Flowserve check valves have the same failure modes as the Kerotest check valves.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change did not change the RCS PIV leakage criteria. ECCS system operation did not change. ECCS flows was verified/balanced to obtain the flows assumed in the accident analysis. The containment isolation function of the check valves was unchanged. Check Valves SI8819A, B, C, D do not have a required isolation time and are not subject to Type C leakage tests.

6G-99-0012
DCP 9800498

DESCRIPTION:

The purpose of this Design Change was to increase the SX Makeup Pump impeller size from 9-9/16" to 9-7/8". Various pump internal part (impellers, wearing rings, lineshaft, strainer basket, and seismic restraint) materials were changed to stainless steel. The SX Makeup Pump Diesel Oil Storage Tank minimum allowable tank level was changed to 47%. The UFSAR description of the maximum water withdrawal rate and fuel consumption rate for the SX makeup pumps were revised to reflect the change in pump performance due to the larger pump impellers.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the SX makeup system does not factor into any initiating event for UFSAR accidents. The changes did not adversely affect the function of the UHS or SX makeup system. The SX makeup system capability was improved by the impeller size change. The change in the SX Makeup Pump Diesel Oil Storage Tank minimum level ensures an adequate volume of diesel fuel is available prior to an event to support makeup to the UHS for at least 72 hours. The changes did not adversely affect the operation of the SX Makeup Pump or SX Makeup system components. The required pump NPSH and submergence are not changed. SX makeup system reliability increased with the longer life impeller materials. The changes did not adversely affect the operation of any equipment important to safety.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because all system pressure boundary components were evaluated for the new maximum pressure and are acceptable. The SX Makeup Pump was seismically qualified for the new materials and operating pressures. The system piping was analyzed for the new design and operating pressures and meets the Asme Code allowables for all applicable loading combinations. The change in system pressure did not adversely affect the ability of motor operated valves, 0SX157A/B and 0SX158A/B, to open or close. No new failure modes were created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change to 47% minimum SX Makeup Pump Oil Storage Tank level matches the Technical Specification acceptance limit.

6G-98-0282
DCP 9800510

DESCRIPTION:

The purpose of this Design Change was to install permanent lead shielding on line 1CV01E-3".

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the lead shielding is a passive component that does not perform any mechanical movement or interact with CVCS or any other SSCs. The operation and intended function of CVCS or any other SSCs remain unchanged by the placement of sheet lead shielding. Therefore, the placement of lead shielding has no affect on the plant operation.

   The affected portion of CVCS piping, and associated piping supports and structures have been evaluated for the installation of lead shielding. The results of the evaluation show that the installation of lead shielding on selected portions of let down line on CVCS has no adverse effect on the piping components, supports and structures.

   Since the radiation shielding is a passive component with no known failure mechanisms, and it does not interact with any other SSCs important to safety, the radiation shielding will not cause the malfunction of other SSCs or equipment important to safety.

   The consequences of an accident cannot be increased since the lead shielding does not introduce any conditions, which would increase the radiation release levels in the plant.

   Radiation shielding attenuates the resultant radiation level and doses to As Low As Reasonably Achievable (ALARA). However, since the placement of lead shielding is localized to small portion of CVCS, the effect on the consequences of the accident is negligible.

   Furthermore, radiation shielding does not interact with any SSCs that mitigates the consequence of the accident. Therefore, the consequence of the accident (off-site dose) is not increased by the placement of lead shielding on CVCS.

6G-98-0282
DCP 9800510
(Cont'd.)

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the lead shield has no known failure mechanisms. The lead shield is a passive component that does not perform any mechanical movement or interact with any other SSCs that may lead to the failure of other SSCs. Therefore, the placement of lead shield does not create any new failure modes, nor affect the failure modes of any other SSCs under applicable operating and accident conditions.

   The thermal induced failure of coating material during normal or DBA condition is incredible. Furthermore, amount of coating material is so minor (i.e., 1/32 inch coating thickness) that the capacity of Fire Zone and the amount of combustible material in the Fire Zone is not adversely affected by the addition of this coating material. As per the coating material and sheet lead manufacturer's the Rad Klad 340 coating material does not contain any halogens that degrades the mechanical properties of stainless steel when in contact, such as chlorine, bromine or iodine.

   Therefore, the coated sheet lead shielding and the sheet lead does not create nor affect the failure modes of CVCS or any other SSCs under any applicable operating and accident conditions.

   The lead shielding only increases the loads on the building structures, which have been addressed in an engineering evaluation and are found to be acceptable. No new failures are introduced as a result of these additional loads. Therefore the possibility for an accident or malfunction remain the same.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because there are no Technical Specification Requirements for the installation or the placement of lead shielding.

6G-99-0011
DCP 9800532

DESCRIPTION:

The purpose of this Design Change was to install new floating Spent Fuel Pool skimmers and Isolation Valves on existing skimmer suction lines 0FC29AA-2" and 0FC29AB-2". The new floating skimmers replaced the original skimmer/strainers 0FC02MA and 0FC02MB, which required mechanical adjustment for changing Spent Fuel Pool levels. The Isolation Valves allow for single skimmer operation. The change also removes the adjustable handle portion of the original 0FC02MA and 0FC02MB mounting brackets and drills anti-siphon holes in the skimmer suction lines.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because installing floating skimmer assemblies does not impact the probability of a skimmer loop piping failure outside the Spent Fuel Pool, which could potentially siphon the pool. This change installs anti-siphon holes in the underwater skimmer suction lines, to prevent accidental siphoning of the fuel pool in the event a floating skimmer head were to sink or detach from the flexible hose, with a coincident skimmer loop piping failure. The Spent Fuel Pool skimmer loop is not relied upon to support the operation of safety related equipment or mitigate the consequences of a Spent Fuel Pool Cooling System (SFPCS) safety-related equipment malfunction.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the component material makeup is acceptable for use in the Spent Fuel Pool. The new floating skimmers are compatible with the existing skimmer pump and filter. The floating skimmers are designed to remain floating, thereby not interacting with the fuel racks. In the unlikely event that a skimmer did sink, the resulting interaction with the fuel racks would be bounded by the analysis for a Fuel Handling Accident inside the Spent Fuel Storage Building defined by the drop in of a Fuel Assembly.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which Technical Specification are based.

6G-99-0144
DCP 9800539 & 9800540

DESCRIPTION:

The purpose of these Design Changes was to make several changes to Gland Water (WG) and Feedwater (FW) piping near the main Feedwater Pumps.

The first change is to add drains to the WG inner seal return lines from the turbine driven FW pumps and the motor driven FW Pump. The drains are located at the low elevation point downstream of the manual isolation valve at the pump.

The second change was to add a 1" bypass line around the FW recirculation lines to the condenser (1/2FW012B,C). Each bypass line will contain two Isolation Valves. This change is only for the turbine driven feedwater pumps.

The third change was to add vibration dampening supports to the gland water return lines near the 1A, 1C, 2A, 2B and 2C FW pumps. (Lines 1WG03AA/AC and 2WG03AA/AB/AC). The 1B pump return line has an existing support that limits vibration to an acceptable amount.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because an increase in heat removal accident can result from an increase in Feedwater Flow, decrease in feedwater temperature or steam blowdown from a broken feedwater line. The addition of the drain lines, bypass lines and the supports cannot affect the control of the feedwater pump, FW heater bypass valves or FW regulating valves such that FW flow increases or FW temperature decreases. Failure of these lines cannot result in an increase in heat removal because back flow from the Steam Generators is prevented by existing FW check and Isolation Valves. Feedwater temperature to the Steam Generators would increase with lower flow. The consequences of increase in heat removal accidents are bounded by the consequences of a steam system piping failure. This change has no effect on the main steam system and cannot affect the consequences of the accident. The feedwater temperature and flow event consequences cannot be increased by this change since there is no ability to reduce feedwater temperature more than previously assumed in the analysis and there is no ability to increase Feedwater Flow to a higher value than assumed in the previous analysis. The new 1" diameter FW bypass lines are in a location where a break could not cause an increase in FW flow. Therefore, there will be no increase in the consequences of this accident.

6G-99-0144
DCP 9800539 & 9800540
(Cont'd.)

A decrease in heat removal accident could result from a loss of Feedwater Flow. The new lines do not increase the probability of a loss of flow since they meet system design requirements and are no more likely to fail than existing lines. The new lines are not in the flow path to the Steam Generators and have no ability to restrict normal Feedwater Flow. The new line bypassing the valves is equipped with two normally closed Isolation Valves in series to provide a primary and secondary barrier against leakage to the condenser. The consequences of the decrease in heat removal events "Loss of Normal Feedwater Flow" and "Feedwater System Pipe Break" are bounded by the consequences of a steam system piping failure. This change has no effect on the main steam system and cannot affect the consequences of the accident. The consequences of the Feedwater Flow accident cannot be increased since the analysis already assumes that all Feedwater Flow is lost. This change has no effect on the operation of the Auxiliary Feedwater System that functions to mitigate the consequences of the accident. Similarly, the feedwater line break accident consequences cannot be made worse since the new lines are no larger than existing lines and are located in an area where they can be isolated from the Steam Generators.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the vibration dampening supports have no effect on system operation other than to reduce displacement due to two-phase flow. The drain and bypass lines are normally isolated and do not affect system operation. During fill and vent of an idle feedwater pump, the drain and bypass lines will be used to ensure that the pump is ready for operation. Mispositioning of these valves would result in leakage and a loss of a small percentage of normal FW flow. Both of these events were previously evaluated.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this change does not affect equipment addressed by the Technical Specifications and does not affect the basis of any Technical Specification.

6G-98-0264, Rev. 1)
DCP 9800560 (Unit 1) & 9800561 (Unit 2)

DESCRIPTION:

The purpose of this Design Change was to install differential pressure indicating switches to monitor performance of the diesel generator lube oil strainers. One differential pressure indicating switch is required for each diesel generator lube oil skid. The design will utilize existing pipe taps used for other instruments (one tap from downstream of lube oil filter differential pressure indicating switch and second tap from upstream of turbo lube oil filter differential Pressure Indicator). The lube oil strainer high differential pressure alarm will be wired parallel with lube oil filter high differential pressure alarm and annunciated on local control panel 1PL07J/1PL08J (Division 1 and 2) and 2PL07J/2PL08J (Division 1 and 2) for Unit 1 and Unit 2 respectively. The new switches will be mounted on instrument stand and will be qualified for structural integrity. The tubing and their supports will also be seismically qualified.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the function of the diesel generators to provide an independent emergency source of power in the event of a complete loss of offsite power is not affected by this change. Installation of differential pressure instrumentation does not affect the operation of the lubricating system. The addition of this instrumentation will provide added assurance that the strainers will perform their function as designed. The design enhancement increases the performance of the diesel generator lube oil system by monitoring differential pressure across the strainer for indication of a clogged strainer. The design uses instrumentation identical to the one that is used for measuring differential pressure across filters and other strainers in the diesel generator lubrication system. As a result, the probability and consequences of a malfunction of equipment important to safety is not increased by this change.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the modification to the Diesel Generator Lube Oil System will not reduce the capability of the system to respond to an accident or transient. No functions or operating characteristics will be altered. As a result of this design enhancement, there will be improvement in monitoring of clogged lube oil strainers. The modification did not create the possibility of an accident or malfunction of a type different from those evaluated in the SAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which Technical Specifications are based.

6H-99-0393
DCP 9800567

DESCRIPTION:

The purpose of this Design Change was to perform the following changes to the Unit 2 Main Steam Power Operated Relief Valve (MS PORVs) actuators:

1. Replaced the accumulator pneumatic pressure switches.

2. Increased the open pressure switch, close pressure switch, open relief valve, and close relief valve setpoints.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the probability of an accident is unchanged by the design changes. The modification did not significantly change the actuator weight, thus the MS piping stresses were not adversely affected. The function and operation of the actuator pressure switches were not changed. The replacement pressure switches were recommended by the valve actuator OEM and are designed to have less setpoint drift and are less susceptible to vibration. The improved pressure switch reliability should improve MS PORV reliability. The ability of the MS PORVs to function to mitigate the consequences of an accident was not affected. The Steam Generator Tube Rupture (SGTR) accident analysis assumes an MS PORV fails open on the ruptured SG (off-site dose case) or fails closed (margin to overfill case). The result of this analysis was unchanged by the modification. Therefore, the consequences of a SGTR were not increased. The actuator changes have been seismically and environmentally qualified.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because this design change was installed to improve pressure switch reliability. The function and operation of the MS PORVs was not changed. The new pressure switches have the same failure modes as the replaced pressure switches. No new failure modes were created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which Technical Specifications are based.

6G-99-0156
DCP 9800578 & 9800579

DESCRIPTION:

The purpose of this Design Change was to install drain lines on 1/2ASG3A-4" upstream of Isolation Valves 1/2MS163 and Check Valves 1/2MS162. The Auxiliary Steam (AS) system supplies start up steam to Gland Sealing (GS) Steam System. The activity provided the means to drain the condensate accumulation in the AS line upstream of Isolation Valves 1/2MS163.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the affected portion of the AS System is in the Turbine Buildings Therefore, the probability of high energy line break (HELB) affecting safety equipment is not affected. This portion of AS is not described or specifically referred to, in any of the accident and transient events described in the SAR. The new drain lines are designed to the same standards as the existing system piping. Therefore, the probability of a steam system piping failure is not increased.

   These lines are not part of a radioactive boundary. A failure of the drains is bounded by the existing analysis in UFSAR Sections 15.1.5 and 15.1.6. The activity does not increase the consequences of any accident or transient evaluated in SAR.

   The activity does not increase the potential for equipment failures or malfunctions from the water hammer. Rather the activity is intended to decrease or eliminate the probability of equipment failure from water hammer events associated with the use of the affected portion of the AS System. Failure of the affected piping will not result in a HELB near equipment important to safety.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the intended function, the methods of performing its function, and the operating condition and parameters of the AS remain unchanged as a result of the activity. Therefore, the activity does not create the possibility of an accident or transient of a different type than previously evaluated. The affected portion of the AS System is not important to safety. The interaction of the AS System with the MS/GS systems is not changed. Therefore, the activity will not create the possibility of a different type of malfunction of equipment important to activity will safety than previously evaluated.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the activity does not affect or alter the margin of safety as defined in the basis for any Technical Specification.

6G-99-0018
DCP 9800599

DESCRIPTION:

The purpose of this Design Change was to install two disk spring washers under the actuator stem nut of Masoneilan Model 37 and 38 actuators for Air-Operated Valves (AOVs). These washers compensate for air diaphragm shrinkage and thereby more effectively secure the diaphragm to the actuator stem nut. The activity was a recommendation from Masoneilan Incorporated.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because there is no change to the operation of the AOVs described in the SAR. The change does not impact:

    - Stroke time or,
    - Stroke length or,
    - The ability of the valves to reposition or,
    - Fail-safe positions for the AOVs.

    Therefore, there is no change in probability of occurrence or consequences or an analyzed accident.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the addition of the spring washers does not introduce a new malfunction. If the spring washers become loose, the failure mechanism is identical to the current failure mechanism.

3. The margin of safety, as defined in the bases for any Technical Specification, is not reduced because the change does not affect any parameters upon which the Technical Specifications are based.

6G-99-0010
DCP 9800610

DESCRIPTION:

The purpose of this Design Change was to add permanent lead shielding to a portion of lines 1RH01BA-12", 1RH01BB-12", 1RH01CA-16", and 1RH01CB-16" to reduce radiation dose rates. The shielding will be in the form of coated lead sheets.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the installation of the lead shielding did not increase the probability of lines cracking or breaking since the affected lines were qualified for all loading conditions. Therefore, the probability of a decrease in reactor coolant inventory and Auxiliary Building flooding was not increased. The RH system continues to provide low head safety injection and heat removal as designed. Auxiliary Building flooding was not worsened since no additional moderate energy crack locations were created and the function of the leak detection sumps was not affected. The lead is shielded from contact with the stainless steel pipe so the possibility of stress corrosion cracking was not increased. The shielding is secured to the piping by stainless steel banding that is loaded well below its carrying capacity.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the design of the shielding ensures that it is well supported and the piping has been qualified for the new loads. The function of the RH system was not changed. Therefore, there will be no new accidents or malfunctions.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this change does not affect any Technical Specifications or Bases.

6G-98-0299
DCP 9800615

DESCRIPTION:

The purpose of this Design Change was to install a low point drain line and valves on the Main Steam (MS) Drip Leg Drain Header and shows the "as-built" configuration of a similar drain line that is installed on the Manual Drain Header, but was not shown on the design drawings. The MS lines have three automatically operated lowpoint drip legs to collect condensate and remove it from the MS System by means of a level controller and a level control valve. From the original design, steam leakage passes one of the level control valves, condenses, and collects in the piping downstream of the valves. As it condenses, a water column is created. This allows the formation of a steam bubble. Once the water column height is sufficient, it collapses the steam bubble causing a waterhammer event. As condensate builds up in low portions of the drain line, water slugs are formed. When one of the level control valves cycle as a result of condensate in a drip leg, the water slugs are moved through the pipe and waterhammer occurs. The design change drains condensate from the header prior to one of these events occurring, preventing the waterhammer and associated pipe/support damage. UFSAR Figure 10.3-1, Sheet 1 of the Main Steam System is revised to show the changes.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the additional piping elements are designed to the same pressure and temperature requirements as the non-safety portions of the MS system, where the drains are attached. Pipe stress and pipe support loads were reviewed and do not exceed design allowables. The portion of the MS system, that includes postulated high-energy pipe break locations, is not affected by the activity. Therefore, the probability of an accident is not increased. The replacement piping is qualified to the same requirements as other non-safety portions of the MS system. Therefore, the impacted steam lines do not affect the ability of the MS system to perform its design function. The new piping and valves are located downstream of the postulated MS break location and do not increase the size of the postulated break. The change does not add any new system interfaces or alter any existing system interfaces. No equipment important to safety is affected by this design change. The portion of the MS system affected by this change is non-safety related, non-seismic, and performs no safety function. Failure of the affected portion of the MS system is completely bounded by a failure of a main steam line, and could not lead to a more severe accident or more severe effects on any equipment important to safety. The affected piping does not function to mitigate the consequences of an accident. Operation of the MS Isolation Valves is not adversely affected.

6G-98-0299
DCP 9800615
(Cont'd.)

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the new components are designed to the system design temperature and pressure requirements and are fabricated using the same materials and grade of materials as the existing MS system, to which they are attached. The pressure rating for all new valves and piping components meets or exceeds that of the existing components in the affected portion of the system. No equipment important to safety is adversely affected and no system interactions are added or revised. Failure modes of the new piping and valves are the same as existing piping and valves. Therefore, the change will not create the possibility of an accident or malfunction different from those evaluated in the SAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which Technical Specifications are based.

6G-99-0039
DCP 9800624 & 9800625

DESCRIPTION:

The purpose of these Design Changes was to replace the existing Diesel Generator Exhaust Stack Rupture Disks. The new rupture disks will have a burst pressure of 2.25 psig (minimum)/3.25 psig (maximum) at 40° F. The currently installed rupture disks (2.5 psig burst pressure at 650 ° F) have too high of a burst pressure based on their temperature dependence and the measured disk temperature following engine operation. The new burst pressure is sufficiently high that it will not spuriously burst during normal diesel operation and low enough that it will burst before engine performance is adversely affected by a blocked exhaust stack.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the Diesel Generator Exhaust Stack Rupture Disks cannot initiate any accidents listed in Chapter 15 of the UFSAR (i.e., tornadoes or loss of offsite power). The diesel generators provide emergency power to equipment required to mitigate the consequences of an accident. Installation of the new rupture disks will improve the reliability of the diesel generator stop provide this function. The setpoint of the new rupture disks will ensure that diesel generator performance is not adversely affected should the exhaust stack become crimped.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the function of the diesel generator exhaust stack rupture disks is to provide an exhaust path for diesel generator exhaust should the normal exhaust path become blocked due to tornado, missile or other event. The new burst pressure will allow this function to be performed. No new failure modes are created by this change. There are no new or changed interactions with other systems.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because Technical Specifications 3.8.1 and 3.8.2 provide requirements based on the status of the diesel generators. The new rupture disks will ensure that the diesels are capable of performing their design function should the exhaust stack become crimped.

6G-99-0095
DCP 9900018 & 9900021

DESCRIPTION:

The purpose of this Design Change was to install protective pipe sleeves for Jockey Pumps 0FP06PA and 0FP06PB to protect the pump shafts from being exposed to the constant flow of water.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the activity did not remove or alter the design and capability of Fire Protection System. The fire hazards, Fire Protection System, and the compensatory measures to be implement when required are not changed or affected. The fire mitigation or the shutdown capability of the Jockey Pumps.

   Therefore the probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report Is not increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report Is not created because the pipe sleeves perform passive function of protecting the pump shaft from being exposed to the constant flow of water. The pipe sleeves do not interact with the Jockey Pumps or any other SSCs important to safety. The intended function and the method of performing the function remains unchanged. The activity does not change the operating parameter of the Jockey Pumps nor the FP Water Supply System.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the activity does not affect or alter the margin of safety as defined in the basis for any Technical Specification.

6G-98-0156, Rev. 2
DCP 9900022 & 9900023

DESCRIPTION:

The purpose of this Design Change was to rework a Temporary Alteration to make it a permanent installation for monitoring the pipe surface temperature as described below. This information will be used to perform surveillances in order to comply with NRC Bulletin 88-08. The RH/RY Temperature Monitoring System (TMS) includes the following:

1. Installed RH/RY RTDs by Temp. Alt. 93-1-003 and 93-2-022
   a. Three (3) strap on RTDs on Loops 1 and 3 Residual Heat Suction Line
   b. Two (2) strap on RTDs on the Pressurizer Aux. Spray Line
2. Install new Junction Box 1JB5001R and 2JB5002R to enclose the data logger unit 1/2UU-RH0500 and modern.
3. Replace field wiring.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the system is to be used for temperature monitoring the RH/RY piping. The mounting of the RTDS, cables and enclosures (junction boxes) have been evaluated. The addition of the RH/RY TMS cannot affect normal operation or design basis functions of the RH/RY systems. The field wiring is designed in accordance with electrical separation and seismic requirements and therefore no new failure modes would result from this new wiring. This change has been designed such that the attachment of the temperature elements to each pipe and the wiring will not affect the structural integrity of the RH/RY or any components that are part of the RCS pressure boundary, or any associated RCS piping. The change does not affect plant operations of any existing modes because this design does not change the function of the RH/RY systems.

6G-98-0156, Rev. 2
DCP 9900022 & 9900023
(Cont'd.)

The temperature monitoring system is intended to reduce the probability of a catastrophic pipe failure, by detecting minor operating anomalies before a major problem occurs. By use of the temperature monitoring system, the System Engineers can make trend changes in operation, especially startup and shutdown and make informed decisions to effectively schedule inspections and maintenance. The new system has been designed such that the attachment of the temperature elements to each RY Piping (Seismic Calculation 7.16.10-BYR-98-142) and the electrical installation, will not affect the integrity of the SSCs. Thus, this change does not increase the probability or consequences of an accident occurring.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because this design change does not adversely impact UFSAR accident-related systems structures, and components (SSCs). The addition of the RH/RY monitoring system does not introduce any adverse interactions between any SSCs.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the addition of the RH/RY monitoring system does not affect any parameters upon which Technical Specifications are based.

6G-99-0050
DCP 9900026

DESCRIPTION:

The purpose of this Design Change was to add C02 ground differential relays to protect SAT/UAT at GND fault on 4.16/6.9kV side.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because SAT/UAT neutrals are grounded through a limiting current resistors. Therefore, in case of a ground fault, current in the circuits will be limited proportionately and may not be sufficient to activate existing phase to phase fault monitoring differential over current relays. This condition may damage the SAT/UAT Buses or may develop in a serious fault, extensively damaging the affected equipment. This modification adds ground differential over current relays to protect the equipment at a fault magnitude lower than phase to phase fault. Still their functions and failures are not different than the existing protective scheme and would not prevent safe shutdown of the reactor or safety SSCs from performing their safety functions. This modification does not interface with any fluid, ventilation and radiation systems components such as piping, valves, pumps, dampers, fans, and rad monitors etc. Therefore consequences of an accident resulting in an off-site dose does not increase and the probability of occurrence of an accident or a malfunction of equipment important to safety previously evaluated in the SAR does not increase.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because added relays, CTS and the protective circuits components are of the same high quality as previously used. Their functions and failures are not different than the existing circuit components. Combustible loading of the added cables in affected Fire Zones is acceptable and documented per NEP 04-07. New non-safety related panel to be added in AEER will be mounted using approved seismic mounting details. This modification has no interface with any fluid, ventilation, and radiation detection and monitoring systems. Therefore possibility for a malfunction or an accident of a different type than any evaluated previously in the SAR is not created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this modification does not impact any existing Tech. Specs and the associated surveillances. (Refer Tech. Specs. 3.8 and Bases for 3.8). Therefore, the margin of safety as defined in the basis for Tech. Specs is not reduced.

6G-99-0042
DCP 9900083

DESCRIPTION:

The purpose of this Design Change was to remove abandoned flow element 1FE-MS257 from the 10" second stage reheat steam line to the Moisture Separator Reheater. This annubar flow element was added by modification M6-1-89-013 to assist in MSR effectiveness testing. At that time, the performance of the MSR was not meeting expectations due to excessive moisture carryover. The annubar was later abandoned and classified as an "installed spare". The connection between the header line, 1MS04BC-10" and the annubar is leaking. The annubar will be removed and a blind flange will be installed. This will eliminate the steam leak problem.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because Steam System Piping Failures and High Energy Line Break events result from failure of the steam system pressure boundary. The blind flange to be installed in place of the flow element 1FE-MS257 meets the design requirements of line 1MS04BC-10" and is designed to maintain the system pressure boundary. In addition, the blind flange weight is essentially offset by removal of the annubar. This ensures that the structural integrity of the piping system is maintained. Consequences of a small steam system piping failure are bounded by a failure of a major main steam line per section 15.1.5 of the UFSAR. Line 1MS04BC-10" is located on the east side of the Turbine Building away from safety related equipment and openings into the Auxiliary Building. The function of equipment important to safety would not be degraded even if the blind flange were to fail. Removal of the annubar reduces the probability of the annubar failing and entering the MS System.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the function of the Reheat Steam Line is not changed and new system interactions are not created. There are no new failure modes and existing failure modes are not changed. The new blind flange provides a simple pressure boundary function for the main steam system. There is no new malfunction associated with this item. The annubar being removed was not used for several years and was never relied upon to perform a safety function.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the function of flow element 1FE-MS257 is not reflected in the basis of any Technical Specification and the function of the Main Steam System is not affected.

6G-99-0067
DCP 9900101

DESCRIPTION:

The purpose of this Design Change was to install a temporary support on line 1CC04AB-12" and valve 1CC158. The clamp/cable will provide a method of restraining the ¼" line to the header until weld repairs can be made. This section of line will remain in service during the time clamp/cable is installed. This line is the Unit 1 CC return header from the 1B RHR Heat Exchanger. This work activity will be done while Unit 1 is in modes 5, 6, or defueled. These contingencies will be in place in the event that the ¼" line fails. The installation of this assembly is a contingency to prevent further degradation of the weld connection.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the effects on the system as a result of the temporary repair was evaluated and determined to not constitute an unreviewed safety question. The engineering review of the evaluation included the determination that the temporary repair would not adversely affect the material strength and seismic requirements of the piping system. The installation of the temporary mechanical restraint assembly to prevent the rupture of a ¾" high point vent. Approved contingency actions will be in place to restore the system's pressure boundary.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the affected component/system are reviewed to determine the overall impact resulting from the temporary repair. If required LCOAR conditions can be established as necessary to ensure that the correct level of administrative control is applied to the temporary repair. The altered configuration of the system was evaluated and determined to not represent an unreviewed safety question.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because no Tech. Specs. are directly affected by this change.

6G-99-0068
DCP 9900102

DESCRIPTION:

The purpose of this Design Change was to install a linestop fitting to facilitate the installation of a linestop. This modification will also temporarily install a linestop in pipe 2HD14AA 18" which will serve as a shell side inlet isolation boundary. This pipe supplies water from flash tank 2A (2HD02TA) to Low Pressure Drain Cooler 21A (2CB01AA). Once repairs are completed to the Drain Cooler, the linestop fitting will be removed. The final plant configuration will include a linestop nozzle, completion plug, and a blind flange.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the installation is qualified for the design pressures and is not expected to cause a loss of condenser vacuum. During the linestop installation, the condenser can be isolated via manual valves. The linestop equipment is designed to be installed to provide positive isolation on the inlet side and act as a boundary to prevent a loss of condenser vacuum. Systems required for a turbine or reactor trip are not affected and the consequences of the trips remain the same.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the only transient possible would be a loss of condenser vacuum. No other credible accidents or transients are created. Failure of the linestop equipment could cause a plant trip due to loss of condenser vacuum followed by a Turbine Trip. This would be similar to a failure of any piping or other failures that could cause a loss of condenser vacuum.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because no Technical Specifications are affected by the change or temporary condition.

6G-99-0098
DCP 9900161 & 9900162

DESCRIPTION:

The purpose of this Design Change was to install new vent lines 1SIJ4A-1, I SIJ5A-1, 1SIJ6A-1, 2S1J4A-1, 2S1J5A-1 and 2S1J6A-1 on existing Safety Injection Lines 1S134A-8, 1S105AB-8, 2S134A-8 and 2S105AB-8. Gate type vent valves 1S1096, 1S1097, 1S1098, 2S1096, 2S1097 and 2S1098 are installed on the vent lines. The vent lines and valves will be installed using a hot tap technique that permits installation in an operable pressurized line.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the installation of the vent lines and valves does not degrade the integrity of the ECCS pressure boundary and does not reduce the capability of the ECCS system to perform its design accident mitigation function. The new vent valves will facilitate keeping the ECCS piping full of water thus reducing the possibility of ECCS equipment malfunctions due to water hammer, cavitation or pumping of non-condensable gases into the RCS. As a result, the probability and consequences of an accident or malfunction of equipment important to safety is not increased by this change.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the completed installation of the vent lines and valves will satisfy all applicable Asme Code design requirements. The new piping and valves have the same potential failure mechanisms as the existing system piping and valves. The modification does not create the possibility of an accident or malfunction of a type different from those evaluated in the SAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the installation of the vent lines and valves maintain the pressure boundary integrity of the ECCS system in accordance with Asme Code design criteria while not creating any new operating limitations or failure mechanisms for the system. The vent lines and valves are intended to support maintaining a water filled system consistent with the requirements of LCO 3.5.2, 3.5.3 and Surveillance Requirements 3.5.2.3. Therefore, the margin of safety associated with the Tech. Specs. are not reduced.

6G-99-0110
DCP 9900207

DESCRIPTION:

The purpose of this Design Change was to modify the Nonessential Service Water (WS) pump (0WS01PA, 0WS01PB, and 0WS01PC) upper motor bearing cooling and pump seal supply lines. The previous installation had the supply lines fed from the WS pump discharge header, downstream of the WS strainers. The supply lines, which branched into individual supplies to each WS pump, were cut and capped. New supply lines were routed from the discharge of each WS pump, through a strainer and a cyclone separator, before splitting to supply the motor bearing coolers and pump seals. New drain lines were added for each strainer and cyclone separator. There is no adverse effect on the operation of the WS system, as the change enhances the cooling flow to the motor cooler and pump seal.

SAFFTY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the WS System provides heat removal for the balance of plant equipment. No equipment important to the safe shutdown of the reactor is served by the WS System. The WS system is not safety-related since all essential loads, and those required for the safe shutdown of the reactor, are served by the Essential Service Water (SX) System. Accordingly, the WS system is designated Safety Category 11, Quality Group D. None of the loads served by WS affect the plant.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the piping system changes are confined to the Circulating Water Pump House (CWPH). None of the SSC in the CWPH is capable of creating an accident or transient that would impact nuclear safety or the safe shutdown of the reactors. In addition, none of the WS system loads affect the plant safety.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because Tech. Specs. are not affected as the WS system provides heat removal for the balance of plant equipment. No equipment important to the safe shutdown of the reactor is served by the WS System. The WS system is not safety-related. The WS system is designated Safety Category II, Quality Group D. None of the loads served by WS affect the safety of the plant.

6G-99-0149
DCP 9900230

DESCRIPTION:

The purpose of this Design Change was to install a linestop fitting to facilitate the installation of a linestop. This modification will also temporarily install a linestop in pipe 2HD14AB-18" which will serve as a shell side inlet isolation boundary. This pipe supplies water from flash tank 2B (2HD02TB) to Low Pressure Drain Cooler 21B (2CB01AB). Once repairs are completed to the drain cooler, the linestop fitting will be removed. The final plant configuration will include a linestop nozzle, completion plug, and a blind flange.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the installation is qualified for the design pressures and is not expected to cause a loss of condenser vacuum. During the linestop installation, the condenser can be isolated via manual valves. The linestop equipment is designed to be installed to provide positive isolation on the inlet side and act as a boundary to prevent a loss of condenser vacuum. Systems required for a turbine or reactor trip are not affected and the consequences of the trips remain the same.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the only transient possible would be a loss of condenser vacuum. No other credible accidents or transients are created. Failure of the linestop equipment could cause a plant trip due to loss of condenser vacuum followed by a Turbine Trip. This would be similar to a failure of any piping or other failures that could cause a loss of condenser vacuum.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because no Technical Specifications are affected by the change or temporary condition.

6H-99-0310
DCP 9900258 & 9900259

DESCRIPTION:

The purpose of this Design Change was to eliminate the wiring for the Close Intercept Valve (CIV) function associated with the Turbine Digital Electro-Hydraulic Computer (DEHC). The change will be accomplished by lifting and abandoning the field cable conductors at 1/2MS01JG and 1/2PA22J. The CIV function is inhibited previously, so eliminating the associated wiring does not affect operation of the plant or turbine controls or protection.

The design change also revises the Load Drop Anticipator (LDA) circuitry associated with the DEHC Overspeed Protection Controller (OPC) to eliminate the LDA signal from initiating closure of the Turbine Governor Valves (GVs) and Intercept Valves (IVs). The change was performed by internal wiring changes to the DEHC (1/2PA22J). The LDA function anticipates a potential overspeed that could result from opening of the generator output breaker at load. However, the Generator protection logic initiates a Turbine Trip on loss of load. Therefore, the LDA function to reduce steam flow by shutting the GVs and IVs is overridden by the Turbine Trip initiated on loss of load. Therefore, eliminating LDA will have no affect on turbine operation or protection.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the change does not involve or interface with any systems, structures, or components important to safety. The activity does not alter the initial conditions used in the UFSAR analysis for any accidents. The components involved do not initiate any accidents or mitigate the consequences of any accident analyzed in the SAR. The function of any system, structure or component important to safety is not altered by this change.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because this design change does not adversely impact UFSAR accident-related systems, structures, and components or SSCs. The change does not introduce any adverse interactions between any SSCs.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which Technical Specifications are based.

6G-99-0240
DCP 9900289 & 9900290

DESCRIPTION:

The purpose of this Design Change was to increase Motor Operated Valves (MOV) 1/2SI8802A and B, Safety Injection (SI) to the Hot Legs Isolation Valves (HLIV), overall actuator gear ratio to increase the operator motor gearing capability.

The ComEd pressure locking prediction methodology was used to determine the required force to unseat the valves under the postulated scenario using conservative assumptions. As documented in pressure locking calculation BRW 96-015 / BYR96-238, an overall actuator gear ratio of 59.4-1 provides the required minimum margin (capability) to open the valve under the postulated scenario while not exceeding the maximum valve stroke time listed in the UFSAR. This design change will increase the stroke time of the valve from a range of 6.4 to 7.8 seconds to approximately 13.5 seconds. The UFSAR maximum stroke time is 15 seconds.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the valve stroke time increase will not affect the ability of the valve to isolate for containment isolation because the valve is normally de-energized closed and the valve will still close within the required UFSAR maximum stroke time of 15 seconds. The activity has no impact on the Reactor Coolant Pressure Boundary or main steam piping nor could these valves cause a fuel handling accident, therefore, the change will not increase the probability of occurrence of any accident or transient.

   The actuator gear ratio change will not increase the consequences of any accident because the change will not affect the function of the valve. The change will give the valve more thrust capability to open and close. The maximum operating stroke time for ECCS motor operated valves up to 10 inches in size is given in UFSAR Table 6.3-1 as 15 seconds. The new stroke time for valves 1/2SI8802A and B will be 13.5 seconds. The new stroke time is less than 15 seconds and therefore, the consequences of a LOCA are not increased.

   For a Fuel Handling Accident inside Containment and for a Steam Piping Failure, the function of the 1/2SI8802 valves is to maintain containment integrity. Since these valves will be closed and will stay closed during the Fuel Handling Accident and Steam Piping Failure Accident, the consequences of these accidents are not increased.

6G-99-0240
DCP 9900289 & 9900290
(Cont'd.)

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the change will not create the possibility of a different accident or transient because the function of the valve and operator is not being changed. The ability of the valve to perform its function is not being changed.

   Valve setup will not be changed with installation of the new overall actuator gear ratio. The function of the valve is also not changed. The 7 second delay initiation of hot leg recirculation based on the slower valve stroke time will not affect this function. The new stroke time remains less than the UFSAR maximum stroke time (15 seconds) for these valves. The activity will not adversely affect other SI or Reactor Coolant System components. Therefore, the change will not create the possibility of a different type of equipment malfunction than previously evaluated.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because no Technical Specifications or Bases are affected by the activity.

6G-99-0197
DCP 9900339, Rev. 1

DESCRIPTION:

The purpose of this Design Change was to provide an alternate detail for installation of a threaded seal plug in the closure cap at the top of the Control Rod Drive Mechanism (CRDM) rod travel housing. This seal plug will be inserted into a threaded opening that is intended for the installation of a lifting eyebolt or a venting tool. The seal plug will be seal welded after it is threaded into the cap. The seal plug will have no effect on the operation of the associated CRDM or of the control rods. If there is ever a need to Install the venting tool or the eyebolt, the seal weld will have to be removed and the seal plug unthreaded from the opening. Industry experience has shown that leakage can develop past the tapered vent plug installed in the closure cap. The leakage of borated Reactor Coolant is undesirable and, over time, could lead to damage of other components on the CRDM assembly. In addition, limits are placed on Reactor Coolant System leakage by the Technical Specifications.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the seal welded seal plug provides an additional boundary against reactor coolant leakage past the vent plug seal. Therefore, the probability of a decrease in reactor coolant inventory is slightly reduced. The seal plug does not adversely affect the operation of, or create affect failure modes of, the Rod Cluster Control Assemblies (RCCAs) or of the Control Rod Drive Mechanism (CRDM). The seal plug seal weld is not a structural weld and does not affect the strength of the threaded joint. The seal plug joint was analyzed by Westinghouse to ensure that joint integrity will be maintained and the seal plug will not be ejected due to RCS pressure forces. Therefore, the probability of a malfunction of an RCCA or ejection of a RCCA component is not increased. There is no credit taken for venting the reactor through the CRDM rod travel housing top cap in any accident scenario. Any reactor coolant leakage past the vent plug and / or the new seal plug is bounded by the large loss of coolant accident described in the UFSAR. Therefore, installation of the seal plug and seal weld will have no impact on the consequences of these accidents.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the vent path through the rod travel housing top cap Is normally sealed with a vent plug during operation. In case of leakage past this vent plug seal, a secondary seal can be added to provide the same function as the vent plug. The secondary seal In the form of a seal plug and seal weld will perform the same function as the original equipment and will not interfere with activities performed when the reactor is operating.

6G-99-0197
DCP 9900339, Rev. 1
(Cont'd.)

With the unit shutdown, venting through the top cap vent plug can be accomplished with the added difficulty of first removing the seal weld and the seal plug. This will not lead to an accident or transient of a different type since any required venting can be performed as needed.

The functions and failure modes of the new seal plug and seal weld are similar to that of the vent plug. Malfunctions of these items are also similar. Since the new seal plug and seal weld are qualified for this application, there will be no new type of malfunction.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because no Technical Specifications, Technical Specification Bases or Surveillances are affected by the addition of the seal plug and seal weld. There is no adverse effect on the RCS pressure boundary.

6G-98-0032
DRP 7-142

DESCRIPTION:

The purpose of this UFSAR change was to provide clarification to the current (UFSAR) description of the Residual Heat Removal (RHR) System's Suction Relief Valves (i.e., 1/2RH8708A/B) relief capacities. The relief capacity of the valves described on page 5.4-39,Revision 5 should be more appropriately listed as "minimum" values. The change is a administrative clarification and does not perform any physical changes to the plant.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the change only provides clarification to the words contained in the UFSAR defining the "minimum" relief capacity of the RHR System relief valves. The numbers stated in the UFSAR are the functional requirements prescribed by the system design. The values are confirmed by Westinghouse letter SD/SA-M-258, dated May 9, 1978, Attachment 1 (Westinghouse Proprietary Class 2) and confirm the safety system is capable of meeting the system design. No physical changes are being made to the system.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the change only provides clarification to the words contained in the UFSAR defining the "minimum" relief capacity of the RHR System relief valves. No physical changes are being made to the plant nor are changes made to the manner the plant is operated. Therefore, the possibility of an accident or malfunction not previously evaluated does not exist.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because no Technical Specifications were impacted by the changes.

