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SUBJECT: Forwards marked-up list w/status of implementation of TMI action plan items at plant,per 890414 request.

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NIAGARA MOHAWK

NIAGARA MOHAWK POWER CORPORATION/301 PLAINFIELD ROAD, SYRACUSE, N.Y. 13212/TELEPHONE (315) 474-1511

April 18, 1989
NMP2L 1195

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Re: Nine Mile Point Unit 2
Docket No. 50-410
NPF-69

Gentlemen:

Your letter dated April 14, 1989 requested us to review the status of implementation of TMI Action Plan Items at Nine Mile Point Unit 2. Enclosed is a marked-up list with the status of each of the TMI Action Plan Items you requested. It should be noted that in some cases Niagara Mohawk may have taken exception to or provided an alternate means of complying with the requirements stated in NUREG 0737 and NUREG 0737, Supplement 1. Therefore, the bases for determining that an item is complete is as described in Section 1.10 of the Nine Mile Point Unit 2 Final Safety Analysis Report. The information contained herein is correct to the best of my knowledge.

Very truly yours,

NIAGARA MOHAWK POWER CORPORATION

L. Burkhardt, III
Executive Vice President
Nuclear Operations

PEF/mjd
7140G

xc: Regional Administrator, Region I
Mr. R. A. Capra, Director
Ms. M. M. Slosson, Project Manager
Mr. W. A. Cook, Resident Inspector
Records Management

A003
1/1

NINE MILE POINT UNIT 2
DOCKET NO. 50-410
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ISSUE NUMBER	MULTI-PLANT ACTION NO.	ISSUE TITLE	LICENSEE IMPLEMENTATION STATUS **
I.A.1.1.1		SHIFT TECHNICAL ADVISOR - ON DUTY	C
*I.A.1.1.2		SHIFT TECHNICAL ADVISOR - TECH SPECS	C
*I.A.1.1.3	F001	SHIFT TECHNICAL ADVISOR - TRAINED PER LL CAT B	C
*I.A.1.1.4		SHIFT TECHNICAL ADVISOR - DESCRIBE LONG TERM PROGRAM	C
I.A.1.2		SHIFT SUPERVISOR RESPONSIBILITIES	C
I.A.1.3.1	F002	SHIFT MANNING - LIMIT OVERTIMES	C
I.A.1.3.2	F002	SHIFT MANNING - MIN SHIFT CREW	C
I.A.2.1.1		IMMEDIATE UPGRADING OF RO & SRO TRAINING AND QUAL. - SRO EXPER	C
I.A.2.1.2		IMMEDIATE UPGRADING OF RO & SRO TRAINING AND QUAL. - SRO'S BE RO'S 1 YR.	C
I.A.2.1.3		IMMEDIATE UPGRADING OF RO & SRO TRAINING AND QUAL. - 3 MO. TRAINING	C
I.A.2.1.4	F003	IMMEDIATE UPGRADING OF RO & SRO TRAINING AND QUAL. - MODIFY TRAINING	C
I.A.2.1.5		IMMEDIATE UPGRADING OF RO & SRO TRAINING AND QUAL. - FACILITY CERTIF.	C
I.A.2.3		ADMINISTRATON OF TRAINING PROGRAMS	C
I.A.3.1.1		REVISE SCOPE & CRITERIA FOR LICENSING EXAMS - INCREASE SCOPE	C
I.A.3.1.2		REVISE SCOPE & CRITERIA FOR LICENSING EXAMS - INCREASE PASSING GRADE	C
I.A.3.1.3.A		REVISE SCOPE & CRITERIA FOR LIC. EXAMS - SIMULATOR PLANTS WITH SIMULATORS	C
I.A.3.1.3.B		REVISE SCOPE & CRITERIA FOR LICENSING EXAMS - SIMULATOR - OTHER PLANTS	C
*I.B.1.2		EVALUATION OF ORGANIZATION & MANAGEMENT (ISEG HAS BEEN IMPLEMENTED)	C
I.C.1.1		SHORT-TERM ACCIDENT & PROCEDURES REVIEW - SB LOCA	C
I.C.1.2.A	F004	SHORT-TERM ACCIDENT & PROC. REVIEW - INADEQ. CORE COOL. REANAL. GUIDE.	C
I.C.1.2.B	F004	SHORT-TERM ACCIDENT & PROC. REVIEW - INADEQ. CORE COOL. REVISE PROC.	C
I.C.1.3.A	F005	SHORT-TERM ACCIDENT & PROC. REVIEW - TRANS. & ACCDTS. REANAL GUIDE. (PROC. GEN. PKG.)	C
I.C.1.3.B	F005	SHORT-TERM ACCIDENT & PROC. REVIEW - TRANSIENTS & ACCDTS. REVISE PROC. (UPGRADED EOP'S)	C
I.C.2		SHIFT & RELIEF TURNOVER PROCEDURES	C
I.C.3		SHIFT-SUPERVISOR RESPONSIBILITY	C
I.C.4		CONTROL-ROOM ACCESS	C

*Indicate differences between operating reactor and NTOL requirements.

** Implementation Status Code

C Completed
N/C No Change Required
N/A Not Applicable

7138G

NINE MILE POINT UNIT 2
DOCKET NO. 50-410
NPF-69

ISSUE NUMBER	MULTI-PLANT ACTION NO.	ISSUE TITLE	LICENSEE IMPLEMENTATION STATUS
I.C.5	F006	FEEDBACK OF OPERATING EXPERIENCE	C
I.C.6	F007	VERIFY CORRECT PERFORMANCE OF OPERATING ACTIVITIES	C
*I.C.7.1		NSSS VENDOR REV. OF PROC - LOW POWER TEST PROGRAM	C
*I.C.7.2		NSSS VENDOR REV. OF PROC - POWER ASCENSION & EMER. PROCS.	C
*I.C.8		PILOT MON OF SELECTED EMERGENCY PROC FOR NTOLS.	C
I.D.1	(See F008 & F071)	CONTROL-ROOM DESIGN REVIEWS (ENTER DATA FOR MPA F008 & MPA F-071)	C
I.D.2.1	F009	PLANT-SAFETY PARAMETER DISPLAY CONSOLE - DESCRIPTION	C
I.D.2.2	F009	PLANT-SAFETY PARAMETER DISPLAY CONSOLE - INSTALLED	C
I.D.2.3	F009	PLANT-SAFETY PAR. DISPLAY CONSOLE - FULLY IMPLE.	Prior to Startup Following 1st Refuel Outage - Software Changes and Test
*I.G.1.1		TRAINING DURING LOW-POWER TESTING - PROPOSE TESTS	C
*I.G.1.2		TRAINING DURING LOW-POWER TESTING - SUBMIT ANAL. & PROCS.	C
*I.G.1.3		TRAINING DURING LOW-POWER TESTING - TRAINING & RESULTS	C
II.B.1.1		REACTOR-COOLANT SYSTEM VENTS - DESIGN VENTS	N/C
II.B.1.2	F010	REACTOR-COOLANT SYSTEM VENTS - INSTALL VENTS (LL CAT B)	N/C
II.B.1.3	F010	REACTOR-COOLANT SYSTEM VENTS - PROCEDURES	N/C
II.B.2.1		PLANT SHIELDING - REVIEW DESIGNS	C
*II.B.2.2		PLANT SHIELDING - CORRECTIVE ACTIONS TO ASSURE ACCESS	C
*II.B.2.3	F011	PLANT SHIELDING - PLANT MODIFICATONS (LL CAT B)	C
*II.B.2.4		PLANT SHIELDING - EQUIPMENT QUALIFICATION - NOT TRACKED AS A TMI ACTION ITEM	C
*II.B.3.1		POSTACCIDENT SAMPLING - INTERIM SYSTEM	N/A
*II.B.3.2		POSTACCIDENT SAMPLING - CORRECTIVE ACTIONS	N/A
*II.B.3.3		POSTACCIDENT SAMPLING - PROCEDURES	C
*II.B.3.4	F012	POSTACCIDENT SAMPLING - PLANT MODIFICATIONS (LL CAT B)	C
II.B.4.1	F013	TRAINING FOR MITIGATING CORE DAMAGE - DEVELOP TRAINING PROGRAM	C
*II.B.4.2.A	F013	TRAINING FOR MITIGATING CORE DAMAGE - INITIAL	C
*II.B.4.2.B	F013	TRAINING FOR MITIGATING CORE DAMAGE - COMPLETE	C
II.D.1.1		RELIEF & SAFETY VALVE TEST REQUIREMENTS - SUBMIT PROGRAM	C
*II.D.1.2.A		RELIEF & SAFETY VALVE TEST REQUIREMENTS - COMPLETE TESTING	C
*II.D.1.2.B	F014	RELIEF & SAFETY VALVE TEST REQUIREMENTS - PLANT SPECIFIC REPORT	C
II.D.1.3		RELIEF & SAFETY VALVE TEST REQUIREMENTS - BLOCK-VALVE TESTING	N/A
*II.D.3.1		VALVE POSITION INDICATION - INSTALL DIRECT INDICATIONS OF VALVE POS	C
*II.D.3.2		VALVE POSITION INDICATION - TECH SPECS	C

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ISSUE NUMBER	MULTI-PLANT ACTION NO.	ISSUE TITLE	LICENSEE IMPLEMENTATION STATUS
*II.E.1.1.1 (SEE NOTE 1)	F015	AFS EVALUATION - ANALYSIS	N/A
*II.E.1.1.2 (SEE NOTE 2)	F015	AFS EVALUATION - SHORT TERM MODS	N/A
*II.E.1.1.3 (SEE NOTE 3)	F015	AFS - LONG TERM MODS	N/A
II.E.1.2.1.A		AFS INITIATION & FLOW - CONTROL GRADE	N/A
II.E.1.2.1.B	F016	AFS INITIATION & FLOW - SAFETY GRADE	N/A
II.E.1.2.2.A	F016	AFS INITIATION & FLOW - FLOW INDICATION CONTROL GRADE	N/A
*II.E.1.2.2.B		AFS INITIATION & FLOW - LL CAT A TECH SPECS	N/A
*II.E.1.2.2.C	F017	AFS INITIATION & FLOW - SAFETY GRADE	N/A
*II.E.3.1.1		EMERGENCY POWER FOR PRESSURIZER HEATERS - UPGRADE POWER SUPPLY	N/A
*II.E.3.1.2		EMERGENCY POWER FOR PRESSURIZER HEATERS - TECH SPECS	N/A
II.E.4.1.1		DEDICATED HYDROGEN PENETRATIONS - DESIGN	N/C
*II.E.4.1.2		DEDICATED HYDROGEN PENETRATIONS - REVIEW & REVISE H2 CONTROL PROC	N/C
*II.E.4.1.3	F018	DEDICATED HYDROGEN PENETRATION - INSTALL	N/C
II.E.4.2.1-4		CONTAINMENT ISOLATION DEPENDABILITY - IMP. DIVERSE ISOLATION	C
*II.E.4.2.5.A		CONTAINMENT ISOLATION DEPENDABILITY - CNTMT PRESS. SETPT. SPECIFY PRESS.	C
*II.E.4.2.5.B		CONTAINMENT ISOLATION DEPENDABILITY - CNTMT PRESS. SETPT. MODS	C
II.E.4.2.6	F019	CONTAINMENT ISOLATION DEPENDABILITY - CNTMT PURGE VALVES	C
II.E.4.2.7	F019	CONTAINMENT ISOLATION DEPENDABILITY - RADIATION SIGNAL ON PURGE VALVES	C
*II.E.4.2.8		CONTAINMENT ISOLATION DEPENDABILITY - TECH SPECS	C
*II.F.1.1	F020	ACCIDENT - MONITORING - PROCEDURES	C
*II.F.1.2.A	F020	ACCIDENT - MONITORING - NOBLE GAS MONITOR	C
*II.F.1.2.B	F021	ACCIDENT - MONITORING - IODINE/PARTICULATE SAMPLING	C
*II.F.1.2.C	F022	ACCIDENT - MONITORING - CONTAINMENT HIGH-RANGE MONITOR	C
*II.F.1.2.D	F023	ACCIDENT - MONITORING - CONTAINMENT PRESSURE	C
*II.F.1.2.E	F024	ACCIDENT - MONITORING - CONTAINMENT WATER LEVEL	C

NOTE 1 - THE ITEM LISTED IS FROM NUREG-0737, ENCLOSURE 2 AND IS APPLICABLE TO NTOL'S ONLY.
NOTE 2 - THE ITEM LISTED IS FOR ALL PLANTS (OPERATING REACTORS AND NTOL'S)
NOTE 3 - THE ITEM LISTED IS FOR ALL PLANTS (OPERATING REACTORS AND NTOL'S)

ISSUE NUMBER	MULTI-PLANT ACTION NO.	ISSUE TITLE	LICENSEE IMPLEMENTATION STATUS
*II.F.1.2.F	F025	ACCIDENT - MONITORING - CONTAINMENT HYDROGEN	C
*II.F.2.1		INSTRUMENTATION FOR DETECT. OF INADEQUATE CORE COOLING - PROCEDURES	C
*II.F.2.2		INSTRUMENTATION FOR DETECT. OF INADEQUATE CORE COOLING - SUBCOOL METER	N/A
*II.F.2.3	F026	INSTRUMENTATION FOR DETECT. OF INADEQUATE CORE COOLING - DESC. OTHER	N/C
*II.F.2.4	F026	INSTRU. FOR DETECT. OF INADEQUATE CORE COOLING - INSTALL ADD'L INSTRU.	N/C
*II.G.1.1		POWER SUPP. FOR PRESSURIZER RELIEF, BLOCK VALVES & LEVEL IND. - UPGRADE	N/A
*II.G.1.2		POWER SUPP. FOR PRESSURIZER RELIEF, BLOCK VALVES & LEVEL IND. - TECH SP	N/A
II.K.1 (Oper. Reactors Only)		IE BULLETINS - 79-05, 79-06, 79-08	C
*II.K.1.5		IE BULLETINS - REVIEW ESF VALVES	C
*II.K.1.10		IE BULLETINS - OPERABILITY STATUS	C
*II.K.1.20		IE BULLETINS - PROMPT MANUAL REACTOR TRIP	N/A
*II.K.1.21		IE BULLETINS - AUTO SG ANTICIPATORY REACTOR TRIP	N/A
*II.K.1.22		IE BULLETINS - AUX. HEAT REM SYSTEM, PROC	C
*II.K.1.23		IE BULLETINS - RV LEVEL, PROCEDURES	C
*II.K.2.2		ORDERS ON B&W PLANTS - PROCEDURES TO CONTROL AFW IND OF ICS	N/A
*II.K.2.8		ORDERS ON B&W PLANTS - UPGRADE AFW SYSTEM	N/A
II.K.2.9	F027	ORDERS ON B&W PLANTS - FEMA ON ICS	N/A
II.K.2.10	F028	ORDERS ON B&W PLANTS - SAFETY-GRADE TRIP	N/A
II.K.2.11	F029	ORDERS ON B&W PLANTS - OPERATOR TRAINING	N/A
II.K.2.13	F030	ORDERS ON B&W PLANTS - THERMAL MECHANICAL REPORT (CE & W PLANTS ALSO)	N/A
II.K.2.14	F031	ORDERS ON B&W PLANTS - LIFT FREQUENCY OF PORV'S & SV'S	N/A
II.K.2.15		ORDERS ON B&W PLANTS - EFFECTS OF SLUG FLOW	N/A
II.K.2.16	F032	ORDERS ON B&W PLANTS - RCP SEAL DAMAGE	N/A
II.K.2.17	F033	ORDERS ON B&W PLANTS - VOIDING IN RCS (CE & W PLANTS ALSO)	N/A
II.K.2.19		BENCHMARK ANALYSIS OF SEQUENTIAL AFW FLOW TO ONCE THROUGH STM GEN.	N/A
*II.K.2.20	F035	ORDERS ON B&W PLANTS - SYSTEM RESPONSE TO SB LOCA	N/A
*II.K.3.1.A	F036	B&O TASK FORCE - AUTOMATIC PORV ISOLATION DESIGN	N/A
*II.K.3.1.B		FINAL RECOMMENDATIONS, B&O TASK FORCE - AUTO PORV ISO TEST/INSTALL	N/A
II.K.3.2	F037	B&O TASK FORCE - REPORT ON PORV FAILURES	N/A
II.K.3.3	F038	B&O TASK FORCE - REPORTING SV & RV FAILURES AND CHALLENGES	C

ISSUE NUMBER	MULTI-PLANT ACTION NO.	ISSUE TITLE	LICENSEE IMPLEMENTATION STATUS
II.K.3.5.A	F039	B&O TASK FORCE - AUTO TRIP OF RCP'S PROPOSED MODIFICATIONS	N/A
II.K.3.5.B	F039	B&O TASK FORCE - AUTO TRIP OF RCP'S MODIFICATIONS	N/A
II.K.3.7		B&O TASK FORCE - EVALUATION OF PORV OPENING PROBABILITIES	N/A
II.K.3.9	F040	B&O TASK FORCE - PID CONTROLLER MODIFICATION	N/A
II.K.3.10	F041	B&O TASK FORCE - PROPOSED ANTICIPATORY TRIP MODIFICATIONS	N/A
II.K.3.11		B&O TASK FORCE - JUSTIFY USE OF CERTAIN PORV	N/A
II.K.3.12.A		B&O TASK FORCE - ANTICIPATORY TRIP ON TURBINE TRIP PROPOSED MODS	N/A
II.K.3.12.B	F042	B&O TASK FORCE - ANTICIPATORY TRIP ON TURBINE TRIP INSTALL MODS	N/A
II.K.3.13.A	F043	B&O TASK FORCE - HPCI & RCIC SYSTEM INITIATION LEVELS ANALYSIS	C
II.K.3.13.B	F043	B&O TASK FORCE - HPCI & RCIC INITIATION LEVELS MODIFICATION	C
*II.K.3.14	F044	B&O TASK FORCE - ISO CONDENSER ISOLATION ON HIGH RAD	N/A
II.K.3.15	F045	B&O TASK FORCE - MODIFY HPCI & RCIC BRK DETECTION CIRCUITRY	C
II.K.3.16.A	F046	B&O TASK FORCE - CHALLENGE & FAILURE OF RELIEF VALVES STUDY	C
II.K.3.16.B	F046	B&O TASK FORCE - CHALLENGE & FAILURE OF RELIEF VALVES MODIFICATIONS	N/C
II.K.3.17	F047	B&O TASK FORCE - ECC SYSTEM OUTAGES	C
II.K.3.18.A	F048	B&O TASK FORCE - ADS ACTUATION STUDY	C
II.K.3.18.B	F048	B&O TASK FORCE - ADS ACTUATION PROPOSED MODIFICATIONS	C
II.K.3.18.C	F048	B&O TASK FORCE - ADS ACTUATION MODIFICATIONS	C
*II.K.3.19	F049	B&O TASK FORCE - INTERLOCK RECIRCULATORY PUMP MODIFICATIONS	N/A
*II.K.3.20		B&O TASK FORCE - LOSS OF SVC WATER AT BRP	N/A
II.K.3.21.A	F050	B&O TASK FORCE - RESTART OF CSS & LPCI LOGIC DESIGN	C
II.K.3.21.B	F050	B&O TASK FORCE - RESTART OF CSS & LPCI LOGIC DESIGN MODIFICATIONS	C
II.K.3.22.A	F051	B&O TASK FORCE - RCIC SUCTION VERIFICATION PROCEDURES	C
II.K.3.22.B	F051	B&O TASK FORCE - RCIC SUCTION MODIFICATION	C
II.K.3.24	F052	B&O TASK FORCE - SPACE COOLING FOR HPCI/RCI LOSS OF AC POWER	C
II.K.3.25.A		B&O TASK FORCE - POWER ON PUMP SEALS PROPOSED MODIFICATIONS	N/C
II.K.3.25.B	F053	B&O TASK FORCE - POWER ON PUMP SEALS MODIFICATIONS	N/A
II.K.3.27	F054	B&O TASK FORCE - COMMON REFERENCE LEVEL FOR BWRS	N/C
II.K.3.28	F055	B&O TASK FORCE - QUALIFICATION OF ADS ACCUMULATORS	C
*II.K.3.29	F056	B&O TASK FORCE - PERFORMANCE OF ISOLATION CONDENSERS	N/A
II.K.3.30.A		B&O TASK FORCE - SCHEDULE FOR OUTLINE OF SB LOCA MODEL	C

