Thomas N. Mitchell
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March 6, 1998

Docket No. 50-278

License No. DPR-56

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

SUBJECT: Peach Bottom Atomic Power Station, Unit 3
Jet Pump Riser Piping

REFERENCE: Letter from G. A. Hunger, Jr. To U. S. Nuclear Regulatory Commission, dated December 22, 1997

Dear Sir:

The purpose of this letter is to provide you with an update regarding the operating strategy for Peach Bottom Unit 3, consistent with the agreement communicated to you in the Reference letter. Due to a delay in our readiness to implement permanent repairs to the subject jet pump risers, a minor change to the Unit 3 operating period has been incorporated into the 10CFR 50.59 Safety Evaluation associated with this issue. This did not require a technical re-analysis, but did necessitate a minor adjustment to the Safety Evaluation, which was reviewed and approved by the Plant Operations Review Committee (PORC) today. A copy of the evaluation is enclosed for your information.

If you have any questions, please do not hesitate to contact us.

Sincerely,

TNMitchell

T. N. Mitchell
Vice President
Peach Bottom Atomic Power Station

enclosure

OAL/GLJ/TNM:ljp

A001/1

cc: H. J. Miller, Administrator, Region I, USNRC
A. C. McMurtray, USNRC Senior Resident Inspector, PBAPS

CCN 98-14018

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PDR ADOCK 05000278
P PDR

Effective Date: 08-21-97

PORC / SQR / Chesterbrook Review & Approval Form

******************Section 1 - Description of Item ****************

1. Item(s) - (Procedure or Document #, draft rev #, Title): 10CFR50.59 REVIEW FOR JET PUMP THERMAL SLEEVE CRACKING, UNIT 3, REV. 3

Documents to be superseded: 10CFR50.59 REVIEW FOR JET PUMP THERMAL SLEEVE CRACKING, UNIT 3, REV. 2

2. Prepared By: KAREN TOM Group: PEDC Ext: 807:4706

3. Indicate Approval Organization(s) req'd: CB ____ LGS ____ PBAPS X

4. Full Description of Activity or Change: REVISE SPECIFIED OPERATING CONDITIONS TO INCLUDE ADDITIONAL HOURS BASED ON REVISED GENE ANALYSIS, WHICH UTILIZED ACTUAL JET PUMP DRIVE FLOW DATA OBTAINED FROM RT-R-02F-900-3.

5. If a TC is being fully incorporated in this change, list TC#(S) here (if multiple documents approved on this form, state which procedures incorporate which TC's):

************ Section 2 - Review and Planning of Activity ************

6. Applicable items in AG-CG-4-2 table are satisfied and commitments annotated - Preparer's Initials: KT

6a. For Admin Procedures, AA-C-5 checklist completed and attached - Preparer's Initials: N/A

7. Mechanisms in place to implement training - Preparer's Initials: KT Training Review (LGS only): ____

8. Procedure Cause Codes (see back) - List the program that initiated this change : OTH

8a. List below the program that failed to identify the change before the one in step 8:

Program: ISI Give Basis for determination: IVVI PER SPEC. M-733 (ISI)

9. 10 CFR 50.59 Review - check appropriate item and attach review as necessary:
Determination Only ____ NO Review Req'd per LR-C-13 (Minor Non-Technical Revision) ____
No Review Req'd (provide basis in descrip) ____ Safety Eval (PORC req'd) ✓ USQ: Yes ____ No ✓

10. 10 CFR 50.54 Reviews (EP, QA, Sec Plan changes or for LGS only, environmental impact) - attach review

11. Document Services Instructions: Procedure Use (circle one): I II III No Change

	Normal Distribution		Expedited Distribution (24 hrs)		Hold - Contact:

Need by date: ________ Other Instructions:

Specific Effective Date if req'd: ________ (Weekly tests performed daily should have a Monday eff date)