6G-98-0106, Rev. 1
DRP 7-166

DESCRIPTION:

The purpose of this UFSAR change was to change Section 5.2.5.8 valve numbers (1/2RC8069A/B) which are the inner and outer Isolation Valves as part of the Reactor Vessel flange o-ring system. The flange leakoff system is designed to monitor RCS leakage past the two independent sets of o-rings that form part of the RCS boundary during normal operation. Plant operators are directed by plant procedures to manipulate these valves in the event that there is leakage past the o-rings. This manipulation would allow the leakage to be quantified and corrective actions initiated. The activity revised the UFSAR description of the sequence in which the valves are manipulated to reflect the original design function of the system. The change does not physically change or alter the valves or their function.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the activity did not impact any accidents or malfunctions previously evaluated in the SAR. The change did not alter the function of the valves. The valves are not credited as part of the RCS pressure boundary or for mitigation of any Design Basis Accidents. Therefore, the probability or consequences of any analyzed accident are not increased by this change.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because this change did not change the intent or function of the valves. The valves are not assumed to function during or after a Design Basis Accident and the potential failure or mis-manipulation of these valves would not lead to any new or different accident scenarios. Therefore, there is no possibility for a new or different type of accident.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this activity does no affect any parameters upon which plant Technical Specifications are based.

6G-98-0215
DRP 7-167

DESCRIPTION:

The purpose of this UFSAR change was to revise the relative humidity of the Exhaust Air from the Fuel Handling Building may be greater than 70% and revises the description for the Fuel Handling Building charcoal filter efficiencies and laboratory testing.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the UFSAR changes have no affect on normal system operation or initiation of the system to route the exhaust through the charcoal absorbers. The change in calculated relative humidity of the exhaust air affects the rated efficiency and laboratory test requirements of the charcoal absorbers. These UFSAR changes have no affect on plant operation because the revised efficiencies and test criteria were already used in the plant accident analysis and test procedures. The higher maximum relative humidity in the FHB has no adverse impact on equipment important to safety located in the FHB. Electrical equipment located in the FHB is designed to operate in the higher humidity environment.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the UFSAR changes are within the design capabilities of the Fuel Handling Building Exhaust System (FHBES), and describe the assumptions used in the accident analysis and SER. The FHBES will not be operated in a different manner such that the possibility of a new accident or malfunction will be created. Thus, the UFSAR changes do not create the possibility of an accident or malfunction of a different type from those previously evaluated.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the UFSAR changes are consistent with the accident analysis assumptions on charcoal filter efficiencies.

6G-98-0277
DRP 7-177

DESCRIPTION:

The purpose of this UFSAR change was to revise the UFSAR to indicate the use of the Emergency Personnel Hatch for access to and from the Containment Building during Modes 1, 2, 3, and 4 of plant operation. Currently, UFSAR section 6.2.6.2 inaccurately describes the Emergency Personnel Airlock as solely used for emergency egress from containment. UFSAR section 3.8.1.1.3.5.1 refers to the Emergency Personnel Airlock as "an Emergency Escape Hatch". Station personnel have been using this airlock as a means of Containment entry since at this location personnel are shielded from the neutron dose.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because use of the personnel air lock does not interact with the nuclear fuel or fuel handing equipment, nor are any reactor coolant boundaries or steam piping affected. Therefore, there is no increase in the probability of an accident resulting in a radioactive release inside the Containment Building. Administrative procedure control access to the containment during all modes of plant operation. The procedure requires at least one airlock door closed at all times to ensure containment integrity and restrict radioactive material release from containment. Additionally, the mechanical interlock between the airlock doors ensures that only one door is open at a time. During refueling outages, when core alterations or irradiated fuel movement inside containment are not in progress, the airlock can be opened, in compliance with Technical Specifications, since there is not potential for a LOCA, steam piping failure or fuel handling accident. Therefore, the consequences of an accident will not be increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the use of the Emergency Personnel Airlock during all modes of operation does not initiate any accident which result in the release of radioactivity inside containment nor introduce any new failure mechanisms. The design of the airlock doors ensure that only one door can be opened at a time. Therefore containment integrity is maintained as required by the Technical Specifications. The change to the UFSAR is not in conflict with any Technical Specifications.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change to the UFSAR does not affect any parameters upon which Technical Specifications are based.

6G-98-0283
DRP 7-180

DESCRIPTION:

The purpose of this UFSAR change was to revise the description of the performance analysis of the Residual Heat Removal (RHR) System and design of the RHR Heat Exchanger in Section 5.4

The change is being made to resolve questions from the 1997 Byron RHR SSFI. UFSAR Sections 5.4.7.1, 5.4.7.2.2, Tables 5.4-7 and 5.4-8 provide information that implies the RHR Heat Exchanger design is based on the heat load and temperatures that exist 40 hours after the reactor is shutdown. The RHR Heat Exchanger Design is based on selection of a heat exchanger large enough to support the desired cooldown time of less than 36 hours with two cooling trains running. The heat exchanger heat removal capacity and operating conditions were then back-calculated using the selected heat exchanger UA value and assumed inlet temperature that would be representative of the shutdown conditions. The heat removal capacity and inlet and outlet temperature for the RHR Heat Exchanger listed in UFSAR Table 5.4.8 are for an assumed inlet temperature of 105° F and a RCS inlet temperature of 137° F. Additionally the UFSAR incorrectly indicates that the decay heat generation at 40 hours after shutdown is 29.95 X 10E6 BTU/hr. The decay heat input used in the RHR performance analysis is 60.5 X 10E6 BTU/hr at 40 hours after shutdown.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the changes to the UFSAR description of the performance evaluation of the RHR System and design of the RHR Heat Exchanger do not increase the probability of an accident. The UFSAR changes do not adversely impact any SSC that are involved in the initiation of any accident. The consequences of any accident are unchanged. The capability of the RHR System to deliver water and remove accident heat is unchanged. The changes reflect the actual design basis of the RHR System and RHR Heat Exchanger and have no impact on the normal plant cooldown. A confirmatory analysis using the method provided in NRC Branch Technical Position ASB 9-2 was performed to validate the Residual Heat Decay Values used in generating the UFSAR Cooldown Curves.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the UFSAR changes do not change the function or design basis for any system or component. Therefore, new accidents or malfunctions are not credible.

<u>6G-98-0283</u>
<u>DRP 7-180</u>
<u>(Cont'd.)</u>

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the changes to the UFSAR do not affect any parameters upon which Technical Specifications are based.

6G-98-0071
DRP 7-185

DESCRIPTION:

The purpose of this UFSAR change was to remove outdated references to specific amounts of electrical cable inventories within Containment (i.e., Table 6.1-5), relying instead on an updated calculation (i.e., 3C8-1280-001) to track the current values. Additionally, to correct reference in UFSAR Section 7.3.1.2.4.e.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because removing outdated information from the UFSAR Table will have not impact on either the probability nor consequences of an accident or malfunction of equipment important to safety. The accidents impacted by the outdated information (i.e., those Design Basis Accidents that require initiation of the containment spray system) are addressed independently by calculation 3C8-1280-001. This calculation addresses hydrogen generation as the result of the interaction between caustic solution from the containment spray system with the electrical cable insulation. The calculation is updated as the electrical systems and associated insulation inventories are modified.

   Additionally, the correction to reference to UFSAR Section 7.3.1.2.4.e will have no impact on either the probability of occurrence or the consequences of a accident or malfunction of equipment important to safety.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the administrative changes do not create any new accidents or malfunctions of a type different than those evaluated in the SAR. The SAR requires the determination of the amount hydrogen generated a the result of failure of a cables during a Design Basis Accident and the current analyses remain bounding. The analyses demonstrate that the equipment meets all Design Basis requirements.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which Technical Specifications are based.

6G-98-0074
DRP 7-190

DESCRIPTION:

The purpose of this UFSAR change was to update Section 9.5.8, to clarify the impact of a System Auxiliary Transformer (SAT) fire on the Emergency Diesel Generator (EDG) operability. The current discussion in the UFSAR implies the SAT deluge system provides a function to support EDG operability. Analyses have determined that, independent of the SAT deluge system, the environmental conditions necessary to transport SAT fire combustion products to the EDG air intakes would dilute the concentration of these non-combustibles such that EDG would be unaffected.

In addition, the discussion of the physical orientation of the EDG air intakes on the Auxiliary Building roof incorrectly implies that the A Train EDG intake is nearest to the SATs. This discussion is being eliminated to correct the implication.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because correcting misinformation in the UFSAR will have no impact on either the probability nor the consequences of an accident or malfunction of equipment important to safety. The intention of the DRP is to ensure the description of the plant is correctly reflected in the SAR and to eliminate potential confusion with respect to the relationship between the SAT deluge system and the EDG. The DRP does not involve any physical change to the plant and does not impact the EDGs ability to mitigate the consequences of an accident.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the DRP primarily focuses on ensuring the SAR sections accurately reflect the as-built configuration of the EDGs and support equipment. No new administrative controls are introduced nor physical changes made to the plant. The operation of the EDGs are not impacted in any manner and the ability for the EDGs to mitigate an accident in the event of loss of offsite power is unchanged. Therefore, there is no possibility to create an accident or malfunction of a different type than was previously analyzed.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change did not affect any parameters upon which Technical Specifications are based.

6G-98-0136
DRP 7-229

DESCRIPTION:

The purpose of this UFSAR change was to enhance the description of the Loose Parts Monitor (LPMS) System description. The activity upgraded the current non-safety related LPMS. The change only affected LPMS equipment in Auxiliary Electric Room Cabinet. No additional changes are being made with this change. This change was evaluated due to the explicit description of the system in the UFSAR, which required update to match the change. The change updated equipment that is now obsolete and enhanced the overall capability of the system to identify and locate potential loose parts. The change also allowed for immediate analysis of data, which previously was only available offsite.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because there are no accidents or anticipated transients which are affected by this change. In addition, there are no accidents which require the LPMS to function or are dependent upon the LPMS to prevent or mitigate the accident. The change does not adversely impact any of the functions of the system. Therefore, there is no increase in the probability or consequence of any accident previously evaluated in the Safety Analysis Report.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because this change does not impact the RCS or the LPMS functions so as to create the possibility of an accident or malfunction of a type different from those previously evaluated in the Safety Analysis Report. This change merely updates and refurbishes the current LPMS. All required functions of the LPMS are maintained by the replacement and there are no new failure modes or malfunctions identified.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this activity was evaluated against plant Technical Specifications and it was determined that there was no adverse affect on any Technical Specifications.

6G-98-0151, Rev. 1
DRP 7-241

DESCRIPTION:

The purpose of this UFSAR change was to reflect the addition of a Vibration Monitoring (VMS) System to the Byron Unit 1 and Unit 2 Reactor Coolant Pumps (RCPs). The modification provided local indication of RCP shaft vibration levels and has the ability to store vibration data for each RCP for diagnostic evaluation, analysis, tracking and trending. Three seismically qualified non-contact proximity probes will be installed for each pump on the 3 seal housing (two for shaft displacement and one for shaft speed). Four additional accelerometers are to be installed to monitor the upper and lower bearings. The modification included the installation of junction boxes, conduit, supports, and wiring. Part of the safety evaluation for the modification was included in Safety Evaluation, 6G-98-0156, RH/RY Temporary Alteration since the RCP monitoring equipment will use some of the same cable routing and will therefore include the same cable separation criteria as the previously evaluated activity.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the change did not affect any offsite power, control, instrumentation or protection schemes. The power requirement of the equipment has been evaluated to be within the capacity of the associated buses. Fault coordination between breakers is not affected.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the change has been appropriately analyzed with regard to impact on associated electrical distribution systems and all equipment is seismically mounted. Structural integrity of all associated systems is unaffected.

3. The margin of safety as defined in the bases for any Technical Specification is not reduced because the activity does not change Technical Specification Bases parameters.

6H-98-0196
DRP 7-253

DESCRIPTION:

The purpose of this UFSAR change was to clarify the following UFSAR sections concerning Emergency Diesel Generator (EDG) operation:

8.3.1.1.2.2 Clarifies the EDG operation with respect to the responsibilities of the Nuclear Station Operator (NSO) in the Main Control Room and the Equipment Operator (EO) locally at the EDG.

9.5.5 Revises Jacket Water (JW) Cooler and system performance data based on performance data received from Cooper-Bessemer reflecting currently installed JW coolers.

9.5.6 Revises the incorrectly stated size of the Starting Air System Receiver to the correct size.

9.5.7 Revises Lube Oil (LO) Cooler and system performance data based on performance data received from Cooper-Bessemer reflecting currently installed LO coolers.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the activity corrects configuration, design, and performance data and clarifies station operational policies within the UFSAR. No changes are being made to any operational parameters associated with the EDGs under all operating modes. The EDGs remain capable of performing the designated safety functions.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the changes do not impact the reliability of the EDGs in any manner. No new failure mechanisms or modes are introduced for the EDGs and its support equipment/subsystems as a result of the changes.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which Technical Specifications are based.

6G-98-0177, Rev. 1
DRP 7-258

DESCRIPTION:

The purpose of this UFSAR change was to reflect the addition of restriction orifices to the Essential Service Water (SX) System blowdown lines to ensure that blowdown flow is limited to analyzed acceptance levels in the Byron Ultimate Heat Sink (UHS) Design Basis analyses. The restriction orifices were installed in the 6 inch blowdown lines and are sized to limit SX blowdown flow to less than 600 gpm under all conditions.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the restriction orifices were installed to perform the same design function as previously installed throttled Isolation Valves. The new orifices removed the potential for mechanical failures or human errors in ensuring there is not excess blowdown flow from the UHS following a Design Basis Accident. The installation relocated the safety/non-safety boundary to assure system integrity is always maintained and would not initiate any Design Basis Accident. In addition, this portion of the SX System performs no function to mitigate the consequences of an accident. Therefore, the probability or consequences of any accident previously evaluated in the Safety Analysis Report are not increased by this activity.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the activity added no new system interactions or functions. The activity did not remove any existing system interactions or functions. The ability of the SX System and the UHS to perform their design functions was unaffected by this change. Therefore, the change did not adversely impact systems or functions so as to create the possibility of an accident or malfunction of a type different from those previously evaluated.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this activity was evaluated against plant Technical Specifications and it was determined that the Technical Specifications provide a margin or safety. The change was implemented to help maintain that margin of safety.

6H-99-0184
DRP 7-272

DESCRIPTION:

The purpose of this UFSAR change was to review the Charcoal Filter Unit HVAC System for the Volume Reduction (VR) System. The charcoal unit will no longer be used, tested, or maintained due to non-use of the VR System. The evaluation was performed to amend the UFSAR to reflect current non-use of the unit.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because Rad Waste Volume Reduction System (VRS) is not being used at Byron and Braidwood. The change is to revise the UFSAR to indicate the Charcoal Filter HVAC System for VRS would not require to be used, tested, or maintained due to non-use of VRS.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the non-use of VRS at Byron and Braidwood had been incorporated into UFSAR revision previously. The change is to indicate that the Charcoal Filter Unit HVAC System for the VRS would not be required since the VR System will not be used at Byron.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the VR System is not used at Byron. Thus, the Charcoal Filter Unit HVAC System for the VRS would not be required to be tested, used or maintained.

6G-98-0304
DRP 8-014

DESCRIPTION:

The purpose of this UFSAR change was to revise the following UFSAR sections:

- Revise UFSAR Section 6.2.6.1 changing the nominal Type A test pressure from "45 psig" to "48 psig for Unit 1 and 45 psig for Unit 2".
- Revise UFSAR Table 6.2-58 to indicate the normal position of valves 1CC9437A/B is "N/A".
- Revise UFSAR Table 6.2-58 to indicate the valves 1PR033A,B,C,D and 1PR002E,F are inside containment versus outside containment.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the changes have no impact on either the probability or the consequences of an accident or malfunction. The higher test pressure is an administrative correction to a previously analyzed containment pressure response for the replacement Steam Generators. The design functions for the revised CC and PR Valves remain unchanged as well as the positions of the valves during an accident.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because there are no accidents impacted by the administrative changes. There are no accidents of a different type than previously evaluated created by the changes.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the changes to the UFSAR do not affect any parameters upon which Technical Specifications are based.

6G-98-0296
DRP 8-019

DESCRIPTION:

The purpose of this UFSAR change was to update "UFSAR" Figures 5.2-5, 9.3-6, 9.3-7, 9.3-8, 9.3-9, and 11.2-8 to reflect the "As-Built" plant configuration. Drawings A-701, A-703, A-704, and A-705 are issued as "For Record" drawings. P&ID Drawings M-48 Sheet 7, M-70 Sheet 2, M-82 Sheet 1, and M-82 Sheet 5 are revised to reflect the "As-Built" plant configuration.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because Auxiliary Building and Containment drain lines are not water sources for a design basis flooding event. The drain piping is not physically rerouted by this change, so this change will not impact other systems, which are considered sources of flooding. Therefore, the drain piping cannot affect the probability of Auxiliary Building or Containment flooding. Based on review of the Auxiliary Building and Containment Flooding Analysis, updating the affected drawings will not impact the consequences of a flooding event or impact off-site dose.

   Auxiliary Building and Containment flood levels are not impacted by the drawing changes. Existing piping single line drawings are not impacted. Therefore, the probability of a malfunction of equipment important to safety is not increased. Since Containment and Auxiliary Building Flood Levels are not impacted by this change, the consequences of a Containment or Auxiliary Building pipe rupture and resultant flooding event is not increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the floor and equipment drain drawing changes do not adversely impact the WE, WF, RE, or RF Systems. The piping systems remain seismically qualified as originally designed. No physical in-plant piping changes result from this change. No new openings in walls or floors are created. Therefore, this change does not create the potential for a different type of flooding event or new type of accident.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which Technical Specifications are based.

6G-99-0028
DRP 8-032

DESCRIPTION:

The purpose of this UFSAR change was to remove setpoint specific information (225 steps) and replace it with text that describes the function (indications that shutdown rod not fully withdrawn). This cycle specific value is an input to the "Rod Off Top" alarm.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the probability of an accident or malfunction is not increased because the alarm setpoint is passive and does not affect control or operation of the Rod Cluster Control Assemblies (RCCAs). Assuming the accident has occurred, the alarm setpoint will still fully function to detect and alarm any condition where a shutdown rod is not fully withdrawn. The alarm does not affect the consequences of any of the rod misoperation transients.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the alarm setpoint is passive and does not affect control or operation of the RCCAs. It does not affect the operation of any other system such that any new operating regimes would be created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the alarm still alerts the operator to a condition where a shutdown rod is not fully withdrawn with the same DRPI precision. Fuel failures do not occur when alignment maintained with 24 steps.

6G-99-0040
DRP 8-033

DESCRIPTION:

The purpose of this UFSAR change was to revise the Generating Station Emergency Plan (GSEP) Rev. 99-G1, Byron and Braidwood Annex to the Emergency Plan (99-BYR1 and 99-BYA1), and the UFSAR to implement a single centralized Emergency Operations Facility (EOF) at Downers Grove for all ComEd Generating Stations.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the location of the offsite emergency response facilities has no impact on equipment important to safety. The consequences of an accident or malfunction will neither increase nor decrease as a result of this move.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the location of the offsite emergency response facilities does not factor into any accidents or malfunctions. Moving the EOF to Downers Grove does not create any new accidents or malfunctions.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the emergency plan and offsite emergency response facilities are not referenced in the Technical Specifications. Moving the EOF to Downers Grove does not reduce the margin of safety. This revision has been evaluated in accordance with 10CFR50.54(g), and does not decrease the effectiveness of the emergency plan.

6H-99-0320
DRP 8-040

DESCRIPTION:

The purpose of this UFSAR change was to modifies the section of the UFSAR which describes Byron and Braidwood Stations implementation of the requirements of NRC Regulatory Guide 1.82, Rev. 0. Specifically, the response to Regulatory Position, Item 10 on page A1.82-3 of the B/B UFSAR is revised to account for the potential that the Containment Spray (CS) Nozzle orifices may not be the most limiting restriction in the systems drawing a suction on the containment recirculation sump during a Design Basis Accident. Also, the size of a particle capable of passing through the sump screen is updated.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the probability of any accident or equipment malfunction is not increased since this change has no effect on any of the initiating factors for any accidents. The consequences are not increased since the design function of the ECCS is maintained based on the physical properties of the debris generated, the available openings through the valves, the flowpath through two centrifugal pumps (Charging and Safety Injection) before reaching the throttle valves, and the high flow velocities at the restrictions within the valves.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because there is no change to any component/structure that could create a possibility of a different type of malfunction or accident. DRP 8-040 revises the UFSAR to reflect the potential configuration of the throttle valves in the ECCS injection lines. This change does not create the possibility of a different type of equipment malfunction.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the original documents that discussed compliance with NRC Regulatory Guide 1.82 Revision 0, indicated that the sizing of the recirculation sump screen was in compliance with the specific recommendations of the Regulatory Guide. That is, the screen openings were smaller than any opening in the systems served by the pumps that take suction from the Containment Recirculation Sumps. This information is currently documented in the UFSAR, Appendix A, section "NRC Regulatory Guide 1.82". Byron SER, Section 6.2.2 states "The applicant's sump design conforms to the guidelines in Regulatory Guide 1.82 except that the floor in the vicinity of each sump is level and does not slope gradually down away from the sump to assist in preventing heavier debris from accumulating at the sump". Additionally, another screen was required to be added (this is the existing outer screen) to achieve lower flow velocities.

6H-99-0320
DRP 8-040
(Cont'd.)

Thus, compliance with the requirements of Regulatory Guide 1.82 was part of the basis for NRC approval of the Byron/Braidwood Design. This compliance resulted in establishing an implicit margin of safety.

The analysis performed in support of this safety evaluation indicates that the change does not result in a discernible reduction in the margin of safety. The debris that would reach the valves through a torturous path (If it is not pulverized by the RH, SI or CV pumps) is either small and pliable or brittle, and would be swept through the valves. Thus, the design function of the valves is maintained.

The margin of safety is determined by the design and qualification of plant equipment, the operation of the plant within analyzed limits, and the point at which protective actions are initiated. T.S. 3.5.2 addresses "ECCS - Operating," and T.S. 3.5.3 addresses "ECCS - Shutdown." All assumptions made in the bases for these ECCS-related specifications are unaffected by the change to the UFSAR. ECCS Pumps remain fully operational, ECCS flow is unaffected, and the resultant accident mitigation consequences and associated margins of safety, specifically the acceptance criteria required by 10 CFR 50.46, are unchanged. There are no design changes or plant equipment performance parameter changes associated with this change. No setpoints are affected, and no change is being to plant operational limits as a result of this change.

Since this change does not result in a discernible reduction in the margin of safety, additional compensatory measures, beyond existing design features and procedural requirements are not needed. As discussed in the body of this evaluation, provisions are in place to control materials taken inside containment and inspections are performed to verify containment and sump cleanliness after outage activities. Additionally, the design of the containment sump and screens exceeds the minimum recommendations given in NRC Regulatory Guide 1.82 Revision 1. Any future physical change, which would result in different type materials added permanently to containment, will be evaluated in accordance with the provisions of 10CFR50.59.

6G-99-0086
DRP 8-043

DESCRIPTION:

The purpose of this UFSAR change was to delete the reference to the tie (interlock) between the low steamline input and the Loop Stop Isolation Valves in the description of inputs to the Steamline Isolation Function in Table 7.3-2 (page 7.3-68) "Instrument Operating Conditions For Isolation Functions". In addition, during the course of the review of the UFSAR, an incorrect reference was found on Figure 7.2-1 (Sheet 7 of 18) for continuation to the Safety Injection and Steamline Isolation function from a Low Steamline Pressure input. The sheet referenced for continuation is sheet 3, which is incorrect because sheet 8 is where the continuation of the signal is shown on the functional diagrams.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because since Byron and Braidwood were not licensed for N-1 Loop operation, the feature for inhibiting the low steamline pressure input for the affected loop based on loop stop valve interlocks was removed during initial construction. This change does not result in physical changes to the plant and is in accordance with the licensing basis (Technical Specification Section 3.4.17 requires the Cold and Hot Leg Loop Stop Isolation Valves to be open and de-energized while in Modes 1-4.) Also, the change of the reference sheet number to Figure 7.2-1 is considered an administrative change to reflect the proper continuation sheet. Based on the above, this change will not affect the ability of SSCs to perform their safety function in order to mitigate the consequences of an accident. Furthermore, this change has no impact on equipment operational parameters, failures, malfunction initiators or previously analyzed accidents therefore; the activity will not increase the probability of occurrence or the consequences of a malfunction of equipment important to safety.

6G-99-0086
DRP 8-043
(Cont'd.)

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the removal of the loop stop interlock tie is considered an omission from original construction because the plants were not licensed to operate with N-1 loops in operation where this feature would be desirable. Therefore, since this change does not result in physical changes to the plant and is in accordance with the licensing basis (Technical Specification Section 3.4.17 requires the Cold and Hot Leg Loop Stop Isolation Valves to be open and de-energized while in Modes 1-4), the activity will not create the possibility of an accident or transient of a different the than an previously evaluated. In addition, the change of the reference sheet number to Figure 7.2-1 is considered an administrative change to reflect the proper continuation sheet. Based on the above, this change will not affect the ability of SSCs to perform their safety function. Therefore, the activity will not affect equipment failures or malfunctions or create new failure modes.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because Technical Specifications 3.4.17 supports four loop operation and requires all the Cold and Hot Leg Loop Stop Isolation Valves to be open and de-energized while in Modes 1-4. Therefore, this change is consistent with the current licensing basis and the margin of safety is unaffected as defined by Technical Specifications.

6H-99-0371
DRP 8-051

DESCRIPTION:

The purpose of this UFSAR change was to revise UFSAR to clarify that in the event one of the RH Pumps cannot be realigned to the Hot Legs through MOV SI8840 at the time of Hot Leg switchover, flow from one of the SI pumps to the Hot Legs is adequate to preclude boron precipitation in the core.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the UFSAR changes will reduced the consequences of a malfunction. The changes were made in response to WOG guidance for emergency response procedures. The changes do not impact the probability of occurrence of an accident or malfunction, since they provide alternatives to an assumed failure.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the UFSAR changes provide guidance to respond to an assumed malfunction and do not affect the probability of a malfunction. The assumed malfunction was already evaluated in the UFSAR, however, the response to the malfunction was not clear.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the changes pertain to maintaining redundant flow paths following an assumed equipment malfunction. The changes actually increase the margin of safety by providing guidance for dealing with an assumed failure.

6G-99-0103
DRP 8-055

DESCRIPTION:

The purpose of this UFSAR change was to revise Section 6.2 to reflect the results of Westinghouse Calculation CN-CRA-99-018, "Byron/Braidwood Unit 2 MSLB Containment Response for the Original Steam Generators", Rev. 0. In effect, UFSAR Section 6.2 will reflect a peak containment air temperature from a Steam Line Break of 330.8° F for Unit 2.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the probabilities are not increased since this change has no effect on any of the initiating factors for a Steam Line Break Accident or Steam System Piping Failure, and the EQ temperature analysis remains valid. The consequences are not increased since the containment and other equipment used for mitigation remains below the design temperatures throughout the transient.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because there is no change to anything that could create a possibility for a different type of malfunction or accident. This is a change to the containment response to a MSLB inside containment. The reanalysis of the containment response to a SLB Accident does not create the possibility of a different type of equipment malfunction.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the margin of safety for Technical Specification LCO 3.6.5 is the difference between the design temperature of 280° F and the failure point of the containment structure and liner. No physical changes are being made to the containment structure or liner to affect the failure point. The reanalysis does not result in the 280° F design temperature being exceeded.

6G-99-0101
DRP 8-056

DESCRIPTION:

The purpose of this UFSAR change was to reflect Technical Specification Change (TSC Byron PR 98211 and Braidwood ITR 98-109). The change revises the Design Features Section 5.6.2, "Fuel Storage Drainage". Section 5.6.2 currently states that the Spent Fuel Storage Pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 423 feet 2 inches. This elevation will be revised to 410 feet 0 inches for both stations.

Change section 9.1.3.2, "System Description" suction line elevation location from four feet to approximately 6½ feet. Also, where the cooling water return line terminates above the stored Fuel Assemblies the word "approximately" will be added prior to 6 feet.

In addition Section 9.1.3.3; "Spent Fuel Pool Dewatering Protection" will be clarified to address the effects of both the SFP cooling and skimmer loop configurations.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because this change does not involve an increase in the consequences of an accident previously evaluated. As documented in NUREG-876, Byron SER, Section 9.1-3, page 9-5, the anti-siphon protection design of the SFP cooling and clean-up piping was reviewed and found to be acceptable stating that all connections to the spent-fuel pool are either near the normal water level or are provided with anti-siphon holes to preclude possible siphon draining of the pool water. This review is applicable to Braidwood as documented in NUREG-1002, Braidwood SER. The anti-siphon attributes employed in the SFP skimmer loops at Braidwood, (under consideration at Byron), are similar in design as well as their submergence levels previously evaluated for the SFP cooling loops. The changes revises the SFP inadvertent drain limit from approximately 423 feet to 410 feet to bound the failure effects of both the SFP cooling and skimmer loops, while considering any misoperation or failure scenario. The revised value meets the SRP acceptance criteria of maintaining at least 10 feet above the active fuel ensuring that adequate radiation shielding is maintained as previously analyzed. There is no physical or operational change being made which would alter the sequence of events, plant response or conclusions of the affected analysis. There is no change in the type or amount of any effluents released, and no change in either the onsite or offsite dose consequences as a result of this change.

   Therefore, based on this evaluation this change does not involve a significant increase in the probability or consequences of an accident.

6G-99-0101
DRP 8-056
(Cont'd.)

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because this change does not involve an increase in the probability or possibility of an accident previously evaluated. The initial conditions of the limiting dewatering incidents involve initiating circumstances/failures such as accidental gate openings, gate seal failures, or an open transfer tube. Specifying a revised inadvertent drain limit, which meets the SRP acceptance criteria, is unrelated to the probability of occurrence of the precursors or initiating events. These initiators are not affected by the SFP cooling or skimmer loop piping/component failure scenarios. There is no change being made to the approved design, nor is there any operational change being made which would increase the probability of occurrence.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this change specifically identifies the SFP level sufficient to ensure that the SRP acceptance criteria for inadvertent draining are met while accounting for the failure effects of both the SFP cooling and skimmer loops. The most limiting postulated SFP dewatering incidents involve SFP drainage to either a dry transfer canal, a dry transfer canal and cask fill area, or a dry transfer canal and cask fill area which additionally communicates through an open transfer tube to an empty refuel cavity. The initial conditions of the dewatering incident analysis and resultant water levels over the spent fuel are not affected by this SFP skimmer/cooling loop issue. These incident initiators are not effected by the SFP cooling or skimmer loop failures, therefore, preserving the previously analyzed and approved margin for these dewatering incidents.

6G-99-0106
DRP 8-057

DESCRIPTION:

The purpose of this UFSAR change was to revise Table 5.2-1 to reflect correct code year and addendum for the Loop Stop Isolation Valves (LSIVs). Table 5.2-1 does not currently include information regarding the LSIVs. Other RCS valves are included in the table.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because this change documents the code year for the LSIVs in the UFSAR. The valve design is not changed and the valve is adequate for the system conditions. Therefore, there is no impact on the probability of any accident or equipment malfunction. The code year information can have no effect on the amount of radioactivity released to the environment. Therefore, there are no changes to accident consequences.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because this is a non-functional change. There are no impacts on plant operations or on system interfaces. No new accidents or transients will be created. This change has no impact on valve or system function. The operating conditions of the RCS are not affected. No different malfunctions of equipment will result.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the documentation change to reflect the Asme Code year does not affect the ability of the LSIVs to be placed in the open position and to have power removed. Also, the change has no impact on the ability of the LSIVs to isolate or to remain isolated.

6G-99-0119
DRP 8-060

DESCRIPTION:

The purpose of this UFSAR change was to reflect the addition of the replacement of the obsolete ammonium hydroxide pumps in accordance with DCP 9900201 and ECN 001471M. DRP 8-060 clarifies the statement regarding chemical feed addition prior to wet lay-up in UFSAR Section 10.4.7.3.4 by deleting the words "under low load conditions." This UFSAR change will reflect a change to the Operating Procedures. This UFSAR change is also required to enable replacement of existing obsolete ammonium hydroxide pumps with a more reliable model under DCP 9900201.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the Category II D Chemical Feed System does not cause an accident. The probability of malfunction of Steam Generator Tubes is not increased by the change because the change reflects current Operating Procedures, which control secondary water chemistry. The consequences of a Steam Generator Tube Rupture are therefore unaffected.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the secondary water chemistry is controlled by Operating Procedures. The change reflects the current practice of chemical addition for wet lay-up of the Steam Generators and the ammonium hydroxide pump replacement with a more reliable pump.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the secondary side chemistry program implementation is unaffected by this change.

6G-99-0121
DRP 8-062

DESCRIPTION:

The purpose of this UFSAR change was to accurately describe the current operation of the Laboratory Ventilation (VL) system in regards to the relative pressure differentials between various areas/rooms. Some current UFSAR statements exist that describe VL System operations that are not obtainable. Plant operation is acceptable with the current operation of the VL System but doesn't match the current UFSAR statements.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the VL System is not an initiator for any UFSAR accidents or transients, and does not interface with equipment important to safety in such a way as to impact its failure. This change to the UFSAR description will have no impact on the probability of any accident or transient described in the SAR. The VL System is powered by non-1E power and mitigation functions Therefore, the consequences of all accidents, transients, and equipment malfunctions are unaffected by this change.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the VL System is a non-safety related ventilation system which serves various laboratory rooms within the Auxiliary Building during normal operation for personnel comfort and ALARA purposes. The spread of airborne activity will be prevented by utilizing the fume hoods and proper ALARA practices. Any contaminated air that would infiltrate into the Auxiliary Building from the applicable laboratory areas will be filtered by the VA System. The VA System is at least as efficient as the VL System, and is a safety related exhaust/filtration system. Operation of the VL System cannot impact or cause any accidents. Thus, a new accident or transient is not created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the VL System is not associated with any Technical Specification. The VA System Technical Specifications are not affected in any way.

6G-99-0124
DRP 8-064

DESCRIPTION:

The purpose of this UFSAR change was to reflect the replacement of American Air Filter P/N 275-107-937 as currently specified on the design drawings with American Air Filter P/N 709-117-360. This required change to UFSAR Tables 9.4-13, 9.4-3, and 9.4-27 for the VE, VI, and VV Systems indicate alternate synthetic fiber filter media, to specify nominal dP, and to clarify efficiency is 90-95%.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the function of the VE System and the electrical equipment served by the VE System is not affected by the activity. The replacement filters are equivalent fit, form, and function, and do not adversely impact the seismic qualification of the filter bank frame.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because there is no change to the function of the VE System or new failure modes created by the activity.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because there is no adverse affect on the reactor trip switchgear, 1E 125 VDC distribution panels, or instrument inverters by the replacement activity.

6G-99-0206
DRP 8-065

DESCRIPTION:

The purpose of this UFSAR change was to remove an obsolete reference and include the latest current reference. The change was to remove IEEE 450-1980 and update it with IEEE 450-1995 and thereby bring the UFSAR up to date with current standards. IEEE 450 deals with the inspection of batteries.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the update of references in the UFSAR does not affect or increase the probability or the consequences of any accident. It also does not affect any equipment functions. The UFSAR was updated to reflect current industry standards and correct outdated references.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the change was an update to an outdated industry standard referenced in the UFSAR. The reference is used for procedures that inspect, conduct measurements, and test batteries.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because Technical Specifications and their associated surveillance requirements and Bases already reference IEEE 450-1995. The Technical Specification Bases explicitly references IEEE 450-1995.

6G-99-0205
DRP 8-066

DESCRIPTION:

The purpose of this UFSAR change was to revise Table 9.4. The changes are:

1. providing clarifying information in the component, malfunction and result sections of the failure analysis for all sections of the Table 9.4 series,

2. expanding upon components referenced and developing the required malfunction and result sections for each new component identified, as appropriate, and,

3. providing consistent format from system-to-system throughout the table and to the extent possible. The intent of the revision is to provide expanded and corrected descriptions to existing failure scenarios and, expand upon the analysis as required to provide for consistency through the table, not to provide a detailed failure analysis of every system component. Corrected statements are acceptable and conform to the original as built design requirements.

The results currently provided in the UFSAR are somewhat ambiguous and leave room for interpretation. The changes in the activity will better describe actual system response (i.e. - actual results) to component failures inherent to the original design.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the change being implemented is a UFSAR text change only that clarifies the actual response of originally installed systems to postulated failures at the system level that could be classified as a "normal failure."

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because no physical changes are being made in the plant that would create the possibility of causing an accident or malfunction that has or has not been previously evaluated. The change is a UFSAR text change that will enhance and clarify existing information that describes the response of equipment following an HVAC component failure or system transient.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because no changes are being made to Basis documents that affect margin of safety and no unreviewed safety questions were identified in the attached evaluation. This change is a text change only to provide clarifying information.

6G-99-0128
DRP 8-076

DESCRIPTION:

The purpose of this UFSAR change was to reflect the addition of modifying the operation of the non-safety related Containment Refuel Machine and Spent Fuel Pool Bridge Crane to allow a wider range of speeds for the bridge, trolley and hoist, of the fuel handling cranes. The change increased the speed of the cranes, while maintaining all interlocks and operational requirements for movement of Fuel Assemblies and are as follows:

Refuel machine bridge from 40 ft/min to 53 ft/min.

Refuel machine trolley from 22 ft/min to 26 ft/min.

Spent Fuel Pool Bridge Crane trolley from 30 ft/min to 40 ft/min.

Spent Fuel Pool Bridge Crane hoist (with load) from 21 ft/min to 22 ft/min.

Spent Fuel Pool Bridge Crane hoist (no load) from 21 ft/min to 33 ft/min.

This safety evaluation addressed the UFSAR changes and changes to the equipment.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the likelihood of a mechanical failure is not increased because the change to increase the range of allowed speeds is within the design criteria for the equipment. The likelihood of an operator error is not increased because the number of Fuel Assembly manipulations is not changed; the equipment the operator uses is not changed; the interlocks on the equipment are not changed; and the same skills of the operator will be used to avoid/prevent errors.

   The consequences do not increase because this change does not result in either an increase in the source term for the radioactive release, an increase of the number of fuel pins damaged, degradation of the radiation detection system, or degradation of the FB ventilation for SFP. Since none of these are affected by this tooling change, the consequences of the accident are not affected.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the changes only affect the speed of assembly movement and no other functions or safety features of the fuel handling equipment are affected. The equipment is designed to operate at the speeds specified in this change and-therefore, will be operated within its design basis. No new failures or malfunctions of the equipment are expected.

6G-99-0128
DRP 8-076
(Cont'd.)

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because no Tech. Specs. bases, or SAR analyses are affected by this change.

6G-99-0186
DRP 8-077

DESCRIPTION:

The purpose of this UFSAR change was to correct the Circulating Water (CW) Pump and Circulating Water Makeup Pump flow rates listed in UFSAR Table 2.4-16. The CW Pump flow was changed from 211,000 gpm/pump to 214,500 gpm/pump and the CW Makeup Pump flow was changed from 30,000 gpm/pump to 24,000 gpm/pump.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the probability of a CW line break, expansion joint failure, or loss of condenser vacuum was unchanged by correcting the CW Pump and CW Makeup Pump flow values in the UFSAR. The arrangement of the River Screen House (RSH) is the function of safety-related equipment is not impaired should a failure of a CW Makeup Pump or component occur. Changing the CW Makeup Pump flow value in the UFSAR does not affect these flood protection features. In the event of a circulating line break which cannot be isolated, the Turbine Building could theoretically be flooded to grade level at Byron. Damage to Turbine Building equipment will not prevent safe shutdown of the plant because changing the CW Pump flow value did not affect the results of these analyses. The change had no affect on equipment failures or malfunctions. No new failure modes were created by correcting the pump flow rates in UFSAR Table 2.4-16. The CW System is not required to maintain the reactor in safe shutdown condition or mitigate the consequences of any malfunction of equipment important to safety.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because changing the CW Pump and CW Makeup Pump flow values in UFSAR Table 2.4-16 does not create the possibility of an accident or transient of a different type. The change does not adversely affect CW System operation. The CW System design is based on the revised flow values.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because no Technical Specifications were affected.

6G-99-0135
DRP 8-081

DESCRIPTION:

The purpose of this UFSAR change was to reflect a change to the Normal Containment Purge Isolation Valves (1/2VQ001A, 1/2VQ001B, 1/2VQ002A, 1/2VQ002B) during Modes 5 and 6. Normally, these valves are closed with power removed from the actuators in Modes 1 through 4. The effect of de-energizing these valves in the closed position during Modes 5 and 6 is that they will not be used for containment purge operation. Containment purge operation in Modes 5 and 6 will be performed by the Containment Mini-Purge System.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the VQ System does not cause any accident or anticipated transient discussed in the UFSAR. The system is implied to function as a result of a safety injection actuation signal to automatically close any open VQ System Containment Isolation Valves because they could have an impact on the release of radioactive products offsite.

   The activity will not increase the offsite dose due to the accidents listed above. If the normal purge system is operating, the volumetric flow rate is 40,000 cfm. The Mini-Purge System is designed for only 3000 cfm. Therefore, the transport time and volume of air are significantly lower than if the normal containment purge system were in operation for a specified time period. In addition, if the Mini-Purge System Isolation Valves are open, they will receive an automatic closure signal in the event of a safety injection actuation. The Normal Purge System Isolation Valves will already be in the closed position.