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ISSUE NUMBER	MULTI-PLANT ACTION NO.	ISSUE TITLE	LICENSEE IMPLEMENTATION STATUS
II.K.3.30.B	F057	B&O TASK FORCE - SB LOCA MODEL, JUSTIFICATION	C
II.K.3.30.C		B&O TASK FORCE - SB LOCA METHODS NEW ANALYSES	C
II.K.3.31	F058	B&O TASK FORCE - COMPLIANCE WITH CFR 50.46	C
*II.K.3.40		B&O TASK FORCE - RCP SEAL DAMAGE - COVERED BY II.K.2.16 AND II.K.3.25	N/C
*II.K.3.43		B&O TASK FORCE - EFFECTS OF SLUG FLOW - COVERED BY II.K.2.15	N/A
II.K.3.44	F059	B&O TASK FORCE - EVALUATE TRANSIENT WITH SINGLE FAILURE	C
II.K.3.45	F060	B&O TASK FORCE - ANALYSES TO SUPPORT	C
II.K.3.46	F061	RESPONSE TO LIST OF CONCERNS FROM ACRS CONSULTANT	C
*II.K.3.57	F062	IDENTIFY WATER SOURCES PRIOR TO MANUAL ACTIVATION OF ADS	C
III.A.1.1		EMERGENCY PREPAREDNESS, SHORT TERM	C
III.A.1.2.1		UPGRADE EMERGENCY SUPPORT FACILITIES - INTERIM TSC OSC & EOF	C
III.A.1.2.2 (SEE F063, F064, F065)		UPGRADE EMERGENCY SUPPORT FACILITIES - DESIGN - INCORP. INTO F063/F064/F065	C
III.A.1.2.3 (SEE F063, F064, F065)		UPGRADE EMERGENCY SUPPORT FACILITIES - MODS INCORPOR. INTO F063, F064 & F065	C
III.A.2.1	F067	UPGRADE PREPAREDNESS - UPGRADE EMERGENCY PLANS :TO APP. E. 10 CFR 50	C
III.A.2.2	F068	UPGRADE PREPAREDNESS - METEOROLOGICAL DATA	C
*III.D.1.1.1		PRIMARY COOLANT OUTSIDE CONTAINMENT - LEAK REDUCTION	C
*III.D.1.1.2		PRIMARY COOLANT OUTSIDE CONTAINMENT - TECH SPECS	C
III.D.3.3.1		INPLANT RAD. MONIT. - PROVIDE MEANS TO DETER. PRESENCE OF RADIOIODINE	C
III.D.3.3.2	F069	INPLANT RAD. MONIT. - MODIFICATIONS TO ACCURATELY MEAS. IODINE	N/A
III.D.3.4.1	F070	CONTROL ROOM HABITABILITY - REVIEW	C
*III.D.3.4.2	F070	CONTROL ROOM HABITABILITY - SCHEDULE MODIFICATIONS	C
*III.D.3.4.3		CONTROL ROOM HABITABILITY - IMPLEMENT MODIFICATIONS	N/A
MPA-F008	F008	I.D.1.1 DETAILED CONTROL ROOM DESIGN REVIEW PROGRAM PLAN	C
MPA-F063	F063	III.A.1.2 TECHNICAL SUPPORT CENTER	C
MPA-F064	F064	III.A.1.2 OPERATIONAL SUPPORT CENTER	C
MPA-F065	F065	III.A.1.2 EMERGENCY OPERATIONS FACILITY	C
MPA-F071	F071	I.D.1.2 DETAILED CONTROL ROOM REVIEW (FOLLOWUP TO F-8)	Prior to Startup from 1st Refuel Outage-Hardware
MPA-B072	B072	NUREG-0737 TECH SPECS (GENERIC LETTERS 82-16 & 83-02)	C
MPA-B083	B083	TECH SPEC COVERED BY GENERIC LETTERS 83-26 & 83-37 FOR NUREG-0737	C
67.4.1	G001	REACTOR COOLANT PUMP TRIP (GENERIC LETTER 85-12)	N/A

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April 13, 1989

DOCKET NO(S). 50-410

Mr. Lawrence Burkhardt III
Executive Vice President Nuclear Operation
Niagara Mohawk Power Corporation
301 Plainfield Road
Syracuse, New York 13212

SUBJECT: NIAGARA MOHAWK POWER CORPORATION
NINE MILE POINT NUCLEAR STATION UNIT 2

The following documents concerning our review of the subject facility are transmitted for your information.

- [ ] Notice of Receipt of Application, dated ____________.
- [ ] Draft/Final Environmental Statement, dated __________.
- [ ] Notice of Availability of Draft/Final Environmental Statement, dated ___________.
- [ ] Safety Evaluation Report, or Supplement No. ____________________ dated __________.
- [ ] Environmental Assessment and Finding of No Significant Impact, dated ___________.
- [ ] Notice of Consideration of Issuance of Facility Operating License or Amendment to Facility Operating License, dated __________.
- [x] Bi-Weekly Notice; Applications and Amendments to Operating Licenses Involving No Significant Hazards Considerations, dated April 5, 1989 [xxxxxxxxxx] May 5, 1989.
- [ ] Exemption, dated __________.
- [ ] Construction Permit No. CPPR-___________, Amendment No. ____________ dated __________.
- [ ] Facility Operating License No. ___________, Amendment No. ___________ dated _________.
- [ ] Order Extending Construction Completion Date, dated __________.
- [ ] Monthly Operating Report for ________________ transmitted by letter dated __________.
- [ ] Annual/Semi-Annual Report- ____ ________________________ transmitted by letter dated __________.

Office of Nuclear Reactor Regulation.

Enclosures:
As stated

cc: See next page

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DATE	4/13/89						

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cc:

Mr. Troy B. Conner, Jr., Esq.
Conner & Wetterhahn
Suite 1050
1747 Pennsylvania Avenue, N.W.
Washington, D.C. 20006

Mr. Richard Goldsmith
Syracuse University
College of Law
E. I. White Hall Campus
Syracuse, New York 12223

Resident Inspector
Nine Mile Point Nuclear Power Station
P. O. Box 99
Lycoming, New York 13093

Mr. Gary D. Wilson, Esquire
Niagara Mohawk Power Corporation
300 Erie Boulevard West
Syracuse, New York 13202

Mr. Peter E. Francisco, Licensing
Niagara Mohawk Power Corporation
301 Plainfield Road
Syracuse, New York 13212

Ms. Donna Ross
New York State Energy Office
2 Empire State Plaza
16th Floor
Albany, New York 12223

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, Pennsylvania 19406

Charlie Donaldson, Esquire
Assistant Attorney General
New York Department of Law
120 Braodway
New York, New York 10271

Mr. Richard M. Kessel
Chair and Executive Director
State Consumer Protection Board
99 Washington Avenue
Albany, New York 12210

Mr. Richard Abbott, Unit 2 Station
Superintendent
Nine Mile Point Nuclear Station
Niagara Mohawk Power Corporation
P. O. Box 32
Lycoming, NY 13093

Mr. James L. Willis, General Supt.,
Nuclear Generation
Nine Mile Point Nuclear Station
Niagara Mohawk Power Corporation
P. O. Box 32
Lycoming, New York 13093

NUCLEAR REGULATORY COMMISSION

[Docket No. 50-313]

Arkansas Power & Light Co.; Arkansas Nuclear One, Unit 1; Denial of Amendment to Facility Operating License and Opportunity for Hearing

The U.S. Nuclear Regulatory Commission (the Commission) has denied, in part, a request by the Arkansas Power & Light Company (AP&L or the licensee) for an amendment to Facility Operating License No. DPR-51 issued to the Arkansas Nuclear One, Unit 1 (ANO-1), located in Pope County, Arkansas.

During an AP&L review of the High Pressure Injection (HPI) system, it was discovered by the licensee that a postulated break of an HPI injection line, just upstream of the reactor coolant system cold leg connection and downstream of the first check valve, could constitute a small break loss of coolant accident (LOCA) not currently enveloped by the approved 10 CFR 50.46 and Appendix K analyses. Subsequent Babcock and Wilcox (B&W) analysis determined that the ANO-1 HPI system might not be able to provide adequate core cooling should the break occur at high power operation. B&W then further determined that the HPI system would provide adequate core cooling at up to 74% of full power based on a best estimate analysis.

The purpose of the licensee's amendment application dated March 23, 1989 was to temporarily reduce the authorized power level for ANO-1 to the 74% level for a period of time until an acceptable hardware modification could be achieved.

The staff reviewed the licensee's request for amendment and supporting analysis provided by B&W and determined that the amendment was partially acceptable. Specifically, the staff approved the licensee's request to operate at a lower power level, but only would approve 50% of full power based on the nature of the B&W analysis. The power level above 50% was rejected pending the submittal of a full, Appendix K, LOCA analysis for the postulated accident. Further the licensee had requested in the suggested wording of the proposed amendment that it would return to an authorized 100% power level upon approval and implementation of a permanent modification to address the problem of the unanalyzed postulated break. This change was rejected and instead a maximum duration of 50 equivalent full power days for continuation of operation was authorized by the NRC in the amendment. This was necessary because of limitations of existing fuel related analyses related to the reload methodology. Further, future modifications to the authorized power level will require separate routine amendment applications. Notice of Issuance of that amendment will be published in the Commission's biweekly Federal Register notice.

The licensee was notified of the Commission's partial denial of the proposed amendment by a letter transmitting Amendment No. 119.

By May 5, 1989, the licensee may demand a hearing with respect to the denial described above. Any person whose interest may be affected by this proceeding may file a written petition for leave to intervene.

A request for hearing or petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Service Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street NW., Washington, DC, by the above date.

A copy of any petitions should be also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to Mr. Nicholas S. Reynolds Bishop, Cook Purcell & Reynolds, 1400 L Street NW., Washington, DC 20005-3502, attorney for the licensee.

For further details with respect to this action, see (1) the application for amendment dated March 23, 1989, and (2) the Commission's letter to the licensee dated March 29, 1989, and (3) the Commission's Safety Evaluation dated March 29, 1989, issued with Amendment No. 119 to DPR-51.

These documents are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room located at Tomlinson Library, Arkansas Tech University, Russellville, Arkansas 72801. A copy of item (2) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Document Control Desk.

Dated at Rockville, Maryland, this 29th day of March, 1989.

For the Nuclear Regulatory Commission

Jose A. Calvo,

Director,

Project Directorate—IV, Division of Reactor Projects—III, IV, V and Special Projects, Office of Nuclear Reactor Regulation.

[FR Doc. 89-8046 Filed 4-4-89; 8:45 am]

BILLING CODE 7590-01-M

[Dockets Nos. 50-275 and 50-323]

Pacific Gas & Electric Co.; Issuance of Amendments to Facility Operating Licenses

The United States Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 34 and 33 to Facility Operating Licenses Nos. DPR-80 and DPR-82, issued to the Pacific Gas and Electric Company (the licensee), which revised the Technical Specifications (TS) for operation of the Diablo Canyon Nuclear Power Plant, Units Nos. 1 and 2 (DCNPP), located in San Luis Obispo County, California. The amendments are effective as of the date of issuance.

The amendments changed the DCNPP Combined Technical Specifications by revising TS 2.2.1, "Reactor Trip System Instrumentation Setpoints," Table 2.2-1, "Reactor Trip System Instrumentation Trip Setpoints," Items 13 and 14 to reduce the steam generator water level low and low-low setpoints from 15 to 7.2 percent of the narrow range span. Also, the associated TS bases were changed.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations. The Commission has made appropriate findings, as required by the Act and the Commission's regulations in 10 CFR Chapter I, which are set forth in the license amendments.

Notice of Consideration of Issuance of Amendments to Facility Operating Licenses and Opportunity for Hearing in connection with this action was published in the Federal Register on June 23, 1988 at 53 FR 23708. No request for hearing or petition to intervene was filed following this notice.

Also in connection with this action, the Commission prepared an Environmental Assessment and Finding of No Significant Impact which was published in the Federal Register on March 23, 1989 at 54 FR 12032.

For further details with respect to this action, see (1) the application for amendments dated April 18, 1988, (2) Amendments Nos. 34 and 33 to Licenses

Nos. DPR-80 and DPR-82, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room 2120 L Street NW., Washington, DC 20555, and at the California Polytechnic State University Library, Government Documents and Maps Department, San Luis Obispo, California 93407. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Director, Division of Reactor Projects—III, IV, V and Special Projects.

Dated at Rockville, Maryland, this 27th day of March, 1989.

For the Nuclear Regulatory Commission.

Harry Rood,

Senior Project Manager, Project Directorate V, Division of Reactor Projects—III, IV, V and Special Projects.

[FR Doc. 89-8047 Filed 4-4-89; 8:45 am]

BILLING CODE 7590-01-M

Biweekly Notice; Applications and Amendments to Operating Licenses Involving No Significant Hazards Considerations

I. Background

Pursuant to Public Law (Pub. L.) 97-415, the Nuclear Regulatory Commission (the Commission) is publishing this regular biweekly notice. Pub. L. 97-415 revised section 189 of the Atomic Energy Act of 1954, as amended (the Act), to require the Commission to publish notice of any amendments issued, or proposed to be issued, under a new provision of section 189 of the Act. This provision grants the Commission the authority to issue and make immediately effective any amendment to an operating license upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

The biweekly notice includes all notices of amendments issued, or proposed to be issued from March 13, 1989 through March 24, 1989. The last biweekly notice was published on March 22, 1989 (54 FR 10831).

Notice of Consideration of Issuance of Amendment to Facility Operating License and Proposed no Significant Hazards Consideration Determination and Opportunity for Hearing

The Commission has made a proposed determination that the following amendment requests involve no significant hazards consideration. Under the Commission's regulations in 10 CFR 50.92, this means that operation of the facility in accordance with the proposed amendments would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The basis for this proposed determination for each amendment request is shown below.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination. The Commission will not normally make a final determination unless it receives a request for a hearing.

Written comments may be submitted by mail to the Regulatory Publications Branch, Division of Freedom of Information and Publications Services, Office of Administration and Resources Management, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and should cite the publication date and page number of this Federal Register notice. Written comments may also be delivered to Room P-216, Phillips Building, 7920 Norfolk Avenue, Bethesda, Maryland from 7:30 a.m. to 4:15 p.m. Copies of written comments received may be examined at the NRC Public Document Room, the Gelman Building, 2120 L Street NW., Washington, DC. The filing of requests for hearing and petitions for leave to intervene is discussed below.

By May 5, 1989, the licensee may file a request for a hearing with respect to issuance of the amendment to the subject facility operating license and any person whose interest may be affected by this proceeding and who wishes to participate as a party in the proceeding must file a written petition for leave to intervene. Requests for a hearing and petitions for leave to intervene shall be filed in accordance with the Commission's "Rules of Practice for Domestic Licensing Procedures" in 10 CFR Part 2. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or an Atomic Safety and Licensing Board, designated by the Commission or by the Chairman of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition and the Secretary or the designated Atomic Safety and Licensing Board will issue a notice of hearing or an appropriate order.

As required by 10 CFR 2.714, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following factors: (1) The nature of the petitioner's right under the Act to be made a party to the proceeding; (2) the nature and extent of the petitioner's property, financial, or other interest in the proceeding; and (3) the possible effect of any order which may be entered in the proceeding on the petitioner's interest. The petition should also identify the specific aspect(s) of the subject matter of the proceeding as to which petitioner wishes to intervene. Any person who has filed a petition for leave to intervene or who has been admitted as a party may amend the petition without requesting leave of the Board up to fifteen (15) days prior to the first prehearing conference scheduled in the proceeding, but such an amended petition must satisfy the specificity requirements described above.

Not later than fifteen (15) days prior to the first prehearing conference scheduled in the proceeding, a petitioner shall file a supplement to the petition to intervene which must include a list of the contentions which are sought to be litigated in the matter, and the bases for each contention set forth with reasonable specificity. Contentions shall be limited to matters within the scope of the amendment under consideration. A petitioner who fails to file such a supplement which satisfies these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing, including the opportunity to present evidence and cross-examine witnesses.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held.

If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for hearing. Any hearing held would take place after issuance of the amendment.

If the final determination is that the amendment involves a significant hazards consideration, any hearing held would take place before the issuance of any amendment.

Normally, the Commission will not issue the amendment until the expiration of the 30-day notice period. However, should circumstances change during the notice period such that failure to act in a timely way would result, for example, in derating or shutdown of the facility, the Commission may issue the license amendment before the expiration of the 30-day period, provided that its final determination is that the amendment involves no significant hazards consideration. The final determination will consider all public and State comments received before action is taken. Should the Commission take this action, it will publish a notice of issuance and provide for opportunity for a hearing after issuance. The Commission expects that the need to take this action will occur very infrequently.

A request for a hearing or a petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Service Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street NW., Washington, DC, by the above date. Where petitions are filed during the last ten (10) days of the notice period, it is requested that the petitioner promptly so inform the Commission by a toll-free telephone call to Western Union at 1-(800) 325-6000 (in Missouri 1-(800) 342-6700). The Western Union operator should be given Datagram Identification Number 3737 and the following message addressed to (*Project Director*): petitioner's name and telephone number; date petition was mailed; plant name; and publication date and page number of this Federal Register notice. A copy of the petition should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to the attorney for the licensee.

