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APR 19 1974

Docket No. 50-277

Philadelphia Electric Company
ATTN: Mr. J. L. Hankins
Vice President
2301 Market Street
Philadelphia, Pennsylvania 19101

Change No. 2
License No. DPR-44

Gentlemen:

We have completed our evaluation of your "Report on Hydraulic Shock Suppressors (Snubbers)" for Peach Bottom Units 2 and 3, dated March 14, 1974. Included with this report were proposed Technical Specifications establishing limiting conditions for operations (LCO), surveillance requirements, bases, and reporting requirements for the hydraulic snubbers used in Peach Bottom Units 2 and 3.

Additional Technical Specifications changes have been included in this Technical Specification Change No. 2. These additional changes fall into three categories; category A changes are required by the Directorate of Licensing; category B changes are required to provide additional clarification of the Technical Specification conditions and category C changes have resulted from operating experience gained during initial operation of Peach Bottom Unit 2.

Based on our evaluation, a copy of which is enclosed as Enclosures 2 and 3, we conclude that the changes described therein do not involve a significant hazard consideration and that there is reasonable assurance that the health and safety of the public will not be endangered.

Accordingly, pursuant to Section 50.59 of 10 CFR Part 50, the Technical Specifications appended to License No. DPR-44 are hereby changed as indicated in Enclosure 1.

Sincerely,

Original Signed by
R. C. DeYoung

R. C. DeYoung, Assistant Director
for Light Water Reactors, Group 1
Directorate of Licensing

Enclosures:
As stated

C/P

OFFICE	L:LWR 1-2	L:LWR 1-2	~~RO~~ Concurred by telephone	~~L~~	L:LWR 1-2	
SURNAME	RPowell:ew	WButler	~~HThornburg~~	~~DSkovholt~~	RCDeYoung	
DATE	4/19/74	4/19/74	~~4/ /74~~	~~4/ /74~~	4/19/74	

Form AEC-318 (Rev. 9-53) AECM 0240 GPO C43 16 81465-1 520-284

APR 1 9 1974

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ENCLOSURE 1

TECHNICAL SPECIFICATIONS CHANGE NO. 2 FOR
THE PEACH BOTTOM ATOMIC POWER STATION UNITS 2 & 3
APPENDIX A, DPR-44

The following changes are to be made to the Technical Specifications for Operating License DPR-44, Appendix A, revised October 1973. The changes which are to be made in the Technical Specifications are underlined.

Item 2-1 MAIN STEAM LINE HIGH RADIATION SET POINT

Category A

a) Page 38, Table 3.1.1	Change 7X to 3X
b) Page 48, Line 23	Change seven to three
c) Page 61, Table 3.2.A	Change 7X to 3X and add reference to note 8
d) Page 63	Add following note 8 8. At a radiation level of 1.5 times the normal rated power background an alarm will be tripped in the control room to alert the control room operators to an increase in the main steam line tunnel radiation level.
e) Page 90, 5th line from bottom of page	Change 7 to 3

Item 2-2 PUMP DISCHARGE PRESSURE INTERLOCK

Category C

a) Page 67	Change RHR (LPCI) Pump Discharge Pressure Interlock trip level setting range from 50 $\pm$ 5 psig to 50 $\pm$ 10 psig
b) Page 67	Change Core Spray Pump Discharge Pressure Interlock level setting range from 185 $\pm$ 5 psig to 185 $\pm$ 10 psig.
c) Page 69	Add a range of $\pm$1 to the Recirculation Pumps A and B, d/p trip level setting.

Item 2-3 STANDBY LIQUID CONTROL SYSTEM

Category B

Add a Section 3.4.C.3 under limiting condition for operation on page 117 as follows.

3.4.C.3 If the concentration, volume, or temperature is outside the limits established by 3.4.C.1 or 3.4.C.2, the reactor shall be shut down in accordance with 3.4.D unless the concentration, volume and temperature is restored within allowable limits within the same time period.

Item 2-4 3.5J LOCAL LHGR

Category A

On page 133a insert between L_T and L under 3.5J Local LHGR the Total Core Length

= 12 feet Unit 2

= 12.167 feet Unit 3

Item 2-5 3.7A PRIMARY CONTAINMENT

Category B

On page 166 add a section 3.7.A.3 as follows.

