PHILADELPHIA ELECTRIC COMPANY

2301 MARKET STREET

P.O. BOX 8699

PHILADELPHIA, PA. 19101

(215) 841-5001

SHIELDS L. DALTROFF
VICE PRESIDENT
ELECTRIC PRODUCTION

July 5, 1985

Docket Nos. 50-277
50-278

Mr. Samuel J. Collins, Chief
Projects Branch 2
Division of Reactor Projects
U. S. Nuclear Regulation Commission
Region I
631 Park Avenue
King of Prussia, PA 19406

Dear Mr. Collins:

Your letter dated May 1, 1985 forwarded Combined Inspection Report 50-277/85-08 and 50-278/85-08 for Peach Bottom Atomic Power Station. The need to delay our response to this report was discussed with J. Gallo of your staff on May 28, 1985 and found acceptable. Within the scope of this inspection no violations were observed; however, the inspector expressed concern that the analyzed number of reactor vessel cycles was exceeded without an engineering evaluation. A review of this matter by our engineering specialists indicates there is a qualitative basis for concluding that the combined fatigue usage factor for the Peach Bottom reactor vessel is well within the design analysis.

The inspector's concerns are restated as follows along with our responses.

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1. Concern:

The inspector noted several inconsistencies in the following documents. The General Electric Specification No.21A1111, Rev. 9, "Reactor Pressure Vessel", indicates that the reactor vessel was designed and analyzed for two safety relief valve (SRV) blowdowns during the 40 year lifetime while the original FSAR indicates two such cycles, the updated FSAR four such cycles, and the plant Surveillance Test, ST 12.4, "Reactor Pressure Vessel Transients-Cycles Record", eight such cycles.

Response:

There are inconsistencies in the documents identified by the inspector. Table 4.2.4 of the original Peach Bottom FSAR identifies the various types of cycles assumed in the original reactor vessel fatigue study. The updated FSAR (UFSAR) contains typographical errors and omissions on Table 4.2.4, including showing four (4) safety relief valve blowdowns. The correct value is two (2) as listed in the original FSAR Table 4.2.4. This and other typographical errors in Table 4.2.4 will be corrected, as indicated in Attachment 1 to this letter, when the next annual UFSAR revision is submitted in July 1985.

The eight cycles listed in the plant Surveillance Test (ST 12.4) were based on the information contained in GE SIL No. 318. This SIL is further discussed in the response to Concern 2.

2. Concern:

The licensee indicated that GE SIL No.318 provided the basis for using eight SRV blowdown cycles; however, after reviewing the SIL, the inspector disagreed that it provides a basis for changing the number of cycles without further analysis.

Response:

It was not our intention to revise the original design analysis. There may have been a misunderstanding relating to the intended use of GE SIL No. 318. Our interpretation of the SIL is that it provides a method of estimating the

transient cycle accumulation rate over an average reactor's lifetime. The SIL also identifies generic design frequencies for twelve categories of transients. The guidance contained in the SIL regarding the design frequencies of SRV blowdowns indicated that a procedural action limit for surveillance procedure ST 12.4 of eight blowdowns over the life of the plant was appropriate. This was based on the design frequency of 0.20 events per year as stated in the SIL, and a 40-year design life expectancy for the vessel.

3. Concern:

The inspector noted that according to ST 12.4 information, Unit 2 had experienced its second SRV blowdown in March 1974 and that Unit 3 had experienced its second SRV blowdown in July 1976. The inspector determined that licensee action was warranted in both cases because the cycles had reached the design basis for Units 2 and 3, respectively. The inspector further determined that licensee action was also warranted in 1979 when GE called attention to the problem in GE SIL No. 318, which was then used by the licensee to alter the allowed value to eight SRV blowdown cycles. In 1981 when, according to licensee records, Unit 2 experienced its ninth SRV blowdown, the licensee again took no action.

Response:

We agree that we may have been remiss in not taking action earlier; however, considering the number and severity of the thermal cycles assumed in the original reactor vessel fatigue study, the assessment of cycle history did not appear to be necessary during the early portion of the vessel's 40-year design life.

Several years ago, Procedure ST 12.4 was initiated as a mechanism to provide a periodic update of the vessel transient cycle history and to require an engineering evaluation if a specific thermal cycle assumption is approached. The records of transient cycles history for each unit were reviewed, and the cycles were recorded in the first ST 12.4 which was completed on July 16, 1984. The surveillance procedure identified the total number of safety relief valve blowdowns to be in excess of the procedural action limit for SRV blowdown transients.

