Docket Nos. 50-352
and 50-353

August 20, 1992

Mr. George J. Beck
Manager-Licensing, MC 52A-5
Philadelphia Electric Company
Nuclear Group Headquarters
Correspondence Control Desk
P.O. Box No. 195
Wayne, Pennsylvania 19087-0195

Dear Mr. Beck:

SUBJECT: 24-MONTH FUEL CYCLE, LIMERICK GENERATING STATION, UNITS 1 AND 2 (TSCR NO. 92-01-0) (TAC NOS. M83334 AND M83335)

The Commission has issued the enclosed Amendment No. 56 to Facility Operating License No. NPF-39 and Amendment No. 21 to Facility Operating License No. NPF-85 for the Limerick Generating Station, Units 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated May 8, 1991 and supplemented by letter dated May 15, 1992.

These amendments change the TSs to 1) revise the channel calibration frequency for the peak acceleration seismic monitoring recorder mounted on the reactor vessel head flange from 18 to 24 months and 2) revise the frequency of surveillance testing of the main steam safety relief valves from 18 to 24 months.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,
S. Barber for
Richard J. Clark, Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 56 to
License No. NPF-39
Amendment No. 21 to
License No. NPF-85
2. Safety Evaluation

cc w/enclosures:
See next page

DISTRIBUTION:

Docket File	MO'Brien(2)	CGrimes, 11E21	SNewberry
NRC & Local PDRs	RClark/JShea	ACRS(10)	JNorberg
PDI-2 Reading	OGC	OPA	
SVarga	DHagan, 3206	OC/LFMB	
JCalvo	GHill(4), P1-22	EWenzinger, RGN-I	
CMiller	Wanda Jones, 7103	BRuland, RGN-I	

CP-1

*Previously Concurred

For CM

OFC	:PDI-2/LA	:PDI-2/PM	:OGC	:SICB/C*	:C/EMCB*	:D/PDI-2
NAME	:MO'Brien	:RClark:tlc:		:SNewberry	:JNorberg	:CMiller
DATE	:7/30/92	:07/30/92	:8/ /92	:07/21/92	:07/20/92	:8/19/92

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

August 20, 1992

Docket Nos. 50-352
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Mr. George J. Beck
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Philadelphia Electric Company
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SUBJECT: 24-MONTH FUEL CYCLE, LIMERICK GENERATING STATION, UNITS 1 AND 2 (TSCR NO. 92-01-0) (TAC NOS. M83334 AND M83335)

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These amendments change the TSs to 1) revise the channel calibration frequency for the peak acceleration seismic monitoring recorder mounted on the reactor vessel head flange from 18 to 24 months and 2) revise the frequency of surveillance testing of the main steam safety relief valves from 18 to 24 months.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

G. S. Barber for

Richard J. Clark, Senior Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 56 to License No. NPF-39
   Amendment No.21 to License No. NPF-85
2. Safety Evaluation

cc w/enclosures:
See next page

cc:

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PHILADELPHIA ELECTRIC COMPANY

DOCKET NO. 50-352

LIMERICK GENERATING STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 56
License No. NPF-39

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Philadelphia Electric Company (the licensee) dated May 8, 1991, as supplemented by letter dated May 15, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-39 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 56 , are hereby incorporated into this license. Philadelphia Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and is to be implemented within 30 days of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James C. Stone

For Charles L. Miller, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the
Technical Specifications

Date of Issuance: August 20, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 56

FACILITY OPERATING LICENSE NO. NPF-39

DOCKET NO. 50-352

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. Overleaf pages are provided to maintain document completeness.*

Remove	Insert
3/4 3-71	3/4 3-71*
3/4 3-72	3/4 3-72
3/4 4-7	3/4 4-7
3/4 4-8	3/4 4-8*

TABLE 4.3.7.2-1

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Triaxial Time-History Accelerographs (T/A's)			
a. Sensors			
1) XE-VA-102 Primary Containment Foundation (Loc. 109-R15-177)	N.A.	SA	R
2) XE-VA-103 Containment Structure (Diaphragm Slab)	N.A.	SA	R
3) XE-VA-104 Reactor Enclosure Foundation (Loc. 111-R11-177)	N.A.	SA	R
4) XE-VA-105 Reactor Piping Support (Mn. Stm. Line 'D,' El 313', in containment)	N.A.	SA	R
5) XE-VA-106 Outside Containment on Seismic Category I Equipment, (RHR Heat Exchanger, Loc. 102-R15-177)	N.A.	SA	R
6) XRSH-VA-107* Foundation of an Independent Seismic Category I Structure (Spray Pond Pump House, El 237')	N.A.	SA	R
b. Recorders (Panel 00C693)			
1) XR-VA-102 for XE-VA-102	N.A.	SA	R
2) XR-VA-103 for XE-VA-103	N.A.	SA	R
3) XR-VA-104 for XE-VA-104	N.A.	SA	R
4) XR-VA-105 for XE-VA-105	N.A.	SA	R
5) XR-VA-106 for XE-VA-106	N.A.	SA	R

*Includes sensor, trigger, recorder, and backup power supply.