Nontimely filings of petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for hearing will not be entertained absent a determination by the Commission, the presiding officer or the presiding Atomic Safety and Licensing Board, that the petition and/or request should be granted based upon a balancing of factors specified in 10 CFR 2.714(a)(1)(i)-(v) and 2.714(d).

For further details with respect to this action, see the application for amendment which is available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street NW., Washington, DC, and at the local public document room for the particular facility involved.

Baltimore Gas and Electric Company, Docket Nos. 50-317 and 50-318, Calvert Cliffs Nuclear Plant, Unit Nos. 1 and 2, Calvert County, Maryland

Date of amendment request: November 1, 1988.

Description of amendment request: The following proposed changes to the Technical Specifications (TS) are in response to the Baltimore Gas and Electric Company (BG&E, the licensee) submittal dated November 1, 1988. The proposed changes would (1) modify the Unit 2 TS 3/4.4.9, "Pressure/Temperature Limits," by raising the minimum pressurization temperature by 20°F, for pressures between 20 and 530 psia, to a minimum temperature of 90°F; (2) change the Units 1 and 2 TS in accordance with the guidance provided in NRC Generic Letter (GL) 87-09, "Sections 3.0 and 4.0 of the Standard Technical Specifications (STS) on the Applicability of Limiting Condition for Operation and Surveillance Requirements."

Basis for proposed no significant hazards consideration determination

To correct the nonconservative value for the MPT currently provided in the Unit 2 TS Figure 3.4-2c, "Reactor Coolant System Pressure Temperature Limitations for 10 to 40 Years of Full Power Operation," of TS 3/4.4.9, "Pressure/Temperature," the licensee has requested Change No. 1 to raise the Unit 2 MPT to 90°F for reactor coolant system (RCS) pressurizer (PZR) pressures between 20 and 530 psia.

The MPT for a reactor vessel is dependent upon the value of the maximum Nil Ductility Transition Temperature (NDTT) for the vessel and its associated components. The maximum reactor vessel or vessel component NDTT at Unit 2 is +30°F for the vessel flange. The original design code for the reactor vessel, Section III of the 1965 Edition with Addenda through Winter 1967 of the ASME Code, requires the MPT to be equal to the sum of the maximum NDTT plus 60°F. Therefore, the MPT between PZR pressures of 20 and 530 psia at Unit 2 is 90°F. TS Figure 3.4-2c currently sets this MPT at the nonconservative value of 70°F.

The licensee evaluated this proposed change against the standards of 10 CFR 50.92 and has determined that the amendment would not:

(i) Involve a significant increase in probability or consequences of an accident previously evaluated.

The Change No. 1 proposed increase of the MPT for PZR pressures from 20 to 530 psia from 70°F to 90°F represents an additional restriction over the TS requirement currently in effect. Furthermore, the 90°F value to which MPT is to be changed is the minimum temperature that was permitted by the reactor vessel construction code to provide adequate brittle fracture protection to the vessel and its components. Finally, this restrictive change will not affect any other plant operations, equipment or accident analyses.

Consequently, this proposed change would not result in any increase in the probability or consequences of previously evaluated accidents.

(ii) Create the possibility of a new or different type of accident from any accident previously evaluated.

This proposed change does not alter any plant operability requirements, other than conservatively shifting the MPT by 20°F, surveillance testing, maintenance, or system design functions. Furthermore, this change reduces the likelihood of brittle fractures of the reactor vessel or of its components which are accidents that are not included in the facility's design basis events.

Thus, this change would not create the possibility of any new or different type of accident.

(iii) Involve a significant reduction in a margin of safety.

This proposal does not alter any plant operational requirements or restrictions other than raising the MPT by 20°F in order to comply with the reactor vessel construction code to ensure that the RCS is at high enough temperature before it is pressurized in order to provide brittle fracture protection to the reactor vessel and its components. Therefore, this proposed change will not involve any reduction in any margin of safety.

Finally, on March 6, 1986, the NRR published guidance in the Federal Register (51 FR 7751) concerning examples of amendments that are not likely to involve a significant hazards consideration.

This change is consistent with one of the examples provided: "(ii) A change that constitutes an additional limitation, restriction, or control not presently included in the Technical Specifications."

Due to all of the above, the NRC staff proposes to determine that the change requested for the Unit 2 TS Figure 3.4-2c involves no significant hazards consideration.

In accordance with the guidance provided in GL 87-09 the licensee has requested Change No. 2 to the Units 1 and 2 TS to address problems arising from potentially unnecessary operational restrictions that result from the current construction of TS 3/4.0, "Applicability."

Specifically, the Generic Letter suggested changing TS 3.0.4, TS 4.0.3, and TS 4.0.4 TS 3.0.4 deals with entry into an operational mode when the LCO's associated with a particular operational mode are not met. The present specification is inconsistent with operational requirements for several plant systems, prohibiting plant start-up with inoperable equipment though continued power operation would be permitted with the same equipment in an inoperable condition.

TS 3.0.4 would be revised to permit entry into an operational mode with specific equipment/system conditions for which the Action Statements of the applicable Limiting Condition for Operation (LCO) would allow continued plant operation for an unlimited period of time. Consequently, entry into an operational mode would be prohibited only when an equipment/system condition exists for which the applicable LCO Action Statement would require shutdown within a specific time interval.

TS 4.0.3 deals with the licensee's failure to perform a surveillance requirement. Failure to perform a surveillance requirement results in the failure to demonstrate that a structure, system, or component is operable. Currently, if a licensee were to miss a surveillance requirement, the construction of this TS would require plant shutdown. If the licensee were to immediately attempt the performance of the missed surveillance to prevent shutdown, the quality of plant control could be degraded. Thus, GL87-09 provides some flexibility to permit the licensee to perform the surveillance in a reasonable period of time.

TS 4.0.3 would be revised to permit up to 24 hours delay in implementing Action Statement requirements to permit completion of the missed surveillance requirement.

TS 4.0.4 deals with entry into an operational mode and satisfaction of the surveillance requirements associated with the Limiting Condition for Operation (LCO) for a given mode. The present TS 4.0.4, due to its construction, could conflict with TS 4.0.3 and possibly prevent passage through or to operational modes as required to comply with TS LCO Action Statement requirements or in the converse. TS 4.0.3 could prevent entry into a mode for which a surveillance is required to demonstrate operability when that surveillance can only be performed in that particular mode.

TS 4.0.4 would be revised to ensure that it "shall not prevent passage through or to operational modes as required to comply with Action Statement" requirements.

The changes, proposed by the licensee to the Units 1 and 2 TS 3.0.4, 4.0.3 and 4.0.4, are in accordance with the guidance provided in GL 87-09. In addition, the licensee proposed changes to the following TS to delete the non-applicability of TS 3.0.4 to these TS in order to fully implement the changes to TS 3.0.4 as recommended in GL 87-09:

TS 3.3.3.2, "Incore Detectors"
TS 3.3.3.3, "Seismic Instrumentation"
TS 3.3.3.4, "Meteorological Instrumentation"
TS 3.3.3.7, "Fire Detection Instrumentation"
TS 3.3.3.9, "Radioactive Gaseous Effluent Monitoring Instrumentation"
TS 3.3.3.10, "Radioactive Liquid Effluent Monitoring Instrumentation"
TS 3/4.7.9, "Sealed Source Contamination"
TS 3/4.7.11, "Fire Suppression Systems"
TS 3.7.11.2, "Spray and/or Sprinkler Systems"
TS 3.7.11.3, "Halon Systems"
TS 3.7.11.4, "Fire Hose Stations"
TS 3.7.11.5, "Yard Fire Hydrants and Hydrant Hose Houses"
TS 3/4.7.12, "Penetration Fire Barriers"
TS 3.11.1.1, "Liquid Effluents—Concentration"
TS 3.11.1.2, "Dose"
TS 3.11.1.3, "Liquid Radwaste Treatment System"
TS 3.11.2.1, "Gaseous Effluents—Dose Rate"
TS 3.11.2.2, "Dose-Noble Gases"
TS 3.11.2.3, "Dose-Iodine-131 and Radionuclides in Particulate Form"
TS 3.11.2.4, "Gaseous Radwaste Treatment System"
TS 3.11.2.5, "Explosive Gas Mixture"
TS 3.11.2.6, "Gas Storage Tanks"
TS 3.11.3, "Solid Radioactive Waste"
TS 3.11.4, "Total Dose"
TS 3.12.1 (Radiological Environmental Monitoring Program)
TS 3.12.2, "Land Use Census"
TS 3.12.3, "Interlaboratory Comparison Program."

The licensee evaluated these proposed changes against the standards of 10 CFR 50.92 and has determined that the amendments would not:

(i) Involve a significant increase in probability or consequency of an accident previously evaluated * * *

The change to Specification 3.0.4, allowing mode changes while in ACTION STATEMENTS that allow continued, unlimited operation, does not affect the probability or consequences of any accident previously evaluated. Since continued, unlimited operation is allowed in either of the modes involved in the mode change, the only difference is that now the mode change is allowed to happen.

The change to Specification 4.0.3, allowing 24-hours to complete missed Surveillance Requirements, does not effect the consequences of previously evaluated accidents, but it may slightly increase the probability of an accident previously evaluated by increasing the time between surveillances. However, the frequency of missed Surveillance Requirements is very low and it is overly conservative to assume that systems or components are inoperable when a Surveillance Requirement has not been performed. Also, by not shutting down the plant, accidents that might occur as a result of the transient are avoided. Therefore, overall, the change does not increase the probabilities significantly, if at all.

Clarification of Specification 4.0.4 for mode changes as a consequence of ACTION requirements does not affect the probability or consequences of previously evaluated accidents. It is not the intent of Specification 4.0.4 to prevent passage through or to operational modes to comply with ACTION requirements. The change resolves potential conflicts between Specifications 4.0.3 and 4.0.4.

Consequently, these proposed changes would not result in any increase in the probability or consequences of previously evaluated accidents.

(ii) Create the possibility of a new or different type of accident from any accident previously evaluated * * *

This change does not add or modify any plant equipment. Therefore, the only possible accidents are still those previously evaluated.

Thus, these changes would not create the possibility of any new or different type of accident.

(iii) Involve a significant reduction in a margin of safety * * *

The change to Specification 3.0.4 reduces the margin of safety in those specifications that allow for continued, unlimited operation and which did not have an exception to 3.0.4 prior to the change. However, as discussed in the NRC staff position in the generic letter, for an LCO that has ACTION requirements permitting continued operation for an unlimited period of time, entry into an operational mode should be permitted in accordance with those ACTION requirements. Therefore, these ACTION requirements provide an acceptable level of safety for continued operation, and there is not a significant reduction in the margin of safety.

Deleting the exception to Specification 3.0.4 in the specifications that allow for continued, unlimited operation does not affect any margin of safety. Prior to this change, Specification 3.0.4 did not apply, as indicated by the exception, and mode changes could be made. With this change, Specification 3.0.4 applies, but, since these specifications allow for continued, unlimited operation, mode changes can still be made.

The proposed change to Specification 4.0.3 would allow time to complete a missed surveillance test and avoid a forced power reduction. Since the majority of surveillances are completed successfully, this avoids potentially unnecessary transients and reduces the potential for plant upset and challenges to safety systems. Therefore, no reduction in a margin of safety results.

Deletion of the statement that exceptions to Specification 4.0.3 are stated in the individual specifications does not effect margin of safety since no such statements exist.

Clarification of Specification 4.0.4 for mode changes as a consequence of ACTION requirements does not affect margin of safety. As pointed out in the NRC staff position on this area in Generic Letter 87-09, it is not the intent of Specification 4.0.4 to prevent passage through or to operational modes to comply with ACTION requirements. The change resolves potential conflicts between Specifications 3.0.4 and 4.0.4.

The staff has reviewed the licensee's no significant hazards consideration determination analysis. Based upon this review, the staff believes that the licensee has met the three standards.

Based upon the above discussion, the staff proposes to determine that these proposed changes do not involve a significant hazards consideration.

Local Public Document Room location: Calvert County Library, Prince Frederick, Maryland.

Attorney for licensee: Jay E. Silbert, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street NW., Washington, DC 20037.

NRC Project Director: Robert A. Capra.

Carolina Power & Light Company, et al., Docket Nos. 50-325 and 50-324, Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina

Date of application for amendments: September 27, 1983.

Description of amendment request: The proposed amendment would change the Technical Specifications (TS) to (1) revise TS Section 3/4.3.2 to include Limiting Conditions for Operation and Surveillance Requirements to ensure the capability of the main stack monitor signal circuitry to isolate containment purge and vent valves; and (2) revise pages affected by the above TS changes and other editorial and formatting changes.

Basis for proposed no significant hazard consideration determination: The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92(c). A proposed amendment to an operating licensee involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The Carolina Power & Light Company (CP&L) has reviewed the proposed changes to TS Section 3/4.3.2 and associated revision to the affected TS pages, and has determined that the requested amendments do not involve a significant hazards consideration. The licensee's analysis is reproduced below:

(I) Change of TS Section 3/4.3.2

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated. The proposed changes do not affect the function or physical nature of any component or system associated with the probability of a Design Basis Accident (DBA) or Transient Analysis. The nonsafety-related main stack radiation monitor is in addition to safety related signals from the low reactor water level instrumentation and high containment pressure instrumentation for which there are existing Technical Specifications. Thus, the main stack monitor signal provides additional assurance that, when necessary, primary containment will be isolated. Further, this function provides additional assurance that the consequences of an accident will be mitigated such that radiological effluents released to unrestricted areas will be kept as low as is reasonably achievable.

2. The main stack radiation monitor and associated signal circuitry are nonsafety-related. The nonsafety-related circuitry is electrically isolated from the existing safety-related isolation logic circuitry. Thus, a failure of the nonsafety-related main stack monitor and/or the associated nonsafety-related circuitry will not affect the existing safety-related isolation signals and therefore, will not create the potential for a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in the margin of safety. The main stack radiation monitor signal setpoints are listed and controlled in the Brunswick Offsite Dose Calculation Manual (ODCM). Changes to this document are submitted to the Staff as part of the Semi-Annual Radioactive Effluent Release Report in accordance with BSEP TS 6.13.2. As noted in an NRC letter dated June 3, 1988, the setpoints are based on the guideline values of 10 CFR Parts 20 and 50, which are more conservative than those of 10 CFR Part 100. Based on this fact, the proposed amendment actually augments the margin of safety.

II. Revision of pages affected by changes to TS Section 3/4.3.2

1. The changes are editorial only and make no changes to the technical content or requirements of the Technical Specifications. Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The changes do not affect the function or physical nature of any component or system. Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The changes are administrative only and as such are not applicable to any safety parameter. Therefore, the proposed amendment does not involve a significant reduction in the margin of safety.

The staff has made a preliminary review of the CP&L determinations and is in agreement with them. Accordingly, the Commission proposes to determine that these changes do not involve a significant hazards consideration.

Local Public Document Room location: University of North Carolina at Wilmington, William Madison Randall Library, 601 S. College Road, Wilmington, North Carolina 28403-3298.

Attorney for licensee: R.E. Jones, General Counsel, Carolina Power & Light Company, P.O. Box 1551, Raleigh, North Carolina 27602.

NRC Acting Project Director: Edward A. Reeves.

Carolina Power & Light Company, et al., Docket Nos. 50-325 and 50-324, Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina

Date of application for amendments: February 1, 1989.

Description of amendment request: The amendments would delete references to instrument tag numbers from the technical specifications and provide other editorial and administrative revisions. Carolina Power & Light Company (CP&L) the licensee for the Brunswick Steam Electric Plant, Units 1 and 2, (BSEP) divided the changes into seventeen change categories. For ease in noticing, the staff grouped these into five board categories.

Category 1 changes would delete instrument tag number throughout the technical specifications, delete the words "Instrument Number" from column headings, replace instrument tag numbers with the words "Transmitter," "Trip Logic," "NO17 Instrument Loop," "Remaining Instruments Logic"; delete Footnote (a) from Table 3.3.5.3-1; delete footnote (b) from table 3.3.6.1-1; and combine footnote ## and ### into footnote (c) on page 3/4 3-26.

Category 2 changes would replace existing numerical and symbolic footnote notations with alphabetical notations; replace the word "Condition" with the phrase "Operational Condition," or "Action" with "Actions," or "Table Notations" with "Notes," or "Action Statements" with "Actions"; change the item notation in Table 3.3.7-1 from alphabetical to numerical, and add the title "Actions" to the top of action table associated with Table 3.3.7-1.

Category 3 changes would delete footnotes no longer necessary. Specifically, footnotes would be deleted from technical specifications dealing with a one time hydrogen injection test authorized in Amendment 131 (Unit 2 only). Footnote ** would be deleted from surveillance requirement 4.1.3.5.b, (Unit 1 only) and footnote * would be deleted from surveillance requirement 4.5.3.1.c (both units).

Category 4 changes would manipulate footnotes and tables (ie. turn the tables, add appropriate headings, double-space, put parentheses around the footnotes notations, and rearrange the footnotes into alphabetical order).

Category 5 changes would repaginate existing pages to accommodate delection of information discussed above and eliminate the current "a" pages.

Basis for proposed no significant hazard consideration determination: The Commission has provided standards for determining whether a no significant hazard consideration exists as stated in 10 CFR 50.92(c). A proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The Carolina Power & Light Company (CP&L) has reviewed the proposed changes to the technical specifications and has determined that the requested amendment does not involve a significant hazards consideration.

The licensee has provided the following no significant hazards consideration rational for the Category 1 changes.

Delete Instrument Tag Numbers:

1. The proposed changes does not involve a change in the design, operation or testing of any plant system. It deletes information not required to be included in the Technical Specifications, thereby potentially reducing both NRC and CP&L administrative effort involved in keeping the Technical Specifications updated. No new equipment will be installed, nor will any new or different operational modes be created. The instrument tag numbers will be addressed in plant procedures and updated therein, as necessary. The tag number lists will be adequately controlled through 10CFR50.59. Therefore, this change has not effect on the probability of an accident, nor does it impact the consequences of any accident previously evaluated.

2. The proposed change deletes information not required to be addressed in the Technical Specifications. It does not reflect a change to the design, operation or testing of plant equipment; it only administratively deletes the instrument tag numbers from the Technical Specifications. The instrument tag numbers will be maintained and updated in the plant procedures. Therefore, no new or different accident possibilities are created.