************** Section 3 - SQR/RS Review and Approval *************

12. Cross-Discipline Review - perform per Exh AG-CG-4-2. Attach add'l sheets as necessary.

SQR XD	Name	Site/Group	Initials	Date	SQR XD	Name	Site/Group	Initials	Date
	KAREN TOM	Preparer/PEDC	KT	3/6/98					
✓	Alex Psaros	PEPR	AP	3/6/98					

Preparer's Initials that all comments resolved: KT SQR Initials if no-cross discipline review: ____

13. ~~SQR~~/Mgr Review Signature: McDelowery Date: 3/6/98

13a. Quality Reviewer Signature (for Quality admin procedures): ____ Date: ____

14. Responsible Superintendent (Sr Mgr or Dir for CB) Review/Approval - check appropriate box:

Reviewed and Approved ✓ Reviewed Only, PORC req'd ____ Reviewed Only, Pl Mgr Approval req'd (LGS only) ____

RS (Sr Mgr or Dir for CB) Signature: [signature] Date: 3/6/98

PORC / SQR / Chesterbrook Review & Approval Form

Item(s) From Line 1 - (Procedure or Document #, draft rev #, Title):

10CFR50.59 REVIEW FOR JET PUMP THERMAL SLEEVE CRACKING, UNIT 3, REV. 3

*********************** Section 4 - PORC Review ***********************

15. PORC Review

X	Approval Recommended
	Approval Recommended w/ below resolution
	Reviewed Only, no action taken
	Remanded

Items for Resolution: ______________________________

Resolution Checked by: Date:

PORC Mtg# 98-009 PORC Date: 3-6-98

PORC Chairman Signature: [signature]

16. Walkaround PORC (limited use only - see AG-CG-4):

PORC Chairman Approval for walkaround: Walkaround #: Date:

Initials signifying acceptance by PORC members/alternates (ensure at least 4 individuals of which at least 2 are members):

____ ____ ____ ____

PORC Chairman Approval of Item: Date:

17. Plant Manager (as necessary - see Exhibit AG-CG-4-2):

Plant Manager: [signature] Date: 3-6-98

NQA/QR Approval: ______________ Date: ________

Vice President: ______________ Date: ________
(for Directives only)

18. PORC Minutes (req'd for all items presented to PORC or PORC Walkaround, whether approved or not. Provide electronic copy to PORC Sec'y - see Exhibit AG-CG-4-2 for minutes requirements).

*******************Procedure Cause Codes************************

The following cause codes are to be used when answering items 8 and 8a on first page of form:

CTP - Commitment Tracking/Operating Experience Program	PEP(*) - Performance Enhancement Program
DBD - Design Basis Document Program	PPIS(*) - Procedure Performance Improvement System
DCR - Design Change Request Program	SET - Setpoint Change Program
DEC - Design Equivalent Change Program	TCP(*) - Temporary Change Process
LIC - Licensing Program (e.g. Tech Spec, UFSAR)	TPA - Temporary Plant Alteration Program
MOD - Mod Process (Big)	VEM - Vendor manual Program
MPC - Minor Physical Change Program	OTH - An Unlisted Process - list in basis (step 8a)

(*) - When this program is identified for initiating a change to a procedure, consideration should be given that another process should have or could have initiated this change earlier.

Peach Bottom Atomic Power Station Unit 3
10 CFR 50.59 Review for Jet Pump Thermal Sleeve Cracking
Revision 3

I. Subject

Revision 3 of this 10 CFR 50.59 Review was performed to address changes to the "specified operating condition." Actual jet pump drive flow data obtained from RT-R-02F-900-3 (Reference 23), was utilized in the GENE analysis to justify extending the operating period. This change does not affect system flow leakage rates and the predicted length of the crack at the end of the specified operating period. All other methodologies, inputs and assumptions utilized in the jet pump analysis remain unchanged with the exception of the actual flow history data recorded under RT-R-02F-900-3.

