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Docket No. 50-278

Mr. Edward G. Bauer, Jr.
Vice President and General Counsel
Philadelphia Electric Company
2301 Market Street
Philadelphia, Pennsylvania 19101

Dear Mr. Bauer:

During the current maintenance/refueling outage for the Peach Bottom Atomic Power Station, Unit 3, inspection of recirculation system piping revealed a number of unacceptable ultrasonic indications. You took corrective action to repair this piping and reported the results of the inspection and analysis including repair procedures in letters to the NRC dated August 9, 1983, August 22, 1983, and August 30, 1983.

Based on our review of these submittals and related discussions with our staff, we found that the corrective actions you took were acceptable for plant restart, but that additional requirements should be implemented for operation during the next cycle. The bases for the staff's conclusion are contained in the attached Safety Evaluation.

By letters dated August 9, 1983 and August 24, 1983, you committed to implement these requirements. Based on these commitments and the acceptability of your corrective actions, the Commission has issued the enclosed Confirmatory Order, a copy of which is being filed with the Office of the Federal Register for publication. With the issuance of this letter, plant restart is hereby authorized.

Sincerely,

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Confirmatory Order
2. Safety Evaluation

cc w/enclosures:
See next page

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Final concurrence by phone

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BDLiaw
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SURNAME	RIngram	GGears:cf	JStolz	GLainas	Karen Cyr	DEisenhut	FMiraglia
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
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September 1, 1983

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Docket No. 50-278

Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: PEACH BOTTOM ATOMIC POWER STATION, UNIT 3

Two signed originals of the Federal Register Notice identified below are enclosed for your transmittal to the Office of the Federal Register for publication. Additional conformed copies (6) of the Notice are enclosed for your use.

☐ Notice of Receipt of Application for Construction Permit(s) and Operating License(s).

☐ Notice of Receipt of Partial Application for Construction Permit(s) and Facility License(s): Time for Submission of Views on Antitrust Matters.

☐ Notice of Availability of Applicant's Environmental Report.

☐ Notice of Proposed Issuance of Amendment to Facility Operating License.

☐ Notice of Receipt of Application for Facility License(s); Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility License(s) and Notice of Opportunity for Hearing.

☐ Notice of Availability of NRC Draft/Final Environmental Statement.

☐ Notice of Limited Work Authorization.

☐ Notice of Availability of Safety Evaluation Report.

☐ Notice of Issuance of Construction Permit(s).

☐ Notice of Issuance of Facility Operating License(s) or Amendment(s).

☒ Other: Confirmatory Order confirming licensee commitments on pipe crack related issues.

Division of Licensing, ORB#4
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

OFFICE→	ORB#4:DL					
SURNAME→	RIngram;cf					
DATE→	9/6/83					

NRC FORM 102 (1-76)

Philadelphia Electric Company

cc w/enclosure(s):

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of PHILADELPHIA ELECTRIC COMPANY (Peach Bottom Atomic Power Station, Unit 3)	))))	Docket No. 50-278

ORDER CONFIRMING LICENSEE COMMITMENTS
ON PIPE CRACK RELATED ISSUES

I.

The Philadelphia Electric Company (the licensee) and three other co-owners are the holders of Facility Operating License No. DPR-56 which authorizes the operation of the Peach Bottom Atomic Power Station, Unit 3 (the facility), at steady-state power levels not in excess of 3293 megawatts thermal. The facility is a boiling water reactor located at the licensee's site in York County, Pennsylvania.

II.

During the current 1983 refueling outage at Peach Bottom, Unit 3, augmented inservice inspection was performed on the recirculation and residual heat removal (RHR) system piping in accordance with Office of Inspection and Enforcement Bulletin 83-02. The original sample size was expanded to 112 welds after ultrasonic (UT) indications were reported on welds in the original sampling. This represents a total of 75% of all welds in these systems. A total of 37 welds in the recirculation and RHR systems were not inspected. Examinations of these uninspected welds were either not necessary (e.g., conforming material as per NUREG-0313, Revision 1), or not practicable (e.g., high radiation, etc.).