   The activity will use the Mini-Purge System to ventilate containment when the reactor is shutdown instead of the Normal Purge System.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the activity does not alter the safety functions of the affected equipment. Containment integrity is still maintained because the Normal Purge System Isolation Valves will be shut in all plant modes and the Mini-Purge System Isolation Valves receive the same automatic closure signals as the Normal Purge System Isolation Valves. Containment can still be ventilated during reactor shutdowns with the Mini-Purge System. There is no requirement to ventilate the containment in a certain period of time when the reactor is shutdown.

6G-99-0135
DRP 8-081
(Cont'd.)

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the activity will de-energize closed the Normal Purge System Isolation Valves in all plant modes. This will place these valves in their preferred position for containment isolation and mitigate the failure of these valves to reseal for containment integrity after stroking. The activity credits the Mini-Purge System to ventilate containment during reactor shutdown periods. The Mini-Purge System will reduce the concentration of noble gases within containment prior to and during personnel access. The Mini-Purge System Isolation Valves are designed to meet the requirements for automatic containment isolation.

6G-99-0192
DRP 8-082

DESCRIPTION:

The purpose of this UFSAR change was to provide an alternate method for securing the hex nuts used to attach the containment recirculation sump inner screen (inner most screen). Previously the 3/8" hex nuts were welded to the screen frame. The attachment detail was revised to either tack weld the hex nut to the screen frame or to stake the bolt threads.

SAFETY EVALUATION SUMMARY:

1 The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the change did not affect the Reactor Coolant System pressure boundary. Thus the probability of a LOCA was unchanged. The alternate methods of securing the screen hex nuts had no affect on the function of the sump screens, the recirculation sump, or the ECCS. No new equipment was introduced and no installed equipment was operated in a new or different manner. There was no change in parameters within which the plant is normally operated or in the setpoints that initiate protective or mitigation actions. No new failure modes were introduced. Tack welding the hex nuts or staking the threads performs the same function as welding the hex nuts to the screen frame.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the operation of the sump and sump screens was unchanged by the change to a method of securing the mounting nuts. No equipment important to safety was affected by the change. The sump and sump screens operate as designed. The change did not introduce any new failure mechanisms. Tack welding the hex nuts or staking of threads are effective methods of preventing the hex nuts from coming loose.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because all assumptions made in the bases for the ECCS Technical Specification were unaffected by the design change. ECCS Pumps remained fully operational, ECCS flow was unaffected, and the resultant accident consequences and associated margins of safety were unchanged.

6G-99-0236
DRP 8-083

DESCRIPTION:

The purpose of this UFSAR change was to correct erroneous information contained in the UFSAR identified during the Design Basis Initiative (DBI) effort and documented under DBI Open Item 7 for Byron and Braidwood. UFSAR Figures 12.3-18, 12.3-40 and 12.3-62 were corrected with the actual thickness of the Fuel Transfer Tube used in the shielding evaluations for Zones 14Z.1 and 14Z.9 which are the Piping Penetration areas adjacent to the Fuel Transfer Tube.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the Fuel Transfer Tube is a passive, structural element. The correction of the thickness in the shielding calculation has no impact on the function of the Fuel Transfer Tube or it's structural strength. Revision of the shielding calculation to account for the actual thickness of the concrete walls comprising the Fuel Transfer Tube will not increase the probability of a malfunction of equipment important to safety.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the Fuel Transfer Tube is a passive, structural element, which has no interaction with other systems or components. It cannot, in-and-of-itself, initiate a malfunction of a different type nor can it interact with other systems or components that could potentially initiate a malfunction of a different type. Revision of the shielding calculation to account for the actual thickness of the concrete walls comprising the Fuel Transfer Tube will not increase the possibility of a different type of malfunction of equipment important to safety.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the addendum to the shielding calculation BB-FH-13 has used the source terms associated with the 100 hours in-vessel decay time and demonstrates that the radiation levels associated with the fuel movement are below the levels for the areas adjacent to the Fuel Transfer Tube based on the Zone Classifications stated in UFSAR Table 12.3-1 and the specific design dose rates stated in Table 12.3-2.

6G-99-0188, Rev. 0
DRP 8-086

DESCRIPTION:

The purpose of this UFSAR change was to revise Sections 9.2.5.3.3 and 9.4.6.2.3 to more accurately describe the relationship between the SX Make Up (M/U) Pump diesel engines and their Jacket Water Heaters with respect to engine reliability under low ambient diesel temperature conditions.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because this change does not involve a physical change to the facility, but provides a better description of the SX M/U Pumps and River Screen House Ventilation Systems as they currently exist. Neither the SX M/U Pumps nor the River Screen House Ventilation Systems are initiators of any accidents. Therefore, the probability of occurrence of an accident is not increased. This change does not affect the ability of the SX M/U Pumps to provide design flow under design conditions. The SX M/U Pumps and the River Screen House Ventilation Systems will continue to operate as before. Therefore, this change cannot increase the consequences of an accident.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because this change does not involve a physical change to the facility, but provides a better description of the SX M/U Pumps and River Screen House Ventilation Systems as they currently exist. It does not physically impact the SX M/U Pumps or the River Screen House Ventilation Systems. The systems will be operated. Therefore, there is no possibility of an accident or transient of a different type than any previously evaluated.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this change does not involve a physical change to the facility, but provides a better description of the SX M/U Pumps and River Screen House Ventilation Systems as they currently exist. There is no change in the manner in which the SX M/U Pumps or River Screen House Ventilation Systems operate.

   The SX M/U Pumps will continue to meet their design requirements to withstand all design basis natural phenomena events and combinations of events except for Seismic Events during low Rock River flow or level (loss of SX Makeup Pump Suction), tornado, and river flood.

6G-98-0274
FDRP 18-051

DESCRIPTION:

The purpose of the Fire Protection Report change package was to add antennas for the 900 MHz-radio system to the Auxiliary and Turbine Buildings. The antennas were added to improve reception in the plant.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the addition of the antennas does not affect or impact previously analyzed accident precursors and or creates the possibility of a malfunction of equipment important to safety.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because no new failure modes were created by the addition of the antennas. Existing station controls and procedures restrict the use of radios around certain sensitive equipment that is important to safety.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the addition of the antennas does not affect any parameters upon which Technical Specifications are based.

6G-98-0173 Rev. 1
FDRP 18-065

DESCRIPTION:

The purpose of this Fire Protection Report Change was to include a design change that installed new aboveground concrete encased 2,500 gallon gasoline and 2,000 gallon diesel fuel storage tanks and associated equipment. It also removes existing underground 2,000 gallon gasoline tank (0FS01T) and 1,000 gallon diesel fuel oil tank (0FS02T). The new tanks are provided with secondary containment, leak detection sensors, level instrumentation, fill/overfill protection, new fuel dispensers, emergency shutoff switches, and stairways for use in refilling the tanks. In addition, the new gasoline tank is provided with a Stage II Gasoline Vapor Recovery System. The tanks are installed on top of a new concrete pad surrounded by vehicle barriers. The Fire Protection Report is revised to address replacement of the tanks.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the new fuel storage tanks are designed with features to reduce the potential for a gasoline or diesel fuel fire. The new tanks are equipped with a secondary containment to prevent fuel leakage from being released to the surrounding area. Each tank is equipped with two types of leak detection equipment, a tank level probe and an interstitial sensor. Each tank is equipped with a seven gallon spill/overfill containment, an overfill limiter, and an overfill alarm. The gasoline tank is equipped with a Stage II vapor recovery system to prevent release of gasoline vapors to atmosphere in the area around the tanks. The new tanks are designed and tested to provide 2-hour fire protection and are protected by concrete and steel vehicle barriers. The original gasoline and diesel fuel storage tanks do not have any of the above safety features. Based on the above, the probability of a fire is reduced. Accident consequences are not increased because replacing the tanks will not adversely impact the consequences of a fire or affect off-site dose. The impact of a postulated tank explosion on safety-related structures was evaluated and found to be acceptable. The new tanks are located approximately 490 feet away from the nearest safety-related structure, the Unit 1 Containment. No equipment important to safety is located near the storage tank facility. The nearest safety-related components are underground piping and electrical duct runs located over 200 feet away from the tanks. Changing the tanks from underground to aboveground does not increase the probability of a malfunction of the SX lines. The tanks are not relied upon to mitigate the consequences of a malfunction of equipment important to safety and do not perform a Safety-Related function. The impact of a gasoline spill on Main Control Room Habitability was evaluated and determined that the concentration of gasoline vapors in the Main Control Room will not exceed allowable levels.

6G-98-0173, Rev. 1
FDRP 18-065
(Cont'd.)

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the new storage tank facility will not adversely impact systems or functions so as to create the possibility of an new accident or malfunction because the impact of the tanks on Main Control Room Habitability and on Safety-Related structures was evaluated and determined to be acceptable. In the event of a large gasoline spill, Main Control Room operators may have to don Self-Contained Breathing Apparatus and/or realign Main Control Room Ventilation, per Station Procedures. The safety design features on the new tanks reduce the potential for fuel tank leakage or overfill. The Gasoline Vapor Recovery System reduces the potential for igniting gasoline vapor at the refueling dispenser. The tanks are installed upon a 12" thick reinforced concrete pad, designed above and beyond the requirements specified by the manufacturer. The tank design meets National Fire Protection, International Fire Code Institute, 1994 Uniform Fire Code, Underwriters Laboratories, Underwriters of Canada, Southwest Research Institute Standard 93-01, and BOCA National Fire Prevention Code requirements. In addition, this tank design underwent vehicle impact, ballistic impact, hydrocarbon pool fire, vapor recovery, and hose stream testing. The manufacturer's design incorporates a U.L. 142 listed cylindrical steel tank encased in a six-inch reinforced concrete vault providing a four-hour firewall when tested to U.L. 1709. The 110% secondary containment meets the Environmental Protection Agency regulation E.P.A 40C.F.R. Part 112 for secondary containment. Based on the above information, this change does not create a new accident or malfunction of a type different from those evaluated in the SAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which Technical Specifications are based.

6G-98-0252
FDRP 18-070

DESCRIPTION:

The purpose of this Fire Protection Report change was to update the addition of combustibles in a location east of the Turbine Building. The change adds the necessary information regarding a battery storage facility in the Level B Storage Building. The change added equivalent combustibles of 8 C&D battery cells and 1.6 pounds of cable insulation to the building. The change was done due to the need to provide and acceptable area for long term storage of spare battery cells.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because there is no affect on plant operations and no effect on the analysis or assumptions made in any of the Design Basis Accidents. The Level B Storage Building is not fire rated and it is not located in a fire hazard area as stated in the Fire Protection Report. There is no equipment important to safety located in or near the Level B Storage Building. The building is separated from the Turbine Building by a 12-inch thick structural reinforced concrete wall with a 9-inch thick structural reinforced concrete roof.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the change was an update to the Fire Protection Report to contain the allocation of a spare battery cell storage area. The change does not create the possibility of an accident or malfunction of a different type because it does not affect normal plant operations and it does not interact with any equipment important to safety.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because a change to the Fire Protection Report to contain and describe a remote non-fire rated battery storage location does not affect any parameters upon which the Technical Specifications are based.

6G-98-0254
FDRP 18-072

DESCRIPTION:

The purpose of this Fire Protection Report Change was to include information from a design change that replaced the diesel oil igniter system for both Auxiliary Steam (AS) Heating Boilers with propane gas igniter systems. A 20-pound propane tank is installed and secured in each room to act as a fuel source for each igniter system. Vent lines, for each propane igniter system, are routed to the outside of the building.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the propane igniter system is designed to meet NFPA requirements and the propane tank is secured to support steel. The probability of a fire in either of the Auxiliary Boiler Rooms has not increased and no additional ignition sources have been added. The consequences of a fire in the Auxiliary Boiler Room are unchanged because the Design Basis Fire assumes the entire combustible contents within the room would be ignited and the sprinkler system would be activated. The added propane fire load is within the sprinkler system's capability to mitigate a boiler room fire.

   There is no change to the AS System, steam piping, pressure or temperature. The replacement of the fuel oil igniters with propane igniters does not change the ability of the isolation sensors to detect and initiate protective action. No equipment important to safety is in the Boiler Rooms and the design change does not adversely affect the safe shutdown from a fire.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the propane pilot is designed to NFPA requirements and will operate with improved performance compared to the fuel oil igniter. The main burner of the oil fired Auxiliary Steam Boiler will be unchanged.

   In the event of a fire, the sprinkler system will activate to extinguish the fire as the sprinkler system is designed to meet the requirements of a Group 1 of Extra Hazard Occupancies. Replacement of the fuel oil igniter with a propane igniter and associated propane tank will not change the Group 1 Classification of the Extra Hazard Occupancy or ability to extinguish the Design Basis Fire.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which Technical Specifications are based.

6G-98-0271
FDRP 19-002

DESCRIPTION:

The purpose of the Fire Protection Report update was to reflect a change to the use of an area adjacent to the Instrument and Electrical Maintenance (IM/EM) Shops. The change derated the fire wall in the Service Building from Coordinate 39 to the stairwell along Coordinate D.1 and the wall in the IM/EM Maintenance Shops along Coordinate 37. The wall was derated from Coordinates B to E from a 3 hour fire rating to no fire rating. The area near these walls are being converted to office space and a tool room, which do not require fire rated exterior walls.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the change does not impact any area of the plant where equipment important to safety is located. The change does not impact any assumptions and or analysis developed to determine the probability of occurrence of any accident. The change does not affect any equipment important to safety functions. The change does not affect any of the precursor assumptions made for accident analysis or equipment malfunctions.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the change is an update to the Fire Protection Report that changes the rating of an area in the Service Building that house office space and other administrative support functions. The change does not affect any systems or components used to mitigate Design Basis Accidents.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which Technical Specifications are based.

6G-99-0035
FDRP 19-006

DESCRIPTION:

The purpose of this Fire Protection Report Change was to correct inaccurate information concerning the Fire Protection (FP) System Jockey Pumps and their electric drive motors. The FP Jockey Pumps are described in Appendix A5.4 of the FPR.

a. Page A5.4-27 will be revised to remove information concerning "manufacturer", enclosure", and "frame" for the Jockey Pump electric motor.

b. Page A5.4-28 will be revised to remove information concerning "manufacturer" and "type" and the specified number of stages will be changed from 4 to 21 stages.

Byron PIF B1999-00565 documented that the FPR specifies an incorrect frame size for the Jockey Pump Motor and the specified number of stages for the Jockey Pump should be 21, not 4 as currently stated.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the activity did not increase the fire hazards in any plant area. The activity did not remove or alter any Fire Protection System that affects its design capability. Therefore, the change did not increase the probability of a Design Basis Fire or the probability of malfunction of equipment.

   The activity only corrected information describing the FP Jockey Pumps. The FP Jockey Pumps have always been 21 stage pumps, the FPR has been incorrect in the past when it was described as 4 stage pumps. The information deleted (i.e., manufacturer, frame size, type) has no impact on operation of equipment. The equipment important to safety are the main fire pumps which supply the water demanded from the FP System.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the activity corrected inaccurate information describing the FP Jockey Pumps. The design performance of the Jockey Pumps and the function of the Jockey Pumps are not changed. The FP System was not affected in any way by the change, therefore, the possibility of an accident or malfunction of a different type is not created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because no Technical Specifications were affected by the change.

6G-99-0044
FDRP 19-007

DESCRIPTION:

The purpose of this Fire Protection Report Change was to show information for modification that installed the instrumentation necessary to gather data and evaluate the feasibility of measuring Feedwater Flow rate ultrasonically instead of the current venturi measurement. This modification only allowed test and collection of Feedwater Flow rate data for evaluation.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the change did not involve or interface with any systems, structures, or components important to safety. The activity does not alter the initial conditions used in the UFSAR analysis for any accidents. The components involved do not initiate any accidents or mitigate the consequences of any accident analyzed in the SAR. The function of any system, structure or component important to safety is not altered by this change.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because this design change does not adversely impact UFSAR accident-related systems, structures, and components (SSCs). The change does not introduce any adverse interactions between any SSCs.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the installation of non-intrusive ultrasonic flow instrumentation on feedwater pipe does not affect any parameters upon which Technical Specifications are based.

6G-97-0045
FDRP 19-009

DESCRIPTION:

The purpose of this Fire Protection Report Change was to replace existing Main Control Room furniture with new updated furniture for the Unit (NSO) desks, Center Desks, back to back unit 2 file cabinet, and the tops for 1/2CX05J, 1/2PM14JB, and the breathing air storage cabinet. Remove the no longer used breathing air equipment at the center desk, 1/2CX05J (1/2EA05MA/B) and air storage cabinet. Reroute electrical power and communication lines from control panels to supply local printers. All existing telephones, pages, security sound power phone systems or other communication and computer systems will be restored to their current pieces of Main Control Room furniture unless removed by another design change, MIS equipment replacement, or some other allowable change. The mounted liquid release key box and fire extinguisher, which were mounted on the Center Desk, are relocated to the nearby former breathing air storage cabinet on the Unit 1 side. The abandoned radio phone jacks on the Center Desk are removed. Revisions to the UFSAR and the Fire Protection Report are made to reflect the removal of the breathing air equipment and changes in combustible loading.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of any accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the addition of Main Control Room furniture and removal of Self Contained Breathing Apparatus (SCBA) components does not impact Containment Systems or a Loss of Coolant Accident (LOCA). Therefore, the probability of an accident is not increased. Neither the Main Control Room boundary or the ventilation system is affected by this change. Therefore, the consequences of a LOCA are not impacted because operator doses in the Main Control Room will not be affected.

   The new desks will support existing and newly installed equipment. There are no new failures identified. The affected furniture is installed with seismic mounting and load considerations analyzed. The additional fire load has been analyzed and found to be negligible. The remaining Emergency Air lines will be capped to prevent any opening in the Main Control Room Floor (fire barrier). SCBA with two extra units (per single failure criteria guidelines) is used to provide emergency breathing air.

   Since the Main Control Room is continually manned (and hand held fire extinguishers are available), the probability of a fire is not considered to increase with the addition of Main Control Room furniture and subsequent negligible increase in fire load.

6G-97-0045
FDRP 19-009
(Cont'd.)

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the Main Control Room equipment will still operate as required. The Main Control Room is continually manned and the negligible increase in fire load due to addition of the furniture is acceptable. SCBAs are available to provide self contained breathing air to Main Control Room personnel in the event the air is unbreathable in the Main Control Room, usually in the case of smoke due to fire. All fire barrier paths will remain intact. The affected furniture is installed with seismic mounting and load considerations analyzed. The additional fire load has been analyzed and found to be acceptable. No new accidents or malfunctions are created as a result of this design change.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because Technical Specifications are not affected by this change.

6G-99-0146
FDRP 19-014

DESCRIPTION:

The purpose of this Fire Protection Report Change was to revise the Fire Protection Report (FPR) Fire Hazards Analysis in Section 2.3 and combustible loading in FPR Table 2.2-3 to describe the flammable liquid cabinets containing various flammable or combustible liquids. The flammable or combustible liquids are being evaluated as lube oil, which is conservative since it has a high heat content of 15,000 BTUs/gallon. An exception are the two " Pope Paint " flammable liquid cabinets in Fire Zone 11.2-0, specified as containing paint or paint thinner with a heat content of 139,382 BTUs/gallon.

The changes to the FPR affect Fire Zones 11.2-0, 11.3-0, 11.5-0, 11.5A-0, 11.6B-0, 11.6E-0, 11.7-0, and 14.3-0. The addition of these cabinets will slightly increase the calculated fire loading for the Fire Zones.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the change of adding the flammable liquid cabinets does not affect the probability of a Design Basis Fire. The change does not add an ignition source and the flammable liquids are stored in approved primary containers and UL listed or FM approved storage cabinets that protect the contents and prevent them from being the initiator of a Design Basis Fire.

   The change of adding the flammable liquid cabinets does not affect the consequences of a Design Basis Fire. The additional fire loading resulting from the addition of the flammable liquid cabinets does not significantly change the fire loading in each zone. The increased fire loading is either a minor change from the analyzed loading or the changed value is still within combustible loading values considered to be "low" (less than 100,000 BTUs/sq. ft, ComEd Std. NES-MS-05.1).

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the change of adding the flammable liquid cabinets does not create the possibility of an accident or malfunction of a type different from those previously evaluated. The flammable liquids are stored in approved cabinets that are placed and maintained in accordance with approved practices and procedures. In this configuration, the stored liquids do not constitute a hazard to or create the potential for a different type of accident.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because Technical Specifications are not affected by the proposed change.

6G-99-0190
FDRP 19-017

DESCRIPTION:

The purpose of this Fire Protection Report Change was to correct a typographical error in Section 2.3.11.55. The fire loading in Fire Zone 11.6B-0 (Aux. Building Offices) is 8,800 BTU/ft^2, not 8,200 BTU/ft^2 as currently specified in this FPR section. Combustible Loading Calculation
ATD-0026, Revision 6, specifies a fire load of 8,800 BTU/ft^2, the same as FPR Table 2.2-3 that specifies the same fire load as Calculation ATD-0026.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the change does not affect the zone occupancy, fire detection, or fire suppression capability in the affected zones. The change only adds combustible loading to the zone. However, the low combustible loading described by the subject change, does not challenge the capability of the zone fire protection design features or adversely affect the ability to achieve and maintain safe shutdown of the plant. Therefore, the change did not affect the probability or consequence of a Design Basis Fire.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the Fire Hazard Analysis (FPR Section 2.3) and Safe Shutdown Analysis (FPR Section 2.4) demonstrates that safe shutdown can be achieved and maintained in the event of a Design Basis Fire in this Fire Zone. The additional minor fire loading attributed to this change does not affect this conclusion. This very small fire loading increase will not affect the currently evaluated Design Basis Fire or create a new type of accident or malfunction.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change did not affect any Technical Specification.

6G-99-0170
FDRP 19-018

DESCRIPTION:

The purpose of this Fire Protection Report Change was to delete all references to fire hydrant hose houses. In place of hose houses, the FPR will be revised to describe fire hoses and associated fire fighting equipment staged on a dedicated fire response truck, ready to transport the equipment to fire hydrants when needed. A redundant backup mobile cart was also dedicated and equipped to backup the dedicated truck. This mobile equipment was equivalent to equipment currently stored in three hose houses. The activity allows the existing hydrant hose houses to be removed and eliminate the associated monthly inspections of the hose houses and the hydrostatic testing of hoses contained within them. The hose staged on the mobile truck and cart was inspected and tested in accordance with NFPA Code 1962 for interior hose.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the activity deleted all fire hydrant hose houses. In place of hose houses, fire hoses and associated fire fighting equipment staged on a dedicated fire response truck, ready to transport the equipment to fire hydrants when needed, will be credited. A redundant backup mobile cart is dedicated and equipped to backup the dedicated truck. This mobile equipment will be equivalent to equipment currently stored in three hose houses, as required by BTP CMEB 9.5.1. The dedicated equipped truck can be driven to the needed yard hydrants, necessary equipment unloaded and installed, and hose streams established in the same amount time or less then the current practice of responding to the hose houses for the necessary hoses and equipment. The fire brigade is trained in the use of the fire response truck in quarterly fire drills for fires requiring use of yard hydrants. The activity provides the same equipment and use the same number of fire brigade members as the current hose houses. The inspection and testing requirements for hydrants and fire hose meet the recommendations of BTP CMEB 9.5.1, NFPA, and NEIL insurance standards. Therefore, the readiness and integrity of the equipment is assured. The effectiveness of mitigating a Design Basis Fire is not changed by the activity, therefore the probability and consequence of a Design Basis Fire is not increased.

6G-99-0170
FDRP 19-018
(Cont'd.)

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the activity of changing from yard hydrant hose houses to a mobile dedicated truck or cart to supply the required fire fighting hoses and equipment is consistent with and permitted by the guidelines of NRC BTP CMEB 9.5.1. Once the hoses and equipment are installed on the hydrants, there is no change from the current Fire Protection Program. The manner by which hoses and equipment are brought to the hydrants does not affect fire development or the manner in which it is extinguished. Therefore, the activity, removing the hose houses, and eliminating the inspections associated with the hose houses does not create an accident or malfunction of a different type from that currently considered in the Fire Hazard Analysis.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because Technical Specifications are not impacted by the activity.

6G-99-0191
FDRP 19-019

DESCRIPTION:

This activity made changes to the surveillance procedures associated with Fire Protection features of the plant. The changes affect the frequency that these existing procedures are performed and in some cases, remove components from the scope of the procedure. The affected procedures do not change the configuration of any plant system. The activity changed the surveillance requirements of Section 3.10 of the Technical Requirements Manual (TRM). The Fire Protection Report is also revised to discuss deviations from the NFPA Fire Code and Appendix R because of the modified surveillance frequencies.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the change to the TRM, FPR, and associated surveillance procedures does not change the occupancy or increase the fire hazards in any plant Fire Zone. No ignition sources or combustibles are added or altered as a result of the activity. Fire Protection features are not altered by the activity, nor are compensatory actions changed if a feature is taken out of service. The activity only affects the interval that surveillance procedures are performed and specific components tested. These procedures are not used or credited to mitigate a Design Basis Fire. Therefore, the activity does not increase the probability of occurrence or the consequence of a Design Basis Fire or malfunction of equipment.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the activity makes changes to the surveillance procedures associated with Fire Protection features of the plant. The changes affect the frequency that these existing procedures are performed and in some cases, remove components from the scope of the procedure. The affected procedures do not change the configuration of any plant system. The test/inspection method, system alignment, and required actions of any procedure are not changed by the activity. Therefore, the possibility of an accident or malfunction of a different type is not created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because there is no margin of safety associated with the Administrative Technical Specification requirement for procedures to implementation the approved Fire Protection Program. The changes do not alter the implementation of the approved Fire Protection Program as described in TS 5.4.1. These changes do not affect any margin associated with the approved Fire Protection Program.

6G-99-0195
FDRP 19-022

DESCRIPTION:

The purpose of this Fire Protection Report change package was to replace the existing fire damper, American Warming Model P-475A with new Model N39CH for EPN 2VD54Y. The existing damper consists of two dampers, each having a 1-1/2 hour fire rating with two Electric Thermal Link (ETLs). The new damper consists of one damper having a 3 hour fire rating and has one ETL.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment Important to safety previously evaluated in the Safety Analysis Report is not increased because this change replaces the existing fire damper 2VD54Y with new damper with same fire rating. The function of the fire damper is not changed by this change. In the event of a fire in the Day Tank Room, this fire damper prevents the fire from spreading to the ventilation ducts and Diesel Generator Room for a period of 3 hours. It also allows the carbon dioxide to stay within the Day Tank Room if the CARDOX System were to be activated. The failure of fire damper to function would allow the fire to spread to the Diesel Generator Room through the ventilation ducts and is not changed by this modification. The new damper will still provide the fire barrier of 3 hours. The activity did not create the possibility of a different type of malfunction of equipment important to safety than any previously evaluated.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because this change replaces the existing fire damper, 2VD54Y, with new damper with same fire rating. The function of the fire damper is not changed by this change. No new failure modes are created. Therefore, the activity did not create a possibility of an accident or transient of a different type than any previously evaluated.

3. The margin of safety, as defined in the basis for any Technical Specification. is not reduced because the change does not affect any parameters upon which Technical Specifications are based.

6G-99-0200
FDRP 19-023

DESCRIPTION:

The purpose of this Fire Protection Report Change was to update the Fire Hazards Analysis in Section 2.3 and combustible loading in FPR Table 2.2-3 is revised to describe the fire loading resulting from lead blankets permanently stored on elevation 426 feet of the Unit 2 Containment Building. The lead blankets are coated by a combustible material made from PVC and polyester. The coating is assumed to be entirely made from PVC, which has higher heat content and will yield the more conservative fire loading value.

The changes to the FPR affect Fire Zone 1.3-2, which is described in Section 2.3.1.6 of the FPR. This safety evaluation only evaluates the effect of the stored lead blankets on the conclusions of the Fire Hazard Analysis of the FPR.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the probability of a Design Basis Fire or malfunction is not increased since the activity does not change the zone occupancy, ignition sources, fire detection, or fire suppression capability in the affected zone. The change only adds combustible loading to the zone. The consequences are not increased since the increased fire loading is still within combustible loading values considered to be "low" (less than 100,000 BTUs/sq. ft, ComEd Std. NES-MS-05.1).

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the Design Basis Fire is assumed to affect the safe shutdown equipment within the zone of the fire. The safe shutdown analysis demonstrates that safe shutdown can be achieved and maintained in the event of a Design Basis Fire in any Fire Zone. The additional minor fire loading attributed to this change does not affect this conclusion, affect the currently evaluated Design Basis Fire, or create a new type of accident.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because Technical Specifications are not affected by the change.

6G-99-0181
FDRP 19-024

DESCRIPTION:

The purpose of this Fire Protection Report Change was to revise Fire Protection Report Table 3-1, "NFPA Code Deviation Report" and to discuss three deviations from the recommendations of NFPA 10-1981. The deviations will discuss the variance on the frequency of inspections and maintenance for portable fire extinguishers located in the Containment Buildings when the units are in Modes 1 and 2. The deviations will also state that inspection/maintenance tags will not be attached to the extinguishers located in the Containment Buildings.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the activity does not change any plant component, system, or structure. It does not affect how any plant system is operated. The consequence of a Design Basis Fire in the Containment Building is that all combustibles are consumed by the fire. The consequence of a Design Basis Fire is analyzed in the Safe Shutdown Analysis (SSA) of the FPR. The activity does not impact the FPR SSA because use of the fire extinguishers is not credited in the SSA. Therefore, the probability of a consequence of a Design Basis Fire or malfunction of equipment important to safety is not increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because these changes only affect fire extinguishers located inside the Containment Buildings. The change does not affect the location, operation, or availability of the extinguishers.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because Technical Specifications are not affected by the changes.

6G-99-0084
DCP 9800633

DESCRIPTION:

The purpose of this Design Change Package was to revise the Piping General Notes, M-679, to allow the use of Piping Design Table 0096BB, Polypropylene Lined Pipe as an alternate to Piping Design Table 0090BB. The alternate lining, polypropylene, is equivalent to SARAN for the uses at Byron, as stated in Dow Chemical Resistance Guide (Revised 1995).

These pipes are in the chemical lines for Make Up Demineralizers, Condensate Polishers, Acid Feed, Caustic Feed and Radwaste Systems. This will provide an equivalent alternative when pipes require replacement since SARAN Lined Pipe is no longer manufactured.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because there is no affect on any accident by the equipment affected. There is no affect on any equipment used to mitigate the consequences of an accident. The new piping is equivalent to the existing piping for the applications at Byron.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the new piping is equivalent to the existing piping for the applications at Byron. There is no affect on the operation of any system.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because Technical Specifications are not affected by these changes.

6G-99-0076
DCR 970104 & 990049

DESCRIPTION:

The purpose of these Document Change Requests was to revise several P&ID and single line drawings to reflect additional Fire Protection Valves. The valves are used for venting or draining and range in size from 3/4" to 4". The passive, normally closed valves have no effect on plant operation since they are branches off of main header lines and have no ability to isolate a normal Fire Protection System flowpath. Also, several related changes to Fire Protection drawings covered by DCR 990049 and revision to BOP FP-M1 valve line-up to add the required valve EPNs are included.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the Fire Protection Water Supply System acts to mitigate or suppress fires after they occur. It has no ability to cause a fire. Therefore, the system cannot increase the probability of a fire. Consequences of a fire would result from damage to equipment important to safety such that expected accident mitigation or safe shutdown functions would not occur. The addition of the new pipes/valves addressed by these DCRs will not prevent the Fire Protection System from performing its function since the valves cannot isolate the flow of water through the system and the valves and pipe are designed consistently with the remainder of the system. All of the new valves and lines are in the Turbine Building and are not located near equipment important to safety. A complete failure of one of these lines would cause flooding in the Turbine Building that would be well within the existing analysis for a CW pipe break. The new pipes and valves have no other way to affect equipment and cannot increase the probability of malfunction.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because there are numerous existing vent and drain valves on the Fire Protection System piping. The new valves are designed to the same requirements as the existing system. There is no change in system function since the valves are passive and normally closed valves out of the main flow path. Therefore, there will be no accidents of a different type.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the function of the Fire Protection vent and drain valve does not change the function of the Fire Protection System as described in the basis of any Technical Specification.

6G-99-0210
DCR 980148

DESCRIPTION:

The purpose of this Document Change Request was to incorporate an as-built change for the Reactor Coolant Pump (RCP) Bus Undervoltage (UV) Time Delay Relays. This DCR is clarifying that the maximum time delay setpoint for each RCP Bus UV Time Delay Relay is 0.70 seconds. With time delay relay tolerances included, the maximum allowable value for each RCP Bus UV Time Delay Relay is 0.77 seconds. This change does not affect the overall Reactor Trip System response times assumed in the UFSAR for Loss of Forced Reactor Coolant Flow scenarios, where the RCP Bus UV Reactor Trip is credited.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the function and operation of the RCP Bus UV Reactor Trip circuitry, as described in the SAR, are not being changed. The overall Reactor Trip System response time assumed in the SAR for the RCP Bus UV Reactor Trip is also not being changed. Therefore, the probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the SAR has not been increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the function and operation of the RCP UV Reactor Trip circuitry, as described in the SAR, are not being changed. No new functions or failure modes are created by this change. This change will not affect the overall Reactor Trip System Response Time assumed in the SAR for the RCP Bus UV Reactor Trip. Therefore, the possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report has not been created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because according to the SAR, the overall Reactor Trip System Response Time for the RCP Bus UV Reactor Trip is 1.5 seconds. This change is not affecting this value. Therefore, the margin of safety is not being reduced.

6H-98-0095
DCR 980169 & 980170

DESCRIPTION:

The purpose of these Document Change Requests was to update Westinghouse Reactor general drawings. The change updates the drawings to reflect as design change to provide Core Exit Thermocouple Housing Guide Cone welds. The revised drawings show the new weld detail.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the design change was internal to the Reactor Vessel and it is not part of the pressure boundary. No boundary to radioactive release was affected by the design change. The consequences were not affected. The Core Exit Thermocouple (CETC) Guide Cones are not part of the UFSAR Chapter 15 Accident Analysis.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the function of the cones and dowel pins were not adversely affected by the design change. The specific fillet weld detail was analyzed to provide a better joint design as compared to only the use of dowel pins. The addition of weld material atop the dowel pins will further secure the pins in place. The probability of a malfunction of a different type was not increased.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the design change did not affect any parameters upon which Technical Specification are based.

6G-98-0214
DCR 980181

DESCRIPTION:

The purpose of this Document Change Request was to convert a temporary alteration installed in the Containment Sump System to a permanent plant design. The purpose of the temporary change was to open a Reactor Building Floor Drain System (RF) cleanout connection in the Byron Unit 1 Seal Table Room. This was done to allow the connection to be credited as the design flowpath from the Seal Table Room into the RF System to allow for detection of small RCS leaks.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the change does not have an impact on equipment failures. The alternate drain path will function equivalent to the original design floor drain and the RF System will continue to function as required. The new drain line is fitted with a drain cap with holes to allow a flowpath and thereby ensure that the particular equipment performs as required. The modification does not change the possibility of occurrence of any Design Basis Accident. The installation of this drain line and its corresponding cap does not change any accident analysis or assumptions made for any Design Basis Accident.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the change makes the tested configuration permanent to ensure continued reliability of the original design function. The change also does not affect any accident precursor assumptions. The components associated with this change perform a passive function.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the conversion of this temporary alteration to a permanent plant configuration does not affect any parameters upon which Technical Specifications are based.

6G-98-0245
DCR 980237

DESCRIPTION:

The purpose of the Document Change Request was to reflect the as-built condition of the plant. Auxiliary Building Equipment Drain System (WE) Isolation Valve 2WE035 for the 2A Residual Heat Removal (RH) Pump Drain and 2WE101A, 2A RH Pump Drain Line test connection were added to drawing M-137-1 and S-WE-100-561 to be shown downstream of Valve 2RH8729A.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the change is a correction of a drawing done to reflect the actual configuration of the system and components and thereby correctly reflect the as-built configuration. The correction of the drawing does not affect accident analysis or the postulated precursors of any of the accidents.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the activity to update plant drawings to reflect the as-built configuration of a component and a drain test connection does not impact the possibility of an accident or equipment malfunction.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the update of a drawing to reflect the as-built condition of the plant for an isolation valve and drain line connection does not affect any parameters upon which Technical Specifications are based.

6G-99-0168
DCR 990098

DESCRIPTION:

The purpose of this Document Change Request was to update UFSAR figure 09.01-08 Sheet 1, P&ID M-63-IA, and EVVCS database for Instrument Air (IA) Isolation Valves on the Spent Fuel Pool (SFP) Sluice Gate and the SF Cask Area Sluice Gate. The affect of adding these valves to the documents listed is that they will be described in the SAR. Previously, these valves existed in the plant, but were not shown in the SAR or design documents.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because these normally closed valves will be administratively controlled to ensure that the Spent Fuel Pool dewatering incident probability is not increased. Failure of the gate seal is already evaluated in the SAR and the gate seal will not fail with any greater probability or consequences due to the presence of these valves.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because these normally closed valves in the non-safety related Instrument Air System have their operation administratively controlled such that they cannot create any new accident or transient.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because failure of the SFP gate seals will only allow the water level to drop to the level described in B/B UFSAR section 9.1.3.3. This activity did not change the initial conditions, rate of change, or results of this incident. The SFP level alarms exist to ensure that the initial conditions of the incident are unchanged and this activity had no affect on the operation of the SFP level alarms.

6G-99-0162
DCR 990149

DESCRIPTION:

The purpose of this Document Change Request was to change the position of Safety Injection (SI) manual valves 1/2SI064 from normally open to normally closed on P&IDs M-61 Sheet 6 and M-136 Sheet 6 and in plant procedures. There is no effect on plant operation. Primary containment penetrations P-55 for Unit 1 and Unit 2 are susceptible to overpressurization during certain Design Basis Accidents such as LOCA. This determination was made as part of a review of containment penetrations in response to NRC Generic Letter 96-06. The disposition of the overpressurization concern for these penetrations was to perform an analysis proving that the penetrations and associated piping components could withstand the maximum pressure that would result from the Design Basis Event. As part of this analysis, it was determined that Instrument 1/2PI-0929 could not withstand the maximum pressure and would likely fall and potentially compromise the containment pressure boundary. To prevent the instrument from being exposed to the pressure, Valve 1/2SI064 will be placed in the normally closed position. The valve was evaluated and found capable of withstanding the maximum pressure.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because changing valve SI064 from normally open to normally closed has no impact on the probability of a loss of reactor coolant inventory. The valve is not part of the RCS pressure boundary. Valve SI064 is part of the primary containment pressure boundary due to its location between Containment Isolation Valves. The valve is required to maintain pressure integrity, to ensure a leak tight containment boundary is provided during this accident. This ensures that radiation released to the environment will not exceed approved limits. Placing the valve in the closed position will protect Pressure Indicator 1/2PI-0929 and provide an additional barrier to the release of radiation through containment penetration P-55 should inside containment isolation valve SI8871 fail to seal when required. Changing valve SI064 from normally open to normally closed does not place the valve outside of its intended design function. There are no changes to system process parameters or flow characteristics.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because there is no change to the function of the Safety Injection System. Pressure Indicator 1/2PI-0929 is not used during normal plant operation or during accidents. There is no flow through the instrument line to 1/2PI-0929, so the position change to normally closed will not affect system conditions. There is no change in equipment function and there will be no additional challenges to the system pressure boundary.

6G-99-0162
DCR 990149
(Cont'd.)

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the functions of Pressure Indicator 1/2PI-0929 or valve SI064 are not reflected in the basis of any Technical Specification. The function of the Safety Injection System is not affected.

6G-99-0204
DCR 990188

DESCRIPTION:

The purpose of this Document Change Request was to revise P&I D's M-35 1 and 2, and M120-1, 2A, and 2B to show the Main Steam (MS) Valve 1/2MS020A - D and the 1/2MS022A - D as normally closed.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the effect of the activity will be negligible. These valves are in series with the normally closed 1/2MSO21A-D and 1/2MS023A-D valves. Closing an additional valve in series on the two Main Steam Branch Lines provides double isolation.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because closing an additional isolation valve on a Main Steam Branch Line will not create a malfunction of a different type. A steamline break and the resulting RCS cooldown has been analyzed and this is the only possible failure mode of this activity.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because closing an additional valve in series will ensure the Main Steam Branch Lines are isolated, thus preventing an unmonitored release path from the Containment to the atmosphere. The original design called for single valve isolation, the change provides for dual valve isolation.