Revision 2 of this 10 CFR 50.59 Review was performed to redefine the "specified operating condition". This revision will allow the flexibility to operate at two different Reactor Coolant Recirculation drive flows for specified periods of time and stay within the bounds of the GENE analysis. This change also affects the Reactor Coolant Recirculation system flow leakage rates and the predicted length of the crack at the end of the specified operating period. All other aspects of the 50.59 Review are valid and unchanged.

Revision 1 of this 10 CFR 50.59 Review was required to increase the postulated flow leakage values which were incorrectly presented in Revision 0, due to a computational error found during NCR 97-02899 reviews. All other aspects of the Revision 0 version of this 50.59 Review are valid and unchanged. Post -LOCA LPCI leakage is within the allowed value identified in the SAR.

During Peach Bottom Unit 3 Refueling Outage (3R11) In-Vessel Visual Inspections (IVVI) of this location were conducted per reference 9. Cracks were found in the weld HAZ joining the Recirculation inlet nozzle thermal sleeve to the elbow on three Jet Pump riser assemblies. The cracks were found on the thermal sleeve side of the weld on the risers associated with Jet Pumps 1 and 2 (Nozzle N2E at 150 deg. Azimuth), 9 and 10 (Nozzle N2A 30 deg. Azimuth), and 13 and 14 (Nozzle N2J at 300 deg. Azimuth). The cracks at 30 and 150 degrees are on the "B" loop of the Reactor

Recirculation system and the crack at 300 degrees is on the "A" loop of the Reactor Recirculation system.

This 10 CFR 50.59 Review will address the **INTERIM USE-AS-IS** disposition of NCR 97-02899 for cracks on the Jet Pump riser elbow to thermal sleeve weld heat affected zone (HAZ). The **INTERIM USE-AS-IS** disposition is valid for continued operation within the Reactor Coolant Recirculation drive flow and time constraints evaluated by GENE (Reference 23).

The **INTERIM USE-AS-IS** disposition allows for administratively controlling (Ref. 20) Reactor Coolant Recirculation drive flow for two operating scenarios, taking into consideration actual operation through 0600 hours on March 5, 1998. Additional hours were obtained based on the difference between the bounding drive flow/time constraints and the actual drive flow/time history recorded in RT-R-02F-900-3. These scenarios maintain the safety margin that was the basis for the previous analyses.

Based on a starting date of March 5, 1998 at 0600 hours:

1. Operate at a NOMINAL value of up to 15.75 Mlbm/hr for each recirculation loop for a period of up to 40 hours and at a NOMINAL value of up to 13.85 Mlbm/hr for each recirculation loop for a period of up to 800 hours,

or

2. Operate at a NOMINAL value of up to 15.75 Mlbm/hr for each recirculation loop for a period up to 130 hours and at a NOMINAL value of up to 13.85 Mlbm/hr for each recirculation loop for a period of up to 250 hours.

Operations at these flow rates may happen at any time (i.e. raise and lower reactor power) as long as the total hours are within the specified limits. These scenarios were selected as an operating strategy, hereinafter known as the "**specified operating condition**".

Transients, outside the specified operating condition, such as single loop operation or excursions above nominal values are bounded by the previous analyses (Ref. 1, 21, 22). Extended single loop operation greater than 24 hours will be evaluated by engineering for impact on the specified operating conditions.

The result of operating at higher Reactor Coolant Recirculation drive flows for extended periods of time could reduce the operating period. Additionally, operating at lower Reactor Coolant Recirculation drive flows could extend the operating period. This will require evaluation by engineering for impact on the specified operating conditions to assure compliance with the drive flow/time constraints that has been evaluated by GENE. Drive flow will be monitored and tracked administratively by using plant procedures.

II. Discussion

Jet Pump Configuration

The Jet Pumps are Reactor Vessel Internals and in conjunction with the Reactor Coolant Recirculation system are designed to provide forced circulation to the core for heat removal from the fuel. The Jet Pumps are located in the annulus region between the core shroud and the vessel wall. Since the Jet Pump suction elevation is at 2/3 core height, the reactor core will remain covered to this height even with a complete break of the Recirculation piping as assumed in the design basis accident (DBA). During post-LOCA LPCI operation, the Residual Heat Removal system pumps take suction from the suppression pool and discharge into the core region of the reactor vessel through the recirculation loops (i.e. through the Jet Pumps into the core region). LPCI helps to restore and maintain the coolant inventory in the reactor vessel such that the core is adequately cooled to preclude fuel clad temperature in excess of 2,200 deg. F following a design basis LOCA (Ref. 4).