3. The proposed change has no effect on the design or operation of any plant system. It only deletes references to instrument tag numbers for the Technical Specifications. The instrument tag numbers are not required to be incorporated in the Technical Specifications, and it takes a great deal of effort for both the NRC and CP&L to keep the information updated. The instrument tag numbers will be handled and updated via plant procedures, thereby potentially eliminating the need for several Technical Specification amendment requests per year. Therefore, since the information will continue to be maintained, only in a different form, there is no impact on the margin of safety of the plant.

Delete the words "Instrument Number" from column headings:

1. The proposed change does not directly affect any equipment or instrumentation. It only deletes the words "Instrument Number" from the column headings of the tables currently listing instruments and their associated instrument tag numbers. The instrument tag numbers are being deleted, as described in Proposed Change Number 3. Therefore, the column headings no longer need to reference the instrument numbers. Thus, the proposed change does not change the probability of any accident previously evaluated.

2. The proposed change is administrative in nature. It deletes column headings that are no longer necessary because the referenced information is being deleted as described in Proposed Change No. 3. No equipment or instrumentation is being changed or affected. Therefore, no new or different accident possibilities are created.

3. The proposed change does not affect any instrumentation or equipment. It is administrative in nature since it is being made only to provide consistency with the information provided in the associated columns. Therefore, there is no decrease in the margin of safety.

Replace instrument tag numbers with "Transmitters," "Trip Logic," "N017 Instrument Loop," and "Remaining Instrumentation," and combine footnotes on page 3/4 3-26:

1. The proposed change does not involve a change in the design, operation or testing of any plant system. It deletes information not required to be included in the Technical Specifications, thereby potentially reducing both NRC and CP&L administrative effort involved in keeping the Technical Specifications updated. The instrument tag numbers will be addressed in plant procedures and updated therein as necessary. Therefore, this change has no effect on the probability of an accident, nor does it impact the consequences of any accident previously evaluated.

2. The proposed change deletes information not required to be addressed in the Technical Specifications. It does not reflect a change to the design, operation or testing of plant equipment; it only administratively deletes the instrument tag numbers from the Technical Specifications. The instrument tag numbers will be maintained and updated in the plant procedures. Therefore, no new or different accident possibilities are created.

3. The proposed change has no effect on the design or operation of any plant system. It only deletes references to instrument tag numbers for the Technical Specifications. The instrument tag numbers are not required to be incorporated in the Technical Specifications, and it takes a great deal of effort for both the NRC and CP&L to keep the information updated. The instrument tag numbers will be handled and updated via plant procedures, thereby potentially eliminating the need for several Technical Specification amendment requests per year. Therefore, since the information will continue to be maintained only in a different form, there is no impact on the margin of safety of the plant.

The following is a combination of two determinations (12 and 16) from the licensee.

Delete footnote (a) from Table 3.3.5.3-1 and footnote (b) from Table 3.3.6.1-1:

1. The proposed change deletes a footnote which was meant to clarify the list of tag numbers associated with Items 9 and 10 of Table 3.3.5.3-1 and Items 1 and 2 of Table 3.3.6.1-1. The tag numbers associated with these items are being deleted, as described elsewhere in this submittal. The definition of instrument functions are required to be listed in the Technical Specifications. Deletion of this footnote will not affect the operation or testing of the instrumentation; therefore, it will not change the probability of an accident, nor will it change the consequences of any accident.

2. The proposed change deletes a footnote which clarifies a list of tag numbers associated with Items 9 and 10 of Table 3.3.5.3-1 and Items 1 and 2 of Table 3.3.6.1-1. The tag numbers are being deleted from the Technical Specifications, as described elsewhere in this submittal. Deletion of this footnote will not impact the operation or testing of the instrumentation, and therefore will not create the possibility of a new or different type of accident.

3. The proposed change deletes a footnote which becomes unnecessary once the instrument tag numbers are deleted from the Technical Specifications. The tag numbers

are being deleted from the Technical Specifications, as described elsewhere in this submittal. The change is administrative since the tag numbers are not required to be listed in the Technical Specifications. The footnote provides a clarification to the list of instruments associated with Items 9 and 10 of Table 3.3.5.3-1 and Items 1 and 2 of Table 3.3.6.1-1. Thus, this footnote is no longer necessary once the tag numbers are deleted. Since the change is administrative, there is no impact on the margin of safety.

The licensee has provided the following no significant hazards consideration rationale for the Category 2 changes:

Replace existing numerical and symbolic footnote notation with alphabetical notation:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the Technical Specifications. The content of the footnotes has not changed unless specified elsewhere in this enclosure. The changes to the footnote or footnote table have been made to provide clarity and consistency to the Technical Specifications. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided elsewhere in the table and in the Technical Specifications. It does not represent a change in the content of the footnote. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency and clarity within the table and the Technical Specifications. It does not involve a change in the content of the footnotes. Therefore, there is no impact on the margin of safety.

Replace the word "condition" with the phrase "Operational Condition," or "Action" with "Actions," or "Table Notations" with "Notes" or "Action Statements" with "Actions":

1. The proposed change does not involve a change in design, operation or testing of any plant system. It is an administrative change intended to provide consistency throughout the Technical Specifications. Therefore, it has no effect on the probability of an accident, nor does it impact the consequences of any accident previously evaluated.

2. The proposed change is administrative in nature, intended only to provide consistency within the Technical Specifications. It does not change the design or operation of any plant system. Therefore, it does not create the possibility of a new of different kind of accident.

3. The proposed change does not affect system operation or design. It only provides consistency in terminology with other sections of the Technical Specifications. For this reason, it has no impact on the margin of safety of the plant.

The following is a combination of two licensee determinations (7 and 8).

Table 3.3.7.1-1 notation changes and add the title "Actions":

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the table. The content of the items has not changed unless specified elsewhere in this enclosure. It provides a missing title to the Action table associated with Table 3.3.7-1. It does not affect the design or operation of any plant system, nor does it change the content of the actions listed. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided elsewhere in the table and in the Technical Specifications. It does not represent a change in the content of the item. It merely adds a missing title. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency and clarity within the table and Technical Specifications. It does not involve a change in the content of the items. It only provides a missing title. Therefore, there is no impact on the margin or safety.

The licensee has provided the following no significant hazards consideration rationale for Category 3 changes.

Delete Footnotes for H Injection Test (Unit 2 only):

1. The proposed change deletes a footnote which no longer applies. The footnote was added to support a one-time hydrogen injection test, which was completed on January 5, 1987. No additional testing is planned. Thus, the proposed change has no effect on the probability of an accident, nor does it affect the consequences of any accidents.

2. The referenced footnote no longer applies to BSEP-2. The hydrogen injection test was successfully completed on January 5, 1987. Thus, the footnote is no longer necessary, and deletion of it will not create the possibility of a new or different type of accident.

3. Footnotes (7) and (i) were aded to support a one-time hydrogen injection test, which was completed on January 5, 1987. No additional testing is planned; therefore, the footnotes no longer apply and should be deleted. This deletion has no impact on the margin of safety.

Delete Foonote ** from Surveillance Requirement 4.1.3.5.b (Unit 1 only):

1. The proposed change deletes a footnote which no longer applies. The footnote was added to allow a one-time extension in the surveillance interval for Surveillance Requirement 4.1.3.5.b. The extension expired after the Spring 1981 outage; the footnote no longer applies. Thus, the proposed change has no impact on the probability or consequences of an accident.

2. The referenced footnote no longer applies to BSEP-1. The surveillance interval extension expired after the Spring 1981 outage. Thus, this footnote is no longer necessary. Therefore, its deletion will not create the possibility of new or different type of accident.

3. Footnote ** was added to the Technical Specifications to allow a one-time extension of a surveillance interval which expired after the Spring 1981 outage. Therefore, this deletion has no impact on the margin of safety of the plant.

Delete Foonote * from Surveillance Requirement 4.5.3.1.c on Page 3/4 5-6.

1. The proposed change deletes a footnote which no longer applies. The footnote was added to allow a one-time postponement of a flow test of the core spray. The extension expired on October 30, 1985 for BSEP-1 and November 15, 1984 for BSEP-2; therefore, the footnote no longer applies. Thus, the proposed change has no impact on the probability or consequences of an accident.

2. The referenced footnote no longer applies. The flow test extension interval expired on October 30, 1985 for BSEP-1 and on November 15, 1984 for BSEP-2. Thus, this footnote is no longer necessary. Therefore, its deletion will not create the possibility of a new or different type of accident.

3. Footnote * was added to the Technical Specifications to allow a one-time extension of a flow test requirement which expired on October 30, 1985 for BSEP-1 and on November 15, 1984 for BSEP-2. Therefore, this deletion has no impact on the margin of safety of the plant.

The following Category 4 determinations were made by the licensee.

Manipulate the footnote tables (i.e., turn the tables, add appropriate headings, double-space the footnotes, put parentheses around the footnote notation, and rearrange the footnotes into alphabetical order) and turn the tables upright:

1. The proposed change is an administrative change to the Technical Specifications to provide consistency throughout the Technical Specifications. The content of the footnotes and items in the table has not changed unless specified elsewhere in this enclosure. The changes to the footnote or footnote table have been made to provide clarity and consistency to the Technical Specifications. Therefore, it does not involve a significant increase in the probability of an accident, nor does it involve a change in the consequences of an accident previously evaluated.

2. The proposed change is purely administrative. It will provide consistency with other entries provided elsewhere in the table and in the Technical Specifications. It does not represent a change in the content of the footnote or items. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change is an administrative change. It will provide consistency and clarity within the table and the Technical Specifications. It does not involve a change in the content of the footnotes. Therefore, there is no impact on the margin of safety.

Lastly, Category 5 determinations are as follows:

Repaginate to accommodate tag number deletions:

1. The proposed change is administrative in nature. It has no impact on the design or

operation of any safety system; it only repaginates the affected section of the Technical Specifications to accommodate deletions on previous pages and to eliminate "a" pages. Therefore, the proposed change does not have any affect on the probability or consequences of any accident previosly evaluated.

2. The proposed change is administrative in nature. It's only purpose is to repaginate a section of the Technical Specifications where information is being deleted which is addressed by other proposed changes provided elsewhere in this submittal. Therefore, it does not create the possibity of a new or different kind of accident.

3. Repagination of this section has no bearing on the design or operation of any system. It is purely administrative. Thus, it does not impact the margin of safety of the plant.

The staff has reviewed the CP&L determinations and is in agreement with them. The instrument tag numbers will still be controlled by the licensee via a licensee controlled document subject to 10 CFR 50.59. The licensee stated that the one-time Unit 2 hydrogen injection test took place in January 1987, and the special footnotes are no longer necessary. The one-time Unit 1 extension in the surveillance interval for surveillance requirements 4.1.3.5.b expired after the Spring 1981 outage and is no longer necessary. The footnotes associated with surveillance requirement 4.5.3.1.c, which deals with the core spray system flow test, is no longer necessary because the tests were conducted within the time periods specified. Lastly, all other changes are administrative in nature. Accordingly, the commission proposes to determine that these changes do not involve a significant hazards consideration.

Local Public Document Room location: University of North Carolina at Wilmington, William Madison Randall Library, 601 S. College Road, Wilmington, North Carolina 28403–3297.

Attorney for licensee: R. E. Jones, General Counsel, Carolina Power & Light Company, P.O. Box 1551, Raleigh, North Carolina 27602.

NRC Acting Project Director: Edward A. Reeves.

Commonwealth Edison Company, Docket Nos. 50–237 and 50–249, Dresden Nuclear Power Station, Unit Nos. 2 and 3, Grundy County, Illinois

Date of application for amendment request: February 22, 1989.

Description of amendment request: The proposed amendment would revise the License Condition of Section 3.H of the Dresden 2 License and Section 3.G of the Dresden 3 License, and would delete all setpoints of the fire protection Technical Specifications (Section 3/4.12) and revise Sections 6.1.C, 6.1.G.1.a and 6.1.G.2.a of Appendix A of both licenses. Generic Letter 86–10, dated April 24, 1986, and Generic Letter 88–12, dated August 2, 1988, from the NRC provided guidance to the licensee to request removal of the fire protection Technical Specifications. The licensees' proposed amendment is in response to these Generic Letters.

Basis for proposed no significant hazards consideration determination: The staff has evaluated this proposed amendment and determined that it involves no significant hazards consideration. According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated; or

2. Create the possibility of a new or different kind of accident from any accident previously evaluated; or

3. Involve a significant reduction in a margin of safety.

The proposed revision to the License Condition is in accordance with the guidance provided in Generic Letter 86–10 for licensees requesting removal of fire protection Technical Specifications. The incorporation of the NRC-approved Fire Protection Program, and the former Technical Specification requirements by reference to the procedures implementing these requirements, into the Final Safety Analysis Report (FSAR) and the use of the standard License Condition, on fire protection, will ensure that the Fire Protection Program, including the system, the administrative and technical controls, the organization, and the other plant features associated with fire protection will be on a consistent status with other plant features described in the FSAR. Also, the provisions of 10 CFR 50.59 would then apply directly for changes the licensee desire to make in the Fire Protection Program. In this context, the determination of the involvement of an unrevised safety question defined in 50.59(a)(2) would be made based on the "accident * * * previously evaluated" being the postulated fire in the fire hazards analysis for the fire area affected by the change. Hence, the proposed License Condition establishes an adequate basis for defining the scope of changes to the Fire Protection Progam which can be made without prior Commission approval, i.e., without introduction of an unreviewed safety question. The revised License Condition or the removal of the existing Technical Specification requirements on fire protection does not create the possibility of a new or different kind of accident from those previously evaluated. They also do not involve a significant reduction in the margin of safety since the License Condition does not alter the requirement that an evaluation be performed for the identification of an unreviewed safety question for each proposed change to the Fire Protection Program. Consequently, the proposed License Condition or the removal of the fire protection requirements does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed modification of the Administrative Control Section of the Technical Specifications (Section 6) includes the review of the Fire Protection Program and implementing procedures and the submittal of recommended changes to the Off-site Review and Investigative Function as one of the responsibilities of the On-site Review and Investigative Function. In this manner, the Fire Protection Program will be addressed by administrative control requirements that are consistent with other programs addressed by License Conditions. These changes are administrative in nature and do not impact the operation of the facility in a manner that involves significant hazards consideration.

The proposed amendment includes the removal of fire protection Technical Specifications in four areas: (1) Fire detection systems, (2) fire suppression systems, (3) fire barriers, and (4) fire brigade staffing requirements. While it is recognized that a comprehensive Fire Protection Program is essential to plant safety, many details of this program that are currently addressed in Technical Specifications can be modified without affecting nuclear safety. With the removal of these requirements from the Technical Specifications, they have been incorporated into the Fire Protection Program implementing procedures. Hence, with the additions to the existing administrative control requirements that are applicable to the Fire Protection Program and the revised License Condition, there are suitable administrative controls to ensure that the licensee initiated changes to these requirements, that have been removed from the Technical Specifications, will receive careful review by competent individuals. Again, these changes are administrative in nature and do not impact the operation of the facility in a manner that involves significant hazards consideration.

Based on the preceding assessment, the staff believes the proposed amendment involves no significant hazard consideration.

Local Public Document Room location: Morris Public Library, 604 Liberty Street, Morris, Illinois 60450.

Attorney for licensee: Michael I. Miller, Esquire; Sidley and Austin, One First National Plaza, Chicago, Illinois 60603.

NRC Project Director: Daniel R. Muller.

Detroit Edison Company, Docket No. 50-341, Fermi-2, Monroe County, Michigan

Date of amendment request: March 10, 1989.

Description of amendment request: The amendment proposes revisions of Technical Specification Section 4.3.8.2.c to allow a one-time extension for the disassembly and inspection of the turbine control valves, high pressure turbine stop valves until the first refueling outage, currently scheduled to begin in September 1989.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92(c)) for a proposed amendment to a facility operating license. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated: or (3) involve a significant reduction in a margin of safety.

The licensee has evaluated the proposed change against the above standards as required by 10 CFR 50.92. The licensee concluded that:

(1) The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The turbine was first rolled on September 26, 1985. The valves will have only experienced operating conditions for approximately 24 months by the beginning of the first refueling outrage. Therefore, in actuality, the valves will be inspected prior to accumulating the amount of wear presently permitted by the Technical Specification. This does not represent any increase in the probability of an accident. Additionally, the protection provided by the overspeed protection system is not needed to protect safety related components, equipment or structures from turbine missiles. Since extending the first interval does nothing to the consequences of an accident, this change will not change the consequences of an accident. Thus, there is no increase in the probability or consequences of any accident previously evaluated.

(2) The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated. A one-time extension to the surveillance interval for turbine valve disassembly and inspection does not create any new modes of operation or testing. The weekly high pressure turbine stop, low pressure turbine stop, high pressure turbine control and low pressure turbine intercept valves cycling surveillance is not changed by this proposed amendment. Therefore, no new or different kind of accident from any accident previously evaluated has been created.

(3) The proposed change does not involve a significant reduction in the margin of safety. As stated above, the valves will actually experience less operating time betwen inspections than what is presently permitted by Technical Specifications and the overspeed protection system is not needed to protect safety-related components, equipment or structures from turbine missiles. Therefore, the margin of safety wil not be reduced by approval of this change request.

The staff has reviewed the licensee's evaluation and concurs with it. On the basis of the above consideration, the staff proposed to find that the changes do not involve a significant hazards consideration.

Local Public Document Room location: Monroe County Library System, 3700 South Custer Road, Monroe, Michigan 48161.

Attorney for licensee: John Flynn, Esq., Detroit Edison Company, 2000 Second Avenue, Detroit, Michigan 48226.

NRC Project Director: Theodore R. Quay, Acting.

Duke Power Company, et. al., Docket Nos. 50-413 and 50-414, Catawba Nuclear Station, Units 1 and 2, York County, South Carolina

Date of amendment request: June 19, 1987 as supplemented March 10, 1989.

Description of amendment request: The proposed amendments would revise the Technical Specifications (TSs) to add changes required by NRC Generic Letter (GL) 85-09, "Technical Specifications for Generic Letter 83-28, Item 4.3." Item 4.3 of GL 83-28 established the requirement for automatic actuation of shunt trip attachment on reactor trip breakers.

The specific changes would:

(1) Add a new Action Statement 12 to Item 19 "Reactor Trip Breakers" of TS Table 3.3-1.

(2) Add Item 21 "Reactor Trip Bypass Breakers" and its associated Action Statement 13, to TS Table 3.3-1.

(3) Add a new Table Notation 14 to Item 1 "Manual Reactor Trip" of TS Table 4.3-1.

(4) Add Item 21 "Reactor Trip Bypass Breakers" and its associated Table Notations 7, 15 and 16, and modify Table Notation 11 of TS Table 4.3-1.