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The licensee also provided information on welds on the Reactor Water Clean-up (RWCU) and Core Spray Systems. Both systems have had substantial portions of their piping replaced with conforming material to reduce susceptability to intergranular stress corrosion cracking (IGSCC). A total of 25 non-conforming welds have not been inspected in these two systems. The licensee provided technical justification for not inspecting these welds. The staff has reviewed the licensee's submittal and has concluded that the uninspected welds are not likely to be cracked to the extent of compromising the safety of the plant.

Overall, out of a total of 112 welds inspected, a total of 15 were found to show linear indications: ten 12-inch riser to elbow welds, and five residual heat removal (RHR) system 20-inch suction line welds. All indications were in the weld heat-affected-zone. In the 10 12-inch riser welds, all indications are oriented in an axial direction. The deepest indication reported in the 12-inch riser welds is 92% of wall thickness with a length of about 0.75 inches. The deepest indication in the large-size pipe welds is 40% of the wall thickness in a 20-inch RHR weld with a length of about 32 inches. In the five 20-inch RHR welds, the indications are oriented predominatly in a circumferential direction.

Evaluation by the licensee, submitted by letters dated August 9, 1983, August 22, 1983 and August 30, 1983, indicates that the projected crack sizes, due to intergranular stress corrosion cracking (IGSCC) and fatigue

crack growth, in two of the 20-inch RHR welds at the end of an 18-month fuel cycle would be within the ASME Code limits.

The licensee's evaluation also showed that the 10 12-inch riser welds and the three 20-inch RHR suction line welds required repair for continued service because their calculated projected cracks would exceed the Code limits at the end of an 18-month fuel cycle.

However, all five RHR suction line welds and the ten 12-inch riser welds were repaired using a weld overlay process. The minimum overlay thickness for the riser welds is 0.25-inch and for the 20-inch RHR weld it varies from 0.25-inch to 0.5-inch. The overlay thickness is designed to meet the Code allowable limits in the new ASME Code Section XI IWB 3600 assuming the presence of through-wall cracks. The length of the overlay varies approximately from 4 to 7 inches and is designed to reduce the stress at the end of the overlay acting on the crack location. RHR welds 10-0-7, 10-0-10, and 10-0-15 are bi-metallic welds (carbon steel to stainless steel) and the overlay is applied only to the stainless steel side of the welds. Personnel of Region I confirmed that the weld overlay repairs were performed in accordance with the qualified and approved procedures consistent with ASME Code requirements.

The staff has reviewed the licensee's submittals including analysis of weld overlay design and the calculation of IGSCC crack growth, based

on current crack growth data, to support the continuing service for an 18-month fuel cycle with the 15 overlay repaired welds. Region I personnel confirmed that the licensee's UT procedures, calibration standards, equipment and IGSCC detection capabilities were acceptably demonstrated in accordance with IE Bulletin 83-02.

The licensee's overlay design analysis performed by General Electric is based on the conservative assumption that all cracks are through-wall cracks. Consequently, its analysis did not depend on and assumptions concerning UT sizing and the IGSCC crack growth rate. The required minimum overlay thickness for each defective weld is calculated by using the methodology allowed in the new ASME Code Section XI IWB 3600 to meet the required Code safety margin. For normal and upset condition, a safety margin of three is required and for the faulted and emergency condition, a safety margin of 1.5 is required. Because the acceptable flaw in the normal condition based on new IWB 3600 is more limiting, the acceptable flaw for the faulted condition need not be considered.

For RHR welds 10-0-5 and 10-0-6, the allowable flaw depth based on new IWB 3600 is conservatively calculated to be 75.5% and 78.1% of wall thickness respectively. This calculation assumes the flaws to be fully circumferential in length and through the original pipe thickness in depth. This assumption is very conservative because the worst reported UT indication in these two welds is about 40% through-wall in depth and less than half of the full circumference in length. The General Electric analysis has shown that a minimum overlay thickness of 0.5 inch is more

than enough to make the assumed through-wall cracks (63% and 65% of the overlay repaired wall thickness) in these two welds meet the new calculated Code allowable limits (75% and 78%). An allowable flaw depth of 82% was similarly calculated for RHR weld 10-0-7. This weld has a worst reported flaw with about 35% of wall thickness in depth and about 7 inches in length. With a minimum overlay thickness of 0.35 inch applied to this weld, the assumed fully circumferential through-wall crack (73% of the overlay repaired wall thickness) is well within the new calculated Code allowable limit (82%).