6H-99-0391
DCR 990213

DESCRIPTION:

The purpose of this Document Change Request was to update the Environmental Qualification Design Basis Document (EQ DBD) for Byron Station (Document No. PMED-BB-EQDBD-00, Revision 0) for the accident temperature parameter and to provide reference to the transient analysis for EQ Zone T3. Affected pages are Pages 14 (Section 1 3.1.2.1), 31 (Section 5.2), 61 (Section 13), and B-17 & 19 of the DBD.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because there is no change to any of the SSCs. This change was incorporated in UFSAR, per DRP 7-040. This change was previously evaluated under SER BRW-SE-1997-201 / 6G-97-0105. Evaluation was done to determine equipment operability as well as consequences of component failure and to show the acceptability of the revised temperature. No additional failures or malfunctions will be introduced by this change.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the change will not create the possibility of an accident or malfunction of a different type from previously evaluated. There is no change to any of the SSCs. The change is to update the EQ DBD to incorporate the revised accident temperature parameter for the Safety Valve Rooms and steam tunnel (EQ Zone T3). There are no new changes the facility or design documents. Previous evaluation (Safety Evaluation BRW-SE-1997-201/ 6G-97-0105) was done to evaluate equipment malfunction. No additional accidents or malfunctions will be introduced by this change.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because there is no change made to the facility or design that is not previously analyzed. The change has been previously evaluated, and found acceptable, and this change is already incorporated into the UFSAR (Ref. DRP 7-040 and Safety Evaluation BRW-SE-1997-201 / 6G-97-0105).

6G-99-0209
DCR 990263

DESCRIPTION:

The purpose of this Document Change Request was to revise drawings M-35-5B and M-35-5C (Unit 1) and M-120-5B and M-120-5C (Unit 2) to accurately reflect the as-built status of the Main Steam (MS) and Extraction Steam (ES) Systems. The changes are being made to show the correct location of ES Drain Piping, from the Cold Reheat Section of the MS System to the Main Condenser. The changes maintain proper configuration control of the plant, and will not affect the operation of the MS/ES Systems, or any other plant systems in an adverse manner.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the portions of the MS/ES System affected by the activity are non-safety related, and do not support any function to those systems important to the safe shutdown of the reactor. Additionally, the change (as-built) is functionally identical to the original design of the plant. Therefore the change will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety as originally evaluated in the SAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the portions of the MS/ES System affected by the activity are non-safety related, and do not support any function to those systems important to the safe shutdown of the reactor. Additionally, the change (as-built) is functionally identical to the original design of the plant.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any equipment listed in the Technical Specifications.

6G-99-0134
DCP 9900351

DESCRIPTION:

The purpose of this Temporary Alteration was to install a mechanical valve blocking device on Chemical and Volume Control System (CV) Valve 2CV112E. The effect of the change will be to lock the valve in the closed position, while the Valve Limitorque actuator is removed for maintenance.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the valve block cannot cause a decrease in reactor coolant inventory because it will not adversely affect the Reactor Coolant Pressure Boundary. The valve block does not increase the potential for a boron dilution event or an increase in reactor coolant inventory because the valve will be locked in the closed position. The valve block does not increase the consequences of the identified accidents because appropriate LCOARs will be entered for the valve's inoperability for the appropriate modes in accordance with the Tech. Specs. and Technical Requirements Manual (TRM).

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the valve block prevents valve 2CV112E from opening, which would be the same effect as the single failure of the valve.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the margin of safety is not impacted because the appropriate LCOARs will be in place for the modes, during which the valve block will be installed.

6H-98-0091
TA 98-2-031

DESCRIPTION:

The purpose of the Temporary Alteration was to install a temporary one ton hoist to the Refueling Machine Monorail and use it as the Auxiliary Hoist to support Control Rod Drive latching activities.

SAFETY EVALUATION SUMMARY:

1 The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the hoist used was identical to the previous hoist, which was Out-of-Service. The temporary installation was tested to the same requirements regarding capacity and speed. With the function being identical, no challenge is present to increase probability of occurrence, consequences of an accident, or malfunction of equipment. The potential accidents reviewed in the UFSAR remain the same.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because installation was identical to the originally installed equipment. The analyses performed in the UFSAR remain valid.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the temporary installation is identical with the original. As no change of scope of work is present, the margin of safety remains the same.

6G-98-0229
TA 98-2-051

DESCRIPTION:

The purpose of this Temporary Alteration was to install a valve blocking device on the Feedwater System (FW) 2FW094 Valve in the open position. This was accomplished by sending a pneumatic signal to the valve positioner, bypassing the pressure transducer and piping. The 2FW094 Valve is the Byron Unit 2 Feedwater High Pressure Heater Blowdown Flow Control Valve. This temporary alteration was installed to allow the Byron Operations Department to properly control secondary side chemistry.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because this change to add a valve block to the 2FW094 Valve did not increase the probability of any accident in the Safety Analysis Report. In addition, the position of this valve is not credited in the safety analysis to mitigate any accidents.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the 2FW094 Valve does not perform any function important to safety, nor did the change affect any other equipment important to safety. This valve does not modulate during normal operations and the alteration did not affect any pressure retaining boundary or affect any assumed system flowrates, normal or post-accident.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this activity did not adversely impact any Technical Specification systems or equipment.

6H-98-0339
TA 98-2-033

DESCRIPTION:

The purpose of this Temporary Alteration was to temporarily disconnect the Reactor Vessel Level Instrumentation System (RVLIS) Train B Sensor 5 and install a temporary resistor in Cabinet 2PA52J. This temporary installation was because Train B Sensor 5 was found failed and not available for level indication. This change allowed the RVLIS System to operate properly with the failed sensor.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the RVLIS System is used for monitoring purposes only. There was no impact on the probability of occurrence of any accident. In addition, even with the failed sensor, Technical Specification minimum requirements for operable sensors were still met and the RVLIS System was fully capable of performing its function and therefore, there was no potential increase in consequences.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the minimum number of Technical Specification required RVLIS Sensors were still available at all times during this installation. In addition, the installation of the resistor in place of the Sensor 5 circuitry ensured that the remaining sensors in Train B were functional. The installation of this resistor did not create a new or different failure mode or malfunction as compared to failure of the sensor itself.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this activity does not adversely impact any Technical Specification systems. RVLIS operability was maintained for all plant modes by this activity.

6G-98-0273, Rev. 1
TA 98-2-067

DESCRIPTION:

The purpose of this Temporary Alteration was to remove the motor actuator from the Unit 2 Essential Service Water (SX) 2A Pump Discharge Header Crosstie Isolation Valve. The motor operator is to be removed with the HBC gear operator left in place to effectively lock the valve stem in place. As a contingency, a manual handwheel is to be mounted on the HBC gear operator to allow the valve to be locally opened if necessary. The station normally runs with the SX Crosstie Valve open to provide maximum operational flexibility of the SX System. The system is designed to operate with the Crosstie Valves open or closed. Each SX Train is designed to be 100% capacity. The Crosstie Valves' safety function is to isolate individual trains of SX to mitigate the consequence of a passive pipe failure. SX Water serves as the emergency backup for Auxiliary Feedwater and an emergency source of fire protection water.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the activity leaves the valve in the required "safe" position. Technical Specifications require the Crosstie Valve to be closed on demand. Two 100% capacity trains of SX are available.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the valve is closed in the required "safe" position. In the unlikely event that the valve is required to open, a handwheel has been added as a contingency. There is no safety-related purpose for opening the valve. No other SSCs are affected by this change.

3. The margin of safety as defined in the bases for any Technical Specification is not reduced because the activity does not change Technical Specification Bases parameters.

6G-98-0303
TMOD 98-0-070

DESCRIPTION:

The purpose of this Temporary Modification was to install a temporary damper block in place of actuator on Miscellaneous Ventilation System (VV) Damper 0VV31Y. The damper block will be installed such that damper is in the closed position. This will defeat the modulation function of the damper and thus defeat the purge and makeup functions for the Main Control Room Offices HVAC System, thus preventing the system from automatically or manually purging the ventilation system with outside air to the areas served.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because failure of the VV System is not a precursor to any SAR accidents. There are no accidents which are more likely to occur due to the change to the Main Control Room Offices Damper. The Main Control Room Offices Ventilation System is not assumed to function following any SAR accidents. The system is non safety related and serves no safety function.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the Main Control Room Offices HVAC System is a non safety-related system and, therefore, has no safety Design Bases. The Main Control Room Offices Ventilation System does not serve any areas that affect the actuation or performance of ESF Equipment or any other equipment important to safety.

6G-98-0303
TMOD 98-0-070
(Cont'd.)

The non safety-related non-seismic Main Control Room offices ventilation system operates on a continuous basis during normal plant conditions. During a loss of offsite power or a high radiation signal form the radiation detectors in the Main Control Room outside air makeup ductwork, the system is shut down. The temporary damper block restrains the damper in its fail-safe position. The system is currently running with the affected damper and associated outside air makeup dampers fully closed. Purging of the system air with fresh outside air is not typically required during wintertime operations, which is when this scheduled maintenance activity on the damper actuator will occur. During emergency operation (radiation accident) of the Main Control Room Ventilation System, the normally open minimum outside air makeup dampers are closed. This temporary change does not affect the performance of these dampers. As a contingency, the control switches on the Main Control Room Offices Control Panel will be maintained in the "closed" position to ensure the subject damper and the outside air ventilation dampers are in their closed position. Therefore, the probability of a malfunction of equipment important to safety is not increased, nor is there the possibility of an accident or malfunction of a different type.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the Main Control Room Offices Ventilation System is not addressed by the Technical Specifications.

6G-99-0005
TA 99-1-002

DESCRIPTION:

The purpose of this Temporary Alteration was to install a jumper to bypass the Reactor Trip Breaker Train A (RTA) Shunt Trip Test Push Button. The normally closed contact of the test button normally allows the 48 volt undervoltage signal from the SSPS Cabinets to reach the Shunt Trip Attachment (STA). When the 48 Volt Undervoltage Signal from the SSPS Cabinet is removed (a signal to trip the Reactor Trip Breaker), the STA drops out and closes a contact which supplies 125VDC to the Shunt Trip Coil of RTA. Due to failure of the test switch, these normally closed contacts are now open, preventing the STA from energizing and removing the trip signal (125VDC) from the Shunt Trip Coil. Installation of these jumpers bypassed these open contacts to restore trip capability to the Reactor Trip Breaker Shunt Trip Coil.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because no accidents were identified that are impacted by installation of this jumper, which jumpers out a passive device (test switch) that maintains circuit continuity and is only required when testing the Reactor Trip Breaker while it is bypassed.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the test switch is a passive device that is normally closed. Installation of the jumper replaced the closed contacts with a piece of wire. Therefore the wire and the contact are electrically equivalent. The probability of the wire failing is considered to be equal to or less than the probability of the switch to fail.

   The only malfunction that would be expected would be for the jumper to short or open. In either case, the STA would de-energize, which would close its contact and trip the Reactor Trip Breaker via the Shunt Trip Coil. Therefore, the consequences of the wire failing were no greater than if the switch failed because ultimately, both would result in a reactor trip.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the installation of a jumper in place of the test switch did not change the function of the reactor trip circuit. The circuit was capable to trip the Rx Trip Breaker. If the jumper were to short or open, the Rx Trip Breaker would trip as described above.

6G-99-0029
TMOD 99-1-010

DESCRIPTION:

The purpose of this Temporary Modification was to install a Temporary Air Jumper from the 3/4" Service Air supply at valve 1SA038A to the supply air connection to the Refueling Machine 1FH01G during Refueling Outage B1R09. This jumper consisted of a 1" diameter hose, a filter to trap particulates and reduce moisture, and an air regulator. The Unit 1 Refueling Machine did not have a permanent supply of air to operate the gripper assembly.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the installation of the air jumper to the Refueling Machine does not adversely affect the fail-safe operation of the Refueling Machine nor affect the probability of occurrence of a dropped Fuel Assembly. The temporary hoses and fittings are designed to be equivalent to the service air piping. Therefore, the probability of a loss of service air due to a hose failure is not increased over that of a system piping failure. The failure modes and effects of the Refueling Machine are not impacted by this activity, as the service air supply to the Refueling Machine is both filtered to reduce moisture and particulates and regulated to the correct working pressure.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because failure of the Refueling Machine is not affected. The limiting accident during refueling operation is that of a dropped Fuel Assembly, resulting in the highest dose released in Containment. Upon a loss of air pressure to the Refueling Machine, the grippers which hold the Fuel Assembly fail to the closed position. The station service air supply is not significantly affected due to the total failure of the hose jumper, as the makeup capability of an individual station air compressor is significantly greater than the losses that could occur from the affected air drop. No changes are made to failure modes of Safety-Related components or systems which utilize compressed air.

   No new failure modes of equipment important to safety have been identified due to the installation of this activity. The fail-safe features of the Refueling Machine are not impacted by this activity. The temporary hose and fittings are rated to a design pressure and temperature that meets or exceeds that of the system piping. Therefore, it is expected that the temporary hose will provide the same degree of reliability as the permanent piping.

6G-99-0029
TMOD 99-1-010
(Cont'd.)

Failure modes are not changed by this activity and neither the Refueling Machine nor the service air system can impair the safe shutdown of the plant. All prior failure modes of the Refueling Machine are unchanged, as upon a loss of air pressure the grippers fail to their closed position. This activity does not affect the consequences of a loss of service air in a manner that results in an increase in those consequences due to the failure of permanent system piping. No unanticipated radiological consequences are affected.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this temporary alteration does not affect any parameters upon which plant Technical Specifications are based.

6G-99-0083
TA 99-2-018

DESCRIPTION:

The purpose of this Temporary Alteration was to install temporary blocking device on the 2B Diesel Driven Auxiliary Feedwater (AF) Pump Cooler Isolation Valve 2SX178 to lock the valve in its closed position. The blocking device consisted of a 2" schedule 40 steel pipe longitudinally cut into two halves and properly clamped around the valve stem to lock the valve in its closed position. With the subject valve block installed on the stem, the block is considered to be an integral part of the valve. Stresses exerted by the valve operator were evaluated and found to be acceptable. This block is used as a backup device to the normal air supply to maintain the valve in its closed position.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because blocking the 2SX178 valve closed does not affect the initiators of any accidents. TS LCO 3.7.5 was be entered for the AF System while this Temporary Alteration was in place, In the unlikely event that the valve block were to catastrophically fail while the 2SX2186 Valve Bonnet was removed, the resultant flow rate out of the SX System, until the valve block was restored, would not result in adverse flooding conditions either in the AF Pump room or in other parts of the Auxiliary Building. Each train of Essential Service Water (SX) is capable of supplying sufficient water to compensate for the leakage that could occur from the opened 2SX2186 valve should a Design Basis Event occur.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because Technical Specification LCO 3.7.5 was entered until the valve blocking device is removed. The SX and AF Systems are designed to give the operator flexibility in aligning the desired pumps and heat exchangers to each unit. The required system functions and operating requirements as defined in the Byron Technical Specifications do not change.

3. The margin of safety, as defined in the basis for any Technical Specification, was not reduced because the action statement for loss of one train of AF has been reviewed and approved by the NRC, and ensures that the remaining train of Auxiliary Feedwater is available and adequate to support the units requirements. The redundant system remained unaffected and available. Therefore, the margin of safety was not reduced beyond what has already been reviewed and approved by the NRC.

6G-99-0091
TA 99-1-020

DESCRIPTION:

The purpose of this Temporary Alteration was to lift power leads to Diesel Generator HVAC (VD) outside air damper 1VD01YB, failing it open, in order to ensure adequate outside air is supplied to the 1B Diesel Generator (DG).

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because failing damper 1VD01YB open does not affect the probability of occurrence or consequences of any accident or equipment malfunction. The VD System is a support system which cools the DG Room to maintain operability of the DG. The DG is required for those accidents which assume a loss of offsite power. This change has no effects on the initiating factors for those accidents. The dampers fail-safe position is full open. Therefore, adequate cooling will still be maintained to the DG during any accident which requires the DG to operate.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because failing damper 1VD01YB open does not affect the probability of occurrence of any accident or malfunction. The VD system is a support system which cools the DG room to maintain operability of the DG. The DG is required for those accidents which assume a loss of offsite power. This change has no effects on the initiating factors for those accidents and no new accidents or malfunctions are created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because failing damper 1VD01YB to the full open position will not reduce the margin of safety as this is its fail-safe position. This position will ensure that full outside air is available to the 1B DG Room to ensure the room is maintained below 132° F.

6G-99-0102
TA 99-0-024

DESCRIPTION:

1. The purpose of this Temporary Alteration was to install a tee and vent valve on Instrument Air (IA) Line leading to the actuator of Gas Decay Tank to Auto Gas Analyzer Inlet Header Isolation Valve 0GW9340C. This temporary modification will be blocked open 0GW9340C to allow grab samples to be taken with the power supply to the solenoid valve for 0GW9340C de-energized.

2. Temporarily revise BCP 300-41 to take grab samples from the Gas Decay Tanks using the manual sample panel with the 0GW9340C valve open.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report was not increased because neither the probability of occurrence nor the consequences of any transient or accident described in the SAR was increased by this change. The ability to provide safe shutdown of the plant was not affected by installing a manual air jumper to maintain 0GW9340C in the open position. The Radioactive Waste Gas (GW) valve would still fail to the close position on loss of Instrument Air. Flow of sample gas to the manual gas sample panel was continued to be controlled administratively by BCP 300-41. Sample gas was routed through the manual sample panel and back to the Waste Gas Header. This change did not create any new accident initiators. The failure modes and effects of the GW System were not impacted by this activity. BCP 300-41 was temporarily revised to control the sample system in this temporary condition.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because by temporarily installing a tee and vent valve to the Instrument Air supply to the 0GW9340C valve, the potential for a new type of accident or malfunction was not created. The system functions were not changed, and chemistry sampling were procedurally controlled. The GW System is not placed into a new configuration. Sample flow was still directed back to the Waste Gas Header.

3. The margin of safety, as defined in the basis for any Technical Specification, was not reduced because the GW System is not required per the Technical Specifications.

6G-99-0107
TMOD 99-2-025

DESCRIPTION:

The purpose of this Temporary Modification was to temporarily bypass Cell 46 of Battery 212 in order to restore full capability to the battery.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the probability of accidents or equipment malfunctions is not increased since the DC batteries are not an initiator for these accidents or malfunctions. The probability of all accidents and malfunctions previously analyzed will remain unchanged. NDIT BYR 99-152 has evaluated the temporarily modified battery and concluded that it will continue to meet its design requirements. The temporarily modified battery will continue to function as designed to mitigate the consequences of all analyzed transients, equipment malfunctions, and accidents.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because jumpering out Cell 46 does not create the possibility of an accident or transient different than any previously evaluated. The jumpers that replace the inter-cell connectors of Cell 46 will meet the original design requirements of the original intercell connectors (ampacity, seismic, safety related, resistance). The seismic qualification of the batteries will not be affected by replacing the inter-cell connectors with jumpers. The seismic qualification of the battery and rack does not credit the intercell connectors for seismic integrity of the battery.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the modified battery will continue to meet its design requirements, as documented in NDIT BYR 99-152, to mitigate the consequences of all analyzed transients and accidents.

6G-99-0148
TMOD 99-1-029

DESCRIPTION:

The purpose of the Temporary Modification (TMOD) was to lift wiring to Fire Protection (FP) Thermal Sensor 1XY-FPL 55 (located at SAT 142-1) in panel 1PA49J and a zener diode was placed across the terminal points to "simulate" a cleared alarm by providing the appropriate voltage drop across the terminal points to emulate the thermal sensor. Alarms EI5 "SAT142-1" at IPM09J and A-4" UNIT 1 AREA FIRE" at OPM01J are presently in alarm due to the failure of 1XY-FP155 in the SAT 142-1 detection system. This input to the alarms was "reset" by lifting leads to 1XY-FP155 and placing a zener diode in its place to simulate a cleared alarm. This was accomplished to remove a Main Control Room distraction.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the device being bypassed is for indication only and is not an initiator of an event and its failure to operate cannot result in increased dose or impede any actions required to mitigate any accident. This TMOD does not interact with any equipment important to safety to the extent that operation would be adversely affected. The TMOD only affects indication and does not affect the Fire Suppression System's automatic actuation for a fire at System Auxiliary Transformer 142-1. The alarm circuitry being modified serves no mitigation functions. This alarm is not assumed or credited to reduce offsite dose (i.e. consequences) during normal operation or following any Design Basis Accident or transient.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the installation of this TMOD does not create any new accidents or transients. The Fire Protection System is not an initiator of any accident or transient. No additional credible accidents or transients can occur due to the installation of the zener diode in place of the thermal sensor. The installation of the zener diode in place of the thermal sensor does not allow operation of the fire detection system in a manner that would create a new equipment malfunction.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the function of the equipment affected by this TMOD is not associated with any Technical Specifications.

6G-99-0196
TMOD 99-2-040

DESCRIPTION:

The purpose of this Temporary Modification (TMOD) was to disable the trolley end of travel limit switch for the U2 Polar Crane (2HC01G) to allow it to travel closer to the Containment Building wall. The effect will be that the mechanical limits will now be limiting the movement of the Polar Crane as opposed to the electrical limit switch. However, administrative control in the form of a card on the control switch and awareness training with each Crane Operator will be used in order to prevent the trolley from powering into the end stops. An additional spotter is not necessary because the limit being disabled is physically nearest the Crane Operator; therefore, the Crane Operator is able to see the trolley movement m relation to the approximate location of the limit switch. The limit switch is approximately two to three feet from the end stop. This TMOD may be installed for any activity that requires the Polar Crane hoist to achieve a position closer to the containment wall in order to support outage activities.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the TMOD will disable the trolley end of travel limit switch for the reverse direction 'm order to allow the hoist to be brought closer to the containment wall for lifting various loads during the outage. This TMOD will only be in affect for lifting non-critical loads. This TMOD will not be installed during the lifting of critical loads, and will not affect operation of the trolley away from the limit switch toward the center of containment. Since the failure of the limits which could result in powering into the end stops, and administrative controls will prevent powering into the end stops for the trolley, the TMOD will not 'increase the probability of occurrence of a malfunction of equipment important to safety. The TMOD will not be installed while lifting or moving any critical loads and will not affect the operation of the Polar Crane over the Reactor Vessel. Therefore, vessel, fuel, and internals integrity will not be adversely affected by the removal of the limit in the reverse direction for the trolley. Lifting and lowering loads closer to the containment wall will be allowed based on the disabling of the track limit switch, but this will not increase the consequences of a malfunction of equipment important to safety.

6G-99-0196
TMOD 99-2-040
(Cont'd.)

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because during refueling, the function of the Polar Crane is to remove the Reactor Vessel head and upper internals assembly. The only critical loads carried by the Polar Crane are the Reactor Vessel head, and upper and lower internals. There are no other loads which, if dropped. would affect the cooling of the reactor or fuel integrity. When carrying the Reactor Vessel head, the operator is restricted in movement to the north-south directions by the high walls enclosed by the Steam Generators. The head must therefore be dropped on the Reactor Vessel to effect reactor coolant or fuel integrity. The results of a load drop analysis for the RESAR-414 Docket were provided in WCAP 9198, January 23, 1978. A load analysis for Byron/Braidwood would be very similar to that analysis. This TMOD will not be installed while lifting the above described critical loads; therefore, the change will not impact the analysis. The TMOD allows loads located closer to the containment wall to be lifted and lowered, but the operation of the trolley away from the wall is unchanged. Therefore, the possibility of an accident or transient different than previously evaluated is not increased. The TMOD will not be installed while lifting or moving any critical loads and will not affect the operation of the Polar Crane over the Reactor Vessel. Therefore, vessel, fuel, and internals integrity will not be adversely affected by the removal of the limit in the reverse direction for the trolley. In addition, the load drop analysis performed by Westinghouse that would be similar for Byron/Braidwood would not be affected. Lastly. lifting and lowering loads closer to the containment wall will be allowed based on the disabling of the track limit switch. but this will not increase the consequences of a malfunction of equipment important to safety in Modes 5, 6 or defueled.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the Polar Crane Trolley or Trolley Track Limit Switch is not discussed in the Basis for any Technical Specification, and the penetration protection TRM Spec. 3.8.a for the Polar Crane power feed is not affected because the power supply and its protection is not modified by this TMOD.

6H-99-0139
BAP 370-3, Rev. 23

DESCRIPTION:

The purpose of this procedure revision was to reflect changes to station commitment records. Commitment record 454-251-87-61400 has been revised to allow the use of nozzle dams or other engineering barriers if Loop Stop Isolation Valves do not close. Commitment Record 454-251-87-02200-01 addressed the same topic and has been closed to the revised Commitment Record 454-251-87-61400.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the procedure addresses only administrative controls for refueling. The revision reflects commitment record changes to allow the use of nozzle dams or other barriers to isolate RCS Loops. The administrative procedure does not change any controls or design analysis documented in the previous evaluation, so equipment important to safe is not affected.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the administrative procedure revision does not change any controls, design analysis documented in the previous evaluation, or manipulate equipment so no new accidents or malfunctions are created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the revision does not affect the basis of any Technical Specification.

6G-99-0153
BAP 1100-3, Rev. 14

DESCRIPTION:

The purpose of this procedure is to provide the contingency actions that must be put in place in order to allow Doors 0DSD432 and/or 0DSD440 from the Main Control Room to the Process Computer Rooms to be blocked open, temporarily expanding the boundaries of the Main Control Room Envelope. Technical Specification required pressures will be maintained.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the failure of the Main Control Room Envelope does not in and of itself directly lead to a LOCA or any other postulated accident. The use of compensatory measures to ensure the Process Computer Room Doors are closed in the event of an accident will ensure that the analyses for applicable accidents are not affected and therefore, the probability of each accident is not affected.

   Communication between the Main Control Room and the Process Computer Room areas is not prohibited during an accident. Although contingency actions will be in place to close the door(s) should a Design Basis Event occur, failure to close the door(s) will not affect Main Control Room dose calculations since damper isolation capability of the Main Control Room HVAC (VC) system remains intact and tripping of the Main Control Room offices HVAC System will cause the Process Computer Rooms and the Miscellaneous HVAC (VV) system duct work to become pressurized to normal Main Control Room pressure. Therefore, the consequences of the accident will not be affected.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because blocking the Main Control Room to Process Computer Room Doors open will not create the possibility of an accident or transient of a different type other than any previously evaluated. It is not expected that these doors could initiate an event of any type.

6G-99-0153
BAP 1100-3, Rev. 14
(Cont'd.)

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because

3.3.7. The VC Actuation system is unaffected by these contingency actions. The VC System will actuate upon an SI or Hi Rad signal as usual.

3.7.10. These contingency actions will not reduce the margin of safety because the Main Control Room Envelope will be verified to be adequate to meet the Technical Specification required differential pressures.

3.7.11. These contingency actions will not reduce the margin of safety of the VC Temperature Control System due to the margin in capacity of the VC Chillers.

6H-99-0121
BAR 0-39-A3 Rev. 0
BAR 0-37-A3, Rev. 0

DESCRIPTION:

The purpose of these procedure revisions was to address appropriate operator response to new alarms being added due to DCPs 9700316 and 9700317. The modification removes several Main Control Room pen chart recorders and replaces them with solid state paperless recorders and a computer data archive collection system. If a recorder communication failure occurs or if power is lost to the data collection unit, a Main Control Room Annunciator will alarm.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the previous safety analysis is bounding because the new procedure revisions do not change or manipulate any plant equipment. The new procedures will be used to direct operators to identify trouble lights and inform staff personnel to assist in troubleshooting.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because no new malfunctions are created by these procedures. The new procedures do not create any new failure modes, Operator responses, or automatic actions.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which Technical Specifications are based.

6H-99-0030
BCP 300-15, Rev. 11

DESCRIPTION:

The purpose of this procedure change was to provide improvement to the temperature setting of the heating block and allows step F3 and F4 to be done in either order for tritium sample preparation.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment Important to safety previously evaluated in the Safety Analysis Report is not increased because the procedure prepares tritium samples for analysis which is not an item required to be reviewed for safety and has no impact on any accidents or plant equipment. The revision for preparation of samples for tritium does not change the original intent or actions of this procedure.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the revision for preparation of samples for tritium does not change the original intent or actions of this procedure.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the revision for preparation of samples for tritium does not changed the original intent or actions of this procedure and has no impact on any margins of safety. Previous revisions to this procedure were validated with unknown samples to show the improved method in the procedure provides acceptable sample results.

6G-99-0080
BOP AP-80, Rev. 3

DESCRIPTION:

The purpose of this procedure change was to incorporate steps to place the Non-ESF 4KV Bus SAT Feed Breakers in the Pull-Out position; when an Emergency Diesel Generator (EDG) is being synchronized to an ESF 4KV Bus during SAT outage live bus transfers. Placing the Non-ESF 4KV Buses in Pull-Out prevents an Auto-Transfer from the UAT to the SAT. This procedure change is applicable when the SAT Cross-tie Links are in the transitional stage of being installed for single SAT operation or removed for two SAT operation.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the probability of losing offsite power under both normal and the switching configuration is essentially the same. Along with the fact that the consequences of an accident or malfunction are reduced due to a higher degree of redundancy of the equipment necessary to mitigate an accident is required at power. While placing the Non-ESF SAT feed breakers in Pull-Out will prevent Auto-Transfer from the UAT to the SAT, it will ensure the integrity of the ESF Equipment and power sources are not compromised. Analysis of the natural circulation capability of the Reactor Coolant System has demonstrated that sufficient heat removal capability exists following RCP coastdown to prevent fuel or cladding damage.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the scenarios envisioned are already enveloped by UFSAR analysis. The activity of placing the Non-ESF 4KV SAT feed breaker in Pull-Out is a conservative measure to ensure that excessive fault currents will not be seen by the ESF Switchgear. The activity could also result in loss of non-emergency AC power coincident with an accident, however this transient has already been analyzed in UFSAR section 15.2.6.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the Non-ESF 4KV Buses are not required to mitigate the consequences of an accident. 4KV circuitry discussed in the Tech. Specs. and/or Bases of Tech. Specs is confined to a discussion of 1E circuitry and the Non-ESF 4KV Sat feed is classified as NON-1E.

6H-99-0107
BOP CV-14, Rev. 14

DESCRIPTION:

The purpose of this procedure revision was to revise the methodology of degassing the RCS via 1/2CV8101, VCT to the Waste Gas Hdr Isol Vlv, by increasing/holding VCT level to/at approx. 90%, decreasing VCT pressure, and cycling 1/2CV8101 until degassing is complete. The current method raises VCT level with 1/2CV8101 open, then closes this valve and decreases VCT level as many times as necessary to degas. Safety evaluation 6G-98-0078 evaluated that it is acceptable for changing the charging pump NPSH and available backpressure to the Reactor Coolant Pump 2 seals with a reduced minimum pressure of 5 psig at the vapor space of the VCT during degassing.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because cycling the vent valve while maintaining level at approx. 90%, decreasing pressure, and increasing temp. will not increase the probability of an accident. The VCT temp. and level be maintained in an acceptable range. If level increases to 95% letdown will automatically divert to the HUT. The pressure will be reduce below the normal operating range. However the reduce pressure will still provide adequate NPSH for the changing pumps and backpressure for the RCP seals. Safety Evaluation 6G-98-0078 determined that the CVCS's ECCS functions or modes of operation are not affected by the procedure change. The ability of the VCT to provide adequate NPSH to CV pumps and RCP 2 seal backpressure are not adversely impacted by the change.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the revised degassing process is essentially the same as the method described in the UFSAR. The gasses purged from the VCT will be processed in the same manner, through the Waste Gas System. The temporary reduction of the vapor space pressure in the VCT during degassing of the RCS will not adversely affect the available NPSH at the CV Pumps during normal plant operations.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the gas mixture of the Waste Gas Decay Tanks will remain within the explosive gas mixture limits during performance of CV-14, Degassing the RCS and Pressurizer. This degas procedure requires Chemistry to increase sampling frequency of the gas decay tanks while degassing.

6G-99-0019
BOP CV-14, Rev. 14

DESCRIPTION:

The purpose of this procedure change was to permit degassing of the RCS via the Volume Control Tank (VCT) to Waste Gas Header Isolation Valve. The procedure change maintains VCT level high and pressure low instead of raising and lowering VCT level. Nitrogen at 15 psig will be used to purge hydrogen from the VCT. Editorial changes to the procedure were also incorporated. The Safety Evaluation resulted from the identified need to change a procedure in the Safety Analysis Report (UFSAR Section 11.3.2.6) that describes the steps to degas the VCT. The change is incorporated in DRP 8-013.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the process is not substantively changed. Cycling the VCT to the Waste Gas Header (as opposed to also raising and lowering VCT level) does not increase the probability of a Waste GAS System leak or malfunction. Since the gases are processed the same way, there is no effect on off-site dose.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because none of the functions of the VCT are revised by this method of degassing.

3. The margin of safety, as defined in the bases for any Technical Specification is not reduced because the degassing operation is technically the same as described in the SAR with a reduced number of required operator actions.

6G-99-0033
BOP CV-14, Rev. 14

DESCRIPTION:

The purpose of this procedure change was to revise the methodology of degassing the RCS via 1/2CV8101, VCT to the Waste Gas Hdr Isol Vlv, by increasing/holding VCT level to/at approx. 90%, decreasing VCT pressure, and cycling 1/2CV8101 until degassing is complete. The current method raises VCT level with 1/2CV8101 open, then closes this valve and decreases VCT level as many times as necessary to degas.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because cycling the vent valve while maintaining level at approx. 90%, decreasing pressure, and increasing temperature will not increase the probability of an accident. The VCT temperature and level will be maintained in an acceptable range. If level increases to 95% letdown will automatically divert to the HUT. The pressure was reduced below the normal operating range. However the reduced pressure still provided adequate NPSH for the charging pumps and backpressure for the RCP seals.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the revised degassing process is essentially the same as the method described in the UFSAR. The gasses purged from the VCT was processed in the same manner through the waste Gas System.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced. The gas mixture of the Waste Gas Decay Tanks remains within the explosive gas mixture limits during performance of BOP CV-14, degassing the RCS and Pressurizer. This degas procedure required Chemistry to increase sampling frequency of the gas decay tanks while degassing.

6H-99-0230

BOP DG-M1	Rev. 14
BOP DG-M1A	Rev. 6
BOP DG-M1B	Rev. 6
BOP DG-M2	Rev. 10
BOP DG-M2A	Rev. 3
BOP DG-M2B	Rev. 3
BAR 1/2PL07J1-D6	Rev. 2
BAR 1/2PL08J-1-D6	Rev. 2

DESCRIPTION:

The purpose of these procedure revisions was to update station procedures and Test Report 2PDS-DG278A/B to incorporate the plant changes made to the Diesel Generators (DG) made per DCP 9800561, "Install Delta P Instrumentation across DG Lube Oil Strainer." The safety evaluation for this activity is accomplished via a validation of previously performed safety evaluation (6G-98-0264, Revision 1) for DCP 9800560 and DCP 9800561. Changes to the UFSAR were proposed by 6G-98-0264, Rev. 1 and are tracked by DRP 7-270.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the change affects no previously analyzed accidents.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the function of the DG Lube Oil System is not changed.

3. The margin of safety, as defined in the bases for any Technical Specification is not reduced because the modification does not change Technical Specification Bases parameters.

6H-99-0040
BOP RC-0, Rev. 11

DESCRIPTION:

The purpose of this procedure revision was to change a procedure described in the UFSAR for filling and venting the Reactor Coolant System (RCS). The change incorporates vacuum filling methodology previously evaluated by Braidwood Safety Evaluation BRW-SE-1998-1837 via validation. The change required to the UFSAR is tracked via DRP 7-261. The procedure is a new method of filling the RCS piping following outages. An air ejector skid will be connected to a flange downstream of the RCS normal bypass line vent valves (1/2RC8029A/B/C/D); with leakage paths and sensitive instrumentation isolated and with the Loop Stop Isolation Valves (LSIVs) closed, a vacuum will be drawn on the isolated RCS loops (one at a time). Filling of the loops is accomplished via the Loop Fill Line from the Chemical Volume and Control System (CV) with the CV Pumps taking suction from the Reactor Water Storage Tank or the Volume Control Tank with water supplied from the Reactor Coolant Make-up System.

SAFETY EVALUATION SUMMARY:

1. Three analyzed events apply to the change.

   a. Reactivity & Power Distribution Anomaly – Dilution during Cold Shutdown
   b. Loss of RH during Shutdown Conditions
   c. Loss of Coolant Accident

   The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because:

   a. the vacuum fill process does not impact equipment necessary to mitigate a RCS dilution event,

   b. the vacuum conditions for the isolated RCS loops (including the pressurizer) will not result in an inadequate net positive suction head for the RH pumps and,

   c the stresses from the vacuum fill process are insignificant as compared with normal operating conditions and level instrumentation has been appropriately addressed by the revised procedure.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the changes do not degrade the operation of any plant equipment as to create an interaction/failure different from those analyzed. Instrumentation/ Equipment potentially affected by the vacuum fill process has been appropriately analyzed and mitigated by the revised procedure.

6H-99-0040
BOP RC-0, Rev. 11
(Cont'd.)

3. The margin of safety, as defined in the bases for any Technical Specification is not reduced because the changes affects no Technical Specification Bases.

6G-98-0027
BOP RC-8A, Rev. 12

DESCRIPTION:

The purpose of this procedure change was to bypass the interlocks on the Loop Stop Isolation Valves (LSIVs) for idle Reactor Coolant System (RCS) Loops. Idle is defined as the Reactor Coolant Pump (RCP) not in operation.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the activity was performed in Modes 5 and 6 when a start up accident is not possible. The interlocks prevent start up of an inactive loop at power. Technical Specifications does not allow operation at power without all 4 RCS Loops and RCPs in operation. This accident was deleted from the UFSAR in Revision 6. An accident or the consequences were not possible when the activity was performed.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because this activity was performed in a condition (Modes 5 and 6) when the start up accident is not possible. No new conditions are created for any other accidents. The LSIVs were used for their designed function of maintenance. The requirements of Technical Specifications were maintained at all times.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the LSIVs were used for their designed maintenance function. The requirements of Technical Specifications were maintained throughout the activity.

6G-98-0097, Rev. 1
BOP RC-9, Rev. 9
BOP RC-9A1, Rev. 0

DESCRIPTION:

The purpose of these procedure changes was to provide for vacuum filling of the Reactor Coolant System (RCS) after loop(s) have been drained. The procedures ensure that all potential leak paths and equipment that might be damaged are isolated prior to initiating the vacuum. The procedures provide for a vacuum to be drawn with the Loop Stop Isolation Valves (LSIVs) closed. Once vacuum of approximately 20 in Hg. is obtained, loop fill via the centrifugal charging pump loops will be accomplished. In addition, minimum RCP seal water flow is maintained during this evolution to ensure continued integrity of the RCP seals is maintained. This activity was performed to reduce critical path time for RCF fill from outages, extend the life of RCP motors and seals, improve RCS chemistry, and reduce personnel radiation exposure.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the activity does not affect, disable, or isolate any safety related equipment. Temporary equipment is installed, operated, and removed per procedure and no permanent changes to plant SSCs are made. In terms of accident probability, the activity does not represent an increase over existing Operating Procedures. In addition, the activity occurs on and only affects isolated portions of the RCS. The systems that control reactor subcriticality, provide core cooling and inventory, and makeup/boration control are unaffected by this activity. The SSCs that are required to mitigate the consequences of any accident are not isolated or affected by this activity. Therefore, the probability or consequences of any accident previously evaluated in the Safety Analysis Report is not increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the activity modifies the process of filling an isolated RCS loop after refueling/outage activities. The created vacuum condition in the RCS has been evaluated and determined to have no impact on any SSCs and there are no new credible failure modes created by this activity. In addition, all safety related equipment will continue to perform its intended design function.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this activity does no affect any parameters upon which plant Technical Specifications are based.

6G-98-0009
BOP TR-9, Rev. 2

DESCRIPTION:

The purpose of this procedure change was to allow the Waste Water Treatment System to process contaminated water and return it to one of either two locations:

1. Back to radwaste (i.e., Release Tank 26T) if the activity level is above the setpoint for 0PR05J, or

2. Discharged to the flume is the activity level is sampled and below the setpoint for 0PR05J.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the equipment addressed by the change is the Waste Water Treatment System (WWTS) and is not safety-related.

   The WWTS involved is being operated in a manner outlined in the ODCM with one exception. In the ODCM, the contamination being addressed by the WWTS originates from primary to secondary leaks. The purpose for the activity is to address secondary contamination originating from the introduction of Na-24 as a tracer for Steam Generator moisture carryover testing. The use of Na-24 for testing moisture carryover has not previously addressed in the ODCM. The introduction of a new source of short lived radioisotopes is bounded by the current analyses for a SG Tube Rupture and the subsequent treatment of contaminated water on the secondary side.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the activity will introduce no new failure modes or alter the consequences of existing failures. Introduction of a new source of radioisotopes into the secondary side is bounded by the current analyses.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not impact any parameters upon which Technical Specifications are based.

6H-99-0397	
BOP VV-13	Rev. 2
BAR 1PM09J-A8	Rev. 2
BAR 1/2PM09J-A7	Rev. 1
BAR 0-33-C7	Rev. 7

DESCRIPTION:

The purpose of these procedure revisions was to modify the above listed procedures above as follows:

Add compensatory measures for the Computer Room Doors being open. These comp measures are being added to BAP 1100-3 in accordance with the 50.59 Safety Evaluation described below

Add a requirement to close the computer room door should a fire occur in the computer room or the Main Control Room. This is one of the comp measures being added to BAP 1100-3.