Basis for proposed no significant hazards consideration determination: The proposed changes to the TSs are submitted by the licensee in response to GL 85-09 which states that: "* * * Technical Specification changes should be proposed by licensees to explicitly require independent testing of the undervoltage and shunt trip attachments during power operation and independent testing of the control room manual switch contracts during each refueling outage. The staff concluded that these tests are necessary to ensure reliable reactor trip breaker operation * * *."

The Commission has provided guidance concerning the application of its standards set forth in 10 CFR 50.92 for no significant hazards consideration by providing certain examples (51 FR 7744). One of the examples of an amendment likely to involve no significant hazards consideration relates to changes that (ii) consititute additional limitations, restrictions, or control not presently included in the TSs. The proposed amendments match the example because they would impose additional limitations for operation and additional surveillance requirements for the reactor trip breaker undervoltage and shunt trip attachments not presently included in the TSs.

The above proposed changes would permit individual testing of the undervoltage and shunt trip attachments and would be in accordance with GL 85.09 for required actions based on generic implications of the Salem ATWS event. Therefore, the staff proposes to determine that the proposed amendments do not involve a significant hazards consideration.

Local Public Document Room location: York County Library, 138 East Black Street, Rock Hill, South Carolina 29730.

Attorney for licensee: Mr. Albert Carr, Duke Power Company, 422 South Church Street, Charlotte, North Carolina 28242.

NRC Project Director: David B. Matthews.

Duke Power Company, Docket Nos. 50-369 and 50-370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of amendment request: January 22, 1989.

Description of amendment request: The proposed change would update pressure and temperature limits in

Technical Specification (TS) 3/4.4.9 for heatup and cooldown of the reactor coolant system, including associated Table 4.4–5 on the withdrawal and examination schedule for reactor vessel material irradiation surveillance specimens. TS Bases 3/4.4.9 would be similarly undated to reference revised heatup and cooldown curves and information associated with their derivation and use.

Basis for proposed no significant hazards consideration determination: TS 4.4.9.1.2 requires that reactor vessel material irradiation surveillance specimens be periodically removed and examined to determine changes in material properties as required by 10 CFR Part 50, Appendix H, in accordance with the schedule in TS Table 4.4.5, and that the results of these examinations be used to update TS figures defining allowable pressure and temperature limits for reactor coolant system heatup and cooldown. Thus, the TSs and Appendix H establish dynamic requirements involving periodic monitoring, evaluation and adjustments for changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region resulting from exposure of these materials to neutron irradiation and the thermal environment. The proposed changes are in accordance with these requirements for periodic updating.

The staff has reviewed the licensee's request for the above amendments and finds the proposed curves to be conservative with respect to the existing pressure-temperature operating limits, and to be based upon results of capsule analyses performed in accordance with NRC approved methods. Therefore, operation in accordance with the updated limits would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated, or (2) create the possibility of a new or different kind of accident from any accident previously evaluated. By adjusting the limits to reflect the change in material toughness due to irradiation, the present margin of safety is not reduced, and therefore, the change would not (3) involve a significant reduction in margin of safety.

Accordingly, the Commission proposes to determine that the proposed amendments involve no significant hazards considerations.

Local Public Document Room location: Atkins Library, University of North Carolina, Charlotte (UNCC Station), North Carolina 28223.

Attorney for licensee: Mr. Albert Carr, Duke Power Company, 422 South Church Street, Charlotte, North Carolina 28242.

NRC Project Director: David B. Matthews.

Florida Power Corporation, et al., Docket No. 50–302, Crystal River Unit No. 3 Nuclear Generating Plant, Citrus County, Florida

Date of amendment request: March 31, 1983 as supplemented June 22, 1983 and revised February 24, 1984, May 31, 1984 and December 31, 1984.

Description of amendment request: The proposed amendment would provide Technical Specifications (TS) for the Reactor Coolant System (RCS) high point vents. These TS define actions to be taken should the RCS vents become inoperable and adds surveillance requirements to ensure vent operability. The RCS vents were installed in response to NUREG–0737 and the guidance provided by Generic Letter 83–37. The addition of this vent system can help to reduce the effects of an accident by venting gases that could inhibit natural circulation core cooling. The action and surveillance requirements provided in this TS proposal would assure the operability of the vents should they be needed.

The March 31, 1983 application, as supplemented June 22, 1983, was previously noticed in the Federal Register on December 21, 1983 (48 FR 56504). As originally submitted the proposed amendment would have allowed indefinite continued operation with one vent inoperable. The current revision requires restoring the inoperable vent to operable status within 30 days or submitting a report and schedule for corrective action within the next 30 days. If the pressurizer vent is inoperable, indefinite continued operation is acceptable provided an alternate vent path is available. In addition, the current revision adds requirements for demonstrating operatility of the vent system block valves and for verifying flow through the vent paths. Because of these revisions, the staff has decided to renotice the proposed amendment.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of criteria for determining whether a significant hazards consideration exists by providing certain examples (51 FR 7750). One of the examples of actions involving no significant hazards consideration is example (ii), "a change that constitutes an additional limitation, restriction, or control not presently included in the technical specifications, e.g. a more stringent surveillance requirement." The proposed change involves additional actions the licensee must take in order to assure the operability of the RCS high point vents. Thus, the proposed changes are in accordance with the above example. Therefore, the staff proposes to determine that the proposed amendment involves no significant hazards considerations.

Local Public Document Room location: Crystal River Public Library, 668 N.W. First Avenue, Crystal River, Florida 32629.

Attorney for licensee: R.W. Neiser, Senior Vice President and General Counsel, Florida Power Corporation, P.O. Box 14042, St. Petersburg, Florida 33733.

NRC Project Director: Herbert N. Berkow.

Florida Power Corporation, et al., Docket No. 50–302, Crystal River Unit No. 3 Nuclear Generating Plant, Citrus County, Florida

Date of amendment request: June 22, 1983, as revised February 24, 1984.

Description of amendment request: The amendment would change Sections 3.6.4.1 and 4.6.4.1 of the Technical Specifications (TS) to require that two containment hydrogen monitors be operable. The current TS require that one hydrogen analyzer and one gas chromatograph be operable. The change also addresses the frequency and method of checking and calibrating the hydrogen monitors. This change request is the result of the installation of two independent, in-place containment hydrogen monitors and of the requirements of Attachment 6 to Section II.F.1 of NUREG–0737.

The June 22, 1983 application was previously noticed in the Federal Register on December 21, 1983 (48 FR 56504). Due to the revised submittal dated February 24, 1984, the staff has determined that the proposed amendment should be renoticed.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determnining whether a significant hazards consideration exists (10 CFR 50.92(c)). A proposed amendment to an operating license for a facility involves no significant hazards considerations if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The amendment request was analyzed in light of the above three criteria. In regard to the first criterion, it was determined that the requested change would not involve a significant increase in the probability or consequences of an accident previously evaluated. The function of the containment hydrogen monitor is to measure the amount of hydrogen in the containment building after an accident. The change would simply replace the current requirement to have a hydrogen analyzer and a gas chromatograph operable with a requirement to have two hydrogen monitors operable.

In regard to the second criterion, it was determined that the proposed change would not create the possibility of a new or different kind of accident from any accident previously evaluated. Since the only real change proposed by this amendment is to replace the requirement for a gas chromatograph and a hydrogen analyzer to be operable with the requirement for two hydrogen monitors to be operable, there will be no risk of a new type of accident resulting from this request.

In regard to the third criterion, it was found that the proposed amendment would not involve a significant reduction in a margin of safety. The requirement for two hydrogen monitors to be operable will result in the ability to measure containment building hydrogen concentration in the event of an accident.

In addition, with respect to all three criteria, the changes to the surveillance requirements were found adequate by the NRC in a letter dated November 1, 1984 and will ensure that the hydrogen monitors will be operable if needed.

Therefore, the staff proposed to determine that the proposed change does not involve a significant hazards consideration.

Local Public Document Room location: Crystal River Public Library, 668 N.W. First Avenue, Crystal River, Florida 32629.

Attorney for licensee: R.W. Neiser, Senior Vice President and General Counsel, Florida Power Corporation, P.O. Box 14042, St. Petersburg, Florida 33733.

NRC Project Director: Herbert N. Berkow.

Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, City of Dalton, Georgia, Docket Nos. 50–321 and 50–366, Edwin I. Hatch Nuclear Plant, Units 1 and 2, Appling County, Georgia

Date of amendment request: February 3, 1989.

Description of amendment request: The amendments would modify the Technical Specifications (TS) for units 1 and 2 to: (1) Change the maximum operating times for certain primary containment isolation valves (PCIVs) to account for a different method of measuring; (2) exclude several unit 1 containment penetrations and PCIVs from the local leak rate test (LLRT) program; (3) revise Unit 1 TS section 4.7.A.2 and Unit 2 TS section 4.6.1.3 to achieve similarity between the two documents, to comply with current 10 CFR Part 50 Appendix J testing requirements, and to specify an allowable leakage; (4) delete penetration 218A from Unit 1 TS Table 3.7–2; and (5) remove the isolation valves associated with the primary feedwater and the torus drainage and purification systems from Unit 2 TS section 3.6.1.2.

Basis for proposed no significant hazard consideration determination: The Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92(c)). A proposed amendment to an operating license for a facility involves no significant hazards considerations if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The licensee's February 3, 1989 submittal provided an evaluation of the proposed changes with respect to these three standards, as follows:

Proposed Change 1 would increase the maximum operating times for 15 PCIVs on Unit 1 and 22 PCIVs on Unit 2 to account for a change in the measurement method from the present "light-to-light" method to a "switch-to-light" method. The change does not involve a significant hazards consideration because:

1. It does not involve a significant increase in the probability or consequences of an accident previously evaluated, because neither plant operation nor design is affected by the proposed change. The use of switch-to-light methodology will ensure the continued operability of the valve at the required level of safety, while meeting ASME Code testing criteria. The revision of maximum operating time does not reflect any change in valve or system operation or design but reflects only a change in testing methodology.

2. It does not create the possibility of a new or different kind of accident from any previously evaluated, because no new modes of plant operation or change in design are involved. This change is based only on a change in valve operating time testing methodology, and the revised maximum operating time is a result of the use of this different technique. It does not represent, nor does it require, any change to actual system or valve operation or design. System response is not altered.

3. It does not involve a reduction in the margin of safety, because the proposed switch-to-light testing methodology and associated revised maximum operating time will ensure the continued operability of the valve at the current required level of safety and consistency with the ASME Code. The new maximum operating time is determined solely from the use of the switch-to-light methodology, while maintaining the same valve and system operation and response, and thereby the same margin of safety.

Proposed Change 2 would delete certain valves and associated penetrations from the Unit 1 LLRT program because the associated piping terminates in the torus below the water line, thereby precluding gaseous leakage. The change does not involve a significant hazards consideration because:

1. It does not involve a significant increase in the probability or consequences of an accident previously evaluated, because the valves proposed for exemption from 10 CFR 50, Appendix J testing do not represent a post-LOCA release pathway to the environment.

2. It does not create the possibility of a new or different kind of accident from any previously evaluated, because these changes do not introduce any new modes of operation. Only testing requirements and acceptance criteria are affected.

3. It does not involve a reduction in the margin of safety, because primary containment integrity will be assured by routine valve surveillance testing per the requirements of ASME Code, Section I, Part IWV–3420.

Proposed Change 3 does not involve a significant hazards consideration because:

1. It does not involve a significant increase in the probability or consequences of an accident previously evaluated, because the change assigns a specified value for testing purposes that will assure the allowable leakage under accident conditions will not be exceeded.

2. It does not create the possibility of a new or different kind of accident from any previously evaluated, because these changes do not introduce any new modes of operation. Only testing requirements and acceptance criteria are affected.

3. It does not involve a reduction in the margin of safety, because primary containment integrity will be assured by leak rate testing of the air lock in accordance with 10 CFR 50, Appendix J requirements.

Proposed Change 4 would delete from TS Table 3.7–2 for Unit 1, a value that had previously been removed from a listing of PCIVs subject to Appendix J leak rate testing. The change corrects an oversight in that the valve should have been deleted from Table 3.7-2 as part of the earlier amendment. The change does not involve a significant hazards consideration because:

1. It does not involve a significant increase in the probability or consequences of an accident previously evaluated, because the penetration has been previously determined not to represent a potential containment leakage path. This change is purely administrative in nature.

2. It does not create the possibility of a new or different kind of accident from any previously evaluated, because these changes do not represent a change to plant design or configuration. This change is purely administrative in nature.

3. It does not involve a reduction in the margin of safety, because this change is purely administrative in nature.

Proposed Change 5 does not involve a significant hazards consideration because:

1. It does not involve a significant increase in the probability or consequences of an accident previously evaluated, because this change merely acknowledges the actual fluid-sealed condition of the subject valves following a postulated LOCA, as allowed by 10 CFR 50 Appendix J, Section III.C.3.

2. It does not create the possibility of a new or different kind of accident from any previously evaluated. Removal of the isolation valves associated with the primary feedwater and the torus drainage and purification systems from the Technical Specification does not involve any physical modification to the plant and, therefore, will not introduce any new modes of plant equipment operation or failure.

3. It does not involve a reduction in the margin of safety. Because the subject valves are in a fluid-sealed condition following a ostulated LOCA, they will perform their intended function whether or not they are considered components of bypass leakage. Therefore, performance of the primary containment system is unaffected by this change.

The staff has considered the proposed changes and agrees with the licensee's evaluation with respect to the three standards.

On this basis, the Commission has determined that the requested amendments meet the three standards and, therefore, has made a proposed determination that the amendment application does not involve a significant hazards consideration.

Local Public Document Room location: Appling County Public Library, 301 City Hall Drive, Baxley, Georgia 31513.

Attorney for licensee: Bruce W. Churchill, Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street NW., Washington, DC 20037.

NRC Project Director: David B. Matthews.

Indiana Michigan Power Company, Docket Nos. 50-315 and 50-316, Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2, Berrien County, Michigan

Date of amendment request: March 4, 1988.

Description of amendment request: The proposed license amendment would change the "Instrumentation" and "Design Features" sections of the Technical Specifications for Units 1 and 2 of the Donald C. Cook Nuclear Plants. The amendment reflect recently completed enhancements to the D.C. Cook meteorological monitoring system. The enhancements were completed in response to concerns raised in a Nuclear Regulatory Commission Technical Evaluation Report issued June 18, 1986. The amendment addresses concerns for adequate instrumentation to provide a representative view of meteorological conditions within the 10 mile Emergency Planning Zone. The proposed amendment utilizes a three tower system to preclude a possible unrepresentative assessment of meteorological conditions due to the Lake Michigan shoreline effects.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a significant hazards consideration exists [10 CFR 50.92(c)]. A proposed amendment to an operating license for a facility in accordance with the proposed amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The licensee has evaluated the proposed change against the above standards as required by 10 CFR 50.92. We have reviewed the licensee's evaluation and concur with it. The licensee concluded that:

1. The change does not involve a significant increase in the probability or consequences of an accident previously evaluated (10 CFR 50.92(c)(1)) because the change will enhance the meteorological monitoring system. The change does not alter or eliminate the functions previously reviewed.

2. The change does not create the possibility of a new or different kind of accident from any accident previously analyzed or evaluated (10 CFR 50.92(c)(2)) because the plant operation and design are not affected by the proposed change. The proposed amendment creates no new accident scenario.

3. The change does not involve a significant reduction in a margin of safety (10 CFR 50.92(c)(3)) because all requirements necessary for safe operation have been retained in the proposed Technical Specifications.

On the basis of the above consideration, the staff proposes to find that the changes do not involve a significant hazards consideration.

Local Public Document Room location: Maude Preston Palenske Memorial Library, 500 Market Street, St. Joseph, Michigan 49085.

Attorney for licensee: Gerald Charnoff, Esq., Shaw, Pittman, Potts and Trowbridge, 2300 N Street, NW., Washington, DC 20037.

NRC Project Director: Theodore R. Quay, Acting.

Niagara Mohawk Power Corporation, Docket No. 50-410, Nine Mile Point Nuclear Station, Unit No. 2, Oswego County, New York

Date of amendment request: December 15, 1988.

Description of amendment request: The proposed amendment would revise the Technical Specification Sections 4.8.4.4, Reactor Protection System Electric Power Monitoring (RPS Logic), and 4.8.4.5, Reactor Protection System Electric Power Monitoring (Scram Solenoids) to change the minimum surveillance test frequency from six to eighteen months. However, testing will be required at each cold shutdown of greater than twenty-four hours if the test has not been performed within the previous 6 months. The purpose of the amendment is to prevent a required plant shutdown solely for the purpose of performing the surveillance test. The test configuration places the plant in a half-scram condition with partial reactor vessel isolation. This condition makes testing at power operation difficult. As a result the reactor is shut down prior to performing the test. By increasing the frequency to 18 months, the test can be performed on a refueling outage interval. Niagara Mohawk has indicated that a net improvement to plant safety can be realized by reducing the frequency of testing.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards in 10 CFR 50.92(c) for determining whether a significant hazards consideration exists. A proposed amendment to an Operating License for a facility involves no significant hazard consideration if operation of the facility in accordance with the proposed amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident from any accident previously evaluated, or (3) involve a significant reduction in a margin of safety. The staff has reviewed the licensee's December 15, 1988 submittal and finds that:

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated because the increase in the surveillance frequency will not have an adverse affect upon the ability of the Reactor Protection System (RPS) and the Nuclear Steam Supply Shutoff System to perform their intended safety functions. The increased frequency has been evaluated with respect to reactor shutdown reliability, incorporating failure probability from industry operating experience, and was found to have a negligible effect with respect to overall plant safety. Further, the proposed change would reduce the amount of time the reactor would be in a half scram condition and vulnerable to challenges to the plant shutdown systems if the testing was performed at power. Although the testing is currently performed at cold shutdown, the margin of safety provided by the Technical Specifications is based on performing the surveillance while at power. Increasing the frequency will also prevent unnecessary cycling of the facility.

2. The proposed amendment does not create the possibility of a new or different kind of accident than previously evaluated because as discussed above, the increase in testing frequency will not adversely affect the Reactor Protection System and Nuclear Steam Supply Shutoff System responses to previously evaluated accidents. The responses remain within previously assessed limits. In addition, no modifications are being made to plant equipment which could create the possibility of a new or different accident.

3. The proposed amendment will not involve a significant reduction in margin of safety because as discussed previously, the change was evaluated with respect to reactor shutdown reliability and found to have a negligible impact with respect to overall plant safety. In addition, the margin of safety provided by the current Technical Specifications is based on performing the test while at power. This places the plant in a half-scram condition increasing the probability of an inadvertent scram. The licensee's submittal provides an evaluation indicating a net improvement to plant safety as a result of decreasing the frequency of placing the plant in the half-scram condition. Therefore, the proposed change will not result in a significant reduction in margin of safety.

Based upon the above, the staff proposes to determine that the proposed amendment will not involve a significant hazards consideration.