The indications reported by General Electric on RHR welds 10-0-10, and 10-0-15 may be overcalls because two independent UT examinations on the same two welds did not find any reportable indications. However, the licensee decided to apply an overlay with a minimum thickness of 0.25 inch to these two welds to increase the safety margin in structural strength and to prevent any potential leakage.

We reviewed the weld overlay design calculation made by General Electric. We concur in their conclusion that the overlay used will provide adequate reinforcement with Code required safety margin for a least the next fuel cycle of operation.

III.

Although the calculations discussed above indicate that the cracks in the 15 overlay repaired welds will not progress to the point of leakage

during the next fuel cycle, and margins are expected to be maintained over crack growth which could compromise safety, uncertainties in crack sizing and growth rate still remain. Further, not all welds were examined, and significant cracks could be present in welds that were not examined.

Because of these uncertainties, we have determined that improvements in the monitoring in the containment for unidentified leakage are required; therefore, new limiting conditions for operation and surveillance requirements have been developed. These enhanced surveillance measures will provide adequate assurance that possible cracks in pipes will be detected before growing to a size that will compromise the safety of the plant.

The staff also has some concern regarding the long-term growth of IGSCC cracks and its effect on the long-term operation of the plant. Therefore, we have determined that plans for corrective action or modification including replacement of the recirculation and other reactor coolant pressure boundary piping systems during the next refueling outage must be submitted for staff review at least three months before the start of the next refueling outage.

By letters dated August 9, 1983 and August 24, 1983, the licensee committed to the above described conditions on leakage monitoring and early submittal of inspection and/or modification plans. I have determined that the public health and safety requires that these commitments should be confirmed by an immediately effective Order.

IV.

Accordingly, pursuant to Section 103, 161i, 161o and 182 of the Atomic Energy Act of 1954, as amended, and the Commission's regulations in 10 CFR Parts 2 and 50, IT IS HEREBY ORDERED EFFECTIVE IMMEDIATELY THAT:

1. The licensee shall operate the reactor in accordance with the present requirements on coolant leakage in Sections 3.6.C and 4.6.C of the Technical Specifications, as modified by Attachment A to this Order.
2. Plans for corrective actions and/or modification, including replacement of the recirculation and other reactor cooling pressure boundary piping systems during the next refueling outage shall be submitted for NRC review at least three months before the start of the next refueling outage.
3. The Director, Division of Licensing, may in writing relax or terminate any of the above provisions upon written request from the licensee, if the request is timely and provides good cause for the requested action.

V.

The licensee may request a hearing within twenty (20) days of the date of publication of this Order in the Federal Register. Any request for a hearing shall be addressed to the Director, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. A copy shall also be sent to the Executive Legal Director at the same address. A REQUEST FOR A HEARING SHALL NOT STAY THE IMMEDIATE EFFECTIVENESS OF THIS ORDER.

If a hearing is to be held, the Commission will issue an Order designating the time and place of any such hearing. If a hearing is held concerning this Order, the issue to be considered at the hearing shall be whether the licensee should comply with the requirements set forth in Section IV of this Order.

This Order is effective upon issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Darrell G. Eisenhut, Director
Division of Licensing

Dated at Bethesda, Maryland
this 1st day of September 1983.

SURVEILLANCE AND LIMITING CONDITION OF OPERATION FOR PEACH BOTTOM ATOMIC POWER STATION, UNIT 3

LIMITING CONDITIONS FOR OPERATION

3.6.C. Coolant Leakage

1. Any time irradiated fuel is in the reactor vessel and reactor coolant temperature is above 212 degrees F, the rate of reactor coolant leakage to the primary containment from unidentified sources shall not exceed 5 gallons per minute. The rate of change of unidentified leakage shall not exceed 2 gallons per minute per 24 hour surveillance period when the reactor is operated in the "Run" mode. In addition, the total reactor coolant system leakage into the primary containment shall not exceed 25 gpm averaged over any 24 hour surveillance period.