Each Reactor Coolant Recirculation loop contains ten Jet Pumps. Recirculated coolant passes down the annulus between the Reactor Vessel wall and the Core Shroud. Approximately one third of the coolant flows from the vessel, through the two external recirculation loops, and becomes the driving flow for the Jet Pumps. Each of the two external recirculation loops discharge high pressure flow into an external manifold from which individual recirculation inlet lines are routed to the Jet Pump risers within the Reactor Vessel. The remaining portion of the coolant mixture in the annulus becomes the suction flow for the Jet Pumps. This flow enters the Jet Pumps at suction inlets and is accelerated by the drive flow. The drive flow and the suction flow are mixed in the Jet Pump throat section. The total flow then passes through the Jet Pump diffuser section into the area below the core (lower plenum), gaining sufficient head in the process to drive the required flow upward through the core.

The recirculation inlet nozzle thermal sleeve is welded to the nozzle safe end at its outer extremity and to the jet pump riser elbow at its inner extremity. The thermal sleeve is designed to provide a pressure retaining flow path for Reactor Coolant Recirculation drive flow to the Jet Pumps. Secondarily, the thermal sleeve reduces temperature variations, and thus thermal loading, on the recirculation inlet nozzle. The thermal sleeve is not a primary pressure boundary.

The thermal sleeve is 10" schedule 40 stainless steel type 304 pipe. The thermal sleeve to riser elbow joint is a field weld, performed during Jet Pump installation into the Reactor Vessel. The welding process was gas tungsten arc with type 308 filler material. During weld preparation, the thermal sleeve was counter-bored for appropriate fit up to the schedule 20 riser elbow. The welds are non-flux, non-creviced, full penetration butt welds (Ref. 8 & 17).

<u>Crack Description/ Geometry</u>

Based on a review by an expert metallurgist from PECON Testing and Laboratories, visual examination of all the Jet Pump indications are characteristic of Intergranular Stress Corrosion Cracking (IGSCC) in the heat affected zone of the austenitic stainless steel circumferential pipe weld. The cracking is away from the toe of the weld (approximately 1/8" to 1/4") and jagged in appearance. The crack ends were intermittent and at the same relative distance from the toe of the weld. No indication of fatigue crack growth was observed in that the crack tips did not turn and follow the toe of the weld, the cracks were jagged and not straight lined, and no crushing of the crack faces was observed. However, boat samples were not obtained of the crack tips to rule out the possibility of fatigue cracking.

The initial visual examinations were performed using modified VT-1 (1 mil wire) standards. Supplemental ultrasonic examinations (UT) were performed at the crack locations and the results are listed below.

Crack on Jet Pumps 1 and 2 Riser
The thermal sleeve-to-elbow weld has a crack from 329.5 deg. through 84.6 deg., looking in the direction of flow. This corresponds to a length of

10.8 +/- 0.39 inches for uncertainty, consistent with BWRVIP protocol. The flaw is on the Thermal Sleeve side of the weld.

Crack on Jet Pumps 9 and 10 riser
The thermal sleeve-to-elbow weld has a crack from 12.1 deg. through 30.2 deg. looking in the direction of flow. This corresponds to a length of 1.7+/- 0.34 inches for uncertainty, consistent with BWRVIP protocol. The flaw is on the Thermal Sleeve side of the weld.

Crack on Jet Pumps 13 and 14 riser
The thermal sleeve-to-elbow weld has a crack from 305.5 deg. through 81.0 deg., looking in the direction of flow. This corresponds to a length of 12.7 +/- 0.34 inches for uncertainty, consistent with BWRVIP protocol. The flaw is on the Thermal Sleeve side of the weld.