Provide guidance that the Main Control Room Envelope may be "impaired' under certain circumstances in accordance with an approved Plant Barrier Impairment (PBI). This impairment would be in accordance with BAP 1100-3.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because revising these procedures will help to ensure that the Main Control Room Envelope is maintained as expected. As discussed in the 50.59 Safety Evaluation which added the pre-evaluated comp measures to BAP 1100-3, the failure of the Main Control Room Envelope does not in and of it self directly lead to a LOCA or any other postulated accident. The use of compensatory measures to ensure the Process Computer Room Doors are closed in the event of an accident will ensure that the analyses for applicable accidents are not affected and therefore the probability of each accident is not affected.

   Communication between the Main Control Room and the Process Computer Room areas is not prohibited during an accident. Although contingency actions will be in place to close the door(s) should a Design Basis Event occur, failure to close the door(s) will not affect Main Control Room dose calculations since damper isolation capability of the VC system remains intact and tripping of the Main Control Room offices HVAC System will cause the Process Computer Rooms and the VV System duct work to become pressurized to normal Main Control Room pressure. Therefore, the consequences of the accident will not be affected.

6H-99-0397

BOP VV-13	Rev. 2
BAR 1PM09J-A8	Rev. 2
BAR 1/2PM09J-A7	Rev. 1
BAR 0-33-C7	Rev. 7

(Cont'd.)

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because revising these procedures will help to ensure that the Main Control Room Envelope is maintained as expected. As discussed in the 50.59 Safety Evaluation which added the pre-evaluated comp measures to BAP 1100-3, blocking the Main Control Room to Process Computer Room Doors open will not create the possibility of an accident or transient of a different type than any previously evaluated. It is not expected that these doors could initiate an event of any type.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because revising these procedures will help to ensure that the Main Control Room Envelope is maintained as expected. As discussed in the 50.59 Safety Evaluation which added the pre-evaluated comp measures to BAP 1100-3, for the following Tech. Specs.

3.3.7 The VC Actuation System is unaffected by these contingency actions. The VC System will actuate upon an SI or Hi Rad signal as usual

3.7.10: These contingency actions will not reduce the margin of safety because the Main Control Room Envelope will be verified to be adequate to meet the Technical Specification required differential pressures.

3.7. 11: These contingency actions will not reduce the margin of safety of the VC Temperature Control System due to the margin in capacity of the VC chillers.

6G-98-0258
BRP 5820-18, Rev. 0
BRP 5820-18T1, Rev. 0

DESCRIPTION:

The purpose of these procedure changes was to change the setpoints for the Unit 1 and Unit 2 Steam Jet Air Ejector (1/2RE-PR027J) to account for current RCS noble gases concentrations. The currently applied SSCR methodology does not allow for timely and accurate changes. 1/2PR027J initiates an actuation of the bypass valves and the off-gas filter unit. The activity is a test or experiment not described in the SAR.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the function of the monitor remains unchanged. The ability of the monitor to correctly and conservatively actuate the bypass valves and the off-gas filter unit is effectively unchanged.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because changing the setpoint for 1/2PR027J does not affect other SSCs. The monitors are recalibrated to detect a leak rate much lower than the leak rate limit of 150 gpd in the Technical Specifications.

3. The margin of safety as defined in the bases for any Technical Specification is not reduced because the activity does not change Technical Specification Bases parameters. Technical Specification 4.4.6.2 described the bases for "Operational Leakage". This TS Bases is unaffected by the change.

6G-99-0051
0BOL 6.8 LCOAR, Rev. 2

DESCRIPTION:

The purpose of this procedure change was to add a clarifying note to Required Action B.1 of 0BOL 6.8 (ITS 3.6.8) to indicate that verification by administrative means that the hydrogen control function is maintained is performed by administratively verifying that one train of RCFCSs and one train of CS are OPERABLE and that the Post-LOCA Purge Unit is available. This is performed by the review documented in Attachment A of 0BOL 6.8.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because this is an administrative change that provides direction to the operators concerning what systems are required in order to conclude that the "hydrogen control function" is maintained should both Hydrogen Recombiners become inoperable, per ITS Spec 3.6.8. No change is being made to the plant. This change will ensure that an alternative gas control mechanism is available for up to 7 days following loss of both hydrogen recombiners. (If this alternate gas control mechanism is not available, the unit must be placed in Mode 3 within 6 hours.) Therefore, the probability or consequences of an accident is not increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because there is no change to any equipment being made by this revision to the BOL. It is an administrative change that will better define the alternate equipment required to be OPERABLE should both hydrogen recombiners become inoperable. Therefore, there is no effect on the probability of the malfunction of equipment important to safety.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this change does not physically impact the plant it only provides better direction for the Operations Department to ensure that they meet the intent of the Technical Specification.

6H-99-0281
1BOL 7.b LCOAR, Rev. 2

DESCRIPTION:

The purpose of this procedure revision was to update 1 BOL 7.b LCOAR - Snubbers, in accordance with Revision 2 of the Technical Requirements Manual for Unit 1.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the procedure change pertaining to the actions required for inoperable Snubbers is an exact word for word duplicate of the revision to the TRM. The Safety Evaluation governing the TRM revision encompasses the procedure change.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the procedure change pertaining to the actions required for inoperable Snubbers is an exact word for word duplicate of the revision to the TRM. The Safety Evaluation governing the TRM revision encompasses the procedure change.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the procedure change pertaining to the actions required for inoperable Snubbers is an exact word for word duplicate of the revision to the TRM. The Safety Evaluation governing the TRM revision encompasses the procedure change.

6H-99-0282
2BOL 7.b LCOAR, Rev. 1

DESCRIPTION:

The purpose of this procedure revision was to update 2BOL 7.b LCOAR - Snubbers, in accordance with Revision 2 of the Technical Requirements Manual for Unit 2.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the procedure change pertaining to the actions required for inoperable Snubbers is an exact word for word duplicate of the revision to the TRM. The Safety Evaluation governing the TRM revision encompasses the procedure change.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the procedure change pertaining to the actions required for inoperable Snubbers is an exact word for word duplicate of the revision to the TRM. The Safety Evaluation governing the TRM revision encompasses the procedure change.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the procedure change pertaining to the actions required for inoperable Snubbers is an exact word for word duplicate of the revision to the TRM. The Safety Evaluation governing the TRM revision encompasses the procedure change.

6G-98-0222
0BOS 0.1-0, Rev. 19

DESCRIPTION:

The purpose of this procedure change was to add a "trigger" tying the operability of 0SX02PB to a river level of at least 671.0', until the "B" intake bay is returned to service as evaluated in Operability Assessment 98-044. This procedure change represents a change to a procedure described in the SAR.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the change is only a change to a procedure not the evaluation of a degraded condition. The procedure change has no effect on operation.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the activity is a change to a passive surveillance to add an additional check of river level. The activity does not change the way SSCs are maintained, operated, or configured.

3. The margin of safety as defined in the bases for any Technical Specification is not reduced because the activity does not change Technical Specification Bases parameters.

6H-99-0013
1BOS 3.1.1-20, Rev. 18

DESCRIPTION:

The purpose of this procedure change was to incorporate Temporary Alteration 99-1-002 which installed a jumper to bypass the Reactor Trip Breaker (RTA) Train A Shunt Trip Test Pushbutton which failed open so the trip signal was not removed.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the affected accidents are a LOCA and Fuel Handling Accident. The removal of if the jumper fails, the shunt trip will open causing the control rods to insert.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the jumper acts as the pushbutton would, if functioning properly, so no new accidents are created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the margin of safety is not affected because the RTA shunt trip test can still be performed to ensure the Reactor Trip Breaker open on demand to comply with Tech. Specs.

6H-99-0018
1BOSR 3.1.5-1, Rev. 1

DESCRIPTION:

The purpose of this procedure change was to incorporate Temporary Alteration 99-1-002 which installed a jumper to bypass the Reactor Trip Breaker (RTA) Train A Shunt Trip Test Pushbutton which failed open so the trip signal was not removed.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because if the jumper fails, the shunt trip will open causing the control rods to insert which is conservative.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the jumper acts as the pushbutton would, if functioning properly, so no new accidents are created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the RTA Shunt Trip Test can still be performed to ensure the Reactor Trip Breakers open on demand to comply with Tech. Specs.

6H-99-0177
1 BOSR 3.2.7-605 A/B, Rev. 3

DESCRIPTION:

Revise slave relay surveillances to reflect DCP that re-wired the Post LOCA Hydrogen Monitor Containment Isolation Valves to be actuated from SSPS relay 605 instead of relay 607.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the previous evaluation addressed the design change to modify the circuitry for Containment Phase A Isolation testing. As a result of the modification, post LOCA Hydrogen Monitor Containment Isolation Valves 1PS230A/B were re-wired to a different relay with the same function. The probability of occurrence is not changed since the previous evaluation addressed the re-wiring and the current revision ensures the proper relay verifies the correct valve responses.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the current procedure revisions implement the modification previously evaluated. The relay re-wiring does not change the Containment Isolation Function of the Post LOCA Hydrogen Monitors so the possibility of an accident is not affected.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the margin of safety is not affected because the post LOCA hydrogen monitor Containment Isolation Valves continue to provide isolation on Phase A signal as designed.

6G-99-0026
1/2BOSR 3.2.9-4, Rev. 2

DESCRIPTION:

The purpose of this procedure change was to implement Westinghouse Technical Bulletin ESBU-TB-98-01-RI recommendation for testing P-4 interlock circuit in the Solid State Protection System cabinet. Revise the 18 month surveillance to take voltage measurements using portable digital voltmeter at the input of universal logic card instead of using the permanently installed voltmeter at the Reactor Trip Breaker cubicle.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the P-4 interlock functions that P-4 performs are not credited in any accident or transient. The test is performed in Mode 4, 5, or 6. In these modes, Main Feedwater and Main Turbine are not in operation.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the testing is performed in Mode 4, 5, or 6 and the functions that the P-4 interlock inputs to are not in operation in these modes. The P-4 interlock functions are not credited in any accident or transient.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the testing is performed in Mode 4,5 or 6 and the functions that the P-4 interlock inputs to are not in operation in these modes. The P-4 interlock functions are not credited in any accident or transient.

6H-99-0063
1/2BOSR 6.3.3-1 , Rev. 1
1/2BOSR 6.3.4-1, Rev. 1

DESCRIPTION:

The purpose of these procedure revisions were to revise Unit 1 & Unit 2 inside and outside containment isolation devices verification procedures to change locations of PR033A, B, C, D and PR002E, F valves from outside containment devices to inside containment devices per UFSAR CHANGE 8-014.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because containment isolation devices are still verified, the change from moving devices from outside containment to inside containment changes verification frequencies. Outside devices are verified monthly and inside devices on a cold shutdown frequency due to operator dose. Requirements to be in a closed position remain the same.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because no change to the requirements are made. Change to survey the valves on a different frequency is due to ALARA concerns and these valves exist inside the containment airlocks. Valves inside containment and inside the airlock are less accessible due to ALARA and less likely of mispositioning.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the margin of safety is not affected because no Technical Specification or basis for Technical Specification is being affected.

6H-99-0370
BVP 800-11, Rev. 4

DESCRIPTION:

The purpose of this procedure revision was to increase the inert gas injection flow limit to 20 cfm.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because this SPP contains administrative controls, appropriate cautions, and limitations which limit the amount of inert gas which can be added to the Main Condenser. In addition, all permanent plant equipment will be operated in accordance with established station procedures

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the activity affects the Main Condenser and the Condenser Offgas System. Analyses have been performed for a complete loss of condenser vacuum. These analyses are bounding for the activity.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the bases for the Technical Specifications are unaffected. The Secondary Water Chemistry Program will not be adversely affected by this activity.

6G-98-0086
1BVS 2.3.5-1, Rev. 12

DESCRIPTION:

The purpose of this procedure change was to evaluate the performance of the replacement Steam Generators. Accuracy of the calorimetric and the RCS flow measurements were performed at 75% and 98% power. Conservative assumptions were made regarding flows and moisture carryover to ensure 100% power would not be exceeded.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because this activity evaluated the performance of the changed out the Steam Generators. The evaluations performed ensure the replacements function as the originals. Additionally, extensive testing was performed to evaluate performance after installation. Therefore, the safety analysis is correct for the new equipment.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because this surveillance was used to verify proper performance. No new failures are created by this procedure.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this activity ensures and evaluates the original design flows for the RCS are maintained. Technical Specification requirements are maintained.

6G-98-0056
1BVS 4.6.2.2-1, Rev. 12

DESCRIPTION:

The purpose of this procedure change was to provide flow through Safety Injection (SI) Check Valve 1SI8956B Check Valve to reseat the valve after flow. This valve failed a leakage test at 380 and must be redone. The maximum amount of flow was less than 20% of the B SI Accumulator.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because this activity was performed in a mode when the accumulators are not required. This was a minimum amount of flow to open the valve, flush the general seating area, and then reseat the valve with the RH pressure in the system. As the pressurizer volume is sufficient to accept the quantity of water, the system remains within the original design criteria.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because this activity was performed and demonstrated the design function of the accumulator. The secondary result was to exercise the check valve. Final operability was determined through performance of the actual 1BVS 4.6.2.2-1 Surveillance. These results demonstrated operability prior to elevating in mode and meeting the requirement for operability in Technical Specifications.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the requirements of Technical Specifications did not require operability of the accumulators. All requirements were met prior to the performance of the temporary procedure. Operability was verified prior to the accumulators being required by Technical Specifications.

6G-99-0094
0BVSR DC-3A, Rev. 0
0BVSR DC-3B Rev. 0

DESCRIPTION:

The purpose of these new procedures was to allow an IEEE-450 based battery test to be performed on either the Switchyard Relay House System 1 Battery [North] (0DC12E) or the System 2 Battery [South] (0DC13E). Application of the procedure executes a full battery discharge test to determine battery capacity based on manufacturer's rating. The procedure also measures cell parameters such as cell float voltages, specific gravity, electrolyte levels, electrolyte temperatures, connection resistance, visual integrity and terminal voltage.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the activity was the implementation of a new procedure. A total loss of Switchyard DC (both Systems 1 and 2) does not result in the loss of offsite power and thus emergency and non-emergency AC power to the plant. During the performance of these procedures a backup battery charger and generator on a trailer are made available as compensatory measures to maintain reliability and availability of the associated DC buses. During the application of the procedure, the subject battery is isolated from its distribution. The application of the procedure surveillance has no direct electrical interaction or specific relationship to the initiation of an accident or a malfunction of equipment important to safety and the probability of occurrence are not increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the new procedure was determined to be consistent with the recommendations and requirements of the Original Equipment Manufacturer (OEM). During the performance of these procedures a backup battery charger and generator on a trailer are made available as compensatory measures to maintain reliability and availability of the associated DC buses. The backup battery trailer has an alternate AC supply capability with its self contained generator and it has the capability to support DC loads longer on battery alone since the battery contained in the trailer is of a higher capacity.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because Technical Specification Bases B3.8.1 states, "The transmission network and Switchyard are maintained in accordance with the UFSAR, and are not governed by the requirements of Technical Specifications."

6G-99-0100
SAAD 1990086 (Process Computer Procedure Change

DESCRIPTION:

Purpose of the change to these procedures is to incorporate the Byron U-1 Replacement Steam Generator (RSG) Moisture Carryover (MCO) value of 0.077% into the plant process computer calorimetric program, the operating daily calorimetric surveillance procedure, and the System Engineering RCS Precision Calorimetric Surveillance Procedure. The revised MCO value was determined from SPP 99-015. The test results have been reviewed and approved by Byron Station PORC . The change will allow for a more representative determination of calorimetric power based on the approved test results. This change will result in a decrease of indicated reactor power of approximately 0.1% RTP. The current MCO value of 0% was put in place to conservatively overestimate reactor power until the results of the Post SGR Startup Tests

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the Steam Generator Moisture Carryover Value is used in the determination of Plant Calorimetric Power. Plant Calorimetric Power is subsequently used for calibration of the power range Nuclear Instrumentation System (NIS) and determination of RCS flowrate. The revised nominal MCO value will provide a more representative value of main steam enthalpy, thereby providing a more representative determination of calorimetric power and RCS flowrate. Since the power range nuclear instrumentation will be calibrated against a more representative calorimetric power indication, the associated reactor trip setpoints will occur as required per the UFSAR and Technical Specifications. The precision RCS flowrate measurement will also be more representative of actual flowrate, therefore, all related assumptions are bounded. Use of the revised nominal MCO value is consistent with the existing Calorimetric Program Design Basis. Therefore, the initial and post accident plant conditions assumed in the accident analysis remain unchanged.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the nominal Unit 1 Steam Generator Moisture Carryover (MCO) value has been revised and validated per Byron Plant Review Report 99-011. Implementation of the revised MCO value will allow the various calorimetric programs to produce a more representative indication of actual reactor power. Use of the revised nominal MCO value is consistent with the existing design basis assumptions related to calorimetric power determination and Reactor Coolant System flowrate determination. Since the plant will continue to operate in accordance with existing design basis assumptions, the possibility of an accident or malfunction different from those previously evaluated will not be created.

6G-99-0100
SAAD 1990086 (Process Computer Procedure Change
(Cont'd.)

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the revised MCO value is a mathematical value used to determine main steam enthalpy for use in the plant calorimetric programs and procedures. The revised MCO value has been validated to more accurately represent actual Unit 1 MCO than the current value resulting in a more accurate/realistic determination of calorimetric power. Therefore, the effect this change has on the calorimetric programs does not result in any new equipment failures or impact any currently analyzed failures in the UFSAR.

   Steam Generator moisture carryover potentially results in increased High Pressure Turbine erosion. However, the revised MCO value of 0.105% is significantly less than the original "design" value of 0.25% for Unit 1 and Unit 2. Therefore, this revised MCO value will not have any increased consequences to plant secondary systems or decrease the margin of safety.

6G-99-0053
1BVSR 5.5.8.SD.3-1, Rev. 2

DESCRIPTION:

The purpose of this procedure change was to incorporate new acceptance criteria into 1BVSR 5.5.8.SD.3-1 to establish an 8 gpm limit for maximum leakage through the related 1SD002A-H, and 1SD005A-D Blowdown (SD) valves per Steam Generator. This leakage limit is for the total combined leakage for applicable valves for each Steam Generator.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the minor increase in blowdown leakage has no effect on the flow or temperature of the Feedwater system so does not increase the probability of occurrence of a Feedwater Malfunction/Feedwater Temperature Reduction. The increase in blowdown leakage decreases the affect of a Feedwater Malfunction as it offsets the affects of increased FW flow. Therefore, the analysis of record remains bounding. For the FW temperature decrease, the limiting DNB is bounded by the Steamline Break Event. In consideration of this there is no increase in the consequences of a Feedwater Malfunction/Feedwater Temperature Reduction accident/transient.

   Excessive Load Increase
   The minor increase in blowdown leakage has no effect on increasing the secondary system steam flow and therefore does not increase the probability of occurrence of an Excessive Load Increase. Since the cooldown produced by the SG Blowdown leakage is a very small fraction of that produced by the excessive load increase itself, the effect in the results of this transient would be insignificant. There is no increase in the consequences of an Excessive Load Increase accident/transient.

   Inadvertent Opening of a Steam Generator or Safety Valve
   The minor increase in blowdown leakage does not cause an Inadvertent Opening of a Steam Generator or Safety Valve and does not increase the probability of occurrence of this accident/transient. The UFSAR analysis for this event is analyzed in the same way as the Steamline Break Event and assumes event initiation at hot shutdown conditions, which bound event initiation at full power conditions. Any SG leakage through the blowdown lines would be bounded by the Steam Line Break evaluation in UFSAR Section 15.1.5. There is no increase in the consequences for Inadvertent Opening of a Steam Generator or Safety Valve accident/transient.

6G-99-0053
1BVSR 5.5.8.SD.3-1, Rev. 2
(Cont'd.)

Steam System Piping Failure
The minor increase in blowdown leakage has no effect on a Steam System Piping Failure and does not increase the probability of occurrence of this accident/transient. In the event of a Steam System Piping Failure the current calculations (CN-TA-97-092) show a resultant flow rate of 770 gpm to the faulted SG and 351 gpm per intact SG. Since the steam lines are isolated fairly quickly, the intact SG flow rate has very little effect on the analysis. Therefore, reducing the AF flow from the design basis value of 1784 to 1752 gpm to the faulted SG (thus allowing for 32 gpm secondary side leakage) the analysis would still bound the actual AF flow of 770 gpm to the faulted SG and 351 gpm per intact SG. This 32 gpm could then be included as margin for allowable leakage through the SG Blowdown lines. This methodology would result in a more severe cooldown than would be the case for actual leakage, since AF water enthalpy would be much less than that of the leak flow. Therefore the current analysis of record remains valid for this event. For the HFP Steamline Break Event, the event is terminated at around 20 seconds. From the time of initiation until termination, the volume removed from the secondary side would be 10.66 gallons. This corresponds to about 72 lbm lost from all four SGs (using a density of 50.65 lbm /cu. ft. at 500 psig). This amount of mass is insignificant compared to the inventory of the secondary side (over 100,000 lbm per SG), so the SG Blowdown leakage has no affect on this analysis. In consideration of this the increase in blowdown leakage does not increase the consequences of a Steam System Piping Failure.

Loss of External Load/Turbine Trip
The minor increase in blowdown leakage does nothing to cause a Loss of External Load/Turbine Trip and does not increase the probability of occurrence of this accident/transient. Following a loss of load/Turbine Trip, the sudden reduction in steam flow will result in increase in pressure and temperature on both the primary and secondary side. The SG leakage in the blowdown lines will reduce the secondary side inventory, which will reduce the SG pressure. While the time required to reach the MSSV setpoint may be increased, the properties of this relief will not be changed. Therefore, for secondary side pressure concerns, the current analysis of record is limiting. The primary side pressure is determined by primary to secondary heat transfer and the pressurizer relief valve properties. None of the properties of the pressurizer safety valves or PORVs are affected by SG leakage. Since the secondary side pressure is reduced by the SG leakage, the heat transfer across the SG tubes is not degraded as much as in the current analysis of record. Moreover, the SG leakage will be removing some of the sensible heat, reducing the severity of the primary side heatup. Finally, the delayed opening of the MSSV is possible, but since this occurs in the first 10 seconds of the event, the small amount of leakage could not have a significant effect on the results. Therefore the current analysis of record remains bounding for this event. There is no increase in the consequences for a Loss of External Load/Turbine Trip accident/transient.

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Inadvertent Closure of Main Steam Isolation Valves
The minor increase in blowdown leakage does nothing to cause an inadvertent closure of Main Steam Isolation Valves and does not increase the probability of occurrence of this accident/transient. This event is bounded by the current Turbine Trip analysis because it assumes termination of all steam flow at event initiation. Any effect on this event by the SG leakage is bounded by the evaluation of the Turbine Trip event. There is no increase in the consequences related to an Inadvertent Closure of Main Steam Isolation Valves accident/transient.

Loss of Condenser Vacuum and Other Events Causing a Turbine Trip
The minor increase in blowdown leakage does nothing to cause a Loss of Condenser Vacuum or other events causing a Turbine Trip and does not increase the probability of occurrence of this accident/transient. This event is bounded by the current Turbine Trip analysis because it assumes termination of all steam flow at event initiation and no steam dump operation. Any effect on this event by the SG leakage is bounded by the evaluation for the Turbine Trip event. There is no increase in the consequences related to a Loss of Condenser Vacuum and other events causing a Turbine Trip.

Loss of Offsite Power/Loss of Normal Feedwater
The minor increase in blowdown leakage does nothing to cause a Loss of Offsite Power/Loss of Normal Feedwater and does not increase the probability of occurrence of this accident/transient. The current record of analysis for the Unit 1 Loss of Normal Feedwater/Station Blackout is Westinghouse Calculation CN-TA-97-096. NFM assessed the impact of the additional blowdown leakage against this calculation as shown in correspondence NFM:PSA:99-013. Both of these events are terminated when the core decay heat falls below the heat removal capability of the Aux Feed system. This is shown by ensuring that the pressurizer does not become water solid, which precludes water relief to the relief tank through the pressurizer safety valves or PORVS. NFM calculated the relationship of the additional 8 gpm leakage from blowdown per Steam Generator against the effects of primary heat removal. They determined that the heat removal capability of Aux Feed is not affected since Aux Feed flow is the same. The time for core decay heat to decrease below the AF heat removal capability remains unchanged. The methodology conservatively showed that the pressurizer does not overfill with 8 gpm leakage from all 4 Steam Generators. There is no increase in the consequences related to a Loss of Offsite Power/Loss of Normal Feedwater accident/transient.

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Feedwater System Pipe Break
The minor increase in blowdown leakage does nothing to cause a Feedwater System Pipe Break and does not increase the probability of occurrence of this accident/transient. This event results in a rapid cooldown of the primary side followed by a heatup after all SGs are degraded. The cooldown part of this event is bounded by the steamline break evaluation. The heatup part of this event was evaluated by NFM in regards to the additional blowdown leakage as shown in correspondence NFM:PSA:99013. NFM evaluated the heatup effects on the primary and determined the margin to hot leg saturation with 8 gpm SG leakage per SG. They concluded that the margin remains less than the margin at the beginning of the heatup portion, and therefore the current analysis of record bounds the 8 gpm leakage per SG. There is no increase in the consequences related to a Feedwater System Pipe Break accident/transient.

Loss of Flow
The minor increase in blowdown leakage does nothing to cause a loss of reactor coolant flow and does not increase the probability of occurrence of a Loss of Flow accident/transient. The loss of flow events are analyzed for their effect on the primary system. The important parameters are RCP characteristics and fuel kinetics parameters. Any SG leakage does not affect these parameters and therefore has no effect on the analysis. There is no increase in the consequences related to a Loss of Flow accident/transient.

Shaft Break/Locked Rotor
The minor increase in blowdown leakage does nothing to cause the breaking or locking of a reactor coolant pump and does not increase the probability of occurrence of a Shaft Break/Locked Rotor accident/transient. The Locked Rotor Event is analyzed for its effect on the Primary System. The important parameters are RCP characteristics and fuel kinetics parameters. Any SG leakage does not affect these parameters and therefore has no effect on the analysis. In NFM:PSA-99-013, NFM has indicated that for the shaft break event the heat transfer in the faulted Loop Steam Generator is reduced due to reduced flow. The heat transfer continues because the reactor coolant in the tubes cools down while the shell side temperature of all the Steam Generators increases (turbine steam flow is reduced to zero upon plant trip). The rapid expansion of the coolant causes an insurge into the pressurizer and an RCS pressure increase. The SG leakage will not reduce the secondary side level to below the SG tubes since this event is terminated within 10 seconds of initiation. Therefore, the heat transfer from the primary to secondary side will not be affected, and the current analysis of record remains bounding. There is no increase in the consequences related to a Shaft Break/Locked Rotor accident transient.

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Rod Withdrawal from Subcritical
The minor increase in blowdown leakage does nothing to affect rod motion and does not increase the probability of occurrence of a Rod Withdrawal from Subcritical accident/transient. This event is sensitive to the reactivity rate of the withdrawn rod. Since the time of the event is less than what is required for any significant heat to be transferred to the secondary side SG leakage has no affect on this event and the current analysis of record remains bounding. There is no increase in the consequences related to a Rod Withdrawal from Subcritical accident/transient.

Rod Withdrawal at Power
The minor increase in blowdown leakage does nothing to affect rod motion and does not increase the probability of occurrence of Rod Withdrawal at Power accident transient. An uncontrolled RCCA withdrawal at power will result in an increase in the core heat flux due to the positive reactivity insertion. Immediately after the rod withdrawal, Steam Generator heat removal rates will lag behind the core power generation rate until the Steam Generator pressure reaches the relief or safety valve setpoint. This leads to an increase in temperature, RCS pressure, and pressurizer water level and pressure. The duration of this event is less than 10 seconds. In this amount of time, the SG leakage would not reduce the SG level below the level of the SG tubes. Therefore, the primary-to-secondary heat transfer would not be significantly different from the analysis of record, which remains the bounding case. There is no increase in the consequences related to a Rod Withdrawal at Power accident/transient.

Rod Cluster Control Assembly Misoperation
The minor increase in blowdown leakage does nothing to affect rod motion and does not increase the probability of occurrence of Rod Cluster Control Assembly Misoperation accident/transient. Rod cluster control assembly misoperation accidents include:

1. One or more dropped RCCAs within the same group,
2. One dropped RCCA bank,
3. Statically misaligned RCCA, and
4. Withdrawal of a single RCCA.

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The dropping of one or more RCCAs within the same group results in a negative reactivity insertion. If the reactivity insertion were detected, reactor trip on negative neutron flux rate would occur. If the reactivity is not detected and the Rod Control System is in manual, reactivity feedback (Doppler and moderator) would cause core power to be restored to an equilibrium condition near the initial power level. The limiting case is when the reactivity insertion does not cause a reactor trip and the Rod Control System is in automatic. Rod withdrawal by the control system to compensate for the negative reactivity insertion could result in power overshoot. The important parameters for this event are rod worth, feedback coefficients, control system parameters, and negative neutron flux rate trip setpoints. The postulated SG leakage does not affect any of the parameters.

The dropping of a RCCA bank results in a negative reactivity insertion that is sufficiently large to cause a reactor trip on negative neutron flux rate. This event is bounded by the dropping of one or more RCCAs within the same group.

The analyses of both the statically misaligned RCCA and the single RCCA withdrawal events do not model the Steam Generators. This event is analyzed using detailed nuclear models and reactor core physics parameters. The analyses do not include inputs or code algorithms that specifically model the Reactor Coolant System Loops or any of the detailed system characteristics of either the primary or secondary-side systems. Therefore, any SG leakage does not impact these events. There is no increase in the consequences related to a Rod Cluster Control Assembly Misoperation accident/transient.

Chemical and Volume Control System Malfunction that results in a decrease in Boron Concentration in Reactor Coolant the Blowdown System has no interface with the Chemical and Volume Control System, therefore, the minor increase in blowdown leakage does nothing to affect Boron Concentration. There is no affect on a Chemical and Volume Control System Malfunction that results in a decrease in Boron Concentration in Reactor Coolant accident/transient. The concern associated with the boron dilution events is the potential change in reactivity in the core associated with the dilution of the Reactor Coolant System. Since the core Boron Concentration, dilution flow rate, Reactor Coolant System Volumes and Alarm Setpoints are not impacted by the SG leakage, there is no impact on the current boron dilution analyses. There is no increase in the consequences related to a Chemical and Volume Control System Malfunction that results in a decrease in Boron Concentration in Reactor Coolant accident/transient.

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Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position
The minor increase in blowdown leakage does nothing to affect the Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position and does not increase the probability of occurrence of this accident/transient. Fuel and core loading errors resulting in loading of one or more Fuel Assemblies into improper positions, loading a fuel rod during manufacturing with pellets of incorrect enrichment, or loading of a full Fuel Assembly during manufacturing with rods of incorrect enrichment could lead to an increase in core peaking. This event is analyzed to determine if the increase in core peaking may result in approach to the DNBR limit during power operation. The SG leakage does not affect core loading or the ability to detect power perturbations. Therefore, the UFSAR analysis remains bounding. There is no increase in the consequences related to an Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position accident/transient.

Spectrum of Rod Cluster Control Assembly Ejection Accidents
The minor increase in blowdown leakage does nothing to affect Rod Cluster Control Assemblies and does not increase the probability of the Spectrum of Rod Cluster Control Assembly Ejection Accidents/transients. The rapid ejection of a RCCA results in a rapid positive reactivity insertion and power excursion. These increases are limited by Doppler feedback due to the increase in fuel temperature and terminated by reactor trip on high neutron flux or high positive rate neutron flux. The important parameters for this event are ejected rod worth, reactivity coefficients, peaking factors, and pressurizer safety valve flow rate. The SG leakage does not impact these parameters. Therefore, the UFSAR analysis remains bounding. There is no increase in the consequences related to Spectrum of Rod Cluster Control Assembly Ejection Accidents/transients.

Inadvertent Operation of Emergency Core Cooling System During Power Operation
The minor increase in blowdown leakage does nothing to affect the Inadvertent Operation of Emergency Core Cooling System During Power Operation and does not increase the probability of this accident/transient. Inadvertent actuation of ECCS at power can increase system pressure and pressurizer level. The important parameters for this event are initial pressurizer level, ECCS injection flow rate and Boron Concentration, RCS flow rate, and RCS volume. The SG leakage does not impact these parameters. The heat removal capability of the secondary side may also affect the amount of pressurizer insurge. For this event, the secondary side level will not be significantly reduced by the SG leakage. Therefore, the SG tubes should remain covered, and the heat transfer will not be significantly retarded. Therefore, heat transferred will not be affected enough to have a significant effect on the pressurizer insurge. There is no increase in the consequences related to Inadvertent Operation of Emergency Core Cooling System During Power Operation accidents/transients.

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CVCS Malfunction that Increases Reactor Coolant Inventory
The minor increase in blowdown leakage does nothing to affect CVCS operation or reactor coolant inventory and does not increase the probability of a CVCS Malfunction that Increases Reactor Coolant Inventory accident/transient. Injection of cold, unborated water is analyzed as the boron dilution event and injection of borated water is analyzed as the inadvertent ECCS actuation event. Evaluations for these events are included in the sections on Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in Reactor Coolant, and Inadvertent Operation of Emergency Core Cooling System During Power Operation. As these other two accidents/transients experience no increase in the consequences due to the increased blowdown leakage there is no increase in the consequences of a CVCS Malfunction that Increases Reactor Coolant Inventory accident/transient.

Inadvertent Opening of a Pressurizer Safety or Relief Valve
The minor increase in blowdown leakage does nothing to affect the Inadvertent Opening of a Pressurizer Safety or Relief Valve and does not increase the probability of this accident/transient. The inadvertent opening of a pressurizer safety or relief valve would result in the depressurization of the Reactor Coolant System. The DNBR will then decrease as a result of pressure reduction. Reactivity feedback determines the power level change with depressurization. With rod control in automatic, power and temperature are maintained until reactor trip occurs on either OTDT or low pressurizer pressure. Power reduction from reactor trip would then result in increasing DNBR. The important parameters are safety and relief valve flow rates, reactivity feedback coefficients and RCS flow rates. The SG leakage does not impact these parameters. Therefore, the UFSAR analysis remains bounding. There is no increase in the consequences related to Inadvertent Opening of a Pressurizer Safety or Relief Valve accident/transient.

Failure of Small Lines Carrying Primary Coolant Outside Containment
The minor increase in blowdown leakage does nothing to affect the Failure of Small Lines Carrying Primary Coolant Outside Containment and does not increase the probability of this accident/transient. The failure of small lines carrying primary coolant outside containment can result in leakage of radioactive primary coolant to buildings outside containment and result in offsite dose release. The important parameters for this event are the leakage rate, the-RCS activity concentration and partition and atmospheric dispersion factors. The SG leakage does not impact these parameters. Therefore, the UFSAR analysis remains bounding. There is no increase in the consequences related to the Failure of Small Lines Carrying Primary Coolant Outside Containment accidents/transients.

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Steam Generator Tube Rupture
The minor increase in blowdown leakage does nothing to affect Steam Generator Tube Ruptures and does not increase the probability of this accident/transient. For the margin to Steam Generator overfill case, leakage through the SG Blowdown valves will decrease SG inventory, and has a second order impact on the time it takes to cool down the RCS. But more importantly, it has a direct impact on the margin to overfill by reducing inventory in the ruptured Steam Generator. Therefore, the UFSAR analysis remains bounding. For the offsite dose case, leakage to the SG Blowdown System reduces the activity available for release through the SG PORVS. Therefore, the UFSAR analysis remains bounding. There is no increase in the consequences related to Steam Generator Tube Rupture accidents/transients.

Loss of Coolant Accident
The minor increase in blowdown leakage does nothing to affect any Loss of Coolant Accidents and does not increase the probability of this accident/transient. The Loss of Coolant Accident was addressed by NFM and documented in NFM:PSA:99012. This document indicates that NFM was requested to evaluate the impact of an 8 gpm leakage per Steam Generator of secondary side water through the Steam Generator Blowdown Valves. Their analysis is as follows. The secondary side of the Steam Generator has little or no impact on the large break LOCA (LBLOCA) analysis. Because at the time of reactor trip the secondary side is isolated and the impact of secondary side on the peak clad temperature (PCT) is negligible prior to the trip. Therefore the LBLOCA is not further discussed here. The secondary side of the Steam Generator is modeled in the small break LOCA (SBLOCA) analysis. The secondary side of the Steam Generator plays a significant role in the outcome of the SBLOCA transient results. NFM provides an attachment to this memo discussing the impact of an evaluation of 8 gpm leakage on the SBLOCA analysis of record documented in above reference. NFM:PSA:99-013 indicated that it is conservatively estimated that the PCT increase for the SBLOCA will be less than 20° F on the analysis of record. Since the SBLOCA analysis of record PCT for Byron Unit 1 and Braidwood Unit 1 is 1695° F (Reference), a 20° F increase in the PCT will not violate the Appendix K criteria of 2200° F for the PCT. It should be recognized that the exact value of the PCT penalty can only be calculated by Westinghouse based on actual computer cases. As stated the 20° F PCT penalty is an arbitrary estimate and the actual value will be significantly less or even perhaps zero degree. This is so because the Steam Generator Tubes remain covered throughout the entire transient and it is determined that even with the 8 gpm leakage it will remain covered. In consideration of this analysis there is no increase in the consequences related to Loss of Coolant Accidents/transients.

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MSLB M&E Analysis
The minor increase in blowdown leakage does nothing to affect a Main Steamline Break. It does not increase the probability of the accident/transient represented by the MSLB M&E Analysis. This analysis calculates the mass and energy released to either containment or into the steam tunnel after a Steam Line Break. This mass and energy release is then used to calculate the pressure and temperature profile of containment and the temperature in the Steam Tunnel. The second portion of this analysis (pressure and temperature responses) is not affected by the SG leakage, since the SG is not used to remove mass or energy from containment or the steam tunnel. The mass and energy released to containment following a Steam Line Break is calculated using the maximum AF flow to the faulted SG to increase the mass and energy that is released. If a leak path is introduced, less mass will be released out the break, so the current analysis assumption for the faulted SG remains conservative. For the intact SGs, the SG Blowdown will increase the amount of heat removed from the primary side, since these SGs have been isolated from the break (the leak flow will remove sensible heat). Since the level in the intact SGs is not expected to fall below the level of the SG tubes, the heat transfer will not be degraded significantly by 8 gpm of leak flow. Therefore, the removal of additional sensible heat will decrease the primary side enthalpy and decrease the enthalpy of the faulted SG, and thus the energy of the break flow. Therefore, the SG leakage is bounded by the current analysis of record for steamline break M&E releases inside containment. For M&E releases outside containment, the temperature in the steam tunnel is largely determined by the energy released out the break. This energy is determined by the initial stored secondary side energy and the primary side energy transferred to the secondary fluid. The amount of initial secondary side energy is not affected by the SG leakage. The transfer of primary side energy to the secondary side could be affected if the SG tubes are uncovered faster, however, the total integrated heat transfer will not change. The intact SG pressure will increase initially with the isolation of the steam lines. However, as additional steam is released (with the failure of the faulted SG MSIV), the faulted SG pressure rapidly decreases. In the limiting case (1.2 sq. ft. break with an MSIV failure), SG tubes are uncovered at 40 seconds. Over this time, the SG pressure ranges from 500 to 11 50 psig. To bound this pressure range, the maximum density will be used (50.65 lbm/cu. ft. at 500 psig). At this pressure, 8 gpm leakage corresponds to 36 lbm. Over this time, roughly 40,000 lbm are blown out the break. Therefore, an additional 36 lbm will be insignificant with respect to the time it takes to uncover the SG tubes. Furthermore, all of the other cases that are close to the maximum temperature also uncover SG tubes in a very short time period (less than one minute). Therefore, the SG Blowdown leakage has an insignificant effect on this analysis. In consideration of this analysis there is no increase in the consequences related to the MSLB M&E Analysis.

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LOCA M&E Analysis
The minor increase in blowdown leakage does nothing to affect a LOCA. It does not increase the probability of the accident/transient represented by the LOCA M&E Analysis. This analysis calculates the mass and energy released to containment after a large break LOCA. This mass and energy release is then used to calculate the pressure and temperature profile of containment after a LOCA. The second portion of this analysis (containment pressure and temperature) is not affected by the SG leakage, since the SG is not used to remove mass or energy from containment. The mass and energy released from containment is affected by the core energy, RCS volume, ECCS flow rate, and break flow rate. After a large break LOCA, the RCS liquid level falls below the SG tubes, so heat transfer is almost completely degraded. Therefore, the performance of the secondary side does not affect the primary side mass and energy released after a large break LOCA, and the current analysis of record remains bounding. In consideration of this analysis there is no increase in the consequences related to the LOCA M&E Analysis.

Offsite Dose Evaluation
The minor increase in blowdown leakage does nothing to create or initiate radioactive release pathways. While this may be a pathway for radionuclides in the plant, this activity does not cause events, which result in radioactive releases. This minor increase in blowdown leakage does nothing to increase the probability of an offsite release affecting the Offsite Dose Evaluation. The Offsite Dose Evaluations in UFSAR Chapter 15 assume zero leakage of the blowdown valves offsite releases are via SG PORVs or containment leakage. The 8 gpm leakage into the Blowdown System, instead of being released to atmosphere, is sent to the blowdown condenser. The water is then normally sent through the SG Blowdown Demineralizers and routed to the Main Condenser. However, upon a high radiation signal, a radiation monitor in the line to the Main Condenser will automatically transfer the blowdown flow to the blowdown monitor tanks. Therefore, this 8 gpm is contained in the Auxiliary Building and is treated by the demineralizers. It would also be expected that some amount of iodine would plate out on system piping, again keeping it from impacting offsite dose. Since the Blowdown System is a closed system, the current offsite dose evaluations in UFSAR Chapter 15 remains bounding. In consideration of this analysis there is no increase in the consequences related to the Offsite Dose Evaluation.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because this minor allowance for increased leakage in the blowdown valves does not result in operating or configuring the plant differently than already analyzed. The minor allowance for increased leakage in the blowdown valves does not cause any damage to plant equipment. The increased leakage through the blowdown valves would not create the possibility of an accident or transient different than previously evaluated.