3.7.A.3 If the primary containment integrity is breached when it is required by 3.7.A.2, that integrity shall be reestablished within 24 hours or the reactor placed in a cold shut down condition within 24 hours.

Item 2-6 PROTECTION FACTORS FOR RESPIRATORS

Category A

On page 257 in Table 6.5.1 make the following changes.

a) Change the protection factor for II Atmosphere supplying respirator 1. Air-line respirator: facepiece, full, D from 500 to 100.

b) Change the protection factor for II Atmosphere supplying respirator 2. Self-contained breathing apparatus, facepiece, full, R from 1000 to 100.

Item 2-7 DESIGN FATIGUE USAGE EVALUATION

Category A

Add the following sections to sections 6.6.2 and 6.7.1.C.1 on pages 259 and 261

a) 6.6.2.7 Design Fatigue Usage Evaluation

1. Monitoring, recording, evaluating, and reporting requirements contained in Section 6.7.1.C will be met for various portions of the reactor coolant pressure boundary (RCPB) for which detailed fatigue usage evaluation per the ASME Boiler and Pressure Vessel Code Section III was performed[1] for the conditions defined in the design specification. The location to be monitored shall be:

a. The feedwater nozzles
b. The shell at or near the waterline
c. The flange studs

2. Monitoring, Recording, Evaluating and Reporting

a. Operational transients that occur during plant operation will, at least semi-annually, be reviewed and compared to the transient conditions defined in the component stress report for the locations listed in 1 above, and used as a basis for the existing fatigue analysis.

b. The number of transients which are comparable to or more severe than the transients evaluated in the stress report code fatigue usage calculations will be recorded in an operating log book. For those transients which are more severe, available data, such as the metal and fluid temperature, pressures, flow rates, and other conditions will be recorded in the log book.

c. The number of transient events that exceed the design specifications quantity and the number of transients events with a severity greater than that included in the existing code fatigue usage calculations shall be added. When this sum exceeds the predicted number of design condition events by twenty five[2], a fatigue usage evaluation of such events will be performed for the affected portion of the RCPB.

Footnote:

1) See paragraph N-415.2, ASME Section III, 1965 Edition

2) The code rules permit exclusion of twenty-five (25) stress cycles from secondary stress and fatigue usage evaluation. (See paragraph N-412(t)(3) and N-417.10(f) of the Summer 1968 Addenda to ASME Section III, 1968 Edition)

b) 6.7.C.1.h Design Fatigue Usage Evaluation Report

In the semi-annual Operating Report, a listing of the number of events identified in 6.6.2.7.2.b above will be tabulated and compared to the design or allowed quantity of comparable or more severe events. In those cases where recalculation of fatigue usage is required per 6.6.2.7.2.c and the calculated usage exceeds two times the design usage limit of the code, the report will define the inservice inspection that will be performed on that portion of the RCPB to monitor for crack

Item 2-8 SEISMIC HYDRAULIC SNUBBERS ON SAFETY RELATED SYSTEMS

Category A

The following limiting condition for operation, surveillance requirements, bases, and reporting shall be included for the seismic hydraulic snubber on safety related systems for Peach Bottom Units 2 and 3.

a) Insert page 234a containing 3.11.D and 4.11.D for Seismic Hydraulic Snubbers on Safety Related Systems

b) Insert page 236a containing the bases for 3.11D and 4.11D

c) Insert page 266a containing the reporting of inspection results

LIMITING CONDITIONS FOR OPERATION

3.11.D. Seismic Hydraulic Snubbers On Safety Related Systems

1. During reactor power operation, the hydraulic snubbers installed on safety related piping systems must be operable except as specified in Spec. 3.11.D.2.

2. If accessible hydraulic snubbers on safety related systems are observed to be inoperable at times other than the prescribed surveillance, continued operation is permissible for a period of seven days. If the defective snubbers have not been repaired within that period the reactor shall be shut down.

3. The reactor shall not be started up with any safety related snubbers known to be inoperable.

SURVEILLANCE REQUIREMENTS

4.11.D. Seismic Hydraulic Snubbers On Safety Related Systems

1. All hydraulic snubbers installed on safety related systems shall be visually inspected at the following frequency:

   a. Whenever the reactor is shut down for 72 hours or longer and the snubbers have not been inspected for 30 days.

   b. The maximum inspection interval of accessible units shall be 6 months.

2. If the results of the inspection performed at the frequency required by Specification 4.11.D indicate reliable snubber operation for one operating cycle, the frequency of inspection shall be decreased to one per operating cycle.