Root Cause

IGSCC is considered to be the most likely initiator of this cracking. The cracking on the Thermal Sleeve is similar to cracking identified to date in other Reactor Vessel Internals. Although not a creviced joint, stainless steel type 304 materials used for the thermal sleeve, in conjunction with past poor water chemistry conditions have made the joints susceptible. Records also indicate the possibility of these being cold sprung during installation thus increasing the residual stresses in the area and increasing the joints susceptibility to IGSCC.

Code Boundary

Jet Pump components are not part of the primary pressure boundary and do not provide a core support function. Jet Pumps are Safety Related and are optionally classified as ASME Section XI components for inspection purposes only. The Jet Pumps provide a Safety Related flow path during LPCI injection.

Flaw Evaluation

The determination of structural integrity was performed by using standard accepted methods for intergranular stress corrosion cracking and fatigue. Although examination of the crack indicates that IGSCC is the sole contributor, fatigue loading was also considered in developing allowable flaw sizes. The source for fatigue crack growth was determined by analytical methods to be low amplitude-high frequency vibration from the high velocity recirculation line flow.

The allowable flaw size at the elbow to thermal sleeve location was determined using standard limit load methodology presented in BWRVIP-41, "BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines" (Ref. 5). Similar methods have been previously used to evaluate other Vessel components such as the core spray lines and shroud. The flaw evaluation methodology used was performed consistent with ASME Section XI, Appendix C requirements (Ref. 7). This evaluation includes the ASME Section XI Safety Factors of 2.77 for Normal and Upset and 1.39 for Emergency and Faulted conditions. Load combinations are in accordance with the UFSAR and BWRVIP-41.

Once the allowable flaw size was determined, the acceptability of an observed flaw was determined by performing a crack growth analysis. This analysis considered both IGSCC and fatigue loading. The IGSCC growth was predicted using the conservative standard of 5×10^{-5} in/hr crack growth rate from each crack tip. This growth is the accepted bounding industry standard for IGSCC in austenitic stainless steels in a BWR environment with normal water chemistry. This is expected to be conservative since the Thermal Sleeve to elbow is a non-creviced weld and PBAPS injects hydrogen into feedwater at a rate equating to 0.3 ppm. The actual growth rate is expected to be on the order of 2.5×10^{-5} in/hr.

Each crack was then evaluated against its susceptibility to fatigue cracking. Fatigue cracking in the riser piping is primarily a result of flow induced vibration caused by the recirculation drive flow. A time history of stress amplitude vs. time for the Jet Pump risers was obtained using baseline testing of a BWR4/251" dia. Reactor Vessel (Browns Ferry Unit 1). During start-up testing at Browns Ferry, strain measurements on the Jet Pump riser braces were obtained at varying power levels and flow conditions. Measurements at the riser brace were scaled to the riser crack location by means of modal shape factors, determined analytically. Data corresponding to 100% core flow at 100% power were used to evaluate the influence of fatigue cracking on the subject risers.

Results of this analysis concluded the N2A riser cracking is small enough that the crack growth rate will not be influenced by fatigue cracking through the next 2 year cycle of full power operation (ΔK is less than ΔK threshold). Therefore, crack growth is limited to IGSCC and crack size will be limited to 3.7 inches by the end of the 2 year cycle and is acceptable to use-as-is.

For the N2E and the N2J risers, the stress intensity range for the assumed loading exceeds the threshold for susceptibility for fatigue cracking (ΔK is greater than ΔK threshold). When applying fatigue crack growth to both thermal sleeve cracks, the lengths would exceed the limit load allowable flaw size by end of cycle.

To mitigate the impact of flow induced vibration on the N2E and N2J Thermal Sleeve cracks, recirculation drive flow will be limited to the specified operating conditions. The predicted end of operating condition flaw sizes are listed below.