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3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because as indicated in the NFM:PSA:99-013 cover letter, NFM has established that there is no reduction in the margin of safety and no increase in consequence.

The SD002 and SD005 valves are indicated as Containment Isolation Valves in Technical Specification Basis Table 3.6.3-1. Also in this table the SD valves are footnoted as, "Not subject to Type C leakage testing." These valve were removed from the Appendix J program as not requiring Type C leakage testing. This amendment was communicated to Dr. Murley (NRC) by T.K. Schuster (ComEd) on 4/20/90. In Attachment A of this correspondence the justification was provided as to why these valves did not require leakage testing for the Containment Isolation Provision. This document indicated the leakage function for these valves was to conserve the Steam Generator secondary side mass (heat sink) in the event of an accident and a phase A initiation. NFM has indicated in NFM:PSA:99-013, that the 8 gpm leakage limit is acceptable for all accident considerations including Steam Generator secondary mass considerations.

As these valves are leak tested in accordance with the Inservice Testing Program, if they fail their leakage test they are declared INOPERABLE. Action 3 of LCO 3.6.3 indicates that the applicable Condition and Required Actions be entered in for systems made inoperable by Containment Isolation Valves. Condition C is applicable for this valve as indicated in Table 3.6.3-1 for the valve being INOPERABLE. The Required Actions (in 72 hrs) is to isolate the affected penetration flow path by use of at least one closed and de-activated automatic or remote manual valve, closed manual valve, or blind flange. In consideration of the above, the valve may declared inoperable due to a Steam Generator mass concern (i.e., > 8 gpm blowdown leakage/Steam Generator). The applicable Technical Specification requires isolation of the pathway because it is a Containment Isolation Valve, even though no Containment Type C leakage testing requirement exists, and it has not challenged the basis for containment protection (isolation). In consideration of the fact that the inoperable valve is not challenging the containment isolation function, there is no reduction in any margin for safety.

6H-99-0336
1/2 BVSR AF-3, Rev. 2

DESCRIPTION:

The purpose of this procedure revision was to add IST testing criteria for the partial stroke testing of Check Valves 1/2AF001 A/B, 1/2AF003A/B, 1/2AF014A-H, and 1/2AF029A/B, by Auxiliary Feedwater (AF) verification of adequate Auxiliary Feedwater Flow through the valves. The revision also includes the addition of a back-leakage test for Check Valves 1/2AF014A-H. This is performed by verification of normal piping temperatures upstream of the referenced check valves, following the shutdown of the Auxiliary Feedwater Pumps.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because this change is to provide IST testing criteria for partial stroke testing of Auxiliary Feedwater Check Valves. Upon the initial start of the Auxiliary Feedwater system on a Unit trip the initial system parameters are the same. Therefore, the consequences of a malfunction of equipment important to safety does not increase.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the revision provides criteria for IST testing of Auxiliary Feedwater Check Valves. It does not significantly alter the procedure from its current status, as the changes basically enhance the data recording portions of the procedure. The Auxiliary Feedwater system will not be operated differently than described in the SAR, therefore no new accidents or malfunctions will be introduced.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which the Technical Specifications are based.

6H-99-0266
1/2BVSR XPT-14, Rev. 2
1/2BVSR 3.1.11-1, Rev. 2

DESCRIPTION:

The purpose of these procedure changes was to incorporate provisions contained in SER for Amendment No. 107. Technical Specification Allowable Value related to Intermediate Range Channel Reactor Trip setpoint from 31.5% to 30% due to NRC approval of Byron Technical Specification Amendment No. 107 were revised. The procedure changes were required to support the amendment.

SAFETY EVALUATION SUMMARY:

This activity revises applicable procedures which describe the Technical Specification Setpoint Allowable Values which were revised due to revision of instrument setpoint accuracy calculations.

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the Allowable Values are used as a basis for channel operability check only and do not change the system function, channel operation, or calibration. The change in the Allowable Values is conservative and does not affect the response of the instrumentation. The systems which are actuated by the corresponding instrument setpoints will operate in the same manner as before and within the design limits of the system.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the Allowable Values are used as a reference basis for instrument channel operability determination and do not have any direct relation to system or instrument channel function. The actions in response to the initiating event in the SAR remain unchanged. There is no physical change to the instrumentation.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the revised Allowable Values are more conservative than the existing Technical Specification Allowable Values.

6G-98-0057
Surveillance procedures which specify steady state frequency

DESCRIPTION:

The purpose of these procedure changes was to establish the steady state administrative limit for DG surveillance testing. This activity is applicable to all surveillance procedures, which specify steady state frequency. Current Technical Specifications permit a frequency range of 60 Hz plus or minus 2%. This evaluation justifies reducing that band to 59.8 to 60.4 Hz in all modes of operation.

Higher or lower frequencies could impact the operation of the ESF equipment powered by the DG. The analyses performed assumed a frequency of 60.0 Hz.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because this activity does not impact the reliability of the switchyard, SATs, or other auxiliary power equipment. There is no possibility these conservative changes could create an initiating event which would increase the probability or consequences of an event.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because this activity did not create any new failure modes or mechanisms for the DGs. The new limits were more conservative than existing Technical Specifications. These limits were achieved due to governor specifications and prior surveillance performance. Therefore, no new accidents or malfunctions were created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the requirements of Technical Specifications were always maintained. The administrative requirements were more restrictive to ensure proper function of the DGs.

6G-98-0079
SPP 97-138

DESCRIPTION:

The purpose of the Special Process Procedure was to inspect the Class 1 Piping for Auxiliary Spray to be tested at normal operating pressure. This test is required every 10 years. This activity also involved the installation of a freeze seal to prevent flow into the pressurizer.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because this activity is performed in Mode 6 when auxiliary spray is not required. With the freeze seal in place, no flow was delivered to the pressurizer. Testing per the Asme Code requirements ensured proper system parameters can be maintained. The analyzed conditions of the Safety Analysis Report previously evaluated the conditions and potential accidents and encompass current operating conditions.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the auxiliary spray is not required in Mode 6. The flow was isolated from the pressurizer by the freeze seal. Reactivity management is unaffected. No conditions exist outside the previously evaluated conditions for safety related systems.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this activity maintained the requirements for Technical Specifications at all times. The testing performed was on equipment not required in Mode 6. Prior to entering modes requiring this equipment, surveillance testing was performed to verify operability.

6G-98-0002
SPP 97-141 and 97-142

DESCRIPTION:

The purpose of these Special Process Procedures was to provide guidance on draining water trapped between Containment Isolation Valves (CIVs). These proposed procedures detail the manipulation and restoration of Containment Isolation Valves (CIVs) and penetration drain valves to drain the trapped water preventing the potential for thermal overpressurization of the penetration piping during a MSLB or a LOCA. The procedures are in response to NRC Generic Letter 96-06.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the affected penetrations and/or equipment does not interface with the primary or secondary plant such that they could initiate an accident or malfunction of equipment important to safety. Therefore the probability of an accident is not impacted by the changes.

   The drained penetrations are not in service and do not provide any safety function during normal plant operation. The procedures are performed in Mode 5 or 6 and the applicable portions of piping affected by these procedures are not required to be in service in those modes. The consequences of an accident are not affected in that the penetrations impacted by the procedures are restored to the normal configuration prior to plant operation.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the affected penetrations are left drained during normal plant operations with the Containment Isolation Valves (CIVs) closed to ensure containment integrity. Draining the penetrations increases the ability of the CIVs to mitigate an accident by eliminating the potential for thermal overpressurization of the penetration piping during a LOCA or MSLB. No new accidents or malfunctions are created by the removal of water trapped between closed CIVs.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not impact any parameters upon which Technical Specifications are based.

6G-98-0016
SPP 98-007

DESCRIPTION:

The purpose of the Special Process Procedure was to record the flowrates of Essential Service Water (SX) to the Reactor Containment Fan Coolers (RCFC) with various Component Cooling Water System configurations and SX Cooling Tower arrangements.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because no accident initiators were affected by the SPP so no new accidents or consequences are possible. The systems involved (SX and CC) were operated within the original design criteria. Data collection was used to evaluate surveillance performance.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the systems involved were operated within their original design. No new unanalyzed conditions were created. Data collection was used to evaluate surveillance performance.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the procedure performed operated the systems within their original Design Bases. The margin of safety in accordance with Technical Specifications was unaffected.

6G-98-0060
SPP 98-017

DESCRIPTION:

The purpose of the Special Process Procedure was to acquire diagnostic test data on Component Cooling System valve strokes in the open and closed directions. This information was used to evaluate the current settings on the valves and was provided to the owner's group testing program, supporting the station's commitment to Generic Letter 96-05.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because this activity was performed during a mode when Component Cooling Flow is not required to the RCPs. With no requirement, any failure would not impact any previous analysis.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because this activity operates the system in its original design basis. Therefore, this operation cannot create any new accidents or malfunctions other than those previously identified and evaluated.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this activity operates the system within the original design and in accordance with Technical Specifications. All operability surveillances verified operability prior to entering modes with additional requirements.

6G-98-0043
SPP 98-023

DESCRIPTION:

The purpose of the Special Process Procedure was to test the "1B Component Cooling Pump/0 Component Cooling Pump Interlock." The procedure verifies the functionality of the interlocks associated with the 1B and the Unit 0 Component Cooling (CC) pumps. This verification was not satisfied during Manual Safety Injection/Phase A testing with the 0CC pump controlled from the remote shutdown panel. The test described by this procedure duplicates the test conditions to support re-verification of this functional requirement. The test inserts a simulated autostart signal for the 0 and 1B CC pumps. The activity represents at test or experiment not described in the SAR.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the test was performed in Mode 5 with all required support functions powered from the unaffected train and the 1B DG available for emergency power. The 1B RH pump remains operable to support TS 3.4.1.4.2 requirements.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because there is no possibility that this test can introduce a failure or problem which can impact Train A equipment reliability.

3. The margin of safety as defined in the bases for any Technical Specification is not reduced because the activity did not change Technical Specification Bases parameters.

6G-98-0115, Rev. 2
SPP 98-046

DESCRIPTION:

The purpose of the Special Process Procedure was to implement the Westinghouse Dynamic Rod Worth Measurement (DRWM) testing program at Byron Station for both Units 1 and 2. The program was for performance of low power physics testing per Westinghouse WCAP13360. Use of this revised testing methodology utilizes a dynamic means of determining control rod worths versus the traditional rod swap technique. The reactivity worth predictions of the control rods are validated by measurement at the beginning of each cycle. Use of the DRWM limits the time that the reactor is in a low power, just-critical condition during physics testing.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the entire DRWM test program is bounded by the assumptions and limits in the UFSAR accident analysis. In addition, the DRWM technique is not an initiator for any accidents. The procedures used for this process are written to ensure the testing is performed in a controlled and safe manner. In addition, the DRWM process and procedures do not affect the integrity of the Fuel Assemblies, control rods, or other reactor internals such that their function in the control of radiological consequences is affected. Therefore, the probability of occurrence or the consequences of an accident previously evaluated are not increased by this activity.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the DRWM process and procedures do not result in different response of safety-related systems and components to accident scenarios than that postulated in the UFSAR. In addition, no new equipment malfunctions have been identified which affect fission product barrier integrity with this change. No new performance requirements or changes are imposed on any equipment important to safety.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this activity was evaluated against plant Technical Specifications and it was determined that there was no impact on any Technical Specifications.

6H-99-0214
SPP 98-073

DESCRIPTION:

The purpose of this Special Process Procedure was to adjust the Unit 1 Pressurizer Spray Valve Bypass Valves to reduce required pressurizer backup heater demand in Mode 1. Bypass spray valves will be throttled and spray line temperature monitored.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because spray flow is not credited in Chapter 15 Analysis. Spray bypass flow is necessary to prevent thermal shock. If the low temperature alarm on the spray line annunciates, this procedure allows spray flow to be reinitiated to heat up the spray line to prevent thermal shock.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because pressurizer spray bypass flow for one train will be isolated for a short period of time. During this period spray line temperatures will be monitored for low temperature. Sprays and heaters remain available to control RCS pressure and prevent pressure transients.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the pressurizer sprays and bypass flows are not used in Chapter 15 for accident mitigation. Changing the position of the bypass flows will not impact the function or availability of the pressurizer sprays or heaters.

6H-98-0166
SPP 98-078

DESCRIPTION:

The purpose of this Special Process Procedure was to provide additional guidance for the activities associated with the removal of a System Auxiliary Transformer (SAT) from service with the associated unit at power and provide contingency actions in the event of a Loss of Offsite Power (LOOP). The removal of a SAT from service at power has been performed previously with SPP 96-064 and the associated Nuclear Design Information Transmittal (NDIT) No. BYR94-019.

Specifically, the new guidance is, while in single SAT operation, to place one or both non-ESF Busses (4.16kV) in Pull-To-Lock with the SAT feed breaker when either:

1. a Emergency Diesel Generator is paralleled to a bus

2. a condensate/condensate booster pump is being swapped; or

3. a heater drain pump is being swapped. This guidance is being added to the SPP as a limitation to ensure the loading of the SAT will not be sufficient to actuate the SAT overcurrent relays.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the additional guidance placed in SPP 98-078 (Unit 1 SAT Outage) is to add procedural limitations to ensure the loading of the SAT will not be sufficient to actuate the SAT overcurrent relays. These additional restrictions reduce the possibility of having the SAT trip on overcurrrent and complicating the recovery. All probabilities for the occurrences or consequences of an accident or malfunction of equipment important to safety were determined to not be increased by the original safety evaluation.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the changes do not adversely affect the operation of any plant SSC important to safety. The change detailed additional restrictions to an activity already determined to not create the possibility of a new accident or malfunction that was not evaluated by the SAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because no Technical Specifications were impacted by the changes.

6G-99-0036
SPP 99-005

DESCRIPTION:

The purpose of the Special Process Procedure was to manually isolate the pressurizer spray valves in Mode 3 to check for leakby. One back-up heater and the modulating heater will be placed on, and then each spray valve manually isolated, one at a time. When the spray valve is isolated, the modulating heater output will drop (amps), and then spray valve leakage can be calculated by converting the decrease in amps to gallons per minute.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because both sprays are assumed to limit the magnitude of RCS pressure increase during loss of secondary loads. The PORVS and PZR safeties are the final line of defense against over pressurization of the RCS. Since this SPP is performed in Mode 3, one spray valve available for pressure control is adequate to mitigate any loss of secondary loads. This SPP does not affect the operation of the PORVs or Safeties. Therefore, there is no increase in the probability of an any accident or RCS pressure transient impacted by this SPP in Mode 3.

   There is no challenge to any of the fission barriers by this SPP, since RCS pressure is controlled by the pressurizer in Mode 3. This SPP does not challenge pressurizer control in Mode 3; therefore, there is no increase in on-site or off-site dose.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the momentary isolation of a spray valve in Mode 3 does not challenge any safety system required to maintain RCS pressure or integrity. The spray valves are relied upon to control pressure during full power and are considered to prevent lifting of the PORVs or PZR Safeties, in Mode 1. However in Mode 3, one spray valve will be adequate to prevent lifting of the PORVs or Safeties, due to the small secondary loads in Mode 3.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because in Mode 3, one spray valve is always available to control pressure. High-pressure transients to the primary are driven by secondary transients resulting in loss of heat sink capability. In Mode 3, one spray valve of 450 gpm, is capable of controlling pressure to prevent PORV opening. If the PORV should open to control pressure, that event creates no malfunction of safety related equipment.

6G-99-0032 & 6H-99-0356
SPP 99-008

DESCRIPTION:

The purpose of this Special Process Procedure was to investigate excessive Unit 1 and 2 HP Turbine Gland Steam (GS) Pressure while at power. The procedure steps isolated the unit's Main Turbine Throttle Valve High Pressure Leak-off Isolation Valves one at a time while monitoring GS System parameter. The same SPP was used for both units.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because only the GS leakoff is affected by this procedure. Closing off the leakoff on the Main Turbine Control Valves does not affect their ability to close in the event of an overspeed condition or reactor trip. Closing the GS leakoff lines only changes the path of the Control Valve leakoff from the condenser to atmosphere assuming the seals on the Throttle Valves leak. The system will be monitored as the leakoff is isolated to monitor for excessive leakoff if it should occur.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because closing the GS leakoff lines only changes the path of the Control Valve leakoff from the condenser to atmosphere assuming the seals on the Throttle Valves leak. The ability of the Throttle valves to close in the event of an overspeed condition or reactor trip remains unaffected by this change in configuration.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because no Technical Specification related components are being manipulated or affected by closing the GS leakoff lines. Operators are monitoring system parameters as the procedure is performed so that action can be taken to prevent the GS System from being overpressurized.

6H-99-0125
SPP 99-015

DESCRIPTION:

The purpose of this Special Process Procedure determined the average moisture carryover from the four Steam Generators by injecting a radioactive tracer into the Steam Generators and sampling for moisture carryover at designated points in the steam cycle.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because no activity in Moisture Carryover testing affects any condition, assumptions, or status of equipment and systems described in UFSAR, Table 15.0-7 "Plant Systems and Equipment Available for Transients and Accident Conditions."

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the few temporary connections made to the plant in order to inject the tracer and sample for carryover are controlled within the procedure and restored within the procedure. These connections do not affect the operation of any plant system. The temporary changes to the Steam Generator Chemistry due to the injection of the tracer into the secondary cycle have been evaluated as being within the EPRI Chemistry Guidelines. The radiation levels in the secondary plant will increase slightly during this test and adequate ALARA controls have been incorporated into this procedure. The increase in secondary plant radiation levels and radioactive concentrations in the SGs does not adversely impact the results of any potential offsite dose calculation, should an accident resulting in an offsite release occur during performance of this test. The operation of a few systems, such as SG blowdown and condenser hotwell makeup & level control, will deviate from the UFSAR descriptions for a short duration during the test. These impacts are within the design basis and do not impact the plant safety analysis.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the conduct of this test does not significantly affect how any plant system is operated. The increase in condenser hotwell level, decrease in S/G blowdown flowrates, and the slight increase in secondary plant radiation levels during performance of this test does not affect the parameters upon which the Technical Specifications are based specifically, for the Specific Activity T/S 3.7.3.

6G-99-0064
SPP 99-021

DESCRIPTION:

The purpose of the Special Process Procedure was to install a supplemental restrains device on the Refueling Machine to prevent a Fuel Assembly from falling onto the lower core plate.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the supplemental restraining device will not adversely affect the ability of the manipulator crane to support the Fuel Assembly or allow Fuel Assembly placement into the core. None of the mitigation systems for the fuel handling accident are affected.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the supplemental restraining device is attached to the upper cable assembly of the Refueling Machine where it provides a restraining function. Upward Polar Crane movement, and ultimately upward Fuel Assembly movement, will be prevented by the installation and operating controls of the SPP. Prior to connecting the Polar Crane to the Refueling Machine upper cable assembly, the Polar Crane will be raised to its top physical limit. The device will be installed such that the Fuel Assembly will be incapable of being raised above its current elevation. This will prevent inadvertent misoperation of the supplemental restraining device. The supplemental restraining device can be removed if the Polar Crane will not allow the Fuel Assembly to be lowered. The B RHR train which is operating during the installation period takes it suction from the 1C hot leg from the opposite side of the core from the activity and will not be affected.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because no Technical Specification or Technical Specification Bases define a margin of safety for this Mode 6 refueling activity.

6G-99-0066, Rev. 1
SPP 99-023

DESCRIPTION:

The purpose of the Special Process Procedure was to lower Stuck Fuel Assembly from Refueling Machine with Polar Crane.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the use of the Polar Crane to lower the Fuel Assembly will not adversely affect the ability of the manipulator crane to support and restrain the Fuel Assembly or to allow Fuel Assembly placement into the core. The Polar Crane capacity meets/exceeds that of the Refueling Machine. None of the mitigation systems for the fuel handling accident are affected. Changing the supporting device from the Refueling Machine to the auxiliary hook of the Polar Crane will have no effect on the consequences of the accident/malfunction, should the assembly fall.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the Polar Crane auxiliary hook attached to the manipulator crane cable provided a restraint function and then the assembly was lowered using the Polar Crane. The B RHR train which is operating during the installation period takes its suction from the opposite side of the core from the proposed activity and will not be affected. The Polar Crane is restricted to vertical motion while it is supporting the Fuel Assembly. The only allowed operation will be vertical motion of the assembly into a vacant lower core plate location chosen to maximize its distance from the other irradiated Fuel Assemblies. Therefore, the assembly will not be moved over irradiated fuel or an unapproved load path. Based on this configuration, the only credible accident is still bounded by the existing fuel handling accident.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the accident described in UFSAR 15.7.4.2.2 's defined as the dropping of a spent Fuel Assembly onto the core during refueling which results in the rupture of the cladding of 314 fuel rods. TS 3.9.7, Refueling Cavity Water Level requires that greater than or equal to 23 feet of water exist above the Fuel Assembly. This activity does not reduce the amount of water through which fission gases would pass.

6G-99-0130
SPP 99-026

DESCRIPTION:

The purpose of the Special Process Procedure was to adjust the manual bypass valves (2RY8050 and 2RY8051) to maintain the spray line temperature above 530° F and the surge line temperature above 517 ° F. This procedure can be performed in Modes 1 (less than 30% power) 2,and 3. The SPP will determine if the internal leakage through the Spray Valves (2RY455B and 2RY455C) maintained the spray/surge line temperatures, then the bypass valves were throttled closed. The spray valves have a internal leakage around the valve seat, and this leakage is estimated to be approximately one gpm for each valve. Operating with the variable heater only is the designed operation of the pressurizer. Closing the bypass valves decreased the amount of spray water that the variable heater must "overcome" to maintain RCS pressure. Additionally less spray flow increased the margin of heat available from the variable heater to maintain RCS pressure.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because both sprays and heaters remain available to control RCS pressure. With both sprays being closed and both bypass valves closed pressure may increase slightly during the execution of SPP 99-026. The SPP procedural guidance required RCS pressure to be maintained between 2225 and 2250 psig during this testing. This did not challenge the PORV setpoint.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the SPP isolated the manual bypass valves, which did not create equipment failures or malfunctions. If the spray line temperature were not maintained by the internal leakage of the spray valves, then the bypass valves would be opened. This would allow for adequate flow to maintain temperature. There are no regulatory requirements for spray valve closure, thus there are no new equipment failure modes created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because closure of the bypass valves will not affect the PORVs or PZR safeties. Temperature monitoring is available to preclude any major temperature transient of the spray or surge lines. The pressurizer control system remains fully functional to control RCS pressure; thus there is no impact on safety related equipment or its ability to function.

6G-99-0104
SPP 99-029

DESCRIPTION:

The purpose of the Special Process Procedure was to check for air in-leakage into the Unit 2 Steam Generator Blowdown (SD) System.

The activity secured Unit 2 Steam Generator Blowdown per an approved operating procedure. A special helium gas leak detection rig was connected to the 2PR27J skid per an approved plant procedure. Valve 2OG020 was cracked open to vent the Unit 2 SD condenser to the Main Condenser. Helium gas was dispensed in short duration puffs at various components in the Unit 2 SD System to determine the source of air in-leakage into the system.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the activity temporarily secures Steam Generator Blowdown to Unit 2 to facilitate identification of air in-leakage into the system. The activity closed the Blowdown Containment Isolation Valves. Therefore, the containment isolation and HELB safety functions are satisfied. The activity did not cause a degradation of these safety functions nor result in an increased challenge of these functions. The activity did not introduce any new failure modes.

   Temporarily securing Unit 2 Steam Generator Blowdown did not cause Steam Generator Chemistry to go beyond specified limits. The blowdown flow path was not secured long enough to cause chemistry to go beyond specified limits.

   The Main Steam Line Break accident assumes Steam Generator Blowdown is secured during the worst case accident. The activity placed the Unit 2 Steam Generator Blowdown System temporarily in this condition.

   The Feedwater System Pipe Break accident assumes Steam Generator Blowdown is isolated due to a low-low Steam Generator water level signal during the accident. The activity placed the Unit 2 Steam Generator Blowdown System temporarily in this condition.

   The Steam Generator Tube Rupture accident credits the Blowdown System as one of three different methods available to place the reactor in a cold shutdown condition. Although the activity temporarily isolated Unit 2 Blowdown, the system was still available for use if needed in the post-SGTR Cooldown.

6G-99-0104
SPP 99-029
(Cont'd.)

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the activity did not create any new equipment failures or malfunctions because the safety functions of containment isolation and high energy line break isolation are still satisfied. The appropriate Steam Generator Blowdown Valves for these automatic signals were in the closed position. This is the required position in the event of a containment phase A or a HELB.

   The activity verified Steam Generator Chemistry within specified limits prior to securing blowdown. The activity had provisions to restore Steam Generator Blowdown in the event that chemistry limits were approached.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the Steam Generator Blowdown System Containment Isolation Valves remained operable during the activity. The activity closes these valves to facilitate identification of the leaking blowdown flow path. Therefore, the Technical Specification acceptance limit was still satisfied.

6H-99-0332
SPP 99-034

DESCRIPTION:

The purpose of this Special Process Procedure was to functionally test a new alignment of the two parallel lube oil strainers by isolating one strainer prior to a cold start of the Diesel Generator (DG). A single strainer alignment during normal operating, hot restart, and cold start conditions was completed on the 2A Diesel Generator in SPP 98-097. This additional testing is being done to verify proper lube oil pressure is available to the 2B, 1A, and 1B Diesel Generators while in a single strainer configuration during a cold start.

Single strainer alignment has been approved by the DG Supplier and has been previously tested in SPP 98-097. A single strainer alignment is being tested on the 2B, 1A, and 1B Diesel Generators to confirm system pressure expectations of a single strainer alignment during a cold start of the engine.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the ability of the DGs to accelerate to rated speed and voltage in less than 10 seconds and operate at loads up to 110% of rated is not affected by this test. The performance of the DG Lube Oil Sub-System to deliver lubricating oil to the DG at adequate pressure will not be affected by this test. The test does not introduce any changes that would challenge engine reliability or capability.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the changes to the DG performed by the SPP do not adversely impact the Lube Oil Sub-System and do not adversely impact the DG. The engine manufacturer stated the engine can perform its design basis function while operating on one lube oil strainer.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which the Technical Specifications are based,

6G-99-0060
Byron Unit 1 Cycle 10 (BY1C10) Core Reload Design

DESCRIPTION:

The purpose of this Safety Evaluation was to review the Byron Unit 1 Cycle 10 (BY1C10) core reload design. The change evaluated encompasses the composite effects of the following changes:

1. Fuel loading pattern
2. Fuel mechanical design changes:
   a. Slightly enlarged grid sleeve on IFM grids
   b. 3-tab inner-to-outer strap joint for Inconel top and bottom non-mixing grids
3. RCCA park position of 225 steps
4. 2 Lead Test Assemblies with features:
   a. Low Tin ZIRLO,
   b. Spring clips,
   c. Pellet theoretical density up to 98%, and
   d. 350 psig backfill pressure in non-IFBA fuel rods

The changes were incorporated into many implementing procedures, including the Byron Curve Book, the Reload Design Key Parameter Checklist, COLR, and DRP 8-032 and 8-039.

6G-99-0060
Byron Unit 1 Cycle 10 (BY1C10) Core Reload Design
(Cont'd.)

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because fuel construction meets all design criteria. The Core Fuel Loading Pattern and changes in operating characteristics do not produce any mechanisms by which any of the evaluated accidents, such as LOCAs or HELBs, etc., can be initiated. The consequences of previously evaluated accidents is not increased because the reload design process confirmed all design parameters satisfy the accident analysis limits and assumptions as documented in the UFSAR or other appropriate evaluations. The analyses included mechanical, nuclear, thermo-hydraulic and transient analyses, which concluded that all core parameter criteria, such as DNB, PCT, and fuel temperature, were met. In addition, the analyses showed that all system performance criteria, such as containment pressure and no water through pressurizer safeties, were met.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because this core reload's fuel mechanical features introduce no new failure modes. The reload key parameters and assumptions meet all standards and criteria. The core operates within pertinent design basis operating limits. Therefore, the cycle specific changes in these parameters introduce no new failure modes.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the core reload design process safety analysis was performed in accordance with NRC approved methodologies and indicates that BY1C10 operates within acceptable limits and the margin of safety is maintained.

6G-99-0061
Byron Unit 1 Cycle 10 (BY1C10) Lead Test Assemblies

DESCRIPTION:

The purpose of this Safety Evaluation was to review the 2 Lead Test Assemblies (LTAs) initially to be loaded into the Byron Unit 1 Cycle 10 (BY1C10) Core. The change evaluated encompasses the change description of the LTA features in the UFSAR (DRP 8-039) and the Technical Specification Bases (006-99) and also addresses the storage of these Fuel Assemblies in the Spent Fuel Pool (SFP). The LTA features are implemented in 82 fuel rods and are:

1. Low Tin ZIRLO (cladding and-skeleton),
2. Spring clips (as opposed to helical springs),
3. Pellet theoretical density up to 98% (formerly 95%), and
4. 350 psig backfill pressure in non-IFRA fuel rods (normally 275 psig).

SAEETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the interaction of these Fuel Assemblies with the core load has been evaluated as acceptable using the reload design methodology as documented by Safety Evaluation 6G-99-0060. Regarding SFP storage, evaluations (using the methodology of WCAP-14416-NP-A) were performed to verify placement acceptance criteria in CAC-97-162 remains valid for the LTAs. Therefore, the consequences of an accident (SFP dilution or Fuel Assembly misplacement) or malfunction of equipment will not be increased. The LTAs were constructed using the same standards as the previously handled and stored Fuel Assemblies. For this reason, the probability of a malfunction of equipment important to safety (Dropped Fuel Assembly) is not increased. The change in the Fuel Assemblies' features also does not impact any evaluated accident's initiating mechanisms.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because these changes do not alter the basic assumptions or acceptance criteria for any accident described in the UFSAR. The LTA fuel mechanical features introduce no new failure modes. Since all possible failure modes, accidents and malfunctions have been previously evaluated, the cycle specific changes in these parameters introduce no new failure modes.

6G-99-0061
Byron Unit 1 Cycle 10 (BY1C10) Lead Test Assemblies
(Cont'd.)

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the applicable acceptance criteria of CAC-97-162 are still met with the LTA Fuel Assemblies. With respect to the SFP Region 1 criteria, the LTAS (4-6 w/o and 98% pellet density) have been analyzed to confirm their reactivity (without IFBA) would still be less than the analyzed assembly with 4.7 w/o, 95% pellet density with no IFBA. Regarding Region 2 criteria, the analysis of record, CAC-97-162, has margin associated with meeting the acceptance criteria. These evaluations demonstrated the Keff requirements are still met and the margin of safety remains intact.

6H-99-0173
Byron Unit 1 Cycle 10 (BY1C10) Core Reload Design

DESCRIPTION:

The purpose of this Safety Evaluation was to review the revised loading pattern (replacement of Fuel Assembly K51F with assembly K32E) that resulted from the failure to locate and remove debris previously observed in K51F. Subsequent evaluations, consistent with the design review processes, were performed to verify the acceptability of the revised loading pattern for all modes of operation during the Cycle 10. The BY1C10 revised Core operating characteristics will be equivalent or less limiting than those previously reviewed and accepted.

The original safety evaluation was performed for the Byron Unit 1 Cycle 10 (BY1C10) Core Reload Design. The change evaluated encompasses the composite effects of the following changes:

1. Fuel loading pattern
2. Fuel mechanical design changes:
   a. Slightly enlarged grid sleeve on IFM grids and
   b. 3-tab inner-to-outer strap joint for Inconel top and bottom non-mixing grids
3. RCCA park position of 225 steps
4. 2 Lead Test Assemblies with features:
   a. Low Tin ZIRLO,
   b. Spring clips,
   c. Pellet theoretical density up to 98%, and
   d. 350 psig backfill pressure in non-IFBA fuel rods.

The changes were incorporated into many implementing procedures, including the Byron Curve Book, the Reload Design Key Parameter Checklist, COLR, and DRP 8-032 and 8-039.

6H-99-0173
Byron Unit 1 Cycle 10 (BY1C10) Core Reload Design
(Cont'd.)

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because fuel construction meets all design criteria. The Core Fuel Loading Pattern and changes in operating characteristics do not produce any mechanisms by which any of the evaluated accidents, such as LOCAs or HELBs, etc., can be initiated. The consequences of previously evaluated accidents is not increased because the reload design process confirmed all design parameters satisfy the accident analysis limits and assumptions as documented in the UFSAR or other appropriate evaluations. The analyses included mechanical, nuclear, thermo-hydraulic and transient analyses, which concluded that all core parameter criteria, such as DNB, PCT, and fuel temperature, were met. In addition, the analyses showed that all system performance criteria, such as containment pressure and no water through pressurizer safeties, were met.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because this core reload's fuel mechanical features introduce no new failure modes. The reload key parameters and assumptions meet all standards and criteria. The core operates within pertinent design basis operating limits. Therefore, the cycle specific changes in these parameters introduce no new failure modes.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the Core Reload Design Process Safety Analysis was performed in accordance with NRC approved methodologies and indicates that BY1C10 operates within acceptable limits and margin is maintained.

6G-98-0083
Byron Unit 2 Cycle 8 (BY2C8) Core Loading Pattern

DESCRIPTION:

The purpose of this Safety Evaluation was to review fuel and core loading design analyses for the Byron Unit 2 Cycle 8 (BY2C8) Core Reload. The BY2C8 Core Reload Design was designed to perform under nominal design parameters, Technical Specifications, and related Bases. The cycle peaking factor limits were increased from 2.5 to 2.6 for FQ and from 1.65 to 1.70 for FNDH which remain with the limits specified in the Byron Unit 2 Core Operating Limits Report (COLR). The BY2C8 core consisted of 89 new Fuel Assemblies, 84 once-burnt assemblies, and 20 twice-burnt assemblies. This activity allowed the refueling and subsequent operation of the Byron Unit 2 Cycle 8 Reactor Core Reload.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because all Byron UFSAR Chapter 15 accidents were evaluated as part of the reload design activity for BY2C8. For all accidents, the reload design did not increase the probability of occurrence of an accident previously evaluated in the Safety Analysis Report. The design and construction of the Core Reload was implemented with the same standards as previously installed equipment. All fuel mechanical, nuclear, thermal-hydraulic, and transient analysis design criteria were met, including initial and revised accident analysis limits and assumptions, including radiological consequences. The consequences of all accidents were determined to be within previous acceptance criteria and not increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the BY2C8 Core Design met all safety parameter limits thus ensuring all pertinent licensing basis acceptance criteria were met. The demonstrated adherence to these criteria precludes new risks to components and systems that could introduce new failure modes or initiating events. All design and performance criteria were met and no new failure modes or limiting single failure mechanisms were created nor did the core operate in excess of pertinent design basis operating limits.

6G-98-0083
Byron Unit 2 Cycle 8 (BY2C8) Core Loading Pattern
(Cont'd.)

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the BY2C8 Core Reload was demonstrated to operate within all safety analysis acceptance limits as defined in the plant Technical Specifications. The design was analyzed in accordance with NRC approved methodologies. The existing Reactor Core Safety Limits and Reactor Core Peaking Factor Limits were all determined to have been met with the reload design.

6G-99-0127
Byron Unit 2 Cycle 9 (BY2C9) Reload Design

DESCRIPTION:

The purpose of this Safety Evaluation was to review the Byron Unit 2 Cycle 9 (BY2C9) Core Reload Design. The change evaluated encompasses the composite effects of the following changes:

1. Fuel loading pattern.
2. Fuel mechanical design changes:
   a. 3-tab inner-to-outer strap joint for Inconel top and bottom non-mixing grids.
   b. Lengthen Guide Tubes and Instrument Tubes by 0.2 inches.
   c. Lengthen Fuel Rod by 0.32 inches.
   d. Shorten end plug by 0.12 inches (previously used design).
   e. Change from P+ to Zirc-4 spring pack with new variable pitch.
   f. Center WABA rods with active fuel region.
   g. Raise bottom Inconel grid 0.7 inches.
   h. Lower first Zircaloy mid-grid 0.3 inches.
   i. IFBA loading of 1.25X with 100 psig backfill pressure.
3. Relocate Secondary Sources into C-8 and N-8.
4. RCCA park position of 231 steps. Change RIL to correspond.
5. Use of 20 WABAs for a second cycle.
6. Elimination of Thimble Plugs for F/As with no other insert.

The changes were incorporated into many implementing procedures, including the Byron Curve Book, the Reload Design Key Parameter Checklist, COLR, and DRP 8-079.

6G-99-0127
BY2C9 Reload Design
(Cont'd.)

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because fuel construction meets all design criteria. The Core Fuel Loading Pattern and changes in operating characteristics do not produce any mechanisms by which any of the evaluated accidents, such as LOCAs or HELBs, etc., can be initiated. The consequences of previously evaluated accidents are not increased because the reload design process confirmed all design parameters satisfy the accident analysis limits and assumptions as documented in the UFSAR or other appropriate evaluations. The analyses included mechanical, nuclear, thermo-hydraulic and transient analyses, which concluded that all core parameter criteria, such as DNB, PCT, and fuel temperature, were met. In addition, the analyses showed that all system performance criteria, such as containment pressure and no water through pressurizer safeties, were met.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because this core reload's fuel mechanical features introduce no new failure modes. The reload key parameters and assumptions meet all standards and criteria. The core operates within pertinent design basis operating limits. Therefore, the cycle specific changes in these parameters introduce no new failure modes.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the core reload design process safety analysis was performed in accordance with NRC approved methodologies and indicates that BY2C9 operates within acceptable limits and margin is maintained.

6G-99-0055
Byron Unit 2 Cycle 8a (BY2CBA) Mid-Load Reload Design

DESCRIPTION:

The purpose of this Safety Evaluation was to review the Byron Unit 2 Cycle 8A (BY2C8A) Mid-Cycle Reload Design. The Reload Design was necessitated by the increase in nominal full power operating temperature (Tref) from 581° F to 583° F. The associated instrument loops requiring re-scaling are:

1. SSCR 99-001 Tave/Delta I Loop 2A (2RC-0411)
2. SSCR 99-002 Tave/Delta T Loop 2B (2RC-0421)
3. SSCR 99-003 Tave/Delta T Loop 2C (2RC-0431)
4. SSCR 99-004 Tave/Delta T Loop 2D (2RC-0441)
5. SSCR 99-005 Tref Converter (2RC-0412)
6. SSCR 99-006 Pressurizer Level Program, Auctioneered High Tave Deviation Alarm, and Turbine Loading Stop (C-16) Setpoint (2RC-0412-8)
7. SSCR 99-007 Steam Dump Control (2FW-0507)

Implementing procedures include:

1. SPP 99-019
2. BY2CBA descendent Byron Curve Book (BCS) figures and tables
3. Reload Design Key Parameter Checklist
4. BAP 560-1
5. BAR 2-18-F.16
6. 2SISR 3.1.7-001/2/3/4
7. 2BOA ELEC-1
8. 2BOA PRI-5
9. 2BOA PRI-6
10. 2BOA SEC-4
11. BOP RY-5

6G-99-0055
Byron Unit 2 Cycle 8a (BY2CBA) Mid-Load Reload Design
(Cont'd.)

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because each of the Analysis of Record above assumed an operating window for T-average between 569.1° F AND 588.4° F. The proposed T-average increase from 581° F to 583° F is still within this operating window and remains bounded by the Analysis of Record. The OTΔT and OTΔT protection system loops will be scaled, considering the increase in T-average, to maintain the nominal trip setpoint with the appropriate temperature penalties. The Reactor Coolant System will be re-scaled to provide its function of maintaining the coolant average temperature at its programmed value (583° F). The programmed value will be changed, but the overall function of the system will continue as designed. The Steam Dump Control System will be re-scaled to maintain the steam dumps design function of providing sufficient steam relief following a lead rejection of up to 50% power in order to avoid a reactor trip. Furthermore, the unit trip controller loop will be re-scaled to maintain its function of bringing the unit to no-load conditions without a significant average temperature undershoot. With respect to core design, an evaluation (Reference 3) was performed to verify the Cycle 8 Design continues to meet key safety parameter limits (Reference 4). The analyses are valid up to a T-average of 588.4° F. Since these changes do not induce or exacerbate any failures on any equipment, the proposed change does not involve an increase in the probability of these accidents, which were previously described in the Safety Analysis Report. The proposed T-average increase from 581° F to 583° F is still within this operating window and remains bounded by the Analyses of Record. Based on being maintained in the analyzed range, the proposed T-average change will not increase the consequences of any of these events.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the proposed change falls within a previously analyzed operating band. From a core design standpoint, the key safety parameter limits are met thus ensuring all pertinent licensing basis acceptance criteria are met. The OTΔT, OTΔT Reactor Control and Steam Dump Control Loops will be re-scaled to maintain their current protection and control functions, respectively. No new performance requirements are being imposed on any system or component such that any design criteria will be affected. No new failure modes or limiting single failures have been created by the proposed change.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the proposed change (Tave increase to 583° F) still meets the acceptance criteria for each Technical Specification (and TRM) LCO. The core design analysis, including the proposed change in T-average, verified the core will operate within safety analysis limits, therefore, subsequent analyses and evaluations.