Local Public Document Room location: Reference and Documents Department, Penfield Library, State University of New York, Oswego, New York 13126.

Attorney for licensee: Mark Wetterhahn, Esquire, Conner & Wetterhahn, Suite 1050, 1747 Pennsylvania Avenue NW., Washington, DC 20006.

NRC Project Director: Robert A. Capra.

Northeast Nuclear Energy Company, Docket No. 50-245, Millstone Nuclear Power Station, Unit No. 1, New London County, Connecticut

Date of amendment request: January 26, 1989.

Description of amendment request: The proposed change to the Technical Specifications would delete the requirement to verify uniformity of air flow distribution across the charcoal absorber banks and HEPA filters of the Standby Gas Treatment System once per operating cycle.

Basis for proposed no significant hazards consideration determination: The licensee has reviewed the proposed changes, in accordance with 10 CFR 50.92 and has concluded and the NRC agrees, that they do not involve a significant hazards consideration in that these changes would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated. The revised surveillance requirement does not adversely affect the consequences of the design basis accidents. In addition, it does not affect the reliability of the testing of the standby gas treatment system since uniform distribution of air across filter banks is only dependent on system geometry. Therefore, it is concluded that previously analyzed accidents are not affected.

2. Create the possibility of a new or different kind of accident from any previously analyzed. Since there are no changes in the way the plant is operated, the potential for an unanalyzed accident is not created. No new failure modes are introduced.

3. Involve a significant reduction in a margin of safety. The proposed requirement does not have any adverse impact on the protective boundaries. Since the proposed change also does not affect the consequences of any accident previously analyzed, there is no reduction in a margin of safety.

Local Public Document Room location: Waterford Public Library, 49 Rope Ferry Road, Waterford, Connecticut 06385.

Attorney for licensee: Gerald Garfield, Esquire, Day, Berry & Howard, Counselors at Law, City Place, Hartford, Connecticut 06103-3499.

NRC Project Director: John F. Stolz.

Pennsylvania Power and Light Company, Docket No. 50-387, Susquehanna Steam Electric Station, Unit 1, Luzerne County, Pennsylvania

Date of amendment request: February 2, 1989.

Description of amendment request: The proposed amendment would revise the Unit 1 Technical Specifications to support the forthcoming Cycle 5 operations, and to make some editorial changes.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92(c)). A proposed amendment to an operting license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) Involve a significant increase in the probability for consquences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The staff has reviewed the licensee's request and concurs with the following basis and conclusion provided by the licensee in its February 2, 1989 submittal in support of its determination that the proposed changes involve no significant hazards consideration.

The following three questions are addressed for each of the proposed Technical Specification changes;

I. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

II. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated.

III. Does the proposed change involve a significant reduction in a margin of safety?

• Specification 2.2.1, Rector Protection System Instrumentation Setpoints.

The change to this specification is a correction of a typographical error on Table 2.2.1-1, Functional Unit 2.b, where footnote "#" has been added onto the allowable value.

I. No. The change is editorial in nature; the footnote is currently applicable to the trip setpoint, and should be applicable to the allowable value. Correct single loop operation limits are provided in Specification 3.4.1.1.2. This change corrects a typographical error in the issuance of Amendment 56 to License No. NPF-14. This correction has no impact on any safety analysis.

II. No. See I above.

III. No. See I above.

• Specification 3/4.2.1, Average Planar Linear Heat Generation Rate

The changes to this specification are editorial in nature in that they reflect the removal of the remaining General Electric fuel from the SSES Unit 1 core.

I. No. The changes to this specification and its associated figures are solely due to the fact that no GE fuel will reside in the Unit 1 Cycle 5 core. All references to the GE fuel and its limits are therefore deleted. The ANF fuel limits remain the same. These editorial changes have no impact on any safety analysis.

II. No. See I above.

III. No. See I above.

• Specification 3/4.2.2, APRM Setpoints

The changes to this Specification are editorial in nature in that they reflect the removal of the remaining GE fuel from the SSES Unit 1 core.

I. No. The changes to this specification are solely due to the fact that no GE fuel will reside in the Unit 1 Cycle 5 core. The definition of "T" for GE fuel is therefore deleted. This editorial change has no impact on any safety analysis.

II. No. See I above.

III. No. See I above.

• Specification 3/4.2.3, Minimum Critical Power Ratio

The changes to this specification correct an administrative error in the issuance of Amendment 72 to License No. NPF-14, and provide new operating limit MCPR curves based on cycle-specific transient analyses.

I. No. The administrative change corrects two pages that were inadvertently reversed in the issuance of Amendment 72; this has no impact on any safety analysis.

Limiting core-wide transients were evaluated with ANF's COTRANSA code * * * and this output was utilized by the XCOBRA-T methodology * * * to determine delta CPRs. Both COTRANSA and XCOBRA-T have been approved by the NRC in previous license amendments. A modified void history correlation was used in the neutronics calculations which ultimately affect the delta CPRs, but this change was needed to achieve the same degree of accuracy for higher fuel exposures as was previously provided for lower fuel exposures * * *. All core-wide transients were analyzed deterministically (i.e., using bounding values as input parameters).

Two local events, Rod Withdrawal Error and Fuel Loading Error, were analyzed in accordance with the methods described in XN-NF-80-19 (A) Vol. 1 * * *. This methodology has been approved by the NRC.

Based on the above, the methodology used to develop the new operating limit MCPRs for the Technical Specifications does not involve a significant increase in the probability or consequences of an accident previously evaluated.

II. No. the methodology described can only be evaluated for its effect on the consequences of analyzed events; it cannot create new ones. The consequences of analyzed events were evaluated in I above.

Regarding the administrative correction, see I above.

III. No. As stated in I above, and in greater detail in the * * *. Reload Summary Report, the methodology used to evaluated core-wide and local transients is consistent with previously approved methods and meets all pertinent regulatory criteria for use in this application. The new void history correlation could be considered an exception since it has not been previously utilized in reload submittals for SSES, but its uses ensures a more accurate result of the higher burnups which will be occurring in Cycle 5.

Based on the above, the use of the methodology used to produce the Unit 1 Cycle 5 MCPR operating limits will not result in a significant decrease in any margin of safety.

Regarding the administrative correction, see I above.

• Specification 3/4.2.4, Linear Heat Generating Rate

All proposed changes to this specification are editorial.

I. No. The proposed changes simply remove all references to GE fuel. This has no impact on safety since it is entirely administrative in nature.

II. No. See I above.

III. No. See I above.

• Specification 3/4.3.6, Control Rod Block Instrumentation

The changes to this specification correct an administrative error in the issuance of Amendment No. 64 to License No. NPF-14 by applying footnote "##" appropriately in trip functions 1a and 2a in Table 3.3.6-2.

I. No. Footnote "##" was inadvertently left off of the two trip functions for which revised limits are required for single loop operation. Placing the footnotes in their appropriate locations ensures a proper cross reference between specifications 3/4.3.6 and 3/4.4.1. This change is editorial in nature and has no impact on any safety analyses.

II. No. See I above.

III. No. See I above.

• Specification 3/4.4.1, Recirculation System

All changes to this specification support single loop operation (SLO).

I. No. The original GE SLO analysis required the adjustment of APRM scram, APRM Rod Block, and Rod Block Monitor setpoints in SLO to bound changes in the assumed drive flow to core flow relationship between two loop and single loop operation. The GE analysis indicated that the two loop to single loop change is typically less then 7% drive flow for a given core flow. SSES-specific data taken by PP&L indicates that an 8.5% drive flow change would bound differences between two loop and single loop operation. Therefore, specifications 3.4.1.1.2a.2, 4, and 6 incorporate setpoint adjustments to account for this 8.5% change.

Specification a.3 is revised to remove the GE fuel reference (an administrative change), and to provide the proper MAPLHGR limit for ANF fuel. LOCA analyses performed by ANF * * * indicate that the two loop MAPLHGR limits are applicable to SLO for ANF fuel.

New specification a.5 proposes new MCPR limits for SLO based on transient analyses performed by ANF for events initiated from SLO conditions * * *. These analyses show that the operating limit MCPR must be increased to a minimum of 1.42 for SLO. A 0.01 constant is added to the two loop operating limit MCPR for low power and low core flow conditions for SLO operating limit MCPR values greater than 1.42.

Based on the above analyses of the non-editorial changes to this specification, appropriate limits have been proposed to assure that SLO will not result in a significant increase in the probability or consquences of any accident previously evaluated. The editorial change has no impact on previous analyses.

II. No. The revised setpoints are based on actual data which makes them more restrictive; the revised limits for MCPR and MAPLHGR are based on approved LOCA and transient analysis methods. Neither of these, nor the editorial change, can create the potential for new events.

III. No. As stated in II. above, the revised setpoints are more restrictive and more accurate, and therefore cannot result in a significant reduction in any safety margin. The revised MCPR and MAPLHGR limits are based on analyses which ensure that no significant reduction in safety margins has occurred based on their inputs, applied conservatisms, and calculational methodologies as documented in this proposal. The editorial change has no safety impact.

Based on the above considerations, the Commission proposes to determine that the proposed changes involve no significant hazards consideration.

Local Public Document Room location: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, Pennsylvania 18701.

Attorney for licensee: Jay Silberg, Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street NW., Washington, DC 20037.

NRC Project Director: Walter R. Butler.

Pennsylvania Power and Light Company, Docket Nos. 50-387 and 50-388 Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of amendment request: December 12, 1988.

Description of amendment request: The proposed amendments would revise the Unit 1 and the Unit 2 Technical Specifications to reflect revisions to the load profiles of battery banks 1D610, 1D620, 1D630, 1D640, 2D610, 2D620, 2D630, and 2D640. The licensee states that these changes are necessary to accommodate the transfer of control room instrumentation inverter loads from present ac power to battery banks and remove the emergency lighting loads from battery banks. The licensee states that the changes will result in a net reduction in battery loads.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92(c)). A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

The staff has reviewed the licensee's request and concurs with the following basis and conclusion provided by the licensee in its December 12, 1988 submittal.

The proposed change does not:

(1) Involve an increase in the probability or consequences of an accident previously evaluated. FSAR Subsection 8.3.2.1.1.4 states that the station batteries have sufficient capacity without the charger to independently supply the required loads for four hours. The Technical Specifications require that the batteries be surveilled to dummy loads which are greater than the design loads. An assessment has been performed by our engineering department which verifies that the batteries have adequate capacity to power the actual loads on the 125V DC system. The new load profiles contained in the proposed amendment to the Technical Specifications envelop the actual loads.

(2) Create the possibility of a new or different kind of accident from any previously evaluated. As stated in Part (1), the batteries have sufficient capacity to power the actual battery loads thus enabling them to perform their intended function. Any postulated accident resulting from this change is bounded by previous analysis.

(3) Involve a reduction in the margin of safety. IEEE 485 requires that the related battery capacity include a margin of aging of the battery and the temperature of the batteries' environment at the beginning of battery life. This margin allows replacement of the battery when its capacity is decreased to 80% of its rated capacity (100% design load). Our engineering department has determined that with the revised reduced load profiles the Class IE 125V DC batteries will supply their connected emergency loads with greater margins of safety at the battery electrolyte temperatures equal to or greater than 60 °F and with 25% aging margins relative to load as recommended by IEEE-485-1983. With the decreased battery loads it can be concluded that the overall margin of the plant is not diminished.

Based on the above considerations, the Commission proposes to determine that the proposed changes involve no significant hazards consideration.

Local Public Document Room location: Osterhout Free Library, Reference Department, 71 South Franklin Street, Wilkes-Barre, Pennsylvania 18701.

Attorney for licensee: Jay Silberg, Esquire, Shaw, Pittman, Potts and Trowbridge, 2300 N Street NW., Washington, DC 20037.

NRC Project Director: Walter R. Butler.

Power Authority of the State of New York, Docket No. 50-333, James A. FitzPatrick Nuclear Power Plant, Oswego, New York

Date of amendment request: January 17, 1989.

Description of amendment request: The proposed Technical Specification (TS) changes will allow the use of the 6-inch line to Standby Gas Treatment System (SGTS) for inerting and deinerting the primary containment, ensure the integrity and operability of the SGTS if a design basis loss of coolant accident (LCOA) occurs while inerting or deinerting the primary containment, specify the actions required when a containment isolation valve becomes inoperable, restrict the maximum angle of opening of the vent and purge valves to ensure their operability during a design basis LOCA, reflect the addition of new containment isolation valves in the Reactor Building Closed Loop Cooling Water System (RBCLCWS) and the exclusion of these valves from quarterly surveillance requirements, and incorporate an administrative change for consistency.

Specifically, the changes affect pages 183, 183a, 185-186a, 191, 192, and 197. The proposed changes include: (1) The use of the 6-inch line with valve number 27 MOV-121 for inerting and deinerting the primary containment; (2) a monthly surveillance requirement for 12-inch valve number 27 MOV-120; (3) actions to be taken to ensure containment isolation; (4) maximum opening angle for vent and purge valves; and (5) a change to the quarterly surveillance requirements for the RBCLCWS valves.

The proposed change will allow the use of the 6 inch line with valve number 27 MOV-121 for inerting and deinerting the primary containment. If a LOCA occurs while inerting or deinerting the primary containment through the 6 inch line, the maximum flow through this line would be such that the delta P across the HEPA filter assembly of the SGTS will not exceed the design limits. Thus, the integrity and operability of the SGTS is assured.

The proposed change will provide assurance of containment isolation by requiring that at least one isolation valve be operable in each affected penetration that is open. Also, if a valve is inoperable, it will be necessary to either restore the inoperable valve to operable status within 4 hours or isolate the affected penetration within 4 hours.

The proposed change will ensure the operability of the containment vent and purge valves during a design basis LOCA. To ensure that the valves will close under the design basis LOCA loads, the maximum angle of opening for valve numbers 27 AOV-111, 27 AOV-112, and 27 AOV-113 is restricted to 40° and for valve numbers 27 AOV-114, 27 AOV-115, 27 AOV-116, 27 AOV-117, and 27 AOV-118 the maximum opening angle is restricted to 50°. This proposed change will also ensure that the containment vent and purse valves can be opened for other safety related reasons. These reasons may include, but are not limited to, inerting or deinerting the primary containment, maintaining containment oxygen concentration, maintaining drywell and suppression chamber pressures, and maintaining the differential pressure between the drywell and suppression chamber.

The proposed change will reflect the addition of new RBCLCWS isolation valves in the FitzPatrick Technical Specifications and the exclusion of these valves from quarterly surveillance schedule established for primary containment power-operated isolation valves. The proposed alternate surveillance test interval, of cycling the valves whenever the reactor is in the cold shutdown condition for greater than 48 hours if they have not been cycled within the preceding 92 days, is satisfactory because (1) The ability to mitigate the effects of an accident are not affected by an inoperable valve; (2) the limits do not communicate with either the containment atmosphere or reactor coolant pressure boundary; (3) the valves are not required to operate (i.e., close) in the event of an accident; and (4) less frequent testing will reduce the possibility of drywell equipment failure or degradation through overheating caused by interruption of cooling water. The valves cannot be cycled during power operation since the result would be loss of cooling water to vital drywell equipment.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with a proposed amendment would not: (1) Involve a significant increase in the propability or consequences of an accident previously evaluated; or (2) Create the possibility of a new or different kind of accident from any previously evaluated; or (3) Involve a significant reduction in a margin of safety.

The licensee has evaluated the proposed amendment against the standards provided above and has made the following determination:

The proposed change does not involve a significant hazards consideration, as defined in 10 CFR 50.92, because operation in accordance with this change would not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated because:

a. The proposed change will ensure the integrity and operability of the SGTS if a LOCA occurs while inerting or deinerting the primary containment. The use of the 6 inch line will provide this assurance because the maximum flow through this line is such that the delta P across the HEPA filter assembly remains within the design limits.

b. The proposed change will ensure the operability and integrity of the vent and purge valves for closing during a design basis LOCA. In order to ensure that these valves will close again the design basis LOCA loads, the proposed change imposes new restrictions by limiting the maximum angle of opening to specified valves.

c. The proposed change will also provide greater assurance of containment isolation in the event of an accident by imposing new restrictions on isolation and restoration of inoperable valves to operable status.

d. The proposed change will increase containment isolation dependability by addition of remote manual RBCLCWS valves. The fail open/as-is design of these valves ensures a continous supply of cooling water during both normal and accident conditions. Surveillance testing cannot cause an accident because testing will be performed during plant shutdowns when the RBCLCWS is not required to cool the drywell atmosphere. Surveillance tests for the new valves will be conducted more frequently than the tests previously conducted on the manual RBCLCWS valves. This will further ensure increased valve reliability.

2. Create the possibility of a new or different kind of accident from any previously analyzed, because:

The proposed change will impose restrictions to ensure operability and integrity of the SGTS while inerting or deinerting the primary containment if a design basis LOCA occurs at that time. Furthermore, the proposed change will also impose additional restrictions to assure containment isolation by limiting the maximum angle of opening of the vent and purge valves and requiring actions to restore inoperable valves to operable status, or isolate each affected penetration, within 4 hours.

The change from manual to remote manual RBCLCWS containment isolation valves does not create the possibility of a new or different type of accident, because a continuous supply of water is assured during both normal and accident conditions by the fail open/as-is design of these new containment isolation valves. RBCLCWS containment isolation valve operability tests cannot create a new or different type of accident because testing will be performed during reactor shutdowns. This will preclude the possibility that their failure to reopen following a test conducted during power operation could precipitate drywell equipment degradation due to loss of drywell atmosphere cooling and high drywell temperatures.

3. Involve a significant reduction in the margin of safety because:

The proposed change will ensure the operability of SBGTS during inerting or deinerting if a design basis LOCA occurs at that time, operability of the vent and purge valves during a design basis LOCA, and isolation of the primary containment by imposing additional restrictions. The addition of remote manual RBCLCWS containment isolation valves will increase the margin of safety by increasing the extent to which the FitzPatrick plant complies with General Design Criterion 57 of Appendix A to 10 CFR Part 50. Compliance with this criterion improves containment isolation dependability. The ability of the plant to mitigate the effects of an accident are not affected by the failure of these valves.

The staff has reviewed the licensee's no significant hazards consideration determination. Based on the review and above discussion, the staff proposes to determine that the proposed changes do not involve a significant hazards consideration.

Local Public Document Room location: State University of New York, Penfield Library, Reference and Documents Department, Oswego, New York 13126.

Attorney for licensee: Mr. Charles M. Pratt, 10 Columbus Circle, New York, New York 10019.

NRC Project Director: Robert A. Capra

Virginia Electric and Power Company, Docket Nos. 50-338 and 50-339, North Anna Power Station, Units No. 1 and No. 2, Louisa County, Virginia

Date of amendment request: October 19, 1984.