TABLE 4.3.7.2-1 (Continued)

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
c. Triaxial Seismic Trigger (S/T)			
1) XSH-VA-001 (Activates Items 1.b.1) thru 5) above)	N.A.	SA	R
2. Triaxial Peak Recording Accelerograph (P/A's)			
a. XR-VA-151 Reactor Equipment (Top of reactor vessel head)	N.A.	N.A.	*
b. XR-VA-152 Reactor Piping (Mn. Stm. Line 'D,' El 313', in containment)	N.A.	N.A.	R
c. XR-VA-153 Reactor Equipment Outside Containment (RHR Heat Exchanger, Loc. 203-R15-201)	N.A.	N.A.	R
3. Triaxial Seismic Switches			
a. XSHH-VA-001 Primary Containment Foundation (Loc. 118-R16-177)	N.A.	SA	R
4. Triaxial Response Spectrum Analyzer (RSA)	N.A.	SA	R

* The calibration frequency for this instrument is once per 24 months.

3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2 The safety valve function of at least 11 of the following reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift settings:*#

4 safety/relief valves @ 1130 psig $\pm$1%
5 safety/relief valves @ 1140 psig $\pm$1%
5 safety/relief valves @ 1150 psig $\pm$1%

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the safety valve function of one or more of the above required safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With one or more safety/relief valves stuck open, provided that suppression pool average water temperature is less than 105°F, close the stuck open safety/relief valve(s); if unable to close the stuck open valve(s) within 2 minutes or if suppression pool average water temperature is 110°F or greater, place the reactor mode switch in the Shutdown position.

c. With one or more safety/relief valve acoustic monitors inoperable, restore the inoperable acoustic monitors to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.1 The acoustic monitor for each safety/relief valve shall be demonstrated OPERABLE with the setpoint verified to be 0.20 of the full open noise level## by performance of a:

a. CHANNEL FUNCTIONAL TEST at least once per 31 days, and a
b. CHANNEL CALIBRATION at least once per 18 months**.

4.4.2.2 At least 1/2 of the safety relief valves shall be removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations at least once per 24 months, and they shall be rotated such that all 14 safety relief valves are removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations at least once per 54 months.

---

* The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

** The provisions of Specification 4.0.4 are not applicable provided the Surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

\# Up to 2 inoperable valves may be replaced with spare OPERABLE valves with lower setpoints until the next refueling.

\## Initial setting shall be in accordance with the manufacturer's recommendation. Adjustment to the valve full open noise level shall be accomplished during the startup test program.

REACTOR COOLANT SYSTEM

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.3.1 The following reactor coolant system leakage detection systems shall be OPERABLE:

a. The primary containment atmosphere gaseous radioactivity monitoring system,

b. The drywell floor drain sump and drywell equipment drain tank flow monitoring system,

c. The drywell unit coolers condensate flow rate monitoring system, and

d. The primary containment pressure and temperature monitoring system.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.*

ACTION:

With only three of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous radioactive monitoring system, primary containment pressure and temperature monitoring system and/or the drywell unit coolers condensate flow rate monitoring system is inoperable; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The reactor coolant system leakage detection systems shall be demonstrated OPERABLE by:

a. Primary containment atmosphere gaseous radioactivity monitoring systems-performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

b. The primary containment pressure shall be monitored at least once per 12 hours and the primary containment temperature shall be monitored at least once per 24 hours.

c. Drywell floor drain sump and Drywell equipment drain tank flow monitoring system-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION TEST at least once per 18 months.

d. Drywell unit coolers condensate flow rate monitoring system-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

*The primary containment atmosphere gaseous radioactivity monitor is not required to be OPERABLE until OPERATIONAL CONDITION 2.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

PHILADELPHIA ELECTRIC COMPANY

DOCKET NO. 50-353

LIMERICK GENERATING STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 21
License No. NPF-85

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Philadelphia Electric Company (the licensee) dated May 8, 1991, as supplemented by letter dated May 15, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-85 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 21 , are hereby incorporated into this license. Philadelphia Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and is to be implemented within 30 days of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

For James C. Stone

Charles L. Miller, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the
Technical Specifications

Date of Issuance: August 20, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 21