6G-99-0154
DCP 9900233

DESCRIPTION:

The purpose of this Safety Evaluation was to provide generic routing criteria for telephone and computer lines in free air in the plant (excluding the Containment Building). The effect on plant equipment/systems is negligible due to the constraints put into the cable routing /installation notes.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because no accidents are affected as a result of this change because the installation criteria precludes interaction with any equipment required to mitigate the consequences of an accident. The cable does not interface with any SSC's assumed to function during or after an accident condition. Any equipment attached to the cables will not be used to mitigate any Design Basis Accidents, nor will it be an initiator of any event. No interaction with SSC's important to safety exists for this change, therefore, the probability of occurrence of a malfunction of these SSC's is not increased by this change. Installation, operation, or failure of this cable or its connected equipment does not affect the availability of components of any ECCS systems assumed in the safety analysis nor their ability to meet their safety function. Therefore, the consequences are not increased. The use of this cable and the equipment attached to it is not assumed or credited to reduce offsite dose during normal operation or following any Design Basis Accident or transient.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the cable is being installed in accordance with installation guidelines such that no possible interactions with SSC's can result from the installation, operation, or the failure of this equipment during accidents. No additional credible accidents or transients can occur due to installation of the cable or equipment. Neither the cable nor the equipment it is connected to actively inter-face with any SSC's assumed to function during postulated accidents. The equipment itself will not be used to mitigate any Design Basis Accident or transients.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the function of the cable/equipment associated with this change is not referenced in any way as the basis for a Technical Specification.

6G-99-0198
DCP 9900233

DESCRIPTION:

The purpose of this Safety Evaluation was to provide direction for installation of communication cable and data transmission cables routing in free air within the Containment Building. This evaluation does not cover use of fiber optic cables.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety is unchanged as a result of this change because the installation criteria precludes interaction with any equipment required to mitigate the consequences of an accident. The cable does not interface with any SSC's assumed to function during or after an accident condition. Any equipment attached to the cables will not be used to mitigate any Design Basis Accidents, nor will it be an initiator of any event.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the cable is being installed in accordance with installation guidelines such that no possible interactions with SSCs can result from the installation, operation, or the failure of this equipment during accidents. No additional credible accidents or transients can occur due to installation of the cable or equipment. This cable/equipment does not interact with any equipment important to safety and therefore, no new failure modes are created that could impact accidents or transient conditions.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the function of the cable/equipment associated with this change is not referenced in any way as basis for any Technical Specification.

6H-99-0083
Temp Procedure for 1BOSR 3.1.5-1, Rev. 1

DESCRIPTION:

The purpose of this Safety Evaluation was to perform a temporary procedure change due to a Train A Reactor Trip Breaker A Shunt Trip Test Pushbutton Failure in the depressed condition. Due to the failure of the test switch, the trip signal to the Reactor Trip Breaker was not removed so a jumper was installed across the Shunt Trip Test Pushbutton. When testing the undervoltage coil, the shunt trip block pushbutton is depressed to ensure the STA coil does not cause the breaker to open. To ensure the block pushbutton functioned properly, it is subsequently tested by depressing the block and simultaneously depressing shunt trip test pushbutton. If the block is faulted, pushing the shunt trip test pushbutton would open the Reactor Trip Breaker. In order to independently test the block pushbutton with the Shunt Trip Test Jumper installed, the jumper must be lifted to ensure a shunt trip signal is simulated.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the Reactor Trip Breakers are relied upon to trip the plant (insert the control rods) during an event that causes the SSPS inputs to exceed trip setpoints or from a manual reactor trip signal. The only postulated equipment failure previously evaluated was a short or an open in the jumper. In either case, this would cause the STA to de-energize, thereby tripping the reactor tip breaker via the shunt trip coil. The probability of occurrence or consequences of an accident or malfunction of this type is not increased by the current screening because the function of the jumper remains the same. The previous evaluation addressed the installation of the jumper and the current evaluation utilized the jumper for routine surveillance testing in lieu of the pushbutton.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the possibility of an accident or malfunction of a different type than evaluated previously is not created because the function of the temporary jumper Is not altered in the surveillance revision. The surveillance revision was needed to allow testing of the undervoltage and shunt trip coils independently using the jumper instead of the shunt trip pushbutton. Since the equipment functions remain the same, no new failure modes are introduced.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the margin of safety, as defined in the basis for Technical Specification 3.3.1 is not affected because the temporary modification does not affect any parameters upon which Technical Specifications are based.

6H-99-0396
See Attached Procedure List

DESCRIPTION:

The purpose of this Safety Evaluation was to revise the procedures listed on the attached list. The revised procedures will provide better operator guidance on determining SX M/U pump operability upon loss of SX M/U pump Jacket Water heaters for different pump ambient temperature conditions. This is in conjunction with UFSAR change DRP 8-076 which provided additional detail with respect to ability of the SX M/U Pumps to start during abnormal conditions. Also, remove the requirement to verify JW temp greater than 75° F from the 0B0SRs as this is not really a condition for starting the pumps. The pumps can be started with any jacket water temperature.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because this change implements improved guidance to the operators for determining SX M/U pump operability as revised in DRP 8-076. The DRP does not involve a physical change to the facility, but provides a better description of the SX M/U Pumps and River Screen House Ventilation Systems as they currently exist. Neither the SX M/U Pumps nor the River Screen House Ventilation Systems are initiators of any accidents. Therefore, the probability of occurrence of an accident is not increased. This change does not affect the ability of the SX M/U Pumps to provide design flow under design conditions. The SX M/U Pumps and the River Screen House Ventilation Systems will continue to operate as before. Therefore, this change cannot increase the consequences of an accident.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because this change implements improved guidance to the operators for determining SX M/U pump operability as revised in DRP 8-076. The DRP does not involve a physical change to the facility, but provides a better description of the SX M/U Pumps and River Screen House Ventilation Systems as they currently exist. It does not physically impact the SX M/U Pumps or the River Screen House Ventilation Systems. The systems will be operated as before. Therefore, there is no possibility of an accident or transient of a different type than any previously evaluated.

6H-99-0396
See Attached Procedure List
(Cont'd.)

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this change implements improved guidance to the operators for determining SX M/U Pump operability as revised in DRP 8-076. The DRP does not involve a physical change to the facility, but provides a better description of the SX M/U Pumps and River Screen House Ventilation Systems as they currently exist. There is no change in the manner in which the SX M/U Pumps or River Screen House Ventilation Systems operate.

The SX M/U pumps will continue to meet their design requirements to "withstand all design basis natural phenomena events and combinations of events except for Seismic Events during low Rock River flow or level (loss of SX Makeup Pump Suction), tornado, and river flood."

Activity Numbers for Safety Evaluation Summary Form

BOP SX-3	Rev. 13	(Remove "maintain ambient above 50^0 F)
0BOSR 7.9.6-1	Rev. 2	(Remove "verify JW $\geq$ 75^0 F)
0BOSR 7.9.6-2	Rev. 2	(Remove "verify JW $\geq$ 75^0 F)
0BOL 7.9	Rev. 2	(Add operability criteria)
BAR 0VHO1J-1-B5	Rev. 1	(Start pps at $\leq$ 40^0 F w/JW htr or $\leq$ 70^0 F w/o JW htr)
BAR 0-38-C4	Rev. 4	(Check RSH temp > 70^0 F if loss of power, if not see BOL)
BAR 1-20-A5	Rev. 2	(Check RSH temp > 70^0 F if loss of power, if not see BOL)
BAR 1-20-A11	Rev. 1	(Check RSH temp > 70^0 F if loss of power, if not see BOL)
BAR 1-20-A12	Rev. 6	(Check RSH temp > 70^0 F if loss of power, if not see BOL)
BAR 1-20-B5	Rev. 2	(Check RSH temp > 70^0 F if loss of power, if not see BOL)
BAR 1-20-B11	Rev .1	(Check RSH temp > 70^0 F if loss of power, if not see BOL)
BAR 1-20-B12	Rev. 1	(Check RSH temp > 70^0 F if loss of power, if not see BOL)
BAR 1-20-Cl2	Rev. 1	(Check RSH temp > 70^0 F if loss of power, if not see BOL)
BAR 1-20-D5	Rev. 1	(Check RSH temp > 70^0 F if loss of power, if not see BOL)
BAR 1-20-D11	Rev. 1	(Check RSH temp > 70^0 F if loss of power, if not see BOL)
BAR 1-20-D12	Rev. 1	(Check RSH temp > 70^0 F if loss of power, if not see BOL)
BAR 2-20-A5	Rev. 2	(Check RSH temp > 70^0 F if loss of power, if not see BOL)
BAR 2-20-A11	Rev. 4	(Check RSH temp > 70^0 F if loss of power, if not see BOL)
BAR 2-20-A12	Rev. 3	(Check RSH temp > 70^0 F if loss of power, if not see BOL)
BAR 2-20-B5	Rev. 2	(Check RSH temp > 70^0 F if loss of power, if not see BOL)
BAR 2-20-B11	Rev. 2	(Check RSH temp > 70^0 F if loss of power, if not see BOL)
BAR 2-20-B12	Rev. 4	(Check RSH temp > 70^0 F if loss of power, if not see BOL)
BAR 2-20-C12	Rev. 3	(Check RSH temp > 70^0 F if loss of power, if not see BOL)
BAR 2-20-D5	Rev. 1	(Check RSH temp > 70^0 F if loss of power, if not see BOL)
BAR 2-20-D11	Rev. 1	(Check RSH temp > 70^0 F if loss of power, if not see BOL)
BOP 199-EA OS	Rev. 16	(If JW not warm, notify SM for LCOAR eval, minimum 2nd floor)

6H-98-0081
ER 9711123

DESCRIPTION:

The purpose of this Engineering Request was to evaluate the installation of two Freeze Seals per ER 9711123. One is to be installed downstream of valve 2RC8842A and the second upstream of Valve 2RC8042A. The safety evaluation for this activity is accomplished via a validation of previously performed safety evaluation TI-95-0147.

TI-95-0147 was written to evaluate freeze seals as applied by BMP 3300-7 "Application of Freeze Seals to all Piping".

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the freeze seals are temporary and the application of BMP 3300-7 administratively controls freeze seal parameters such that the piping parameters assumed in Design Bases accidents are not affected.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the application of BMP 3300-7 administratively controls freeze seal parameters such that the piping parameters assumed in Design Bases accidents are not affected.

3. The margin of safety, as defined in the bases for any Technical Specification is not reduced because the modification does not change Technical Specification Bases parameters.

6G-98-0291
ER 9808072

DESCRIPTION:

The purpose of this Engineering Request was to install a freeze seal on Essential Service Water (SX) Line 1SX96AB-2" to support maintenance activities requiring isolation of Valve 1SX2166A. The freeze seal will be located downstream of the valve on a horizontal portion of the line. The freeze seal process is administered by procedure BAP 1600-16. The procedure is being applied to ER 9808072 and WR 980020373.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the applicable accident (Chapter 15.1, Increase in Heat Removal) has been evaluated to be appropriately bounded by the planned contingencies.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the freeze seal essentially performs the same function as a closed valve. With the planned contingencies, the SX System remains able to perform its design safety function. The process described by BAP 1600-16 adequately minimizes potential equipment failures due to application of the freeze seal.

3. The margin of safety, as defined in the bases for any Technical Specification is not reduced because the activity does not change Technical Specification Bases parameters.

6G-99-0072
ER 9808116

DESCRIPTION:

The purpose of this Engineering Request was to evaluate the temporary freeze seal placed on Chemical & Volume Control (CV) Systems line 2CVG3C-3/4" to support maintenance activities on Line 2CVL3-3/8". This freeze seal made up an OOS boundary and provided isolation while work is performed on line 2CVL3-3/8". Contingency plans are contained in BMP 3300-7 and in the freeze seal evaluation to ensure controls are in place should the freeze seal fails.

SAFETY EVALUATION SUMMARY-.

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the effects on the system as a result of the freeze seal installation were evaluated and determined to not constitute an unreviewed safety question. The engineering review of the freeze seal evaluation included the determination that the freeze seal would not adversely affect the material strength and seismic requirements of the piping system. A freeze seal acts equivalent to an isolation valve and its failure is detected by the initial slow leakage across the seal boundary. Approved contingency actions will be in place to restore the system's pressure boundary. A freeze seal failure is not a precursor to any of the accidents postulated in the UFSAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the affected component/system are reviewed to determine the overall impact resulting from the isolation created by the freeze seal. LCOAR conditions are established as necessary to ensure that the correct level of administrative control is applied to the freeze seal evolution. The altered configuration of the system was evaluated and determined to not represent an unreviewed safety question. Contingencies approved in the freeze seal controls ensure that restoration of the system is immediately performed at the onset of a detected leak. This ensures that a malfunction of a different type is not created. Because the appropriate regulatory requirements, precautions, limitations and prerequisites will still be met, there is no possibility of the occurrence of an accident or malfunction of a type different than those evaluated in the SAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because no Technical Specification are directly affected by this change.

6G-98-0302, Rev. 1
ER 9808888

DESCRIPTION:

The purpose of this Engineering Request was to install a freeze seal on Component Cooling Water (CC) Line 1CC48BA-1" (discharge side of relief valve 1CC9421B) to support maintenance activities of valve 1CC9412B. The 1A Letdown Heat Exchanger was placed Out-of-Service (OOS) during the freeze process. A blind flange was installed when the relief valve was removed. The freeze seal process is administered by procedure BAP 1600-16. The procedure is being applied to ER 9809025 and WR 98002347. The activity was performed during Modes 5, 6 or defueled when the CC System is not required to be operable.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the CC system has no safety-related function in the applicable modes. Administrative controls and contingencies are adequate to maintain analyzed probabilities and consequences.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because installation of the freeze seals serves the same function as a closed valve. The installation of a blind flange was planned following relief valve removal. Auxiliary Building flooding concerns were evaluated and minimized. The process described by BAP 1600-16 adequately minimizes potential equipment failures due to application of the freeze seal.

3. The margin of safety, as defined in the bases for any Technical Specification is not reduced because the activity did not change Technical Specification Bases parameters.

6G-98-0301
ER 9809025

DESCRIPTION:

The purpose of this Engineering Request was to install freeze seals on Safety Injection (SI) System lines 1SI03CB-2" and 1SI03DB-2" to support maintenance activities requiring upstream and downstream isolation of valve 1SI8833B. These freeze seals provide redundant isolation to Check Valves 1SI8905B/C and 1SI8949B/C. The freeze seal process is administered by procedure BAP 1600-16. The procedure is being applied to ER 9809025 and WR 98002347. The activity is scheduled for performance during modes 5, 6 or defueled.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the applicable accident (Chapter 15.6, Decrease in Reactor Coolant Inventory) has been evaluated to be appropriately bounded by the planned contingencies.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because installation of the freeze seals serves the same function as a closed valve. The freeze seals are considered redundant isolation and, therefore, the failure of either or both will not affect RCS inventory. The process described by BAP 1600-16 adequately minimizes potential equipment failures due to application of the freeze seal.

3. The margin of safety, as defined in the bases for any Technical Specification is not reduced because the activity does not change Technical Specification Bases parameters.

6G-99-0009
ER 9809338

DESCRIPTION:

The purpose of this Engineering Request was to evaluate installation of a temporary freeze seal on Essential Service Water (SX) System Line 1SX48A-1 1/2" to support maintenance activities on valve 1SX2085. This line is the return line from the cubicle cooler 1VA05S. The freeze seal made up the Out-Of-Service (OOS), boundary and provided isolation boundary. Contingency plans were available as part of the freeze seal evaluation in the event of a freeze seal failure.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the effects on the system as a result of the freeze seal installation were evaluated and determined to not constitute an unreviewed safety question. The engineering review of the freeze seal evaluation included the determination that the freeze seal would not adversely affect the material strength of the piping system. A freeze seal acts equivalent to an isolation valve and its failure is detected by the initial slow leakage across the seal boundary. A freeze seal never fails in a catastrophic manner and its failure can be detected with plenty of time to implement the approved contingency actions to restore the system's pressure boundary.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the affected component/system are reviewed to determine the overall impact resulting from the isolation created by the freeze seal. An OOS is established to ensure that the correct level of administrative control is applied to the freeze seal evolution. The altered configuration of the system was evaluated and determined to not represent an unreviewed safety question. Failure of a freeze seal is a slow deterioration of the pressure boundary that is detected by evidence of a slowly developing leak. Contingencies approved in the freeze seal controls ensure that restoration of the system is immediately performed at the onset of a detected leak.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the temporary installation of the freeze seals is done while the component/system is OOS. Installation of freeze seals does not affect any parameters upon which Technical Specifications are based

6G-99-0002
WR 990000334

DESCRIPTION:

The purpose of this Work Request was to temporarily expand the Main Control Room Envelope as described in UFSAR section 6.4.2.1 to include the Unit 1 and Unit 2 Process Computer Rooms by the intentional propping open of the Unit 1 and/or Unit 2 Computer Room Door(s), 0DSD432 and 0DSD440.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the change is to temporarily expand the Main Control Room Envelope by propped opening of the Process Computer Room Doors. The failure of the Main Control Room Envelope does not in and of itself directly lead to a LOCA or any other postulated accident. The use of compensatory measures to ensure the Process Computer Room Doors are closed in the event of an accident will ensure that the analyses for applicable accidents are not affected and, therefore, the probability of each accident is not affected.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because compensatory measures to close Computer Room Doors in the event of a LOCA will ensure the original design conditions assumed for the performance of this equipment are unchanged. Conservatism present in chiller sizing calculations provides suitable margin to neglect increased loading on the Main Control Room Chillers due to Process Computer Room Heat Loads. The normal operating temperature range of the Essential Service Water Cooling System during non-accident conditions provides sufficient heat sink capacity to ensure the overloading of chillers, or exceeding performance ratings of the essential service water cooling towers, will not occur.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the Station will ensure compensatory measures required in the event of a fire within the Process Computer Rooms are in place as required by Administrative Technical requirements. In the event any of these compensatory measures fail, the plant is analyzed for safe shutdown in the event that the Main Control Room becomes uninhabitable.

6G-99-0132
WR 990074856

DESCRIPTION:

The purpose of this Work Request was to temporarily remove the "M" hole pin from the Spent Fuel Tool to accommodate the potential failure of the hold down spring bolts on the Fuel Assemblies to be off loaded. This would allow the spent fuel tool to latch into the assembly with these corner clamps having separated from the top nozzle base. The pin would be re-installed after the potential problem is corrected.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because without this change, the ability to properly latch the top nozzle may be more difficult. With the change, the latch is successfully completed and the assembly is securely fastened. Once the latched condition is achieved, the presence of the pin provides no holding function and therefore this change has no effect on the probability of dropping the Fuel Assembly.

   The consequences do not increase because this change does not result in either an increase in the source term for the radioactive release; an increase of the number of fuel pins damaged; degradation of the radiation detection system; or degradation of the FB ventilation for SFP. Since none of these are affected by this tooling change, the consequences of the accident are not affected.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because The purpose and function of the spent fuel tool is to safely transport new and spent fuel in the pool area. The only limiting failure possible with the tool is the failure to hold the assembly, which has been previously addressed. This small mechanical change does not have a different (or more significant) impact than that.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because no Technical Specification Bases, or SAR analyses are affected by this change.

6G-98-0113
OOS 940012511 & 940012434

DESCRIPTION:

The purpose of this Out-of-Service (OOS) was to leave the mechanical condenser tube cleaning system (Amertap System) for Byron Unit 1 and Unit 2 tagged out-of-service for greater than 180 days. Condenser tubes are cleaned during outages by an alternate mechanical method. Out of Service (OSS) requests in place greater than 180 days require a safety evaluation.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the activity does not affect analyzed accidents.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the isolation of the Amertap System cannot add new failure modes.

3. The margin of safety as defined in the bases for any Technical Specification is not reduced because the activity does not change Technical Specification Bases parameters.

6G-98-0232
OOS 940012923 (Unit 1) & 940012924 (Unit 2)

DESCRIPTION:

The purpose of this OOS was to isolate the Boron Thermal Regeneration System (BTRS) from interfacing systems in accordance with On-Site Review (OSR) 92-034. The system was placed in lay-up conditions per instructions in the OSR. The expected continued operation of Byron Station as a "base load" power plant has made the BTRS function of compensating for xenon transients and other reactivity changes as a result of reactor power variations unnecessary.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the BTRS system is in wet layup and isolated from the CV System by redundant valves. This change makes the system no longer a "moderate energy" fluid system.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because no analyzed accidents are affected.

3. The margin of safety, as defined in the bases for any Technical Specification is not reduced because the activity does not change Technical Specification Bases parameters.

6G-98-0233
OOS 980001044

DESCRIPTION:

The purpose of this OOS was to remove the solenoid from valve 1FSV-BR7022 so it can be installed on valve 1SI9978A. The Boron Thermal Regeneration System (BTRS) has been isolated from interfacing systems in accordance with On-Site Review (OSR) 92-034. Valve 1FSV-BR7022 is one of the valves which isolates the BTRS System from the CV System. The BTRS system was placed in lay-up conditions per instructions in the OSR. The expected continued operation of Byron Station as a "base load" power plant has made the BTRS function of compensating for xenon transients and other reactivity changes as a result of reactor power variations unnecessary.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the BTRS system is in wet layup and isolated from the CV System by redundant valves. The removal of a solenoid from one of the "failed" air operated valves used as BTRS isolation does not change the function, position, operation or mode of the previously isolated valve.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because no analyzed accidents are affected.

3. The margin of safety, as defined in the bases for any Technical Specification is not reduced because the activity does not change Technical Specification Bases parameters.

6G-98-0234
OOS 980001045 (Unit 1) & 980001631 (Unit 2)

DESCRIPTION:

The purpose of this OOS was to remove the solenoid from valves 1FSV-BR7002 and 2FSV-BR7002 so the solenoids can be used on operational systems. The Boron Thermal Regeneration System (BTRS) has been isolated from interfacing systems in accordance with On-site Review (OSR) 92-034. Valve 1FSV-BR7022 is one of the valves which isolates the BTRS system from the CV system. The BTRS system was placed in lay-up conditions per instructions in the OSR. The expected continued operation of Byron Station as a "base load" power plant has made the BTRS function of compensating for xenon transients and other reactivity changes as a result of reactor power variations unnecessary.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the BTRS system is in wet layup and isolated from the CV System by redundant valves. The removal of a solenoid from one of the "failed" air operated valves used as BTRS isolation does not change the function, position, operation or mode of the previously isolated valve.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because no analyzed accidents are affected.

3. The margin of safety, as defined in the bases for any Technical Specification is not reduced because the activity does not change Technical Specification Bases parameters.

6G-98-0243
OOS 980002260

DESCRIPTION:

The purpose of this OOS was to take the "B" (one of two) condensing units for the Technical Support Center (TSC) Heating, Ventilation and Air Conditioning (HVAC) System Out-of-Service for repair.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the operation of the system as described in the SAR is not affected. Monitoring by the system engineer during the Summer of 1998 has confirmed the ability of Train A compressor to satisfy design temperatures.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because environmental conditions will not degrade such that TSC equipment is affected.

3. The margin of safety, as defined in the bases for any Technical Specification is not reduced because the activity does not change Technical Specification Bases parameters.

6G-98-0217
OOS 980003840

DESCRIPTION:

The purpose of this OOS was to isolate 3 site boilers used for increasing humidity. The activity will reduce preventive maintenance work load by extending the frequency of pressure vessel inspections required by ASME and IDNS. The activity also isolates associated Turbine Building Drains and the Make-Up Demineralizer Water Source.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because no analyzed accidents are affected by the activity. No postulated accident decreases humidity.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because no new failure modes exist.

3. The margin of safety, as defined in the bases for any Technical Specification is not reduced because the activity does not change Technical Specification Bases parameters.

6H-99-0241
OOS 980012235 & 980012239

DESCRIPTION:

This evaluation was written to support the referenced OOSs being greater than 180 days old. These OOSs are for the 1/2CV8118 valves, which are relief valves on the discharge of the Positive Displacement pumps. The Positive Displacement (PD) Pumps are themselves in a long-term OOS. The OOSs previously in place for the PD Pumps were established since VT-2 pressure testing has not been performed on those pumps. Since the PD Pumps are in a long term OOS, the station has decided not to test their discharge relief valves. The OOSs for the relief valves is to assure the PD Pumps are not returned to service without the relief valves being tested within the Code requirements and periodicity. The previous in place OOS for the Unit 2 PD Pump (940007543) was supported by Safety Evaluation 6G-97-0198. The OOS for the Unit 1 Pump (940013105) was supported by Safety Evaluation Screening 6H-98-0207, which referenced the aforementioned Safety Evaluation, 6G-97-0198.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report Is not increased because there is no change in the probability of occurrence or consequences of an accident of malfunction of equipment important to safety previously evaluated in the original safety analysis because this validation has not changed or impacted those considerations as addressed in the original safety evaluation.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because this validation incorporates the original referenced safety evaluation and has not created the possibility of an accident or malfunction different than the analysis of the original safety evaluation.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this validation incorporates the original referenced safety evaluation and has not reduced the margin of safety as defined in the basis for any Technical Specification different than the analysis of the original safety evaluation.

6G-99-0211
Abnormal Valve Lineup For 2SX173

DESCRIPTION:

The purpose of this Safety Evaluation was to install a temporary system alteration to the Essential Service Water (SX) System cooling subsystem to the 2B Auxiliary Feedwater (AFW) pump. The temporary change will be instituted under a station Work Request package that is being instituted as a result of mechanical binding of the air-operated 2SX173 valve. This mechanical binding was identified during surveillance testing of the 2B AFW pump on 12/22/99. The 2SX173 valve receives an automatic signal to open upon when the 2B AFW pump is started. Once the 2SX173 valve opens, the 2SX178 also opens which creates a flowpath for Essential Service Water through the jacket water and Lube Oil Heat Exchangers and the cubicle cooler for the 2B AFW Pump room. Due to the potential for mechanical binding of the valve which would prevent the 2B AFW Pump from receiving adequate SX cooling, the temporary change is to fail-open the 2SX173 and 2SX178 Valves.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because this change has no impact on the probability of any of the accidents identified above. The temporary change places the 2SX173 valve and the 2SX178 valve in their accident conditions, which does not impact the probability of an accident.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because an accident of a different type is not created by this temporary alteration. The essential service water system will continue to perform its design function. The 2SX173 and 2SX178 valves will be failed in their accident positions. There is no possibility that this configuration could create an accident of a different type.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the temporary change does not impact the ability of the AFW System (TS 3.7.5) or the SX (TS 3.7.8) System to perform their intended design functions. The systems will still be fully capable of performing their intended design functions. Therefore, there is no reduction in the margin of safety as described in the bases for the referenced sections of the Technical Specifications.

6G-98-0250, Rev. 1
Beacon Implementation

DESCRIPTION:

The purpose of this Safety Evaluation was to perform implement the BEACON Surveillance Methodology for Byron Unit 1 and 2 and the replacement of the surveillance codes for INCORE, FOLLOW and TOTE. INCORE and FOLLOW are used for monthly reactivity and power distribution Technical Specification Surveillances. TOTE is used to generate cycle specific burnup and isotopic balances. No modification of the current plant instrumentation and/or other plant components is required for the implementation of BEACON methods. The activity is a change to a procedure described in the SAR.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the BEACON methodology satisfies all fuel mechanical, nuclear, thermal-hydraulic, and transient analysis design criteria. In addition, the BEACON methodology does not change any of the key safety parameter limits or levels of margin as considered in the reference Design Bases evaluations.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because all design and performance criteria will continue to be met and no new failure modes or limiting single failure mechanisms are created.

3. The margin of safety, as defined in the bases for any Technical Specification is not reduced because the activity does not change Technical Specification Bases parameters.

6G-98-0093
Byron Tech Review - BYTR 98-017

DESCRIPTION:

The purpose of this Safety Evaluation was to perform the activity is the in-situ testing of Byron Unit 2 Steam Generator Tubes with identified defects. The tubes will be pressurized up to 6000 psig utilizing Framatome Technologies Inc. procedures. The leak rate of the tubes will be monitored and the tube will be plugged when complete. This activity is to support the collection of leakage data for tubes with defects. The data is factored into conditional monitoring and operational assessments for the specific defect mechanism. The activity represents a test or experiment not described in the SAR.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because analyzed accidents are not affected. The test is performed with the Steam Generator isolated and out of service. The test pressure of 6000 psig is well below expected burst pressures of 10000-11000 psig. Defects may cause a decrease in the expected burst pressure, however the tube is expected to exhibit "leak before break" characteristics. The test fluid volume is small and tube leakage during the test cannot affect surrounding tubes.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the tube is removed from service by an approved plugging technique. Surrounding tubes are protected from impingement by "stabilizing" tubes with circumferential defects.

3. The margin of safety, as defined in the bases for any Technical Specification is not reduced because the activity does not change Technical Specification Bases parameters.

6G-98-0094
Byron Tech Review - BYTR 98-018

DESCRIPTION:

The purpose of this Safety Evaluation was to implement Framatome Technologies Inc. (FTI) procedures to remove a portion of a tube from a Steam Generator and to install a welded plug at the tubesheet end from Unit 2. The activity involves removal of the tube-to-tubesheet weld, relaxation of the expanded tube in the tubesheet, cutting the tube from the inside and pulling the tube through the tubesheet to the primary channel head. The activity is a contingent activity to be implemented upon less than satisfactory eddy current test results. UFSAR Table 5.4-3, Steam Generator Design Data states the number of Steam Generator Tubes. The removal of a Steam Generator tube, therefore, represents a change to the facility as described in the Safety Analysis Report.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the FTI Tube Pull process and subsequent plugging of the tube(s) removed ensure the integrity of the Steam Generator. The activity was performed with the Steam Generator isolated and out of service.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the tube removal and tube plugging processes are limited to the tube selected for removal and do not affect the surrounding tubes, equipment, systems or functions.

3. The margin of safety, as defined in the bases for any Technical Specification is not reduced because the activity does not change Technical Specification Bases parameters.

6G-98-0130
Byron Tech Review – BYTR 98-018

DESCRIPTION:

The purpose of this Safety Evaluation was to remove portions of three tubes from the Unit 2B Steam Generator using Framatome Technologies Inc. (FTI) methodology as described in Document 51-1264562-00, "50.59 Safety Evaluation Input for Tube Pull – Unit 2 for laboratory analysis of anomalous eddy current indications. A portion of each tube will remain intact in the Steam Generator. The affected tubes will be removed from service by the installation of a welded plug at the hot leg tube end and a rolled plug at the cold leg tube end.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the operation of the Steam Generators with the tube remnants isolated by tube plugging does not affect the function of the Steam Generators or other components.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the probability of primary to secondary leakage is not increased. The FTI analysis evaluated both operating and transient conditions and found no significant additional failure mechanism.

3. The margin of safety, as defined in the bases for any Technical Specification is not reduced because the proposed activity does not change Technical Specification Bases parameters.

6H-98-0039
Byron 2nd Interval Inservice Testing Plant for Pumps and Valves

DESCRIPTION:

The purpose of this Safety Evaluation was to make Revision 1 to the Byron 2nd Interval Inservice Testing Plan for Pumps and Valves. With the exception of the following, all changes were either typographical or the addition of clarifying information:

1. The backflow and position indication tests for the following valves are now contained in a Refuel Outage justification and listed in the program as Refuel frequency. This is an administrative change only.

1/2CC9486	1/2CV8113	1/2PS228A	1/2PS230B	1/2SI8968
1/2CC9518	1/2IA091	1/2PS228B	1/2PS231A	1/2WM191
1/2CC9534	1/2PR002G	1/2PS229A	1/2PS231B	1/2W0007A
1/2CS008A	1/2PR002H	1/2PS229B	1/2RY8046	1/2W0007B
1/2CS008B	1/2PR032	1/2PS230A	1/2RY8047	

2. The closure testing (backflow) of valves 1/2CV8546 and 1/2SI8926 is being added to the program. Operability assessment 96-032 was performed to verify that no safety concerns exist due to the previous absence of this test in the IST program.

3. The 1FW043A-D valves were deleted from the program since they were removed from the feedwater system.

4. The partial stroke test of the 1/2FW009A-D and 1/2MS001A-D is being deleted from the program. Stroke testing of these valves at power would result in a reactor trip, and because partial stroke testing at power presents the unwarranted risk of a potential reactor trip. Stroke time testing of these valves will be completed during cold shutdown, as conditions allow, in accordance with OM-10, paragraph 4.2.1.2.

5. The valve table has been revised to correctly reflect the required the five-year test frequency for safety valves 1/2MS013A-D, 1/2MS014A-D, 1/2MS015A-D, 1/2MS016A-D and 1/2MS017A-D. Previously, the table incorrectly indicated a ten-year frequency in accordance with their ASME Class 2 status. This change is an administrative change to correct a typographical error. At Byron, the test is performed at the more conservative frequency of every refueling outage.

6. Valve 1PR002G has been removed from the program because it has been capped and rendered non-operable.

6H-98-0039
Byron 2nd Interval Inservice Testing Plant for Pumps and Valves
(Cont'd.)

7. 0SX127A/B valves and their associated relief request have been removed from the program due to DCP 9700294 having removed them from the essential service water system.

8. Valves 0WW049A/B are electively being added to the IST Program. Prior to DCP 9700294, the Well Water (WW) and the Essential Service Water (SX) systems discharged into the SX basin via a common discharge point. Check valves 0SX127A/B prevented flow from the SX entering the WW system when closed. Valves 0SX127A/B allowed WW flow to the SX basin when open. The modification provided separate discharge points for SX and WW to the SX basin and the 0SX0127A/B valves were re-designated as 0WW049A/B, repositioned slightly, and retained in the WW discharge pathway to the SX basin. Valves 0SX127A/B were Safety-Related, Asme Code Class 3 valves. Because of their function and code class they were required to be within the scope of the IST program. Valves 0WW049A/B are non-Safety-Related and non-Code Class (not Class 1, 2, or 3) components. NUREG-1482, Section 2.2 indicates that non-Code components are not required to be within the scope of the IST program. However, because it is important to ensure the deep well is capable of transferring water to the ultimate heat sink, Valves 0W49A/B are electively being added to the IST Program.

9. Relief request VR-2 was revised per guidance provided in the NRC Safety Evaluation Report issued November 18, 1996. This revision incorporates more aggressive actions to be taken in the event that one member of the CS valve in a given sample group fails its inspection criteria.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the changes involved with converting to the new code do not require physical changes to the plant. Normal plant operations will not be affected. The intent of the Asme Code is to ensure the operational readiness of pumps and valves required to function to shutdown the plant to cold shutdown, maintain cold shutdown, or mitigate an accident. The major changes involve the addition of new testing requirements, which should improve the reliability of the safety-related components when called upon to perform. The additional testing will be performed during plant conditions, which will not adversely impact the systems involved. This additional testing will increase the level of confidence that the safety systems will function as designed. The probability of occurrence or the consequences of an accident will not increase.

6H-98-0039
Byron 2nd Interval Inservice Testing Plant for Pumps and Valves
(Cont'd.)

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the changes involved with converting to the new code do not require physical changes to the plant. The major changes involve the addition of new testing requirements, which should improve the reliability of the safety-related components when called upon to perform. The additional testing will be performed during plant conditions, which will not adversely impact the systems involved. This second interval initial program has been prepared in accordance with Oma-1988, Part 6 and Part 10, as required per 10CFR50.55a(f)(4)(ii). Any deviations from this code have been in relief requests for the NRC review and approval.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change does not affect any parameters upon which Technical Specifications are based.

6H-98-0131, Rev. 1
DCP 980089, 9800090, 9800316, 9800317

DESCRIPTION:

The purpose of this Safety Evaluation was to review the replacement of Main Control Room Recorders with solid state paperless recorders and a computer data collection system and RS-485 communication highway for archival purposes. Furthermore, this validation is to document the acceptability to remove the HEPA Filter differential recorder (i.e., 0PDR-VA030).

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the solid state paperless recorders are seismically qualified per IEEE 344-1975, are compatible with the installed environmental conditions and meet or exceed the RFI/EMF shielding criteria. The recorder display replicates the existing paper recorder visual display and does not increase the need for operator intervention or increase operator burden. The new recorders are have a faster response time and are more accurate. The new recorders do not adversely impact any loop inaccuracies. Failure of the new recorders are similar to failure of the current recorders and do not impact system performance or capability.

   The communication cable connections will be terminated to preclude loop failure if a single recorder is physically removed. Data is redundantly stored either through the on-board floppy or via the communications link to the Data Collection Unit.

   The replacement of the 0PDR-VA030 does not impact the associated Safety Related ESF ventilation systems since the recorder does not perform an active role in the monitored systems operation.

   The installation of new recorders and the removal of 0PDR-VA030 will not impact the performance of any SSCs necessary to mitigate any accidents.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the proposed changes do not adversely affect the operation of any plant SSC important to safety. The installation of solid state recorders does not introduce any system level failure not previously considered. The new recorders have layers of redundancy to ensure data collection and continued operator display.

   The electrical raceway and cable routing were designed and installed in accordance with the appropriate seismic and electrical criteria.

6H-98-0131, Rev. 1
DCP 980089, 9800090, 9800316, 9800317
(Cont'd.)

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because no Technical Specifications were impacted by the changes.

6G-98-0044
DCP 9500393

DESCRIPTION:

The purpose of this Safety Evaluation was to review numerous activities associated with the Replacement Steam Generator Outage (SGRO), at Byron Station, that needed to be completed prior to the beginning of the outage. These activities included replacement Steam Generator (RSG) Transport, RSG haul route upgrades (railroads, roads, bridges, etc.), original Steam Generator (OSG) haul route upgrades, Steam Generator offload system installation, erection of a RSG temporary storage structure, erection of a OSG decontamination facility, and various excavation and construction activities associated with the outside lift system and runway beam support structures.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the activities associated with this change had no impact on any accidents contained in the Safety Analysis Report for Byron Station. This included evaluation of the potential impact from the installation activities on equipment important to safety at Byron Station. The accidents analyzed included TNT explosion, LOOP/LOCA, Seismic Events, tornadoes, and flooding. The evaluation included the equipment important to safety that is located on the Byron Site outside of Containment such as the ultimate heat sink and the offsite power lines. The review concluded that none of the activities resulted in an increase in probability or consequences in the Safety Analysis Report.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the installation and erection of various structures and equipment to support the Unit 1 RSG project did not create any new accidents or malfunctions of equipment of a different type than previously evaluated in the Safety Analysis Report. There were appropriate controls that were placed upon all activities, which precluded the possibility of new or different malfunctions. All activities that are the subject of this evaluation were performed in accordance with the approved plan and procedures.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this activity did not adversely impact any Technical Specification systems or equipment.

6G-99-0052
Improved Technical Specification (ITS) Bases Change to B 3.9.2

DESCRIPTION:

The purpose of this Safety Evaluation was to revise ITS Bases B3.9.2 to clarify when the RWST would be considered a potential dilution source. The RWST would be considered to be potential dilution source if the RWST Boron Concentration is NOT above the Refueling Boron Concentration specified in the COLR. The effect of this change would remove the requirement to close CV112D/E if the RWST Boron Concentration is greater than Refueling Boron Concentration, but less than the RCS Boron Concentration.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the change to the Bases specifies that the RWST would be considered a potential dilution source if the RWST Boron Concentration is NOT above the Refueling Boron Concentration specified in the COLR. UFSAR section 15.4.6 states, "Inadvertent dilution is prevented by administrative controls which isolate the RCS from the potential source of unborated water. CVCS Valves, specified in Technical Specification, will be verified closed and secured in position by mechanical stops or by removal of air or electrical power. These valves block all flow paths that could allow unborated makeup water to reach the RCS. Any makeup which is required during refueling will be borated water supplied from the RWST. In addition the Bases for LCO 3.9.2 states, "This LCO requires that flow paths to the RCS from unborated water sources be isolated to prevent unplanned boron dilution during Mode 6 and thus avoid a reduction in SDM".

   The accident described in 15.4.6 involves the addition of unborated water to the RCS, resulting in a reduction in the Boron Concentration. During Mode 6, Tech. Spec. LCO 3.9.1 requires that the RCS, refueling canal, and the refueling cavity Boron Concentrations be above the limit specified in the COLR for refueling operations. Thus, if the RWST Boron Concentration is above this limit specified in the COLR, there is no credible means in which the RWST can dilute the RCS below this value, nor can the RWST cause a reduction in Shutdown Margin below the minimum required Refueling Boron Concentration.

   The change to the Bases clarifies that the RWST is only a potential dilution source IF its Boron Concentration is below the Refueling Boron Concentration specified in the COLR. 1/2BOL 9.2 specifies that CV112D/E are required to be secured in the closed position IF the RWST Boron Concentration is less than the Refueling Boron Concentration limit of LCO 3.9.1.

6G-99-0052
Improved Technical Specification (ITS) Bases Change to B 3.9.2
(Cont'd.)