Peach Bottom Unit 3 Flaw Evaluation Summary

Location	Current Length* (in.)	Predicted** Length (in.)	Allowable Flaw Length (in.)	Percent of Allowable Flaw Length
JP 1 / 2	11.2	12.6	17.9	70.4%
JP 9 /10	2.1	3.7	17.9	20.7%
JP 13 / 14	13.1	14.9	17.9	83.2%

* Length was used in GE analysis and includes UT uncertainty, reference 1, 2, 3 and 21

** Flaw length predicted to occur at the end of operating period - based on the specified operating conditions. JP 9 / 10 is based on a 2 year normal operating cycle.

Leakage Evaluation

Due to the small crack opening area, any leakage through the cracks will be minimal. Postulated leakage will be approximately 345 GPM per loop for Reactor Coolant Recirculation flow at 15.75 Mlbm/hr loop drive flow and approximately 150 GPM for the inservice LPCI flows post-LOCA. This assumes the crack grows to the predicted flaw length. There is no specified allowable design leakage limit for the Reactor Coolant Recirculation flows and the postulated leakage is negligible when compared to system flows. The original design allowable leakage of 3000 GPM, for Low Pressure Coolant Injection (LPCI), will not be exceeded. Therefore, crack leakage during operations and post accident, for the specified operating conditions, will not impact any ECCS/LOCA analysis (Ref. 4).

III. Determination

1. Does the activity or discovered condition involve a Technical Specifications change or other Facility Operating (or possession only) License amendment?

No. Fracture Mechanics analysis of the cracks and evaluation of potential leakage of Recirculation coolant or LPCI (post LOCA) flow into the annulus region of the Reactor Pressure Vessel (RPV) has confirmed the operability of the subject Jet Pumps, Reactor Coolant Recirculation system and the Low Pressure Coolant Injection (LPCI) mode of the Residual Heat Removal system for the specified operating conditions. This analysis does not necessitate a change to surveillance requirements or limiting conditions of operation of the Jet Pumps, the Reactor Coolant Recirculation system or the LPCI mode of Residual Heat Removal (RHR) system due to the specified operating conditions on Reactor Coolant Recirculation pump flow. Therefore, the continued operation of the Jet Pumps, the Reactor Coolant Recirculation system and the LPCI mode of Residual Heat Removal system as-is does not require a Technical Specification change or any Operating License amendment.

2. Does the activity or discovered condition make changes to the facility as described in the SAR?

Yes. Continued operation of the subject Jet Pumps with cracking as described above is considered a change to the facility as described in the SAR. The original design and analysis of the Jet Pumps consisted of welded, slip joint and bolted connections. There is no consideration for cracking in the original Jet Pump design. Although the subject Jet Pumps are outside of the ASME Section XI boundary, they continue to meet the structural integrity safety margins as defined by ASME Section XI, 1989, Appendix C for the specified operating conditions, including all postulated crack growth.

Potential leakage paths from the floodable inner volume of the Reactor Vessel (e.g. 2/3 core height) during a Recirculation system pipe break and subsequent LPCI reflooding is documented in the SAR. Postulated leakage from the Jet Pump cracks during this condition has been calculated to be approximately 150 GPM for the inservice LPCI loop. This additional leakage is well within the 3000 GPM allowance designed in the LPCI subsystem for potential leakage paths but will be considered a change to the facility as described in the SAR. Additionally, two loops of LPCI flow through one Reactor Coolant Recirculation loop is less than the specified Recirculation flow limits evaluated for the specified operating condition. The associated piping stresses are therefore bounded by evaluated Reactor Coolant Recirculation system operation.

Another potential leakage path from the Jet Pump cracks, during operations, is inside the Reactor Vessel pressure boundary and would not have an unacceptable effect on the system performance of the Reactor Coolant Recirculation system. A computation was performed and has determined that potential leakage through the cracks is insignificant when compared to normal system flow through the riser piping. Since the leakage flow has been determined to be insignificant and contained within the Reactor Vessel pressure boundary this leakage is not considered to be a change to the facility as described in the SAR.