3. Once per operating cycle a 2-1/2" snubber installed in a location representative of the worst environmental conditions shall be removed and disassembled for the purpose of inspection for seal degradation.

T.S. Change No. 2

3.11 Bases

D. Seismic Hydraulic Snubbers On Safety Related Systems

Hydraulic snubbers are necessary on safety related system piping to restrain seismic motion and yet allow unrestrained thermal growth.

Should a defective hydraulic snubber be discovered during reactor operation seven days is a reasonable time for repair without affecting operation. This allowance is justifiable since the probability of a seismic event during this period is considered low.

4.11 Bases

D. Seismic Hydraulic Snubbers on Safety Related Systems

A periodic inspection is required to establish their reliability and the compatibility of the seal materials with the hydraulic fluid and the operating environment.

It is considered that continued proper operation through one operating cycle and satisfactory findings of periodic inspections is sufficient justification to extend the inspection interval. The interval of once per operating cycle will minimize the effects on power operation.

Disassembly of a snubber installed in a location representative of the worst environmental conditions will provide an indication of the degree of hydraulic cylinder seal degradation.

T.S. Change No. 2

6.7.2.C Reporting Results of Inspections of Seismic Hydraulic Snubbers

1. Notification of snubber failure shall be reported as an abnormal occurrence in accordance with 6/7.2.A.

2. If no defective snubbers are identified, a report shall be submitted in accordance with paragraph 6.7.1.c.1.d.

T.S. Change No. 2

ENCLOSURE 2

SAFETY EVALUATION
FOR TECHNICAL SPECIFICATIONS CHANGE NO. 2

CHANGE NO. 2 TO APPENDIX A OF THE LICENSE FOR
PEACH BOTTOM ATOMIC POWER STATION UNIT 2, DPR-44
DOCKET NUMBER 50-277

1.0 INTRODUCTION

The Technical Specification changes included in this evaluation, have resulted from experience gained from operating reactors. An evaluation of each of these proposed changes to the Technical Specifications follows.

2.0 MAIN STEAM LINE HIGH RADIATION SET POINT

On page 38, Table 3.1.1 change Main Steam Line Isolation Valve Closure Trip Level Setting from 7X normal full power background to 3X normal full power background. On page 48 line 23 change seven times normal background to three times normal background. In Table 3.2.A page 61 change High Radiation Main Steam Line Tunnel Trip Level Setting from 7X normal rated full power background to 3X normal rated full power background and add a note 8 reference. On page 63 note 8 for Table 3.2.A as follows:

> Note 8. At a radiation level of 1.5 times the normal rated power background an alarm will be tripped in the control room to alert the control room operators to an increase in the main steam line tunnel radiation level. On page 90 the fifth line from bottom of page change 7 times to 3 times.

EVALUATION

The above changes reduces the trip set point for high radiation level in the main steam line tunnel from 7 times to 3 times throughout the Technical Specification and adds an alarm in the control room at a radiation level of 1.5 times background at rated full power. This reduction of the trip setting was based upon experience gained at the Vermont Yankee reactor and will be consistently applied to all operating and future operating boiling water reactors.

3.0 TABLE 3.2.B INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEM

On page 67 change the trip level settings for the RHR (LPCI) Pump Discharge Interlock from 50 ± 5 psig to 50 ± 10 psig and change the Core Spray Pump Discharge Pressure Interlock from 185 ± 5 psig 185 ± 10 psig. On page 69 add a range of ± 1 to the Recirculation Pump A and B d/p so the trip level setting is 2 psid ± 1 for both pumps.

EVALUATION

Operating experience has indicated the drift of the instruments used for detecting that the core spray and RHR pumps are running is sufficient to exceed the range of ± 5 psi but would not exceed ± 10 psi. Increasing the range from ± 5 psi to ± 10 psi will not affect the interlock function as the rated head and shut off head for both the core spray and RHR pumps are 265 psig and 380 psig and 250 psig and 345 psig respectively.

Addition of range of ± 1 to the recirculation pumps A & B two psid trip level setting was inadvertently omitted from the original Technical Specifications. This ± 1 psi range allowance is required to permit normal drift of the instruments between surveillance.

4.0 3.4. STANDBY LIQUID CONTROL SYSTEM

Add the following to section 3.4 to clarify the limiting condition of operation requirements on page 117.