2. The primary containment (Drywell) sump collection and flow monitoring system shall be operable during reactor power operation. From and after the time that this system is made or found to be inoperable for any reason, reactor power operation is permissible only during the succeeding 24 hours unless the system is made operable sooner. For purposes of this paragraph, the primary containment (Drywell) sump collection and flow monitoring system operability is defined as the ability to measure reactor coolant leakage.

3. If the conditions in 1 or 2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in at least Hot Shutdown within the next 12 hours and in Cold Shutdown Condition within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.C Coolant Leakage

1. Reactor coolant system leakage shall be determined by the primary containment (Drywell) sump collection and flow monitoring system and recorded every 4 hours or less.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
INSPECTION, ANALYSIS AND REPAIR OF THE
PEACH BOTTOM UNIT 3 RECIRCULATION SYSTEM
AND RESIDUAL HEAT REMOVAL (RHR) SYSTEM PIPING

PHILADELPHIA ELECTRIC COMPANY, ET AL.
DOCKET NO. 50-278

Introduction

During the current Peach Bottom Unit 3 refueling outage, Philadelphia Electric Company (PECo) performed augmented inservice inspection on a total of 112 Class 1 austenitic stainless steel piping welds, consisting of 77 welds in the Recirculation system and 35 welds in the RHR system, of which 40 were 12" riser welds and 72 were welds of 20" or larger in diameter. The inspected welds include all the welds (91 welds) that were treated with Induction Heating Stress Improvement (IHSI). The purpose of IHSI is to reduce the potential of susceptible welds to IGSCC. Except for nine welds in the 28" recirculation piping and three welds in the RHR piping, all nonconforming welds in the recirculation and RHR piping systems were inspected in this outage. Ultrasonic examination was performed on each of the 112 welds and additional radiographic examination using MINAC (Miniature Linear Accelerator) was performed on 20 12" riser welds where the riser pipes were cladded.

General Electric (GE) at King of Prussia, Sonic System International (a subcontractor of General Electric) and Southwest Research Institute (SWRE) performed the UT examination for PECo. Region I of NRC has determined that their UT procedures, calibration standards, equipment and IGSCC detection capabilities were satisfactorily demonstrated in accordance with I&E Bulletins 83-02 and 82-03, and the same procedures and techniques were used in this UT examination. The results of the UT examinations indicated that a total of 15 welds consisting of 10 12" riser to elbow welds in the recirculation system and five 20" riser to elbow welds in the RHR system showed reportable linear indications. All indications were in the weld heat-affected-zone (HAZ).

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In the 10 12" defective riser welds, all indications are oriented in axial direction. The worst axial indication was reported in weld 2-AHF-2 with a depth about 92% of wall thickness and a length about 0.75 in. In the five defective 20" RHR welds, the indications are oriented predominantly in circumferential direction. The worst circumferential indication was reported in weld 10-0-6 with a depth about 40% of wall thickness and a length about 32 inches.

General Electric at San Jose performed the flaw evaluation for PECo in accordance with the new Code Section XI IWB 3600. The new Code IWB 3600 provides flaw acceptance criteria for austenitic stainless steel piping based on the net section collapse approach which was approved by the ASME Code main committee this March, and is expected to be published later this year. The results of the evaluation indicate that 13 of the 15 defective welds (10 12" riser welds and three 20" RHR welds) require weld overlay repair because the final flaw sizes of these 13 welds at the end of an 18-month period (12,000 hours) would exceed the new Code allowable limits. However, PECo opted to weld overlay repair all 15 defective welds including the three 20" RHR welds whose final flaw sizes were shown to be within the new Code allowable limits at the end of an 18-month period.

General Electric performed the weld overlay repair on the 15 defective welds. The minimum overlay thickness for the riser welds is 0.25" and for the 20" RHR welds it varies from 0.25" to 0.5". The overlay thickness is designed to meet the Code allowable limits in the new IWB 3600 assuming the presence of through-wall cracks. The length of the overlay varies approximately from 4 to 7 inches and is designed to reduce the stress at the end of the overlay acting on the crack location. RHR welds 10-0-7, 10-0-10, and 10-0-15 are bi-metallic welds (carbon steel to stainless steel) and the overlay is applied only to the stainless steel side of the welds. Region I of NRC has confirmed that the weld overlay repairs were performed in accordance with the qualified and approved procedures consistent with ASME Code requirements.