Description of amendment request: The proposed amendments would revise the North Anna Units 1 and 2 (NA-1&2) Technical Specifications (TS) in order to provide leakage integrity tests of the isolation valves in the containment purge lines and the steam jet air ejector system lines each time the containment integrity is established. The proposed changes will help identify excessive degradation of the resilient seats of the system isolation valves. In addition, the TS format for NA-1 would be revised to be consistent with the NA-2 TS format. The proposed change (leakage integrity tests) is in response to NRC Generic Item B-24.

Basis for proposed no significant hazards consideration determination: The Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples (51 FR 7550). Example (ii) of those examples involving no significant hazards considerations, which states "a limitation, restriction or control not presently included in the technical specifications, e.g. a more stringent surveillance requirement," is applicable to the proposed change regarding the performance of additional surveillance on the isolation valves in the containment purge lines.

In addition, the license proposed to reformat the NA-1 TS to be consistent with the format of the NA-2 TS. This proposed change is in accordance with Example (i) of the Commission's guidance, which states that changes which involve "a purely administrative change to the Technical Specifications: for example, a change to achieve consistency throughout the technical specifications, correction of an error or a change in "nomenclature" do not involve a significant hazards consideration.

Therefore, on the basis of the above, the staff proposes to determine that the proposed amendments do not involve significant hazards considerations.

Local Public Document Room location: The Alderman Library, Manuscripts Department, University of Virginia, Charlottesville, Virginia 22901.

Attorney for licensee: Michael. W. Maupin, Esq., Hunton and Williams, P.O. Box 1535, Richmond, Virginia 23212.

NRC Project Director: Herbert N. Berkow.

Virginia Electric and Power Company, Docket Nos. 50-280 and 50-281, Surry Power Station, Units Nos. 1 and 2, Surry County, Virginia

Date of amendment requests: March 20, 1988.

Description of amendment requests: The proposed Technical Specifications (TS) changes will revise Sections 3.14 and 3.23 by imposing additional system operating restrictions on the Main Control Room and Emergency Switchgear Room (MCR and ESGR) Air Conditioning System.

The MCR and ESGR Air Conditioning System has been modified, as an interim measure, to satisfy the design basis assumptions used in the design of the MCR and ESGR Air Conditioning System until a permanent upgrade is implemented in 1990. The interim modifications were made to maintain acceptable temperatures in the control rooms and in the emergency switchgear/relay rooms under normal operation and accident conditions.

The modified system will require the operation of two chillers, two of the four MCR air handling units, and four ESGR air handling units to maintain design temperatures under maximum heat load conditions. Taking credible single failures into consideration requires that redundant equipment be available during operation. As such, the interim limiting conditions for operation will require that three chillers and eight air handling units be operable when at power operation. Further, the interim limiting conditions for operation will require that both drive motors on each ESGR air handling unit be operable. In addition to the equipment restrictions above, a fire watch will be required during this interim period in both unit's ESGR and Mechanical Equipment Room (MER) #3 to address Appendix R considerations.

Action statements will allow that redundant equipment be inoperable for a period not to exceed seven (7) days facilitate preventative and corrective maintenance. If the inoperable equipment is not returned to operable status within seven (7) days, the appropriate reactor unit(s) must be brought to the shutdown condition. The action statements only allow continued operation (i.e., 7-days window) when sufficient equipment is operable to maintain design room temperatures under maximum design heat loads. The action statements require that the appropriate reactor unit(s) be shut down whenever less than the requisite equipment is operable.

Basis for proposed no significant hazards consideration determination: The Commission has provided standards for determining whether a significant hazards consideration exists (10 CFR 50.92(c)). A proposed amendment to an operating license for a facility involves no significant hazards considerations if operation of the facility in accordance with the proposed amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The licensee has determined and the NRC staff agrees that the proposed amendments will not constitute a significant hazards consideration in that:

(1) The implementation of this modification does not significantly increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety and previously evaluated in the Final Safety Analysis Report.

The installation of the interim modification will ensure that design temperatures are maintained under design basis conditions and credible single failure scenarios; therefore, the main control rooms and emergency switchgear rooms will remain at temperatures which afford habitability and reliable equipment operation. The imposition of interim system operating restrictions will ensure that the requisite equipment is operable to maintain design bulk air temperatures.

(2) The implementation of this modification does not create a possibility for an accident or a malfunction of a different type than any evaluated previously in the Final Safety Analysis Report.

The interim modification will uprate the existing air conditioning system to ensure that it will perform its safety related function of maintaining design temperatures in the main control rooms and emergency switchgear rooms during normal and accident conditions. The design considers and accounts for design basis conditions and credible single failure scenarios.

This interim modification requires manual action be taken to energize redundant mechanical equipment which is consistent with the original design basis. The manual action required as a result of this modification must be taken locally at the equipment. However, adequate time is available for local operation of equipment to be accomplished.

(3) The implementation of this modification does not significantly reduce the margin of safety as defined in the basis of any Technical Specification. Although the interim modification requires additional equipment to operate[,] the main control room and emergency switchgear room air conditioning system will be maintained within its design basis. Therefore, [the interim modification ensures] that safety equipment reliability and control room habitability [are] maintained under normal and accident conditions. This modification restores equipment redundancy and provides single failure protection under credible equipment failure scenarios.

Accordingly, the staff proposes to determine that the proposed changes do not involve a significant hazards consideration.

Local Public Document Room location: Swem Library, College of William and Mary, Williamsburg, Virginia 23185.

Attorney for licensee: Michael W. Maupin, Esq., Hunton and Williams, Post Office Box 1535, Richmond, Virginia 23213.

NRC Project Director: Herbert N. Berkow.

Washington Public Power Supply System, Docket No. 50–397, Nuclear Project No. 2, Benton County, Washington

Date of amendment request: March 3, 1989

Description of amendment request: The proposed amendment would revise several sections of the WNP–2 Technical Specifications to include limiting conditions of operation which would be applicable to the fifth cycle of plant operation.

For the fourth refueling of WNP–2, scheduled to begin in April 1989, the licensee will replace 144 of the General Electric (GE) initial core fuel assemblies with ANF reload fuel. The replacement assemblies will include 136 new assemblies of the 8x8 design currently in place in the core. The average enrichment of these assemblies is 2.62 percent uranium-235. Two of these assemblies are of a different cladding design. Four assemblies, also of the 8x8 design, fabricated for cycle four but not utilized, will be placed in the core with this refueling. These four assemblies are of an average enrichment of 2.64 percent U–235.

Also to be placed in the core as part of this refueling are four assemblies of a nine fuel pin by nine fuel pin design. These assemblies are referred to as lead fuel assemblies (LFA) because they represent a new design for the fuel supplier (ANF). The LFAs are also of two different cladding designs. The average enrichment of the LFAs is 2.53 to 2.59 percent U–235. The average enrichment and enrichment distribution within the lead fuel assemblies have been selected to match as closely as possible the neutron characteristics of the 8x8 assemblies included in the reload. The licensee has stated these LFAs will be placed in core locations which have been analyzed to have sufficient margin such that the LFAs are not expected to be the limiting assemblies in the core on either a nodal or a bundle power basis.

Specifically the proposed license amendment would revise Technical Specification (TS) 3.1.3.4, "Four Control Rod Group Scram Insertion Times, Limiting Conditions for Operation," to include the scram time values used by ANF in transient analyses of the reload fuel.

The amendment would revise TS 3.2.1, "Power Distribution Limits, Average Planar Linear heat Generation Rate, Limiting Condition for Operation," TS 3.2.3, "Power Distribution Limits, Minimum Critical Power Ratio," and TS 3.2.4, "Power Distribution Limits, Linear Heat Generation Rate, Limiting Condition for Operation," to include limits which would ensure protection of the LFAs, and would revise TS 5.3.1, "Design Features, Reactor Core, Fuel Assemblies," to show the presence of these lead assemblies.

The amendment also would make an editorial change to Bases section B 2.1.2, "Thermal Power, High Pressure and Low Flow," to delete reference to a table removed by a previous license amendment, and would revise the Index of the Technical Specifications to include items affected by this amendment.

The license amendment application submittal of March 3, 1989 is composed of three documents: WNP-2 Cycle 5 Reload Summary Report, Technical Report No. WPPSS-EANF-124 along with an attachment which provides the pages of the Technical Specifications to be changed; WNP-2 Cycle 5 Plant Transient Analysis, ANF-89-01, and WNP-2 Cycle 5 Reload Analysis, ANF-89-02. The reload report describes the reload fuel and summarizes the safety analyses. The WNP-2 Cycle 5 Reload Analysis Report is intended to be used in conjunction with ANF Topical Report XN-NF-80-19(P)(A), Volume 4, Revision 1, "Application of the ANF Methodology to BWR Reloads." This topical report gives a detailed description of the methods and analyses used.

Basis for Proposed No Significant Hazards Consideration Determination: The Commission has provided standards for determining whether a significant hazards consideration exists as stated in 10 CFR 50.92. A proposed amendment to an operating license for a facility involves no significant hazards considerations if operation of the facility in accordance with a proposed amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) Involve a significant reduction in a margin of safety.

The proposed amendment to the WNP-2, Technical Specifications to support this reload is very similar to Example (iii) provided by the Commission (51 FR 7751, March 6, 1986) of the types of amendments not likely to involve significant hazards considerations. Example (iii) is an amendment to reflect a core reload where:

(1) No fuel assemblies significantly different from those found previously acceptable to the Commission for the previous core at the facility in question are involved;

(2) No significant changes are made to the acceptance criteria for the Technical Specifications;

(3) The analytical methods used to demonstrate conformance with the Technical Specifications and regulations are not significantly changed; and

(4) The NRC has previously found such methods acceptable.

Items 2, 3 and 4 are adhered to explicitly, while the question of the significance of differences in the lead assemblies may merit further examination. The major difference between the four LFAs and the other 760 assemblies in the reactor core is the 9x9 configuration of the LFAs. The 9x9 array provides a smaller reactor coolant flow pathway between the pins. It also affords a larger cladding surface area. The LFAs will use the same channels as the 8x8 fuel assemblies.

As noted above, the enrichment and fuel placement are designed to match the neutron performance characteristics of the ANF 8x8 fuel. The licensee has determined that the insertion of the four 9x9 LFAs will have negligible effects upon core wide transient performance. From specific analyses of operating limits the licensee developed LHGR and MAPLHGR limits applicable to the LFAs. The analyses demonstrated that the remainder of the cycle five operating limits apply to the four LFAs. Analyses were performed consistently with the ANF methodology.

Because the lead assemblies are designed to match the 8x8 fuel, because the safety limits are analyzable using the methodology applicable to the existing fuel, and because the lead assemblies will be placed in the core at locations where they will not be limiting, the Commission finds that the four LFAs are not significantly different from those assemblies previously found acceptable.

In addition to providing examples of amendments not likely to involve a significant hazards consideration, the Commission has provided standards for determining whether no significant hazards consideration exists (10 CFR 50.92(c)). A proposed amendment to an operating license for a facility involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from an accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

On the basis of the evaluation performed in accordance with 10 CFR 50.92, and the fact that the analytical methods used have been approved previously by the NRC staff and do not provide results significantly different, the licensee has concluded, and the staff agrees, that operation of WNP-2 in accordance with the proposed reload amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated because the transient analyses have been reanalyzed for the reload core. The proposed changes to the Technical Specifications reflect new operating limits associated with the reload core and are based on approved analysis methods and are within the current acceptance criteria;

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated because the operation limitations applied to cycle 5 are identical to previous cycles. The values were derived from NRC qualified codes and by applying the most limiting transients throughout the cycle. These limitations are sufficient to ensure the plant is operated within previously accepted conditions. In addition, no changes sufficient to create a new type of malfunction are contemplated; or

(3) Involve a significant reduction in the margin of safety because the margin of safety for all accidents or operational occurences analyzed for cycle 5 operation is either identical to or more conservative than used for previous cycles.

Based on the above considerations the Commission proposes to determine that the requested changs to the WNP-2 Technical Specifications involve no significant hazards considerations.

Local Public Document Room location: Richland City Library, Swift and Northgate Streets, Richland, Washington 99352.

Attorneys for licensees: Nicholas S. Reynolds, Esq., Bishop, Cook, Purcell and Reynolds, 1400 L Street NW., Washington, DC. 20005-3502 and Mr. G.E. Doupe, Esq., Washington Public Power Supply System, P.O. Box 968, 3000 George Washington, Way, Richland, Washington 99352.

NRC Project Director: George W. Knighton

Previously Published Notices of Consideration of Issuance of Amendments to Operating Licenses and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing

The following notices were previously published as separate individual notices. The notice content was the same as above. They were published as individual notices either because time did not allow the Commission to wait for this biweekly notice or because the action involved exigent circumstances. They are repeated here because the biweekly notice lists all amendments issued or proposed to be issued involving no significant hazards consideration.

For details, see the individual notice in the Federal Register on the day and page cited. This notice does not extend the notice period of the original notice.

Commonwealth Edison Company, Docket Nos. 50-237 and 50-249, Dresden Nuclear Power Station, Unit Nos. 2 and 3, Grundy County, Illinois; Docket Nos. 50-254 and 50-265, Quad Cities Station, Unit Nos. 1 and 2, Rock Island County, Illinois

Date of amendment request: February 17 and 21, 1989.

Brief description of amendment: These amendment requests for Dresden and Quad Cities (respectively), revise the "Administrative Controls" Section (Section 6.0) of Technical Specifications (TS) to include: (1) Removal of station and corporate organization charts, (2) Position title changes for Radiation Protection and Chemistry Technicians and Supervisors to reflect a recent division of the Rad/Chem organization into two separate departments, (3) Changes to most of the station and corporate position descriptions, titles, lines of authority, and responsibilities, and (4) Miscellaneous typographical and editorial changes.

Date of publication of individual notice in Federal Register: March 15, 1939 (54 FR 10762).

Expiration date of individual notice: April 14, 1989.

Local Public Document Room location: For Dresden Station, the Morris Public Library, 604 Liberty Street, Morris, Illinois 60450; for Quad Cities Station, Dixon Public Library, 221 Hennepin Avenue, Dixon, Illinois 61021.

Commonwealth Edison Company, Docket Nos. 50-373 and 50-374, LaSalle County Station, Unit Nos. 1 and 2, LaSalle County, Illinois

Date of amendments request: February 17, 1989

Brief description of amendments: The amendments would revise Section 6.0 of the Technical Specifications by removal of the organizational figures, a position change from Radiation Chemistry Technician to Radiation Protection Technician, several position title changes, and a clarification to the distribution requirements for Onsite Reviews.

Date of publication of individual notice in Federal Register: March 15, 1989 (54 FR 10762)

Expiration date of individual notice: April 14, 1989

Local Public Document Room location: Public Library of Illinois Valley Comminity College, Rural Route No. 1, Oglesby, Illinois 61348.

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

During the period since publication of the last biweekly notice, the Commission has issued the following amendments. The Commission has determined for each of these amendments that the application complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set fourth in the license amendment.

Notice of Consideration of Issuance of Amendment to Facility Operating License and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing in connection with these actions was published in the Federal Register as indicated. No request for a hearing or petition for leave to intervene was filed following this notice.

Unless otherwise indicated, the Commission has determined that these amendments satisfy the criteria for categorical exclusion in accordance with 10 CFR 51.22. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared for these amendments. If the Commission has prepared an environmental assessment under the special circumstances provision in 10 CFR 51.12(b) and has made a determination based on that assessment, it is so indicated.

For further details with respect to the action see (1) the applications for amendments, (2) the amendments, and (3) the Commission's related letters, Safety Evaluations and/or Environmental Assessments as indicated. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document rooms for the particular facilities involved. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Director, Division of Reactor Projects.

Arkansas Power & Light Company, Docket No. 50-313, Arkansas Nuclear One, Unti 1, Pope County, Arkansas

Date of amendment request: November 17, 1986 and Janaury 13, 1989 as supplemented.

Brief description of amendment: This amendment revised the ANO-1 Technical Specifications to reflect changes in reporting requirements of 10 CFR 50.72 and 50.73 in accordance with NRC Generic Letter 83-43.

Date of issuance: March 21, 1989.

Effective date: March 21, 1989.

Amendment No.: 118.

Facility Operating License No. DPR-51: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: February 11, 1987 (52 FR 4403).

The January 13, 1989 submittal provided additional clarifying information and did not change the finding of the initial notice.

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 21, 1989.

No significant hazards consideration comments received: No.

Local Public Document Room location: Tomlinson Library, Arkansas Tech University, Russellville, Arkansas 72801

Baltimore Gas and Electric Company, Docket No. 50-318, Calvert Cliffs Nuclear Power Plant, Unit No. 2, Calvert County, Maryland

Date of application for amendment: December 14, 1988 as supplemented February 17, 1989

Brief description of amendments: This amendment provides a temporary, one-time 28-day extension to the surveillance interval, required by TS Surveillance Requirement 4.6.1.2.d, for the performance of each individual Type B or C containment local leak rate test.

This temporary change shall expire upon reaching 199.9°F average reactor coolant system (RCS) temperature during initial RCS heatup following the Unit 2 Cycle 9 refueling outage and then the specified maximum surveillance interval shall revert to the normally required 24-month period.

Date of issuance: March 15, 1989.

Effective date: March 15, 1989.

Amendment No.: 118.

Facility Operating License No. DPR-69: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: January 3, 1989 (54 FR 4354).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 15, 1989.

No significant hazards consideration comments received: No.

Local Public Document Room location: Calvert County Library, Prince Frederick, Maryland.

NRC Project Director: Robert A. Capra.

Carolina Power & Light Company, et al., Docket Nos. 50-325 and 50-324, Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina

Date of application for amendments: October 28, 1988, as supplemented March 6, 1989.

Description of amendments: The amendment revises the Technical Specifications by modifying footnote * * *, in Table 1.2, "Operational Conditions." The revised footnote allows the reactor mode switch to be placed in the Refuel position while a single control rod is being moved, as opposed to only when being recoupled provided the one-rod-out interlock is operable.

Date of issuance: March 14, 1989.

Effective date: March 14, 1989.

Amendment Nos.: 125 and 155.

Facility Operating License Nos. DPR-71 and DPR-62: Amendments revise the Technical Specifications.

Date of initial notice in Federal Register: January 11, 1989 (54 FR 1019) Additional information of a clarifying nature was submitted by the licensee by letter dated March 6, 1989. The additional information did not alter the action noticed and did not effect the staff's proposed no significant hazards considerstion determination. The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 14, 1989. No significant hazards consideration comments received: No.

Local Public Document Room location: University of North Carolina at Wilmington, William Madison Randall Library, 601 S. College Road, Wilmington, North Carolina 28403-3297.