Add 3.4.C.3 If the concentration, volume, or temperature is outside the limits established by 3.4.C.1 or 3.4.C.2, the reactor shall be shutdown in accordance with 3.4.D unless the concentration, volume, and temperature is restored within allowable limits within the same time period.

EVALUATION

Addition of the section 3.4.C.3 above is to clarify the time permitted to return the liquid control solution to within Technical Specification requirements.

5.0 3.5.J LOCAL LHGR

The fuel element length for Units 2 and 3 are 12 feet and 12.167 feet respectively.

On page 133a under 3.5.J LOCAL LHGR between L_T and L insert for Total Core Length:

= 12 feet Unit 2

= 12.167 feet Unit 3

EVALUATION

The Unit 3 fuel length is 12.167 feet and this value of total core length L_T shall be used to establish the limits in 3.5.J.

6.0 3.7.A PRIMARY CONTAINMENT

On page 166 add 3.7.A.3 as follows to clarify this limiting condition of operation.

3.7.A.3 If the primary containment integrity is breached when it is required by 3.7.A.2, that integrity shall be reestablished within 24 hours or the reactor placed in a cold shut down condition within 24 hours.

EVALUATION

This addition limiting condition of operation is required to clarify and provide the time period permitted for repairing or reestablishing the primary containment integrity.

7.0 TABLE 6.5.1 PROTECTION FACTORS FOR RESPIRATORS

Several of the protection factors which must be used in establishing personnel protection from radiation effects given in Table 6.5.1 have been reduced by Directorate of Licensing to provide an additional safety margin. The following changes should be changed in Table 6.5.1 on page 257.

Change the protection factor for II. Atmosphere - supplying respirator 1. Air-line respirator: facepiece, full, D from 500 to 100.

Change the protection factor for II Atmosphere - supplying respirator 2. Self-contained breathing apparatus, facepiece, full, R from 1000 to 100.

EVALUATION

The change in the protection factors indicated above have been established by the Directorate of Licensing.

8.0 DESIGN FATIGUE USAGE EVALUATION

To provide additional information in the semi-annual report regarding operational transients the following shall be added to section 6.6.2 and 6.7.1.C.1. on pages 259 and 261 of the Technical Specifications.

Add 6.6.2.7 Design Fatigue Usage Evaluation

1. Monitoring, recording, evaluating, and reporting requirements contained in Section 6.7.1.C will be met for various portions of the reactor coolant pressure boundary (RCPB) for which detailed fatigue usage evaluation per the ASME Boiler and Pressure Vessel Code Section III was performed[1] for the conditions defined in the design specification. The locations to be monitored shall be:

   a. The feedwater nozzles
   b. The shell at or near the waterline
   c. The flange studs

2. Monitoring, Reacording, Evaluating and Reporting

   a. Operational transients that occur during plant operations will, at least semi-annually, be reviewed and compared to the transient conditions defined in the component stress report for the locations listed in 1. above, and used as a basis for the existing fatigue analysis.

   b. The number of transients which are comparable to or more severe than the transients evaluated in the stress report Code fatigue usage calculations will be recorded in an operating log book. For those transients which are more severe, available data, such as the metal and fluid temperatures, pressures, flow rates, and other conditions will be recorded in the log book.

   c. The number of transient events that exceed the design specifications quantity and the number of transient events with a severity greater than that included in the existing Code fatigue usage calculations shall be added. When this sum exceeds the predicted number of design condition events by twenty-five,[2] a fatigue usage evaluation of such events will be performed for the affected portion of the RCPB.

Add 6.7.1.C.1.h Design Fatigue Usage Evaluation Report

In the semi-annual Operating Report, a listing of the number of events identified in 6.6.2.7.2.b above will be tabulated

1) See paragraph N-415.2, ASME Section III, 1965 Edition
2) The Code rules permit exclusion of twenty-five (25) stress cycles from secondary stress and fatigue usage evaluation. (See paragraphs N-412(t)(3) and N-417.10(f) of the Summer 1968 Addenda to ASME Section III, 1968 Edition)

and compared to the design or allowed quantity of comparable or more severe events. In those cases where recalculation of fatigue usage is required per 6.6.2.7.2.c and the calculated usage exceeds two times the design usage limit of the Code, the report will define the inservice inspections that will be performed on that portion of the RCPB to monitor for crack initiation.