During the overlay of RHR weld 10-0-6, a circumferential through-wall crack about 2 inches in length was found. PECo indicated that the reason the original UT examination missed this through-wall circumferential crack is due to the presence of an oversized crown on the subject weld which prevented a proper UT examination of the upper portion of the weld. This through-wall crack did not affect the adequacy of the overlay repair because the overlay was designed for a fully circumferential through-wall crack. The licensee submitted by letter dated August 30, 1983, its evaluation and justification for those welds in the Recirculation, RHR, Reactor Water Cleanup (RWCU) and Core Spray Systems which have not been inspected during this outage.

Evaluation

We reviewed the licensee's submittals including General Electric's overlay design analysis to support the continuing service for an 18-month period (12,000 hours) of the 15 overlay repaired welds.

General Electric's overlay design analysis is based on the conservative assumption that all cracks are through-wall cracks eliminates the concern regarding the uncertainties in the UT sizing and the IGSCC crack growth rate because they will not be considered in this analysis. The required minimum overlay thickness for each defective weld is calculated by using the methodology allowed in the new IWB 3600 to meet the required Code safety margin. For normal and upset condition, a safety margin of 3 is required and for the faulted and emergency condition, a safety margin of 1.5 is required. Because the acceptable flaw in the normal condition based on new IWB 3600 is more limiting, the acceptable flaw for the faulted condition need not be considered.

For RHR welds 10-0-5 and 10-0-6, the allowable flaw depth based on new IWB 3600 is conservatively calculated to be 75.5% and 78.1% of wall thickness respectively. This calculation assumes the flaws to be fully circumferential in length and through the original pipe thickness in depth. This assumption is very conservative because the worst reported UT indication in these two welds is about 40% through-wall in depth

and less than half of the full circumference in length. GE's analysis has shown that a minimum overlay thickness of 0.5 inch is more than enough to make the assumed through-wall cracks (63% and 65% of the overlay repaired wall thickness) in these two welds meet the new calculated Code allowable limits (75% and 78%). An allowable flaw depth of 82% was similarly calculated for RHR weld 10-0-7. This weld has a worst reported flaw with about 35% of wall thickness in depth and about 7 inches in length. With a minimum overlay thickness of 0.35 inch applied to this weld, the assumed fully circumferential through-wall crack (73% of the overlay repaired wall thickness) is well within the new calculated Code allowable limit (82%).

The indications reported by GE on RHR welds 10-0-10 and 10-0-15 may be overcalls because both LMT and SWRE had examined the same two welds and did not find any reportable indications. Region I of NRC reviewed all the UT data recorded by the three UT teams and agreed with the LMT and SWRE's conclusion that there are no reportable indications in RHR welds 10-0-10 and 10-0-15. However, PECo decided to apply an overlay with a minimum thickness of 0.25 inch to these two welds to increase the safety margin in structural strength and to prevent any potential leakage.

We reviewed the weld overlay design calculation made by GE. We agree with their conclusion that the overlay used will provide adequate reinforcement with the Code required safety margin for at least the next fuel cycle of operation (12,000 hours).

GE also performed an evaluation of the overlay shrinkage effect by using a finite element model of the Recirculation and RHR piping systems. The shrinkage was conservatively assumed to be 1/4 inch for the riser weld overlay and 3/8 inch for the RHR weld overlay. GE indicated that the maximum calculated nominal stress due to the weld shrinkage is about 5000 psi and 16,000 psi in the Recirculation

and RHR systems, respectively, which is well below the Code allowable value of the yield strength (25 Ksi) of the material. The subject shrinkage stresses are not expected to have any significant deleterious effect on the Recirculation and RHR piping systems.

In its August 30, 1983 submittal pertaining to uninspected welds, the licensee reported that for the Recirculation and RHR piping systems a total of 17 Class 1 non-conforming welds, of which 12 are in the large diameter lines ($\geq$20") and 5 in small diameter lines ($\leq$ 6") were not UT examined during this outage. Ten (8, 28" welds and 2, 22" welds) of the 12 uninspected large diameter welds were located in the high radiation area (up to 10 R/hr on contact) and required removal of large quanties of insulation. These welds were not examined based on ALARA considerations. The other two uninspected large diameter welds (24") in the RHR return loops were located between the penetration and the valve outside the drywell. Because of physical restriction, these two welds cannot be UT examined. Five small diameter uninspected welds were all weldolet welds which are not readily ultrasonically examinable.