FACILITY OPERATING LICENSE NO. NPF-85

DOCKET NO. 50-353

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. Overleaf pages are provided to maintain document completeness.*

Remove	Insert
3/4 3-71	3/4 3-71*
3/4 3-72	3/4 3-72
3/4 4-7	3/4 4-7
3/4 4-8	3/4 4-8*

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SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Triaxial Time-History Accelerographs (T/A's)			
a. Sensors			
1) XE-VA-102 Primary Containment Foundation (Loc. 109-R15-177)	N.A.	SA	R
2) XE-VA-103 Containment Structure (Diaphragm Slab)	N.A.	SA	R
3) XE-VA-104 Reactor Enclosure Foundation (Loc. 111-R11-177)	N.A.	SA	R
4) XE-VA-105 Reactor Piping Support (Mn. Stm. Line 'D,' El 313', in containment)	N.A.	SA	R
5) XE-VA-106 Outside Containment on Seismic Category I Equipment, (RHR Heat Exchanger, Loc. 102-R15-177)	N.A.	SA	R
6) XRSH-VA-107* Foundation of an Independent Seismic Category I Structure (Spray Pond Pump House, El 237')	N.A.	SA	R
b. Recorders (Panel 00C693)			
1) XR-VA-102 for XE-VA-102	N.A.	SA	R
2) XR-VA-103 for XE-VA-103	N.A.	SA	R
3) XR-VA-104 for XE-VA-104	N.A.	SA	R
4) XR-VA-105 for XE-VA-105	N.A.	SA	R
5) XR-VA-106 for XE-VA-106	N.A.	SA	R

*Includes sensor, trigger, recorder, and backup power supply.

TABLE 4.3.7.2-1 (Continued)

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
c. Triaxial Seismic Trigger (S/T)			
1) XSH-VA-001 (Activates Items 1.b.1) thru 5) above)	N.A.	SA	R
2. Triaxial Peak Recording Accelerograph (P/A's)			
a. XR-VA-151 Reactor Equipment (Top of reactor vessel head)	N.A.	N.A.	*
b. XR-VA-152 Reactor Piping (Mn. Stm. Line 'D,' El 313', in containment)	N.A.	N.A.	R
c. XR-VA-153 Reactor Equipment Outside Containment (RHR Heat Exchanger, Loc. 203-R15-201)	N.A.	N.A.	R
3. Triaxial Seismic Switches			
a. XSHH-VA-001 Primary Containment Foundation (Loc. 118-R16-177)	N.A.	SA	R
4. Triaxial Response Spectrum Analyzer (RSA)	N.A.	SA	R

* The calibration frequency for this instrument is once per 24 months.

3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2 The safety valve function of at least 11 of the following reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift settings:*#

4 safety/relief valves @ 1130 psig ±1%
5 safety/relief valves @ 1140 psig ±1%
5 safety/relief valves @ 1150 psig ±1%

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the safety valve function of one or more of the above required safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With one or more safety/relief valves stuck open, provided that suppression pool average water temperature is less than 105°F, close the stuck open safety/relief valve(s); if unable to close the stuck open valve(s) within 2 minutes or if suppression pool average water temperature is 110°F or greater, place the reactor mode switch in the Shutdown position.

c. With one or more safety/relief valve acoustic monitors inoperable, restore the inoperable acoustic monitors to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.1 The acoustic monitor for each safety/relief valve shall be demonstrated OPERABLE with the setpoint verified to be 0.20 of the full open noise level## by performance of a:

a. CHANNEL FUNCTIONAL TEST at least once per 31 days, and a
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4.4.2.2 At least 1/2 of the safety relief valves shall be removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations at least once per 24 months, and they shall be rotated such that all 14 safety relief valves are removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations at least once per 54 months.

* The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.
** The provisions of Specification 4.0.4 are not applicable provided the Surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.
\# Up to 2 inoperable valves may be replaced with spare OPERABLE valves with lower setpoints until the next refueling.
\## Initial setting shall be in accordance with the manufacturer's recommendation. Adjustment to the valve full open noise level shall be accomplished during the startup test program.