3. Does the activity or discovered condition make changes to procedures as described in the SAR?

No. Jet Pump operability is verified daily per Technical Specification requirements. Jet Pump dP measurements are used to determine operability and to calculate core flow and are unaffected by cracks on the Jet Pump risers.

The postulated leakage from the cracks will not manifest itself as an additional uncertainty in core flow measurement during plant operations since the leakage occurs upstream of the Jet Pump flow measurement instrumentation. Furthermore, the flow-biased portions of the APRM and Rod Block functions are not credited in the core reload licensing analysis. Thus the core reload licensing analysis is unaffected.

The flow signal used by the APRM system to establish flow biased rod block and scram trip setpoints is derived from the drive flow transmitters tapped off of the recirculation pipe venturis. The flow value is processed by the APRM flow units prior to use by the APRM system. Per Technical Specification Surveillance SR 3.3.1.1.7, the APRM drive flow signal is adjusted accordingly every 31 days to correspond to the total core flow. Therefore, the postulated leakage that may exist due to the Jet Pump Thermal Sleeve cracks will not impact the accuracy of the APRM flow biased setpoints since the flow signal is gained to correlate to core flow. The procedure that implements this surveillance is ST-I-60A-220-3, Drive Flow/Core Flow Correlation Check. In addition to this surveillance, the relationship between APRM flow and core flow is conservatively checked as part of the weekly APRM gain calibration procedure and as part of GP-2 and GP-5.

Based on the above discussion the activity or discovered condition does not make changes to procedures as described in the SAR?

4. Does the activity or discovered condition involve tests or experiments not described in the SAR?

No. Continued operation of the Jet Pumps, Reactor Coolant Recirculation system and the LPCI mode of RHR with cracks in the Jet Pump Thermal Sleeves does not involve any tests or experiments not described in the SAR. When applying accepted crack growth rates for the specified operating conditions to the flaw sizes identified on the Jet Pump thermal sleeves the flaw size is bounded by the limit load allowable flaw size summarized in reference 1. Therefore, margin exists in the remaining thermal sleeve ligaments to assure structural integrity and systems operability during the specified operating conditions interval. There are no additional tests or experiments involving plant systems or equipment required for verification of this analysis.

Since the answer to question 2 is yes, a Safety Evaluation is required for this proposed activity.

IV. Safety Evaluation

A. Those accidents potentially negatively impacted by this change include those accidents requiring an inner volume containing the core (e.g. 2/3 core height) that can be flooded following a break in the nuclear system process barrier external to the Reactor Vessel. The Abnormal Operating Transients potentially negatively impacted by this change are a Recirculation Pump trip, Restart of an Idle Recirculation Pump, and a Recirculation Flow Control Failure.

A-1 May the proposed activity or discovered condition increase the probability of occurrence of an accident previously evaluated in the SAR?

No. The safety design basis of a Jet Pump assembly is to provide a portion of the floodable inner volume containing the core. LPCI reflooding of the core, post-LOCA, through the Jet Pumps will prevent excessive fuel cladding temperatures ultimately, preventing undue hazard to the health and safety of the public. Initiators, assumed failures and sequences for transients and accidents are not affected. The current condition of the Jet Pumps is not a new accident initiator. GE's review of all postulated load combinations on the Jet Pumps has determined that load combinations including the design basis accident LOCA loads are bounding for all normal, derated, Abnormal Operational Transient, and Accident conditions, including those mentioned in "A" above.

The inner volume is defined as:
1. The Jet Pumps from the Jet Pump Nozzles down to the Shroud support.
2. The Shroud support which forms a barrier between the outside of the shroud and the inside of the Reactor Vessel.
3. The Reactor Vessel wall below the Shroud support.
4. The Shroud up to the level of the Jet Pump Nozzles.

Note: the identified cracks are not part of the inner volume.

A fracture mechanics evaluation at the specified operating conditions, using the crack lengths verified by the UT data (Ref. 2 & 3) and applying DBA loads, has validated the continued structural integrity of the Jet Pump assemblies for all postulated plant conditions. Therefore, there is no increase in the probability of occurrence of an accident previously evaluated in the SAR for the specified operating conditions.