Carolina Power & Light Company, et al., Docket Nos. 50-325 and 50-324, Brunswick Steam Electric Plant, Units 1 and 2, Brunswick County, North Carolina

Date of application for amendments: August 23, 1988.

Description of amendments: The amendments change Technical Specification Tables 3.3.5.7-1 for Units 1 and 2 to reflect the modification of the present fire detection system for the diesel generator cells. The detection system will be modified by replacing the present smoke detectors with a combination of heat and flame detectors.

Date of issuance: March 20, 1989.

Effective date: March 20, 1989.

Amendment Nos.: 126 and 156.

Facility Operating License No. DPR-71 and DPR-62: Amendments revise the Technical Specifications.

Date of initial notice in Federal Register: February 1, 1989 (54 FR 5161)

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 20, 1989.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of North Carolina at Wilmington, William Madison Randall Library, 601, S. College Road, Wilmington, North Carolina 28403-3297.

Duquesne Light Company, Docket No. 50-334, Beaver Valley Power Station, Unit No. 1, Shippingport, Pennsylvania

Date of application for amendments: September 7, 1988.

Brief description of amendments: The amendment imposes requirements on the incore thermocouples and the reactor vessel level indicating system (RVLIS). In addition, the existing specification on the subcooling margin monitor is revised to require at least one channel operable (from zero required).

Date of issuance: March 13, 1989.

Effective date: March 13, 1989.

Amendment No.: 137.

Facility Operating License No. DPR-66: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: October 19, 1988 (53 FR 40985).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 13, 1989.

No significant hazards consideration comments received: No.

Local Public Document Room location: B.F. Jones Memorial Library, 663 Franklin Avenue, Aliquippa, Pennsylvania 15001.

Florida Power and Light Company, et al., Docket No. 50-389, St. Lucie Plant, Unit No. 2, St. Lucie County, Florida

Date of application for amendment: September 1, 1988.

Brief description of amendment: This amendment changed the Technical Specifications associated with the boric acid makeup (BAMU) system. Specifically, the required boron concentration requirements were reduced, the borated water volume was increased, and the requirement to heat trace the BAMU was deleted.

Date of issuance: March 13, 1989.

Effective date: March 13, 1989.

Amendment No.: 40.

Facility Operating License No. NPF-16: Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: October 5, 1988 (53 FR 39169).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 13, 1989.

No significant hazards consideration comments received: No.

Local Public Document Room location: Indian River Junior College Library, 3209 Virginia Avenue, Ft. Pierce, Florida.

Indian Michigan Power Company, Dockets Nos. 50-315 and 50-316, Donald C. Cook Nuclear Plant, Units Nos. 1 and 2, Berrien County, Michigan

Date of application for amendments: December 2, 1986, November 23, 1987 and July 21, 1988.

Brief description of amendments: The amendments modified paragraphs 2.D of the licenses to require compliance with the amended Physical Security Plan. This Plan was amended to conform to the requirements of 10 CFR 73.55. Consistent with the provisions of 10 CFR 73.55, search requirements must be implemented within 60 days and miscellaneous amendments within 180 days from the effective date of these amendments.

Date of issuance: March 15, 1989.

Effective date: March 15, 1989.

Amendment Nos.: 122, 109.

Facility Operating License No. DPR-58 and DPR-74: These amendments revised the licenses.

Date of initial notice in Federal Register: December 30, 1988 (53 FR 53093). The Commission's related evaluation of the amendments is contained in a letter to Indiana Michigan Power Company dated March 15, 1989, and a Safeguards Evaluation Report dated March 15, 1989.

No significant hazards consideration comments received: No.

Local Public Document Room location: Maude Preston Palenske Memorial Library, 500 Market Street, St. Joseph, Michigan 49085.

Mississippi Power & Light Company, System Energy Resources, Inc., South Mississippi Electric Power Association, Docket No. 50-416, Grand Gulf Nuclear Station, Unit 1, Claiborne County, Mississippi

Date of application for amendment: January 26, 1989, as supplemented February 20, March 3, and March 6, 1989.

Brief description of amendment: The amendment provides one-time exceptions to TS 3.0.4 approved for use only during the third refueling outage. The exceptions will allow entry into specified operational conditions without meeting the Limiting Condition for Operation, provided the requirements of the associated action statements are met. Those TS affected are:

a. Residual Heat Removal—Cold Shutdown, TS 3.4.4.9.2, ACTIONs a and c

b. ECCS—Shutdown, TS 3.5.2, ACTION a

c. Suppression Pool, TS 3.5.3, ACTION c

d. Containment and Drywell Isolation Valves, TS 3.6.4, ACTIONs b and c

e. Secondary Containment Automatic Isolation Dampers/Valves, TS 3.6.6.2, ACTIONs b and c

f. Standby Service Water system, TS 3.7.1.1, ACTIONs b, c, and d.

g. Ultimate Heat Sink, TS 3.7.1.3, ACTION a.

h. Control Room Emergency Filtration System, TS 3.7.2, ACTION b.1.

i. Residual Heat Removal and Coolant Circulation—Low Water, TS 3.9.11.2, ACTIONs a and b.

Date of issuance: March 16, 1989.

Effective date: March 16, 1989.

Amendment No. 58.

Facility Operating License No. NPF-29. This amendment revises the Technical Specifications and/or License.

Date of initial notice in Federal Register: February 8, 1989 (54 FR 6199).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 16, 1989.

No significant hazards consideration comments received: No.

Local Public Document Room location: Hinds Junior College, McLendon Library, Raymond, Mississippi 29154.

Northeast Nuclear Energy Company, et al., Docket No. 50-336, Millstone Nuclear Power Station, Unit No. 2, New London County, Connecticut

Date of application for amendment: November 15, 1988 and February 1, 1989.

Brief description of amendment: This amendment allows operation of Millstone unit 2 for Cycle 10. The changes to the Technical Specifications reflect a revised safety analysis that includes the use of fuel designed and fabricated by Advanced Nuclear Fuels Corporation. Fuel designed and fabricated by ANF has not been previously utilized for Millstone Unit 2. The changes to the Technical Specifications also reflect the effects of reduced reactor coolant flow from 340,000 to 325,000 gpm.

Date of issuance: March 20, 1989.

Effective date: March 20, 1989.

Amendment No.: 139.

Facility Operating License No. DPR-65. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: January 24, 1989 (54 FR 3545) and February 15, 1989 (54 FR 6977).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 20, 1989.

No significant hazards consideration comments received: No.

Local Public Document Room location: Waterford Public Library, 49 Rope Ferry Road, Waterford, Connecticut 06385.

Northeast Nuclear Energy Company, et al., Docket No. 50-423, Millstone Nuclear Power Station, Unit No. 3, New London County, Connecticut

Date of application for amendment: June 15, 1988.

Brief description of amendment: The amendment revises Technical Specification Tables 2.2-1 and 3.3-4 to decrease the reactor trip setpoint and the engineered safety features actuation setpoints for auxiliary feedwater initiation identified as steam generator water level low-low from 23.5% to 18.10% of the narrow range instrument span. These changes increase the margin between the steam generator water level low-low trip setpoint and normal operating band and reflects the results of a revised calculation of the errors associated with related instrumentation.

Date of issuance: March 14, 1989.

Effective date: March 14, 1989.

Amendment No.: 31.

Facility Operating License No. NPF-49. Amendment revised the technical Specifications.

Date of initial notice in Federal Register: July 27, 1988 (53 FR 28293).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 14, 1989.

No significant hazards consideration comments received: No.

Local Public Document Room location: Waterford Public Library, 49 Rope Ferry Road, Waterford, Connecticut 06385.

Philadelphia Electric Company, Docket No. 50-352, Limerick Generating Station, Unit 1, Montgomery County, Pennsylvania

Date of application for amendment: December 14, 1988.

Brief description of amendment: The amendment changed the Technical Specifications to permit removal of the Rod Sequence Control System and to reduce the Rod Worth Minimizer low power setpoint.

Date of issuance: March 22, 1989.

Effective date: March 22, 1989.

Amendment No. 17

Facility Operating License No. NPF-39. This amendment revised the Technical Specifications.

Date of initial notice in Federal Register: February 1, 1989 (54 FR 5172).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 22, 1989.

No significant hazards consideration comments received: No.

Local Public Document Room location: Pottstown Public Library, 500 High Street, Pottstown, Pennsylvania 19464.

Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Docket Nos. 50-277 and 50-278, Peach Bottom Atomic Power Station, Unit Nos. 2 and 3, York County, Pennsylvania

Date of application for amendments: October 21, 1988 as supplemented on November 30, 1988.

Brief description of amendments: These amendments revised the minimum count rate required on the source range monitors for the withdrawal of control rods for startup.

Date of issuance: March 15, 1989.

Effective date: March 15, 1989.

Amendments Nos.: 140 and 142.

Facility Operating License Nos. DPR-44 and DPR-56: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: December 30, 1988 (53 FR 53096).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 15, 1989.

No significant hazards consideration comments received: No.

Local Public Document Room location: Government Publications Section, State Library of Pennsylvania, Education Building, Commonwealth and Walnut Streets, Harrisburg, Pennsylvania 17126.

Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Docket Nos. 50-277 and 50-278, Peach Bottom Atomic Power Station, Units Nos. 2 and 3, York County, Pennsylvania

Date of application for amendments: June 12, 1987 as amended February 7, 1989.

Brief description of amendments: These amendments revised the Technical Specification Limiting Condition for Operation, Surveillance Requirements and BASES to reflect the incorporation of Recirculation Pump Trip and Alternate Rod insertion (Injection) features that are consistent with the requirements of 10 CFR 50.62 C(3) and C(5) as reported in the staff's safety evaluation dated December 21, 1988.

Date of issuance: March 22, 1989.

Effective date: Unit 2 prior to startup in Cycle 8; Unit 3 prior to startup in Cycle 8.

Amendments Nos.: 141 and 143.

Facility Operating License Nos. DPR-44 and DPR-56: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: February 17, 1989 (54 FR 7313).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 22, 1989.

No significant hazards consideration comments received: No.

Local Public Document Room location: Government Publications Section, State Library of Pennsylvania, Education Building, Commonwealth and Walnut Streets, Harrisburg, Pennsylvania 17126.

Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, Docket Nos. 50-277 and 50-278, Peach Bottom Atomic Power Station, Unit Nos. 2 and 3, York County, Pennsylvania

Date of application for amendments: February 11, 1982, as amended on August 24, 1983, November 1, 1985, September 30, 1986, September 8, 1987 and September 7, 1988.

Brief description of amendments: These amendments changed the Technical Specification Administrative controls to reflect the addition of restrictions on the use of overtime for plant personnel who perform safety related functions.

Date if issuance: March 22, 1989.

Effective date: March 22, 1989.

Amendments Nos.: 142 and 144.

Facility Operating License Nos. DPR-44 and DPR-56: Amendments revised the Technical Specifications.

Date of initial notice in Federal Register: October 19, 1988 (53 FR 40998) The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 22, 1989.

No significant hazards consideration comments received: No.

Local Public Document Room location: Government Publications Section, State Library of Pennsylvania, Education Building, Commonwealth and Walnut Streets, Harrisburg, Pennsylvania 17126.

Public Service Electric & Gas Company, Docket Nos. 50-272 and 50-311, Salem Generating Station, Unit Nos. 1 and 2, Salem County, New Jersey

Date of application for amendments: January 3, 1989 and supplemented by letter dated February 16, 1989.

Brief description of amendments: These amendments redefined the FULLY WITHDRAWN position of the control rod cluster assemblies from 228 steps to a band between 222 and 228 steps withdrawn. Other changes: deleted Figure 3.1.2 from Unit 1 Technical Specifications; deleted Specification 3.10.5 from Unit 2 Technical Specifications; and incorporated rod testing requirements into Specification 3.1.3.2.2 for Unit 1 and Unit 2 Technical Specifications.

Date of issuance: March 22, 1989.

Effective date: For Unit 1: Effective as of startup from the eighth refueling outage, currently scheduled to begin April 1989. For Unit 2: Effective as of startup from the fifth refueling outage, currently scheduled to begin January 1990.

Amendment Nos. 91 and 66.

Facility Operating License Nos. DPR-70 and DPR-75. These amendments revised the Technical Specifications.

Date of initial notice in Federal Register: February 8, 1989 (54 FR 6208).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated March 22, 1989.

No significant hazards consideration comments received: No.

Local Public Document Room location: Salem Free Public Library, 112 West Broadway, Salem, New Jersey 08079.

Wisconsin Public Service Corporation, Docket No. 50-305, Kewaunee Nuclear Power Plant, Kewaunee County, Wisconsin

Date of application for amendment: November 29, 1988, supplemented November 30, 1988.

Brief description of amendment: The amendment reflected personnel changes, corrected typographical errors, and made minor word changes to clarify the intent of Technical Specifications (TS).

Date of issuance: March 13, 1989.

Effective date: March 13, 1989.

Amendment No.: 81.

Facility Operating License No. NPF-30. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: February 8, 1989 (54 FR 6213).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 13, 1989.

No significant hazards consideration comments received: No.

Local Public Document Room location: University of Wisconsin Library Learning Center, 2420 Nicolet Drive, Green Bay, Wisconsin 54301.

Wolf Creek Nuclear Operating Corporation, Docket No. 50-482, Wolf Creek Generating Station, Coffey County, Kansas

Date of amendment requests: May 7, 1987.

Brief description of amendment: The amendment allows plant operation to continue for 72 hours for diagnosis and repair, for the case where one or more control rod assemblies are electrically inoperable.

Date of Issuance: March 6, 1989.

Effective date: March 6, 1989.

Amendment No.: 27.

Facility Operating License No. NPF-42. Amendment revised the Technical Specifications.

Date of initial notice in Federal Register: May 20, 1987 (52 FR 18991).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated March 6, 1989.

No significant hazards consideration comments received: No.

Local Public Document Room Location: Emporia State University, William Allen White Library, 1200 Commercial Street, Emporia, Kansas 66801 and Washburn University School of Law Library, Topeka, Kansas 66621.

Dated at Rockville, Maryland, this 30th day of March 1989.

For the Nuclear Regulatory Commission.

B.D. Liaw,

Acting Associate Director for Special Projects, Office of Nuclear Reactor Regulation.

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personnel system changes designed to bring about a more participatory work environment (PWE) in a Federal installation. Private sector organizations have used SBP systems and PWE concepts successfully in a variety of settings to achieve greater workforce flexibility, leaner staffing, higher quality output, greater long term productivity, and enhanced employee motivation and commitment.

Participating Organization

Defense Depot Ogden, Utah (DDOU) will be the test site for this demonstration project, which is projected to run for 5 years. DDOU is one of 6 DLA sites performing similar functions.

Types and Numbers of Participating Employees

The demonstration will cover all permanent employees at DDOU, which (as of October, 1988) includes 531 non-supervisory General Schedule (GS) employees, 93 supervisory GS and Performance Management and Recognition (PMRS) employees, 887 non-supervisory Federal Wage System (FWS) employees, and 59 Wage Supervisors, for a total of 1570 eligible employees.

Labor Participation

Employees at DDOU are represented by the American Federation of Government Employees (AFGE) which has exclusive recognition for the total depot work force. AFGE has been involved in the development of the project since its inception and the president of AFGE Local 2721 is a member of the project steering committee.

Methodology

Under the demonstration project, DDOU will implement a system of interrelated organizational structure and personnel system changes. Specific interventions will include:

(a) Restructuring the workforce into teams of multi-skilled individuals, such that each team is responsible for a total product, service, or process; all team members will have equal opportunity to learn and perform all the job skills necessary to achieve the team's objectives.

(b) Classifying positions at the team, rather than the individual, level based on a simplified system that integrates work previously classified under both the General Schedule and the Federal Wage System; work performed by teams will be classified based on 5 levels of difficulty.

(c) Establishing a skill-based compensation system consisting of 5 pay bands; advances within a pay band will be based on the acquisition of necessary skills and knowledge, meeting appropriate minimum time requirements, and having a Fully Successful or higher performance rating.

(d) Implementing a performance evaluation system that uses 3 summary performance levels and incorporates subordinate and peer input; the system will require ratings of both team and individual performance.

(e) Using an incentive/recognition program that includes productivity gainsharing for all employees and the option of paid leave in lieu of monetary incentives.

(f) Establishing a formal job knowledge certification program.

(g) Creating an alternative disciplinary procedure for minor offenses under which managers may substitute non-punitive "letters of discipline" for formal disciplinary procedures.

Training and Implementation

This project will represent a significant change to management and employee practices, and adequate training will be essential to successfully transform programs and attitudes. Training will be provided to all employees in project concepts and changes. Team leaders and managers will receive additional training in leadership skills, group dynamics, team building, problem solving, and communications.

Employee Protection

The project will provide for full employee protection on entry into and exit from the demonstration system. In no case will an individual's current salary be reduced as a result of these changes.

Evaluation

A comprehensive and methodologically rigorous evaluation will be conducted by an external evaluator with Office of Personnel Management (OPM) oversight. The objectives of the evaluation will be to assess project outcomes and determine the applicability of project changes to other Federal installations.

Benefits of the Proposed Project

The project is expected to demonstrate that an SBP system can be successfully implemented in a Federal installation in a way that increases employee involvement in work-related decisions, enhances the flexibility of the workforce, improves quality and on-time performance, and reduces operating costs.

II. Introduction

A. Background

Over the last several years, an innovative approach to employee compensation which encourages and rewards employee growth and skill development has evolved in the private sector. This alternative to traditional compensation systems is known by a variety of names. Frequently called Skill-Based Pay (SBP) or Pay For Knowledge, it bases employee pay on the acquisition of required skills rather than on the performance of a specific job.

Through the integration of SBP plans into the organization, companies such as General Motors, Procter & Gamble, Sherwin-Williams and others have improved the skills and abilities of their work force, enhanced their ability to shift employees according to workload demands, and have reduced total employment levels.

In addition to SBP, industry has introduced team structures and has used participatory work practices to enrich the work experience. These work innovations have improved both employee and organizational performance, e.g., employees gain opportunities for challenge, achievement, growth and development while organizational productivity is enhanced through increased output, quality, lower costs, and improved employee morale and motivation.

The intent of this project is to demonstrate that private sector flexibility in resource management can be applied to the public sector. It is also expected that the creation of an enriched work environment will stimulate the organization to respond creatively to the need for increased efficiency and effectiveness. No aspect of implementation will be effected until all appropriate bargaining responsibilities with the union have been met.

This project is an operational approach to effect major modifications to the existing personnel management system in support of the Defense Logistics Agency's (DLA) Logistics System Modernization Program. These changes are intended to move DLA and its demonstration site, Defense Depot Ogden, Utah toward greater operational flexibility. The plan encompasses 2 successful strategies employed by private industry: skill-based compensation and participatory work practices.