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.3.1 The following reactor coolant system leakage detection systems shall be OPERABLE:

a. The primary containment atmosphere gaseous radioactivity monitoring system,

b. The drywell floor drain sump and drywell equipment drain tank flow monitoring system,

c. The drywell unit coolers condensate flow rate monitoring system, and

d. The primary containment pressure and temperature monitoring system.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.*

ACTION:

With only three of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous radioactive monitoring system, primary containment pressure and temperature monitoring system and/or the drywell unit coolers condensate flow rate monitoring system is inoperable; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The reactor coolant system leakage detection systems shall be demonstrated OPERABLE by:

a. Primary containment atmosphere gaseous radioactivity monitoring systems-performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

b. The primary containment pressure shall be monitored at least once per 12 hours and the primary containment temperature shall be monitored at least once per 24 hours.

c. Drywell floor drain sump and Drywell equipment drain tank flow monitoring system-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION TEST at least once per 18 months.

d. Drywell unit coolers condensate flow rate monitoring system-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

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*The primary containment atmosphere gaseous radioactivity monitor is not required to be OPERABLE until OPERATIONAL CONDITION 2.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS.56 AND 21 TO FACILITY OPERATING

LICENSE NOS. NPF-39 AND NPF-85

PHILADELPHIA ELECTRIC COMPANY

LIMERICK GENERATING STATION, UNITS 1 AND 2

DOCKET NOS. 50-352 AND 50-353

1.0 INTRODUCTION

By letter dated May 8, 1991, as supplemented May 15, 1992, the Philadelphia Electric Company (PECo, the licensee) submitted a request for changes to the Limerick Generating Station, Units 1 and 2, Technical Specifications (TS). The requested changes would 1) revise the channel calibration frequency for the peak acceleration seismic monitoring recorder mounted on the reactor vessel head flange from 18 to 24 months and 2) revise the frequency of surveillance testing of the main steam safety relief valves (SRVs) from 18 to 24 months. The May 15, 1992 letter officially notified us of what we had already noted, namely, that the May 8, 1991 date on the application was a typographical error and that the date on the application should be May 8, 1992. The supplemental letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination. As discussed in PECo's letter of February 11, 1992, because of the economic benefits associated with a longer fuel cycle, they are in the process of changing the operation of each of the Limerick units from the current 18-month refueling cycle to a 24-month refueling cycle. The fuel that was loaded in Limerick, Unit 1, in the spring of 1991 and the fuel loaded in Limerick, Unit 2, in the spring of 1992 is designed for a 24-month fuel cycle. In order to accomodate this change, there are many surveillance requirements in the present TSs that specify an 18-month surveillance interval (as opposed to "each refueling outage") that would have to be changed. This is the first of three separate applications being submitted by the licensee. The other two applications, which will be separate licensing actions, will be submitted in late July to September 1992. These TS changes were evaluated in accordance with the guidance provided in NRC Generic Letter (GL) 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-month Fuel Cycle," dated April 2, 1991.

2.0 EVALUATION

There are a number of instruments located throughout the plant designed to monitor and provide earthquake spectra immediately following an assumed earthquake. The seismic instrumentation is described in Section 3.7.4 of the Updated Final Safety Analysis Report (UFSAR). The overall system meets the

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P PDR

intent of Regulatory Guide (RG) 1.12 (Revision 1). All locations where response spectrum recorders are required by the RG are monitored by time history accelerographs. The seismic monitoring instrumentation consists of the following:

a. Six triaxial time history accelerographs;
b. Three triaxial peak recording accelerographs;
c. One triaxial seismic switch;
d. One triaxial seismic trigger;
e. One response spectrum analyzer;
f. A system control panel which includes seismic event visual and audible annunciators;
g. Cassette recorders and a playback unit; and
h. One uniaxial seismic trigger.

The surveillance requirements on the above instruments are listed in Table 4.3.7.2.1 on pages 3/4 3-71 and 3/4 3-72. In the subject application, the licensee is only proposing to change the surveillance interval on item 2.a, one of three triaxial peak recording accelerographs, designated XR-VA-151.

Seismic instrument XR-VA-151 is a peak acceleration recorder (i.e., Engdahl Enterprises model PAR400) designed to record the peak accelerations in three orthogonal directions that the instrument's mounting location experiences during a seismic event. This recorder is mounted onto the reactor-vessel-head flange using a steel plate and thermal insulating material; the instrument does not contact the flange. XR-VA-151 is a passive instrument which uses the mechanical energy imparted to it during a seismic event to record the data. The acceleration data is recorded on a replaceable medium within the instrument; the data is not transmitted to any other location. The data is retrieved after the seismic event and is used to verify design analyses in support of justifying plant integrity and operability.

TS surveillance requirement 4.3.7.2.1 requires that the seismic monitoring instruments be demonstrated operable by the performance of the channel check, channel functional test, and channel calibration operations at the frequencies shown in TS Table 4.3.7.2-1. This TS Table specifies that only a channel calibration is required for seismic instrument XR-VA-151 at a frequency designated by the notation "R." TS Table 1.1 defines the "R" notation as a frequency of at least once-per-18-months (i.e., 550 days). Therefore, the proposed TS change would annotate the frequency of channel calibration for seismic instrument XR/VA-151 on TS Table 4.3.7.2-1 to indicate that the calibration frequency for this instrument is once-per-24-months.