EVALUATION

The above additions, 6.7.2.7 and 6.7.1.C.1.h, have been added to operating plants Technical Specifications to provide semi-annually information and analyses that assures design fatigue factors are not exceeded and requires additional inservice inspections if design fatigue factors are exceeded.

9.0 SEISMIC HYDRAULIC SNUBBERS

On March 18, 1974, the Philadelphia Electric Company submitted to the Directorate of Licensing a report on the hydraulic shock snubbers used in Peach Bottom Units 2 and 3. Of the 348 hydraulic shock suppressors installed in Peach Bottom Units 2 and 3 none were manufactured by the Bergen-Patterson Company. The Regulatory staff has reviewed this report and the proposed technical specifications covering hydraulic shock suppressors and conclude they are acceptable. The following are the technical specifications which will be included in the Technical Specifications.

Enclosure 3 contains the Technical Specifications for the Limiting Condition For Operation, 3.11.D, the Surveillance Requirements, 4.11.D, the bases and the reporting requirement 6.7.2.C for the Seismic Hydraulic Snubbers On Safety Related Systems.

10. CONCLUSION

On the basis of our evaluation, we have concluded that the Technical Specifications Change No. 2 for the Peach Bottom Atomic Power Station, Units 2 and 3, do not present significant hazard considerations and that there is reasonable assurance that the health and safety of the public will not be endangered by operations affected by these proposed changes. The Technical Specifications changes described above should be reissued to holders of Appendix A to Operating License DPR-44.

Walter R. Butler

Walter R. Butler, Chief
Light Water Reactors Br. 1-2
Directorate of Licensing

Ray Powell

Ray Powell, Project Manager
Light Water Reactors Branch 1-2
Directorate of Licensing

Date: April 19, 1974

ENCLOSURE 3

PBAPS

LIMITING CONDITIONS FOR OPERATION	SURVEILLANCE REQUIREMENTS
3.11.D. Seismic Hydraulic Snubbers On Safety Related Systems	4.11.D. Seismic Hydraulic Snubbers On Safety Related Systems
1. During reactor power operation, the hydraulic snubbers installed on safety related piping systems must be operable except as specified in Spec. 3.11.D.2.	1. All hydraulic snubbers installed on safety related systems shall be visually inspected at the following frequency: a. Whenever the reactor is shut down for 72 hours or longer and the snubbers have not been inspected for 30 days. b. The maximum inspection interval of accessible units shall be 6 months.
2. If accessible hydraulic snubbers on safety related systems are observed to be inoperable at times other than the prescribed surveillance, continued operation is permissible for a period of seven days. If the defective snubbers have not been repaired within that period the reactor shall be shut down.	2. If the results of the inspection performed at the frequency required by Specification 4.11.D indicate reliable snubber operation for one operating cycle, the frequency of inspection shall be decreased to one per operating cycle.
3. The reactor shall not be started up with any safety related snubbers known to be inoperable.	3. Once per operating cycle a 2-1/2" snubber installed in a location representative of the worst environmental conditions shall be removed and disassembled for the purpose of inspection for seal degradation.

T.S. Change No. 2

3.11 Bases

D. Seismic Hydraulic Snubbers On Safety Related Systems

Hydraulic snubbers are necessary on safety related system piping to restrain seismic motion and yet allow unrestrained thermal growth.

Should a defective hydraulic snubber be discovered during reactor operation seven days is a reasonable time for repair without affecting operation. This allowance is justifiable since the probability of a seismic event during this period is considered low.

4.11 Bases

D. Seismic Hydraulic Snubbers on Safety Related Systems

A periodic inspection is required to establish their reliability and the compatibility of the seal materials with the hydraulic fluid and the operating environment.

It is considered that continued proper operation through one operating cycle and satisfactory findings of periodic inspections is sufficient justification to extend the inspection interval. The interval of once per operating cycle will minimize the effects on power operation.

Disassembly of a snubber installed in a location representative of the worst environmental conditions will provide an indication of the degree of hydraulic cylinder seal degradation.

T.S. Change No. 2

6.7.2.C Reporting Results of Inspections of Seismic Hydraulic Snubbers

1. Notification of snubber failure shall be reported as an abnormal occurrence in accordance with 6/7.2.A.

2. If no defective snubbers are identified, a report shall be submitted in accordance with paragraph 6.7.1.c.1.d.

T.S. Change No. 2