Per UFSAR 15.4.6.3 dilution of the RCS during refueling is precluded through administrative control of valves in the possible dilution flow paths. Thus, valves CV112D/E will be closed if the RWST Boron Concentration is below the Refueling Boron Concentration specified in the COLR (causing a potential dilution source from the RWST). Thus, changing the Bases as specified will not increase the consequences of an inadvertent dilution accident since it will continue to be precluded per LCO 3.9.2

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because changing the Bases as specified will not create the possibility of an accident or transient of a different type than previously analyzed. Allowing CV112D/E to remain open when the RWST Boron Concentration is greater than the Refueling Boron Concentration will not create a new accident since the RWST will not be considered a potential source of unborated water, and there is no credible means to use the RWST to reduce the Shutdown Margin in the reactor.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change to the Bases clarifies when the RWST would be considered a potential dilution source. The RWST would be considered to be potential dilution source if the RWST Boron Concentration is NOT above the Refueling Boron Concentration specified the COLR. The effect of this change would remove the requirement to close CV112D/E if the RWST Boron Concentration is greater than Refueling Boron Concentration, but less than the RCS Boron Concentration. Thus, the actions of LCO 3.9.2 would be followed if a chemistry sample determined that the RWST Boron Concentration was below the Refueling Boron Concentration specified in the COLR.

6H-99-0420
ODCM Chapter 12, Rev. 1.6

DESCRIPTION:

The purpose of this Safety Evaluation was to review multiple changes made to the ODCM. The changes made to the ODCM resulted with the implementation of the Technical Requirements Manual (TRM) made from the conversion to ITS.

In addition, several additional administrative changes were performed. In Section 12.1, definitions were updated to fully agree with ITS/TRM definitions. The intent of the definitions was not changed, just the wording, to completely align with ITS/TRM wording. Additionally, section 12.6 was revised to refer to the Independent Technical Review (ITR) and PORC process, replacing the On-Site Review and investigative function. This was performed to agree with the current revision of site procedure BAP 1210-1.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because Chapter 12 of the ODCM documents the station's Radioactive Effluent Technical Standards. No standards listed in the ODCM where changed as a result of this revision. The changes were primarily administrative in nature, and do not change any of the specifications spelled out in the ODCM. Therefore, neither the probability of an occurrence nor the consequences of an accident or malfunction will be affected by the change in format.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because ODCM Chapter 12 lays out methodology for monitoring offsite releases during normal operations, and monitoring the environment to verify compliance with effluent limits. It does not have any affect on plant structures, systems, or components.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because no plant structures, systems, or components were affected by this administrative change to the format of ODCM Chapter 12, Rev 1.6.

6G-99-0077
Off-Site Dose Calculation Manual (ODCM), Rev. 2, April 1999

DESCRIPTION:

The activity is to implement the April 1999, Revision 2 of the Off-Site Dose Calculation Manual (ODCM). The ODCM has been revised to reflect the common methodology used at all ComEd Nuclear Stations for determining the TEDE to the public.

These commonalties are reflected in Chapters 1 through 7 and Appendices A through C of this revision. Currently, these methodologies are stated in Chapters 1 through 9 and Appendices A through E. This revision edits the text for clarity by removing redundant information, site specific criteria, and combining chapters, appendices and tables. Two criteria are added for the station to evaluate the effect of dose to the public. These are the dredging of rivers and storage of radioactive material on site.

The dredging activity is not applicable to Byron Station because the rule applies to navigable rivers. The Army Corp of Engineers dredge rivers for navigation purposes and disposes the soil on non-ComEd sites. The Rock River is not navigable and when dredged the material is disposed of on ComEd property. The second evaluation has already been implemented at Byron Station. During the SGRP, B1R08, an additional TLD and surveillance was added to the RETS/REMP program. The surveillance monitors the exterior of the Old Steam Generator Facility (OSGF) and accumulation of liquids from inside the building.

This activity does not change the current requirements of the RETS/REMP Program. All surveillance criteria remain the same.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the activity is to change the ODCM by editing the text for clarity by removing redundant information, site specific criteria, and combining chapters, tables, and appendices. The changes are editorial. The current RETS/REMP Surveillances remain the same. These changes will not affect any system, structure or component. Therefore, the activity will not increase the probability of occurrence of any accident or transient.

   The activity is to change the ODCM by editing the text for clarity by removing redundant information, site specific criteria, and combining chapters, tables, and appendices. The changes are editorial. The current RETS/REMP Surveillances remain the same. These changes will not affect any system, structure or component.

6G-99-0077
Off-Site Dose Calculation Manual (ODCM), Rev. 2, April 1999
(Cont'd.)

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the changes to the ODCM affect no system, structure or component. These changes will not alter or modify any system, structure or component there by creating any accident or transient of a different type than any previously evaluated. The changes to the ODCM are editing the present format of the current version. This revision edits text for clarity by removing redundant information, site specific criteria and combining chapters, appendices and tables. The additions to this revision has the site evaluate two new ways dose to the public can be increased. These evaluations do not affect any current surveillance requirement already in place at Byron Station.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because:

    1. The activity did not change dose limits to the public or design objectives,

    2. the due dates for reports to the NRC remain the same,

    3. no surveillance requirements will be changed, and

    4. the location of the RETS program will remain in Chapter 12 of the ODCM.

6G-98-0119
On-site Review - OSR 97-157

DESCRIPTION:

The activity is placing the Source Range High Flux at Shutdown (HFS) alarm in the "Block" position during a refueling outage. The reason for the activity is excessive source range noise.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because blocking the alarm during refueling outages (Mode 6) does not affect analyzed accidents. The HFS function is to alarm during an inadvertent dilution event and the potential return to criticality. During Mode 6 there is little potential for either an inadvertent dilution or a return to criticality.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because procedural instructions within the OSR adequately address ensuring the alarm is functional when required (Modes 3-5). The only failure introduced by this change is the possibility of the switch remaining in the blocked position when required.

3. The margin of safety, as defined in the bases for any Technical Specification is not reduced because the activity does not change Technical Specification Bases parameters.

6H-99-0010
PCCIR 6-99-005

DESCRIPTION:

The purpose of this Safety Evaluation was to change the setpoint in the Plant Computer associated with the Unit-1 AR011/12 Rad Monitor. This provides input into the SPDS circular display, to give the Unit Operator a quick status of the unit at a glance. The original 50.59 (6G-97-0255) was used to justify the procedure used to calculate the updated setpoint.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because adjusting the setpoint at which these monitors initiate an actuation does not affect on how the equipment will isolate the containment purge pathway. These monitors will still initiate a containment isolation signal upon alarming. No system, structure or component has been changed by this activity.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because decreasing the setpoint at which these monitors initiate an actuation does not affect on how the equipment will isolate the containment purge pathway. These monitors will still initiate a containment isolation signal upon alarming. No system, structure or component has been changed by this activity.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because these monitors are still set to actuate at 10 mR/hr above background. This is the criteria set in Technical Specification Table 3.3.6-1, note (b).

6H-99-0093
PCCIR 6-99-011

DESCRIPTION:

The purpose of this Safety Evaluation was to change 2RT-ARJ011/12 setpoints to rising containment radiation levels. Per T.S. 3.3.6, the radiation trip setpoints must be within 10 mR/hr but must be far enough away to preclude ESF activation upon normal containment releases. Change setpoints to 34/38 mR/hr for alert and trip setpoints.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the change in setpoints is within the range of valves specified in Technical Specification The change does not alter the function of the instrumentation.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the setpoint change will not increase the possibility of an accident since the reaction of the radiation monitor is not the cause of an accident, but a response to an existing condition. The change does not alter any accident causing equipment.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the margin of safety is not affected since the equipment is providing the same function and the relationship to the background radiation is being adjusted to account for changing activities.

6H-99-0102
PCCIR 6-99-015

DESCRIPTION:

The purpose of this Safety Evaluation was to perform change 1RT-AR011/012 setpoints due to rising Containment radiation levels. Per TS 3.3.6, the rad trip setpoints must be within 10 mR/hr. but must be far enough away to preclude ESF actuation upon normal Containment Releases. Change setpoints to 40/44 mR/hr for the alert and trip setpoints.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the change in setpoints is within the range of values specified in the Technical Specifications, and the change does not alter the function of the instrumentation.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report Is not created because the setpoint change will not increase the possibility of an accident since the reaction of the rad monitor is not the cause of an accident, but a response to an existing condition. This change does not alter an accident causing equipment.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the margin of safety is not affected since the equipment is providing the same function and the relationship to the background radiation is being adjusted to account for change in conditions.

6H-99-0122
PCCIR 6-99-022

DESCRIPTION:

The purpose of this Safety Evaluation was to decrease the 1/2AR011/12 monitor setpoints. The alarm is to be set at 10 mR/hr above background.

SAFETY EVALUATION SUMMARY:

1 The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because decreasing the setpoint at which this monitor initiates an actuation does not affect on how the equipment would isolate the containment purge pathway. These monitors will still initiate a containment isolation signal upon alarming. No system, structure or component has been changed by this activity.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because decreasing the setpoint at which this monitor initiates an actuation does not affect on how the equipment would isolate the containment purge pathway. These monitors will still initiate a containment isolation signal upon alarming. No system, structure or component has been changed by this activity.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because these monitors are still set to actuate at 10 mR/hr above background. This is the criteria set in Technical Specification Table 3.3.6-1, note (b).

6H-99-0171
PCCIR 6-99-024

DESCRIPTION:

The purpose of this Safety Evaluation was to change 1RT-AR011/12 setpoints due to anticipated increased radiation levels because of Unit 1 start-up. Per Technical Specifications 3.3.6, setpoints must be within 10 mR/hr. Change setpoints to 1.33 mR/hr for alert and 37 mR/hr for high.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the change in setpoints is within the range of valves specified in the Technical Specifications. The change does not alter the function of the instrumentation.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the setpoint change will not increase the possibility of accident since the response of the radiation monitor is not the cause of an accident. The response is to an existing condition. This change dos not alter any system, structure or component.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the margin of safety is not affected since the equipment is providing the same function. The setpoints are being adjusted to account for anticipated background dose rate changes.

6H-99-0154 THRU 6H-99-0164

2BISR 3.1.7-001	Rev. 2
2BISR 3.1.7-002	Rev. 2
2BISR 3.1.7-003	Rev. 2
2BISR 3.1.7-004	Rev. 2
2BOA ELEC-1	Rev. 54C
2BOA PRI-5	Rev. 57C
2BOA PRI-6	Rev. 56B
2BOA SEC-4	Rev. 52B
BAR 2-18-E16	Rev. 5
BOP RY-5	Rev. 8
BAP 560-1	Rev. 17

DESCRIPTION:

The purpose of this Safety Evaluation was to revise station procedure as required to reflect Byron Unit 2 increase in operating RCS Average Temperature (Tave) from 581° F to 583° F during Cycle 8. These validations are written against 50.59 Safety Evaluation 6G-99-0055, which evaluated the 2° F Tave increase. By increasing the RCS Tave, it is estimated that an additional 4 MWe can be obtained from Unit 2 Cycle 8 due to improved cycle efficiency.

SAFETY EVALUATION SUMMARY:

1. This safety evaluation was performed for the Byron Unit 2 Cycle 8A (BYC8A) mid-cycle reload design. The reload design was necessitated by the increase in nominal full power operating temperature (Tref) from 581° F to 583° F.

   The associated instrument loops requiring re-scaling are:

   1. SSCR 99-001, Tave/Delta I Loop 2A (2RC-0411)
   2. SSCR 99-002, Tave/Delta T Loop 2B (2RC-0421)
   3. SSCR 99-003, Tave/Delta T Loop 2C (2RC-0431)
   4. SSCR 99-004, Tave/Delta T Loop 2D (2RC-0441)
   5. SSCR 99-005, Tref Converter (2RC-0412)
   6. SSCR 99-006, Pressurizer Level Program, Auctioneered High Tave Deviation Alarm, and Turbine Loading Stop (C-16) Setpoint (2RC-0412-8)
   7. SSCR 99-007, Steam Dump Control (2EW-0507)

6H-99-0154 THRU 6H-99-0164
Various Procedures
(Cont'd.)

Implementing procedures include:

1. SPP 99-019
2. BY2CBA descendent Byron Curve Book (BCS) figures and tables
3. Reload Design Key Parameter Checklist
4. BAP 560-1
5. BAR 2-18-F16
6. 2SISR 3.1.7-001/2/3/4
7. 2BOA ELEC-1
8. 2BOA PRI-5
9. 2BOA PRI-6
10. 2BOA SEC-4
11. BOP RY-5

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because each of the Analysis of Record above assumed an operating window for T-average between 569.1° F AND 588.4° F. The T-average increase from 581° F to 583° F is still within this operating window and remains bounded by the Analysis of Record. The OTΔT and OTΔT Protection System Loops will be scaled, considering the increase in T-average, to maintain the nominal trip setpoint with the appropriate temperature penalties. The Reactor Coolant System will be rescaled to provide its function of maintaining the coolant average temperature at its programmed value (583° F). The programmed value will be changed, but the overall function of the system will continue as designed. The Steam Dump Control System will be rescaled to maintain the steam dumps design function of providing sufficient steam relief following a lead rejection of up to 50% power in order to avoid a reactor trip. Furthermore, the unit trip controller loop will be rescaled to maintain its function of bringing the unit to no-load conditions without a significant average temperature undershoot. With respect to cored design, an evaluation was performed to verify the Cycle 8 Design continues to meet key safety parameter limits (Reference 4). The analyses are valid up to a T-average of 588.4° F. Since these changes do not induce or exacerbate any failures on any equipment, the change does not involve an increase in the probability of these accidents, which were previously described in the Safety Analysis Report. The T-average increase from 581° F to 583° F is still within this operating window and remains bounded by the Analyses of Record. Based on being maintained in the analyzed range, the T-average change will not increase the consequences of any of these events.

6H-99-0154 THRU 6H-99-0164
Various Procedures
(Cont'd.)

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the change falls within a previously analyzed operating band. From a core design standpoint, the key safety parameter limits are met thus ensuring all pertinent licensing basis acceptance criteria are met. The OTΔT, OTΔT Reactor Control and Steam Dump Control Loops will be re-scaled to maintain their current protection and control functions, respectively. No new performance requirements are being imposed on any system or component such that any design criteria will be affected. No new failure modes or limiting single failures have been created by the change.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change (Tave increase to 583° F) still meets the acceptance criteria for each Technical Specification (and TRM) LCO. The core design analysis, including the change in T-average, verified the core will operate within safety analysis limits, therefore, subsequent analyses and evaluations.

6H-99-0375
SAAD 1999-0181

DESCRIPTION:

The purpose of this Safety Evaluation was to review SAAD 1999-0181 – to change the PPC rod control values to be consistent with the Byron Unit 2 Cycle 9 Design requirements/COLR. Specific values pertain to the new rod park position (228→231), steps of overlap (113→115), and steps of separation (115→116).

This evaluation validated the BY2C9 Safety Evaluation 6G-99-0127, which was performed for the Unit 2 Cycle 9 (BY2C9) Core Reload Design. That evaluation encompasses the composite effects of the following changes:

1. Fuel loading pattern.
2. Fuel mechanical design changes:

    a. 3-tab inner-to-outer strap joint for Inconel top and bottom non-missing grids.
    b. Lengthen Guide Tubes and Instrument tubes by 0.2 inches.
    c. Lengthen fuel rod by 0.32 inches.
    d. Shorten endplug by 0.12 inches (previously used design).
    e. Change from P+ to Zirc-4 spring pack with new variable pitch.
    f. Center WABA rods with active fuel region.
    g. Raise bottom Inconel grid 0.7 inches.
    h. Lower first Zircaloy mid-grid 0.3 inches.
    i. IFBA loading of 1.25X with 100 psig backfill pressure.

3. Relocate Secondary Sources into C-8 and N-8.
4. RCCA park position of 231 steps. Change RIL to correspond.
5. Use of 20 WABAs for a second cycle.
6. Elimination of Thimble Plugs for F/As with no other insert.

6H-99-0375
SAAD 1999-0181
(Cont'd.)

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because fuel construction meets all design criteria. The Core Fuel Loading Pattern and changes in operating characteristics do not produce any mechanisms by which any of the evaluated accidents, such as LOCAs or HELBs, etc., can be initiated. The consequences of previously evaluated accidents are not increased because the reload design process confirmed all design parameters satisfy the accident analysis limits and assumptions as documented in the UFSAR or other appropriate evaluations. The analysis included mechanical, nuclear, thermo-hydraulic and transient analyses, which concluded that all core parameter criteria, such as DNB, PCT, and fuel temperature, were met. In addition, the analyses showed that all system performance criteria, such as containment pressure and no water through pressurizer safeties, were met.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because this core reloads' fuel mechanical features introduce no new failure modes. The reload key parameters and assumptions meet all stands and criteria. The core operates within pertinent design basis operating limits. Therefore, the cycle specific changes in these parameters introduce no new failure modes.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the core reload design process safety analysis was performed in accordance with NRC approved methodologies and indicates that BY2C9 operates within acceptable limits and margin is maintained.

6G-98-0198
SSCR 98-029

DESCRIPTION:

The purpose of this Safety Evaluation was to revise the setpoints for the Unit 1 Steam Jet Air Ejector (1PR027J) to account for a changed mixture of noble gases in the Unit 1 Reactor Coolant System following the eighth Unit 1 refueling outage. 1PR027J initiates an actuation of the bypass valves and the off-gas filter unit. The monitor setpoints detect leak rates of the same magnitude, but due to the variance in noble gas concentrations, setpoints need to be adjusted for monitor response. The setpoint changes are a change to a procedure described in the SAR.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the function of the monitor remains unchanged. The ability of the monitor to correctly and conservatively actuate the bypass valves and the off-gas filter unit is effectively unchanged.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because changing the setpoint for PR027J does not affect other SSCs. The monitor has been calibrated (based on an updated RCS noble gas ratio) to detect a 50 gpd leak rate which is 1/3 of the leak rate limit of 150 gpd in the Technical Specifications.

3. The margin of safety, as defined in the bases for any Technical Specification is not reduced because the activity does not change Technical Specification Bases parameters. Technical Specification 4.4.6.2 described the bases for "Operational Leakage". This Technical Specification Bases is unaffected by the setpoint change.

6H-98-0174
BAR RM11-1-2PR27J, Rev. 3
SSCR 98-032

DESCRIPTION:

The purpose of the activities is to make changes to the setpoints for the Unit 2 Steam Jet Air Ejector (SJAE) Alert and High Alarm being controlled by SSCR 98-032.

Following a refueling outage, the concentration and mixture of noble gases in the Reactor Coolant System change. This change requires an adjustment to the 2RE-PR027J (i.e., U-2 SJAE radiation monitor) setpoints which are calibrated to Xe-133, based on a specific noble gas ratio. The monitor setpoints are calibrated to detect leak rates of the same magnitude (i.e., 20 gpd for the Alert Alarm, 50 gpd for the High Alarm), but due to variance in the noble gas concentrations, the setpoints require adjustment.

Furthermore, the setpoint changes are being reflected in the associated Byron Annuciator Response (BAR) and the PCCIR.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the calibration of radiation monitor setpoint is not an accident precursor and therefore has no impact on accident probabilities. The setpoint change does not impact how any SSCs function, nor cause a change in the interaction of any SSCs. The calibration of the radiation monitor setpoint is required to adjust for the variation in the isotopic mixture of noble gases as they currently exist in the RCS, with the net result being no change to the defined alarmed leak rate setpoints. The High Alarm Setpoint remains calibrated to detect a 50 gpd leak rate, one-third of the 150 gpd leak rate limit defined by Technical Specifications. With no net change to the leak rate alarmed setpoints, the consequences of a malfunction of equipment important to safety is not changed.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the calibration of a radiation monitor setpoint based on a updated RCS noble gas ratio such that the resulting setpoint more accurately represents the existing RCS leak rate does not introduce a new accident or malfunction of a different type than previously analyzed in the SAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change calibrates the SJAE radiation monitor to more accurately reflect the current RCS leak rate based on the existing RCS noble gas ratio. This calibration restores the margin of safety which Technical Specifications are based.

6H-99-0311
SSCR 99-030

DESCRIPTION:

Revise Unit 2 Reactor Coolant System Flow Transmitters in order for all instrument channels to indicate 100% flow. The RCS flow indication currently reads slightly less than 100% due to small, incremental reductions in RCS flow caused by Steam Generator tube plugging since initial startup. This scaling change will result in all 12 RCS flow indications to be returned to 100% flow.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because rescaling the RCS flow transmitters has no effect on the probability of an accident since these transmitters cannot initiate the accident. The transmitters are being rescaled and the 90% flow trip setpoints verified to ensure Technical Specification and UFSAR requirements are met. Therefore, the consequences of the accident will remain bounded by the safety analysis.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the RCS flow transmitters can accommodate the rescaling needed for all 12 RCS flow indicators to read 100%. No plant equipment is physically being changed out, only the dP corresponding to 100% flowrate is being revised. Therefore the possibility of an accident or malfunction of a different type than previously evaluated is not created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the Technical Specification required RCS Low Flow Trip Setpoint of 90% of loop minimum measured flowrate (92,850 gpm) remains bounded.

6G-99-0089
Technical Evaluation - BOM 1998-3667-001

DESCRIPTION:

The purpose of this Safety Evaluation was to address the nuclear safety concerns, associated with the Alternate Replacement Part Evaluation (BOM-1998-3667-001), due to Fire Protection Gauge replacements throughout the plant. This safety evaluation evaluated the effect on the Fire Protection Report's (FPR) Fire Hazard Analysis (FHA) for substituting Acrylonitrile Butadiene Styrene (ABS) cased gauges for the originally installed Brass cased gauges (Viking model PI 590). The original Viking supplied pressure gauges are no longer supplied with a brass case. Manufacturing design enhancements have been incorporated into the Underwriters Laboratory listed replacement gauges. This evaluation just reinforced the degree of Byron evaluation to ensure that the increased fire loading would have an insignificant effect on nuclear safety.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the non-safety related component is installed significantly far enough away from safe shutdown equipment. The probability of occurrence is not increased since the increase in fire loading if all the gauges were placed in one zone is less than a transient fire load and the gauges provide no additional ignition source. The consequences of an accident are not increased since the gauges are installed on water filled fire protection piping located within the envelope or adjacent to the existing sprinkler patterns. The probability or consequences of equipment malfunctioning important to safety is not impacted since the Auxiliary and Turbine Building gauges are not located in the proximity of equipment important to safety and the gauges increased the combustible loading in their respective zones by less than 0.1%.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the replacement gauge is identical in form, fit, and function to the original gauge. The replacement gauges are manufactured to the stringent Industry standards in order to be Underwriters Laboratory listed and Factory Mutual approved.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the replacement gauges do not affect any parameters upon which Technical Specifications are based. The Fire Protection's Systems major components are contained in the Technical Requirements Manual (TRM), however the replacement gauges do not affect the function or margin of safety associated with these major components.

6G-99-0015
Technical Specification Bases 3.6.3

DESCRIPTION:

The purpose of this Safety Evaluation was to revise two Technical Specification (TS) Bases Tables developed during the conversion to the Improved Technical Specification Specifically, these two tables were added to the Bases to document the actions required for Containment Isolation Valves (CIVs). TS Bases Table B 3.6.3-1 identifies the valves considered to be CIVs, lists the isolation times, and directs the user to an action table (i.e., TS Bases Table B 3.6.3-2) for determining the appropriate response if a CIV is determined to be inoperable. The change is being made in Table 3.6.3-2 to Action 6 for the mini-purge/post LOCA outboard CIVs (i.e., VQ003, VQ005B, and VQ005C.) Action number 6c is being changed to directly apply to the post LOCA purge system to address the current action requirements for the system. Action 6d is being added to apply to the Mini-Purge System arrangement of a single inboard and dual outboard Isolation Valves to address the current requirements by specifying the need for both outboard Isolation Valves to be inoperable to enter the more restrictive action requirement. For the actions concerning the mini-purge outlet piping and CIV arrangement, the Bases action sections for A.1 and B.1 have been enhanced to address the unique valve arrangement and isolation requirements.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because neither the probability of occurrence nor the consequences of an accident or malfunction of equipment important to safety are impacted by the change. The isolation signals and valve actuation continue to respond exactly the same for the Containment Mini-Purge System and Post LOCA Purge Systems. The changes to the required actions for the Mini-Purge System and Post LOCA Purge Systems address the unique valve configuration of these systems and do not affect either the probability or consequences of an accident or malfunction.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the changes do not impact plant operations. The isolation signals and CIV actuation will continue to respond as previously designed and tested. The changes only impact the operator's response following the declaration of a CIV to be inoperable based on the unique valve configuration of the system. This response will replicate the operator's current response to a inoperable CIV in a two valve series penetration. Therefore, the possibility of a different accident or transient being created does not exist.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the changes do not affect any parameters upon which Technical Specifications are based.

6G-99-0016
Technical Specification Amendment 106

DESCRIPTION:

The purpose of this Safety Evaluation was to delete the Administrative Technical Requirements (ATR) Manual. The deletion of the ATR Manual is necessary because the contents have been relocated to the Technical Requirements Manual (TRM). The TRM is an element of Improved Technical Specifications (ITS), which is the designated repository for all the Limiting Conditions for Operations (LCOs) and associated requirements that do not satisfy or fall within any of the four criteria codified in 10CFR50.36. The TRM is incorporated in the UFSAR by reference, which makes it a licensee-controlled document. Licensee controlled documents can only be revised in accordance with the provisions of 10CFR50.59, which ensures records are maintained. Control will be maintained over future changes.

The deletion of the ATR Manual is necessary to eliminate redundant documents with the same requirements. All items are relocated to the TRM. No change of intent of the requirements is needed. The TRM will be an owner-controlled document, as was the ATR. When ITS is implemented, the ATR is no longer needed. All items are relocated to the TRM. This process was completed during the ITS conversion. All relocations are changes because the format was modified to replicate the ITS format.

The ATR contains requirements that are station commitments. A formal tracking system will maintain the removed requirements from Current Technical Specifications for tracking purposes.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because this change will have no affect on plant operations in any mode because it does not change any function of any SSC. This change will not affect operations because the content of the ATR will remain unchanged during the transfer. All changes will continue to be evaluated by the 10CFR50.59 rule.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because no new failures can occur. With the contents of the ATR remaining unchanged, no different equipment failures are possible. The original assumptions for failures remain valid and will continue to be reviewed by the 10CFR50.59 rule for future changes.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this administrative change merely relocates the requirements. The TRM replaces the ATR in its entirety. Only format changes were made. The margin of safety is unchanged.

6H-99-150, Rev. 1
Westinghouse SECL 99-035

DESCRIPTION:

The purpose of this Safety Evaluation was to review and accept Westinghouse supplied 50.59 for the contingency of loose parts originating from a Loop Stop Isolation Valve (LSIV) in the cold leg piping and to address several identified loose parts. This evaluation and plant specific analyses provide the engineering basis for evaluating the potential effects of the assumed loose parts to determine if continued operation represents an unreviewed safety question.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because operation at Byron Station with potential or identified loose parts will not adversely affect the structural integrity of the Reactor Vessel, Reactor Vessel internals, pressurizer, RCPs, LSIVs or RCS piping. Evaluations have been performed which demonstrate that the mass of the potential or identified loose parts is not sufficient to result in impact loads which could have a detrimental effect on the integrity of these systems. The potential or identified loose parts will not have any adverse effects on the DNB related accidents since any potential flow blockage at the bottom nozzle or the bottom Inconel grid caused by the presence of the loose parts will not significantly affect flow in regions where DNB is a concern. The consequence of a previously analyzed event are dependent on the initial conditions assumed for the analysis and the availability and the successful functions of the equipment assumed to operate in response to the analyzed event. The response of the plant safety systems, when subjected to accident conditions will not be affected to prevent the mitigation of accidents previously evaluated in the UFSAR.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the change does not alter assumptions made in the safety analysis. The change does not have a detrimental impact on the manner in which plant equipment operates or responds to an actuation signal. The potential or identified loose parts do not create the possibility of an equipment malfunction different than any evaluated in the UFSAR.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the margin of safety is determined by the design and qualification of plant equipment, the operation of the plant within analyzed limits, and the point at which protective or mitigative actions are initiated The margin of safety is provided, in part, by the safety factors included in the Asme Code and by the conservatisms inherent in the accident analysis acceptance criteria. The margin of safety is not reduced, since operation with potential loose parts does not impact these factors.

6H-98-0116
WR 980005470

DESCRIPTION:

The purpose of this Safety Evaluation was to review the temporary installation of the Westinghouse Oxide Thickness measurement System components and rigs on top of the spent fuel racks at Byron Station and performance of other Nuclear Fuel Characterization tests and measurements. There were specific restrictions on the installation of the rigs, which were determined via calculation. In addition, the activity allowed for a determination of Nuclear Fuel Characterization (i.e. oxide measurements, Fuel Assembly growth data, and crud measurements utilizing procedures provided by Westinghouse). This information was obtained to determine Byron's susceptibility to nuclear industry fuel performance issues.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the Fuel Handling Accident Inside Spent Fuel Storage Facility was evaluated for impact. The equipment was installed per specific requirements provided by Engineering via Nuclear Design Information Transmittal BRW-DIT-98-0118. The other measurement activities were evaluated versus their potential impact on the plant safety analysis and determined to have no impact. This included the impact on the Spent Fuel Pool inventory, spent fuel exhaust inlets, fuel handling building ventilation and the HEPA and charcoal filters. The activity did not result in an increase in probability or consequences in the Safety Analysis Report.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the activity did not alter any operational or postulated accident loadings. The utilization of the analyzed procedures did not create any new accidents or malfunctions than already analyzed. The potential dropping of loads was analyzed and determined acceptable given the administrative controls that were implemented.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because this activity did not adversely impact any Technical Specification system. Therefore, plant operation remained within the basis of plant Technical Specifications.

6G-99-0031
TRM Change 99-04

DESCRIPTION:

The purpose of this Safety Evaluation was to revise the Technical Requirement Manual (TRM), Section 3.10 "Fire Protection", and to implement the recommendations of the following Fire Protection System review. The Fire Protection (FP) Systems and features installed at Byron Station were reviewed against the Administrative Requirements of Byron TRM Section 3.10, 10CFR50 Appendix R, and NRC Branch Technical Position 9.5-1 Appendix A. The purpose of the changes to the TRM is to clearly specify plant FP equipment subject to the conditions of the TRM Section, eliminate compensatory measures that are not needed, and to specify compensatory measures that are appropriate for the FP function governed by the TRM.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the changes to the TRM do not change the occupancy or increase the fire hazards in any plant Fire Zone and does not degrade any fire protection detection or suppression system protecting safety-related equipment. The changes to the TRM do not physically diminish the Fire Protection Systems provided to ensure post fire safe shutdown capability and the changes to the compensatory measures provide reasonable assurance of post fire safe shutdown when the systems are inoperable. Since the Fire Protection Systems have maintained their capability to protect safe shutdown and safety-related equipment and safe shutdown equipment is not otherwise affected by the changes, the conclusions of the safe shutdown analysis of FPR Section 2.4 are unchanged. Therefore the probability and consequences of a Design Basis Fire are not increased.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because the changes to the TRM affect administrative controls for FP Systems when components should become inoperable. They do not physically add Fire Protection Systems to any areas of the plant or make any changes to how these systems operate. When individual components of the FP Systems become inoperable, compensatory actions are implemented. The changes have not instituted a compensatory action not previously implemented by the current TRM. The compensatory action do not re-align equipment or create conditions not previously considered in the fire hazard analysis.

6G-99-0031
TRM Change 99-04
(Cont'd.)

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because there is no margin of safety associated with the Administrative Technical Specification requirement for procedures to implementation the approved Fire Protection Program. The changes do not alter the implementation of the approved Fire Protection Program as described in TS 5.4.1. However, some changes in the details of procedures will be made to reflect components added or deleted or other approved changes to the TLCO required actions. These changes do not affect any margin associated with the approved Fire Protection Program.

6G-99-0057
TRM 3.7.b

DESCRIPTION:

1. The purpose of this Safety Evaluation was to revise TRM Section 3.7.b to allow on-line testing of snubbers. The change, more specifically, allows the removal of one snubber from a piping system on a voluntary basis for the purpose of testing and reinstalling. A 72-hour timeclock is applicable. The allowance is limited to one snubber unless further justification is provided to allow removal of multiple snubbers.

2. TLCO 3.7.b, Note A is revised to perform an evaluation for a single snubber removal only when a required snubber is identified as inoperable during a visual examination or when a required snubber fails to meet the functional test acceptance criteria.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the absence or removal of a snubber is not the initiator for any accident or transient. When the required snubber is removed, the snubber will be returned to operable status within 72 hours per TRM actions. With a snubber removed, a system will maintain its ability to perform as required during any accident or transient condition within a given analytical boundary (Ref: ComEd Procedure NES-MS-03.2).

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because a required snubber must be returned to OPERABLE status within 72 hours, which is consistent with the pre-change TRM requirements. Any accident possible with a removed snubber is also possible with an installed snubber (no new initiators or assumptions are created). No new equipment is being introduced and installed equipment is not being operated in a different manner.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because snubbers are not addressed in the Tech. Specs., but in the TRM. The TRM provides the required Actions associated with an inoperable snubber. Only when a required snubber cannot be restored to OPERABLE within 72 hours, are the Technical Specifications entered. In this case the TRM provides direction to declare the associated system inoperable and follow the required Actions for that system. The completion time of Condition A remains unchanged (72 hours). In addition, direction to enter the system LCO (Technical Specifications) remains consistent with the current TRM requirements (if a required snubber has not been returned to OPERABLE status within 72 hours). Therefore, the margin of safety as described in the basis for Technical Specifications is not impacted by the changes to the TRM.

6G-99-0111
TRM Appendix 0

DESCRIPTION:

The purpose of this Safety Evaluation was to alter the current station philosophy with respect to Limiting Condition for Operations (LCO) 3.0.6. When the station implemented the Improved Technical Specifications (ITS), the station philosophy regarding T.S. LCO 3.0.6 was prescribed in TRM Appendix 0 - SFDP and did not take complete advantage of the allowances made by the TS. For example, when a SUPPORTED SYSTEM LCO is not met solely due to a SUPPORT SYSTEM LCO not being met, the station took the more conservative position that the SUPPORTED SYSTEM LCO would still be entered, but entry would be delayed by the completion time allowed for the SUPPORT SYSTEM. This change would preclude any entry into SUPPORTED SYSTEM LCO, provided no loss of safety function exists. This change provides the maximum flexibility to the station allowed by the T.S. LCO 3.0.6.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because no accidents are applicable, therefore the consequences of an accident nor the probability of occurrence of an accident will not increase. This change is to the Station's Methodology for implementing the allowances of Technical Specification LCO 3.0.6 as amended by Amendment 106.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because this change is to the Station's Methodology for implementing the allowances to Technical Specification T.S. LCO 3.0.6 as amended by Amendment 106. Changing the methodology will not create the possibility of an accident or transient.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the change implements T.S. LCO 3.0.6 exactly as written and approved by the NRC in Amendment 106.

6G-99-0122, Rev. 1
TRM 3.1.k, Rev. 3

DESCRIPTION:

The purpose of this Safety Evaluation was to support Revision 3 to TRM 3.1.k. This revision adds a second provision that may be opted to take exception to 3.1.g. The provision is sufficient RCS Boron Concentration to maintain keff ≤ 0.987. If this provision is opted, and the RCS Boron Concentration does not meet the requirements, the required actions are to either immediately initiate boration to restore the RCS Boron Concentration or to immediately open the Reactor Trip Breakers. In addition, if the new provision is opted, the new surveillance requirement would be verification of sufficient RCS Boron Concentration every 2 hours. This change can be summarized by the option of a less restrictive condition (more than one bank withdrawn at a time with DRPI inoperable) offset by a much more restrictive condition (a much higher Boron Concentration and the associated much larger shutdown margin).

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence, consequences of an accident, or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased. This change does not increase the probability of a malfunction of equipment because this change does not alter the operation of any system. This change does not increase the probability of any accident because this change provides a more net restrictive plant condition. All related previously evaluated incidents either assume reactor criticality to be an initial condition or a result of the initiating event. Criticality will be precluded under this change by the restrictions on Boron Concentration.

   Should one of these previously evaluated incidents occur, the consequences ill not be increased because this change precludes criticality and it does not impact any of the systems designed to mitigate the consequences of these incidents.

2. The possibility for an accident or malfunction of a different type other than any evaluated previously in the Safety Analysis Report is not created. This change will result in reactor, which is always subcritical, regardless of RCCA position, due to the more restrictive limitation on RCS Boron Concentration. The less restrictive change, the ability to render DRPI inoperable with more than one RCCA bank withdraw at a time does not fundamentally alter the operation of the plant such that a new, unevaluated, accident may occur.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because the TRM change does not affect any parameters upon which Technical Specifications are based.

6G-99-0133
Technical Specification Bases 3.9.4

DESCRIPTION:

The purpose of the Technical Specification Bases Change was to revise the Bases to Technical Specification LCO 3.9.4, Station Procedures BGP 100-6 and BAP 370-3 to ensure consistency with the Fuel Handling Accident Analyses assumptions in the UFSAR and Technical Requirements Manual (TRM) Section 3.9.a. The bases of Technical Specification 3.9.4 contains a statement regarding TRM Section 3.9.a which may be interpreted to prohibit the performance of CORE ALTERATIONS prior to the reactor being subcritical for 100 hours. This statement is inconsistent with TRM Section 3.9.a and fuel handling accident analyses, which prohibits only the movement of irradiated Fuel Assemblies in the Reactor Vessel prior to being subcritical for 100 hours. The purpose of TRM 3.9.a requirements of 100 hours of subcriticality is to limit the consequences of a "Fuel Handling Accident", not of a core reactivity accident. CORE ALTERATIONS are defined to include only those activities, which can affect core reactivity. Therefore, activities which do not impact fuel handling accidents can be performed prior to 100 hours of subcriticality. Therefore, the changes were made to ensure consistency between all documents and bases. In addition, the bases for Spent Fuel Pool cooling remained unchanged since TRM 3.9.a continues to prohibit movement of fuel from the Reactor Vessel to the Spent Fuel Pool until 100 hours of subcriticality are achieved.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or a malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because the likelihood of an operator error is not increased. An operator has to be manipulating a Fuel Assembly for the event to happen. The number of Fuel Assembly manipulations for any given set of operational objectives (e.g., core offload) is not changed. The equipment the operator is using is not changed. The interlocks on the equipment are not changed. The same skills of the operator will be used to avoid/prevent errors. Therefore, since all activities related to fuel handling operations remain unchanged, the probability of any transient or accident occurrence is not increased.

   To increase the consequences of the accident (i.e., increase of the off-site dose), at least one of three things must happen: 1) increase in the source term for the radioactive release; 2) degradation of the radiation detection system; or 3) degradation of the purge system isolation (FB ventilation for SFP). Clearly, the radiation detection and ventilation systems are not impacted by this change. The potential impact on the source term is addressed below.

6G-99-0133
Technical Specification Bases 3.9.4
(Cont'd.)

The analyses performed in accordance with Reg. Guide 1.25 assume a minimum of 100 hours between the time of plant shutdown and the accident. The change will allow CORE ALTERATIONS (including control rod unlatching) to take place prior being shutdown for 100 hours, however, movement of irradiated Fuel Assemblies within the Reactor Vessel will continue to be prohibited prior to being subcritical for a minimum of 100 hours in accordance with the assumptions of the Fuel Handling Accident Analyses. Since the movement of irradiated fuel is the initiating event in the accidents described in 15.7.4.2.2, and the assumptions listed in the accidents analyzed remain unchanged, there is no increase in the consequences of the accidents analyzed in the UFSAR.

Finally, since this change will not result in additional fuel failures or impede the detection or mitigation systems affecting offsite dose calculations, the UFSAR analysis remains valid and bounding for this change.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because there are no credible accidents of a different type that can be created as a result of the change. The requirement for being subcritical for a minimum of 100 hours is related to the movement of irradiated Fuel Assemblies, and the current requirement remains unchanged. Therefore, all assumptions for the accidents analyzed remain unchanged. All requirements specified in Technical Specifications and the TRM governing fuel movement continue to be met, all equipment used to handle fuel and related components remain unchanged, and all procedures used in fuel handling remain unchanged. For these reasons, no new failure modes are created.

3. The margin of safety, as defined in the basis for any Technical Specification, is not reduced because all requirements specified in Technical Specifications and the TRM governing fuel movement continue to be met, all equipment used to handle fuel and related components remain unchanged, and all procedures used in fuel handling remain unchanged. For these reasons, no new failure modes are created.

6H-99-0021
Transfer of Technical Specifications Items to TRM

DESCRIPTION:

The proposed activity is a review of changes made to the Improved Technical Specification (ITS) Technical Requirements Manual (TRM). The changes involve the transfer of items originally designated to be included in the TRM that are now designated to be made a part of other licensee controlled documents. The affected portions of the ITS TRM for this change are identical for Byron and Braidwood. The ITS TRM for both stations was approved by the SER dated 12/22/98.

SAFETY EVALUATION SUMMARY:

1. The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report is not increased because no analyzed accidents apply to the change.

2. The possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report is not created because no technical changes (either actual or interpretational) were made in transferring the relocated items to other licensee controlled documents.

3. The margin of safety, as defined in the bases for any Technical Specification is not reduced because the change does not affect any Technical Specification Bases.