As noted above, this instrument is mounted on the reactor vessel head flange. To get at the instrument to perform a channel calibration not only requires shutting down the plant and deinerting containment, but also entails removal of the drywell head (a major lift), removal of the mirror insulation and waiting for decay of shorter-lived nuclides to be able to access the area. It is only feasible to perform this calibration when the plant is shutdown for refueling. XR-VA-151 is not important to safety in that it is not needed for safe shutdown, nor does the instrument interface with or control any structure, system, or component which is important to safety. XR-VA-151 does not control or initiate any protective or mitigating action. XR-VA-151 does not present the plant operators with any on-line information which is used by the operators for the initiation of any protective or mitigating actions.

The licensee has verified with the manufacturer of the instrument that a calibration interval of 24 months is acceptable. The manufacturer recommends periodic replacement of some of the components (e.g., gasket, O-rings) but this recommended replacement period exceeds 30 months. The licensee has experienced problems with the recorder because of failures of mounting components. The instrument mount was subsequently redesigned and it appears that the problem has been successfully resolved.

As discussed above, seismic instrument XR-VA-151 is a passive device. This instrument does not interface with any other plant system or equipment, nor does this instrument provide on-line operational information to the plant operators. There is no accident previously evaluated which has as its initiator anything that is related to instrument XR-VA-151, its accuracy, or to the frequency of this instrument's surveillance testing. The proposed TS change to the surveillance testing interval for XR-VA-151 will not affect the ability of plant equipment important to safety to bring the plant to a safe shutdown condition, maintain the plant in a safe shutdown condition, or mitigate the consequences of any accident. As a result, the proposed change will not impact on-site or off-site doses resulting from accident-related radiological releases.

The licensee has provided a reasonable analysis to support their position that extending the surveillance interval will not degrade the functionality of the instrument. We conclude that the proposed TS change is acceptable.

The second change requested by the subject application is to revise the frequency of surveillance testing of the main steam SRVs from 18 to 24 months to accommodate a nominal two-year fuel cycle. The design and functioning of the SRVs is described in Section 5.2.2, "Overpressure Protection," of the UFSAR. Although the change was not mentioned in the licensee's submittal, the proposed TS pages submitted with the application noted a change in the last line of Surveillance Requirement 4.4.2.1. The change was to delete the word "tested" which was an obvious typographical error in the present TSs. Removal of the superfluous word does not in any way affect the staff's no significant hazards consideration.

The SRVs are two-stage pilot-operated dual-function safety relief valves manufactured by the Target Rock Company. In the safety mode, the valve opens solely by mechanical means when pressure at the inlet of the valve reaches the set pressure of the valve. In the depressurization mode, the valve is remotely opened by a solenoid valve manifold/pneumatic operator assembly to provide controlled depressurization of the reactor coolant pressure boundary. There are a total of 14 SRVs that all function in the safety mode and have the capability to operate in the depressurization mode via manual actuation from the Main Control Room. Five (5) of the SRVs are allocated to the automatic depressurization system (ADS) which can automatically operate the valves in the depressurization mode to reduce reactor pressure and thus allow the low pressure Emergency Core Cooling Systems (ECCS) to cool the reactor. This change request only pertains to the self-actuating safety mode of the SRVs. The ADS mode of operation, as described in Section 7.3.1.1.1.2 of the UFSAR is not effected by the proposed TS change.

TS surveillance requirement 4.4.2.2 requires that: "At least 1/2 of the safety relief valves shall be removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested and storedin accordance with manufacturer's recommendations at least once per 18 months, and they shall be rotated such that all 14 safety relief valves are removed, set pressure tested, and reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations at least once per 40 months." This change request proposes to change the 18 months to 24 months for testing at least half of the SRVs, and change the 40 months (combination of an 18-month cycle and a 22-month cycle) to 54 months (combination of a 24-month cycle and a 30-month cycle) for testing all 14 SRVs.

The two-stage pilot operated SRVs have shown both setpoint drift high and failure to lift. The NRC has issued a number of information notices describing these problems (IN82-41, IN83-39, IN83-82, IN86-12, IN88-30 and IN88-30, Supplement 1, the latter issued February 2, 1990). On April 24, 1992, the NRC's Office for Analysis and Evaluation of Operational Data (AEOD) issued a special report entitled "Safety and Safety/Relief Valve Reliability," AEOD/S92-02. The study analyzes and evaluates operational experience related to failure and degradations of safety valves and safety/relief valves (SRVs) used on domestic light water reactors.