Within several days an engineering review was initiated to determine the appropriate actions needed to assess the impact on the reactor vessel. Subsequently, the NSSS vendor was requested to perform an evaluation that will calculate an upper limit for the SRV blowdowns based on an allowable fatigue value. Completion of this evaluation is expected by December, 1985. We are considering administrative measures to expedite future requests in a more timely manner for engineering evaluations involving safety and licensing issues.

Table 4.2.4 of the UFSAR identified thirteen categories of thermal design cycles, and a total of more than 13,000 thermal cycles which are assumed in the reactor vessel fatigue study. One category is 200 scram cycles. Two of these scrams are assumed to be simultaneous with a safety relief valve blowdown. Vessel fatigue is calculated based upon all the assumed transient cycles. The total number of thermal design cycles experienced by PBAPS vessels is far less than the total number of thermal design cycles identified in the corrected Table 4.2.4 (Attachment 1). Therefore, the combined fatigue usage factor for the Peach Bottom vessel should be well within the design analysis.

The two (2) SRV transient cycles listed in FSAR Table 4.2.4 are based on the assumption of a design basis scram where both pressure and temperature drop to the levels specified in the design thermal cycle diagram. Most scrams involve pressure drops and temperature changes less than the values specified in the design thermal cycle diagram. Thus, additional thermal cycles can be justified based on the actual operating history.

4. Concern:

During review of the reactor pressure vessel transient surveillance test, the inspector identified two additional discrepancies between the FSAR Table 4.2.4, and GE Specification 21A1111. The FSAR Table identifies 80 cycles as the analyzed total for loss of feedwater heaters while the GE Specification allows 70 cycles for feedwater heater bypass. Further, the FSAR Table identifies 123 hydrostatic tests at 1563 psig as the analyzed limit while the GE Specification allows 3 hydrostatic tests to 1563 psig. Further review appears warranted to resolve the discrepancies between the FSAR and the GE Specifications.

Response:

The two discrepancies identified represent typographical errors in the updated FSAR Table 4.2.4. The correct values are as stated in GE Specification 21A1111. The errors will be corrected as indicated in Attachment 1 in the next annual FSAR revision, to be submitted in July 1985.

5. Concern:

The inspector also determined that Technical Specification 6.10.2.f requires the licensee to keep records of operational cycles for facility components designed for a limited number of transients or cycles. ST 12.4 addresses reactor vessel transients including heatup, cooldown, loss of feedwater heaters, reactor scram, hydrostatic testing less than 1250 psig and stuck open SRV.

FSAR Table 4.2.4., however, discusses additional reactor design cycles such as reactor vessel bolt-up and un-bolt, startup and shutdown. FSAR Appendix C, Section C.5.3.2.2., describes the reactor vessel internals fatigue analysis and the transients considered, including such events as HPCI operation.

Response:

ST 12.4 was not intended to meet the record retention requirement of Technical Specification 6.10.2.f. Plant operating records are available for past reactor transients and operational cycles as required by the Technical Specifications. ST 12.4 only provides an accumulated total of certain types of operational cycles or transients. The Technical Specifications provided by the NRC Staff to Philadelphia Electric Company for use in preparing the Limerick Technical Specification required tracking of the following transients: heatups and cooldowns; loss of feedwater heaters; reactor trip cycles from full power, and hydrostatic pressure tests. The Peach Bottom ST 12.4 was written to track transients listed in the model provided for Limerick. Safety relief valve blowdowns were added to the list as a fifth category by the Plant Operations Review Committee (PORC) because of the desire of plant supervision to follow this transient as a matter of good engineering practice.

Should you require any additional information please do not hesitate to contact us.

Very truly yours,

Attachment

cc: Dr. T. E. Murley, Administrator, Region I, USNRC
T. P. Johnson, Resident Site Inspector

TABLE 4.2.4

REACTOR DESIGN CYCLES (40-YEAR LIFE)

Type of Cycle	Number of Cycles
Bolt up	123
Design hydrostatic tst as 1250 psig	130
Startup (100-F/hr heatup rate)	120
Daily reduction to 75 percent power	10,000
Weekly reduction to 50 percent power	2,000
Control rod worth test	400
Loss of feedwater heaters (80 cycles total)	
Turbine trip at 25 percent power	10
Loss of heating to feedwater heater	70
Scram (200 cycles total)	
Loss of feedwater pumps, isolation valves close	10
Turbine trip, feedwater on, isolation valve stay open	40
Reactor overpressure with delayed scram, feedwater stays on, isolated valves stay open	1
Single relief valve or safety valve blowdown	2
All other scrams	147
Improper start of cold recirculation loop	5
Sudden start of pump in cold recirculation loop	5
Shutdown (100-F/hr cooldown rate)	118
Hydrostatic test at 1,563 psig	3
Unbolt	123