The licensee has inspected 72 large diameter (20" and above) welds and no defective welds were found for welds in pipes with a diameter of 22" or above. Therefore, based on the results of this large sampling, we believe extensive cracking is not likely to be present in those 12 uninspected large diameter welds ($\geq$ 22"). As a compensatory measure to ensure early detection of potential leakage in repaired and uninspected welds, the licensee has installed moisture sensitive tapes on 30 selected large diameter welds including 10 uninspected

large diameter welds inside the drywell and 5 uninspected weldolet welds. Based on the above discussion, we conclude that the 17 uninspected welds in the recirculation and RHR systems are not likely to be cracked to the extnet of compromising the safety of the plant. Even if they were to crack through the wall fully, the resultant leakage would be detected by the moisture sensitive tape leak detection system.

The licensee did not inspect the non-conforming welds in the reactor water cleanup system (RWCU) (6") and Core Spray (10" and 12") piping systems during this outage. There are non-conforming welds in the RWCU system which are not inspectable due to physical limitations. There are 22 non-conforming welds in the two loops of the Core Spray system. The licensee indicated that the portions of the Core Spray piping containing the 22 non-conforming welds were operated at a temperature less than 200°F and are considered to be not susceptible to IGSCC. We do not fully agree with the licensee's justification; however, we do believe the low temperature (≤250°F) portions of piping are less susceptible to cracking.

We also note that the portion of the RWCU piping containing the 3 non-conforming welds in the RWCU piping system is isolable and, therefore, will not impact the integrity of the reactor coolant primary pressure boundary. The licensee also indicated that they will install moisture sensitive tapes on these 3 welds in the RWCU system to ensure early detection of potential leakage. Based on the above discussion, we conclude that the uninspected welds in the RWCU and Core Spray systems are not likely to be cracked to the extent of compromising the safety of the plant. Even if they were to crack through the wall fully, the resultant leakage would be detected by the leak detection system.

Leak Detection

Although a large portion of the nonconforming stainless steel welds in Peach Bottom Unit 3 was examined and all defective welds found were weld overlay repaired, not all welds were examined and significant cracks could be present in welds that were not examined. Because of this concern, it is prudent to improve the requirements for monitoring for unidentified leakage.

The required additional monitoring and tighter limits on unidentified leakage are summarized below:

(1) An additional operational limit on reactor coolant system leakage of an increase in unidentified leakage of two gallons/minute or more within any 24-hour period or less (except during the first 24 hours after startup). On exceeding this limit, the reactor should be placed in a cold shutdown condition within 24 hours for inspection.

(2) The leakage measured from each sump should be recorded every 4 hours.

(3) At least one of the leakage measurement instruments associated with each sump should be operable.

We conclude that implementation of the above measures will provide adequate assurance that possible cracks in pipes will be detected before growing to a size that will compromise the safety of the plant.

Summary and Conclusions

We have reviewed Philadelphia Electric Company's submittals regarding the actions taken or to be taken during this refueling outage on the inspection, analyses and repairs of recirculation and RHR piping system welds in the Peach Bottom Unit 3 plant. This includes a description of the defects found, description of repairs, stress and fracture analysis.

We conclude that the Peach Bottom Unit 3 plant can be safely returned to power and operated in its present configuration at least for one 18-month fuel cycle (12,000 hours) provided that the following items are satisfactorily completed:

(1) The Code-required hydrostatic test and nondestructive examination on overlay repaired welds should be satisfactorily completed prior to startup.

(2) The additional leak detection requirements as listed in the section on Leak Detection should be properly implemented prior to startup.

Nevertheless, we still have concern regarding the long-term growth of small IGSCC cracks that may be present but not detected during this inspection. Therefore, we require that plans for inspection and/or modification of the recirculation and other RCPB piping systems during the next refueling outage be submitted for our review at least three months before the start of the next refueling outage.