The two-stage SRV consists of a pilot stage assembly and a main stage assembly. Principal parts of the pilot stage assembly are the body, main disc, main spring and internal porting for the pilot stage as shown in the attached figure. Principal parts of the pilot stage are the bonnet, the stabilizer disc, the pilot disc, the set pressure spring, and the pneumatic operator. The shape of the pilot disc-seat interface in the SRV is conical. It is much smaller than that of a safety valve. The shape and size of the disc-seat interface affects the forces which tend to oppose the disc lifting on system pressure. The smaller size and conical shape concentrate the forces opposing lift of the pilot disc, while the larger, flat disc-seat interface of the safety valve is less sensitive to these forces.

The SRV operates in a saturated steam environment at a pressure which ranges from atmosphere to about 1100 psig. Operating pressure in a BWR is about 1040 psig. The SRVs are generally arranged into three groups, with a different setpoint for each group. At Limerick, 4 of the 14 SRVs are set to lift at 1130 psig ± 1%, 5 are set to lift at 1140 psig ± 1% and 5 are set to lift at 1150 psig ± 1%. During a mild pressure transient, only the 4 SRVs with the lowest setpoint would be expected to lift to reduce system pressure, reducing the dynamic and thermal loads on the suppression pool.

With respect to operation of the SRVs, when the pressure under the stabilizer disc reaches the setpoint of the valve, it will push the pilot disc upward, releasing the pressure behind the main piston into the discharge of the SRV. The differential pressure across the piston will lift the main disc, relieving reactor coolant system (RCS) pressure. The ADS and manual modes are initiated by the pneumatic actuator raising the pilot stem and can be done at any pressure. The SRVs are installed to prevent overpressurization of the RCS by lifting at the required setpoint to relieve pressure and to maintain RCS pressure boundary integrity by reseating tightly without leakage.

Each of the Limerick units have 14 SRVs. PECo has three (3) sets of SRVs (14 SRVs per set). While not required by the TSs, during each refueling outage, PECo has replaced all 14 SRVs with refurbished, tested and recertified valves and has sent the 14 SRVs removed during the outage to Wyle Laboratories for testing. Limerick, Unit 1 has completed 4 cycles of operation, shutting down March 20, 1992 for the fourth refueling outage and returning to power July 9, 1992. Limerick Unit 2, which was issued a full-power license on August 25, 1989, has completed one full cycle of operation, with the first refueling outage extending from March 22, 1991 to June 5, 1991.

As part of the application of May 8, 1992, PECo provided a Table summarizing the SRV set pressure surveillance results for the 14 SRVs tested following the first three cycles of operation for Unit 1 and the first cycle of operation for Unit 2. The SRVs removed in late April 1992 from Unit 1 were not tested at Wyle Laboratories until the week of June 22, 1992. The data from the latter tests has since been provided to us.

As noted previously, the two-stage, pilot-actuated SRVs have exhibited a number of malfunctions, primarily failure to lift and setpoint drift high (which are essentially the same problem) as well as failure to reseat and pilot leakage. The problems with the SRVs, root cause analysis and common corrective actions have been described in the seven information notices issued by the NRC staff. Following the failure of all 11 SRVs at Hatch Unit 1 to lift at their established setpoint during a pressure transient on July 3, 1982, the BWR Owners' Group (BWROG), together with the Target Rock Company and the General Electric Company (GE), instituted a program to determine the cause of the setpoint drift and to determine a corrective action for the problem. After much investigation, friction in the labyrinth seal area and disc-to-seat sticking were postulated to be the causes of the setpoint drift. Labyrinth seal friction was caused by adverse tolerance buildup during manufacturing and disc-to-seat sticking was postulated to be caused by a film of oxide which covered disc and seat areas exposed to steam. The disc and seat were made of nearly identical materials; therefore, their oxides formed a single film, inhibiting disc movement. Subsequently, PH 13-8 Mo was selected as the material for the disc because it was substantially different from the seat material (Stellite 6 or 6B). The oxides formed by the disc and seat in the steam space were not expected to join at their interface and impede the disc movement. A program of in-plant testing of approximately equal numbers of valves with new discs and old discs began in early 1986 with the installation of the first of the new discs. Limerick, Unit 1 was a participant in this test program. The PH 18-8 Mo was used as the disc material in 7 of the 14 SRVs that operated throughout Cycle 3 (May 22, 1989 to September 7, 1990, about 15 1/2 months of exposure). When tested at Wyle Laboratories, only 3 of the 14 SRVs lifted within the TS required limit of ± 1% of the nameplate setpoint. The results were discussed in LER 50-352/91-015 dated July 5, 1991. The highest percentage drift was 4.07%, which was considerably lower than the highest setpoint drift found in the previous Unit 1 cycles (greater than 11% in cycle 1 and 14.0% in cycle 2) and also significantly lower than the 11.59% highest drift subsequently found for the 14 SRVs exposed during the first cycle of Unit 2.

The initial test results from plants such as Brunswick, Hatch and Fermi in which the SRV discs had been replaced with PH 13-8 Mo material indicated some improvement in the extent of setpoint drift. Subsequent experience, however, indicated that the magnitude of bonding between the disc and seat with the new material was not much less than with Stellite discs. The BWR Owners' Group, GE and Target Rock Company evaluated the experience with the replacement disc material and concluded that the PH 13-8 Mo discs provide no better assurance of lift at the appropriate pressure than the Stellite discs. The recommendation to install the PH 13-8 Mo discs was withdrawn.

The 14 SRVs that were in service on Unit 1 during the third fuel cycle were refurbished and installed on Unit 2 in April 1991 for service during the current second fuel cycle. The 7 SRVs that had PH 13-8 Mo discs were refurbished with Stellite discs. Thus, all of the SRVs at Limerick now have Stellite discs as they did initially.

During the third Unit 1 refueling outage (fall 1990), three of the 14 SRVs had modified valve bodies that were designed to drain condensate away from the main seat. The modification tended to reduce leakage into the suppression pool. During the recently completed fourth refueling outage, all 14 replacement SRVs have the modified valve bodies. Lower tailpipe temperatures indicate reduced leakage through the SRVs.

Industry data on Target Rock two-stage SRVs does not indicate a trend toward negative drift (i.e., decreasing set pressure). Therefore, extending the operating cycle from 18 to 24 months will not increase the probability or occurrences of an inadvertent SRV opening. The issue of concern is whether the upward set-point drift which has been experienced at Limerick and other BWRs is likely to be increased by a longer operating cycle and, if so, what are the likely impacts on plant safety.

Industry data on the Target Rock two-stage SRVs demonstrates that significant drift can occur within a month of service after refurbishment and can lead to the conclusion that setpoint drift magnitude approaches a plateau early in the operating cycle. The pilot disc/seat bonding mechanism described earlier also supports the plateau concept. As the oxide grows and covers the surfaces in contact, further oxidation would be impeded. Therefore, the bond strength, and consequently the set pressure drift magnitude, should approach a plateau.

Based on the conclusion that a drift plateau is approached early in the operating cycle, the proposed extension of the surveillance interval from 18 to 24 months (plus 6 months grace) is not likely to significantly increase the magnitude of setpoint drift.

The 14 SRVs installed at Limerick provide considerably more relieving capacity than is required by the applicable edition of the American Society of Mechanical Engineers (ASME) Code. Only 11 of the 14 SRVs are required to be operable in accordance with TS Section 3.4.2.

As discussed previously, PECo has been removing, testing, and replacing all 14 SRVs at each refueling outage. The PECo staff has informed us that they plan to continue replacing all 14 SRVs at each refueling outage until there is a better resolution of the setpoint drift and seat leakage problems, even though the TSs only require that 50% of the SRVs be set pressure tested each refueling outage (presently every 18 months). In comparison, ASME Code Section XI would only require 20% of the SRVs to be set pressure tested in the same period. ANSI/ASME OM-1-1981, "Requirements for Inservice Performance Testing of Nuclear Power Plant Pressure Relief Devices," as invoked by ASME Code Section XI, requires all SRVs to be tested within a 60-month period with a minimum of 20% tested within any 24 months. The proposed change to TS Section 4.4.2.2 would require 50% of the SRVs to be tested in 24 months, and all SRVs to be tested within a maximum period of 54 months (i.e., 48 months with a 6-month grace period).

As part of the BWROG SRV Setpoint Drift Fix Program, a sensitivity study of BWR plants using the Target Rock Company two-stage SRV was performed by GE (NEDO-22210). This study determined that sufficient overpressure protection margin existed at all plants to tolerate an upward setpoint drift of 10% on each SRV during the limiting pressurization transient. The most severe pressurization transient event was conservatively assumed to be the simultaneous closure of all Main Steam Isolation Valves (MSIVs) with a coincident failure of the MSIV position scram signal (reactor scram subsequently occurs on a high neutron flux signal). GE has performed plant specific evaluations for several BWR IV plants including Limerick. In a study for the Hatch Plant, GE determined that an average drift of 16% above a nominal set pressure of 1100 psig would not cause peak reactor vessel pressure to exceed 1375 psig for the limiting postulated event. In another study for the Brunswick Plant (GE Report No. MDE-46-0386), 4 of 11 valves remained closed to a test pressure of 1250 psig (greater than 20% drift). The average drift for all valves was 13%. Had the most limiting postulated event occurred with the SRVs at the "as-found" set pressures, peak vessel pressure would have remained well below the 1375 psig ASME limit. In fact, only seven of the valves would have had to actuate in order to reverse the pressure transient.

A plant-specific evaluation was performed for Limerick Unit 1 by GE during the 1986 surveillance test outage. The results were issued as GE Report No. MDE-85-0786. A 10% setpoint drift above nameplate set pressures was assumed for all 14 SRVs coincident with the most severe pressurization transient defined above. Under these conditions, peak vessel pressure would have been approximately 1350 psig. Additionally, the impact on fuel thermal margin, Loss of Coolant Accident/Emergency Core Cooling System (LOCA/ECCS) performance, High Pressure Coolant Injection/Reactor Core Isolation Cooling (HPCI/RCIC) operability, and drywell pressure and temperature response was assessed. The evaluation showed that a 10% setpoint drift of all 14 SRVs would not have caused any of the plant safety limits to be exceeded or impact safe plant operation.

As shown in Table 1 of the licensee's application, no more than two SRVs in any set tested have exceeded a 10% setpoint drift and this was with the set of 14 SRVs removed during the Limerick, Unit 1 first refueling outage. This set of valves was only exposed for less than 11 months, from June 21, 1986 to May 15, 1987. The cause of the setpoint drift was primarily corrosion induced bonding between the Stellite pilot valve disc and Stellite seat (LER 50-352/87-034-01 dated August 21, 1987).

Based on the above, we have concluded that the proposed change to TS Section 4.4.2.2 to extend the surveillance interval for the SRVs from 18 to 24 months is acceptable. Even though the SRVs are likely to experience setpoint drift, there is reasonable assurance that the SRVs will provide reactor overpressure protection for the Limerick, Units 1 and 2 Nuclear Steam Supply System.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Pennsylvania State official was notified of the proposed issuance of the amendments. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (57 FR 24675). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

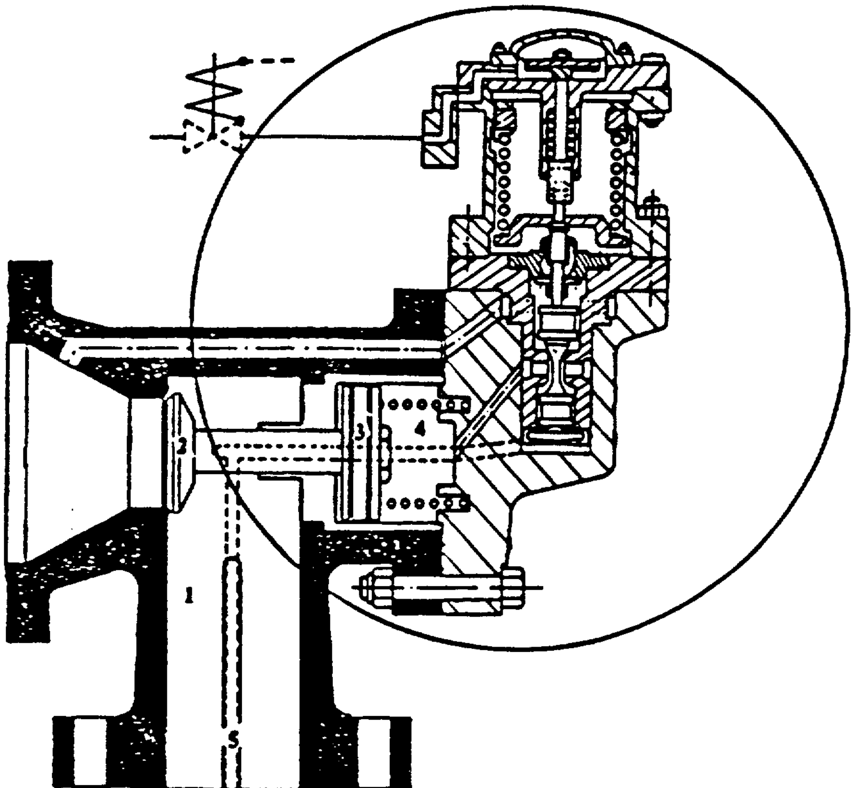
5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Attachment: Two-stage SRV Figure

Principal Contributor: R. Clark

Date: August 20, 1992



Two-Stage SRV

1 Body
2 Main Disc
3 Main Piston
4 Main Spring
5 Pilot Inlet Sensing Tube

Two-Stage SRV, Pilot Stage

1 Pneumatic Actuator
2 Setpoint Spring
3 Labyrinth Seal
4 Pilot Rod
5 Pilot Stage Discharge Tube
6 Main Piston
7 Main Spring
8 Stabilizer Disc
9 Pilot Disc