A-2 May the proposed activity or discovered condition increase the consequences of an accident previously evaluated in the SAR?

No. The consequences of an accident previously evaluated in the SAR have not been increased due to cracks identified on the Jet Pump Thermal Sleeve. The flaw sizes identified on the Thermal Sleeves with calculated crack growth for the specified operating conditions are bounded by the allowable flaw size evaluation summarized in references 1 and 21. The safety function of the Jet Pumps is the passive function of maintaining 2/3 Core coverage, in conjunction with other Vessel Internals, and to provide a flow path for LPCI injection following a design basis accident. This function is an accident mitigator which allows reflooding of the core in the event of a breach in the nuclear system process barrier external to the Reactor Vessel. The bounding design basis accident is the Loss of Coolant Accident (LOCA) as defined in UFSAR Section 14.6.3. Therefore, margin exists in the remaining thermal sleeve ligament to ensure structural integrity and Jet Pump operability through the specified operating conditions. No safety limit will be impacted and no barrier design limits are compromised.

Due to the small total area open to flow at the crack locations, any leakage through the cracks during a LPCI reflood (post LOCA) will be minimal.

Leakage through the cracks, including projected crack growth at the end of the specified operating conditions, is calculated to be approximately 150 GPM for the inservice loop. This leakage is well within the allowable design leakage documented in the SAR for the LPCI mode of operation.

Since the Jet Pump structural integrity is assured and any additional leakage after LPCI reflooding is within existing system margins the existing accident analysis and assumptions are unchanged and valid for the specified operating conditions and the identified condition will not increase any onsite or offsite radiological conditions. Therefore, there will be no increased consequences of an accident previously evaluated in the SAR.

A-3 May the proposed activity or discovered condition create the possibility of a different type of accident than previously evaluated in the SAR?

No. The GE evaluation has supported the operability and the structural integrity of the Jet Pumps in terms of the component's ability to mitigate the consequences of an accident, as described above. Additionally the Jet

Pumps are not accident initiators and no new accident initiators will be created by operating with cracks in the Jet Pump Thermal sleeves for the specified operating conditions. For a change to create the possibility of an accident of a different type, the change must allow for a new fission product release path, result in a new fission product barrier failure mode, or create a new sequence of events that results in fuel cladding failures.

Since the structural integrity of the Jet Pump has been assured and there are no new failures modes introduced, there is no possibility of a different type of accident created other than those currently presented in the SAR.

B. Equipment Important to Safety that is potentially adversely impacted by this change includes the Jet Pump assemblies, LPCI injection capability through the Jet Pump, and the components comprising the Reactor Vessel Internals inner volume as defined in question A-1.

B-1 May the proposed activity or discovered condition increase the probability of occurrence of a malfunction of equipment Important to Safety previously evaluated in the SAR?

No. The safety function of the Jet Pumps is the passive function of maintaining 2/3 Core coverage, in conjunction with other Vessel Internals, and to provide a flow path for LPCI injection following a design basis accident. A fracture mechanics analysis has been performed to demonstrate the structural integrity of the Jet Pumps for the specified operating conditions. Therefore, there is no degradation in the ability of the Jet Pumps to perform their intended design function during the evaluated specified operating conditions. There is no impact on any other Reactor Vessel internals component included in the inner volume boundary which would be affected by cracks found on the Jet Pump Thermal Sleeve. All original design and seismic requirements of the Jet Pump are still met and no additional loads have been imposed. Postulated leakage has been evaluated and system performance of LPCI is determined to be within the allowable leakage limits.

Additionally, ST-O-02F-560-3 and ST-O-02F-550-3 verify the operability of the Jet Pumps by satisfying Technical Specification Surveillance's 3.4.1.1, 3.4.2.2, and 3.4.1.2, during operations greater than 25% reactor thermal power. Existing Off Normal procedure, ON-100, directs operator actions if there are operating symptoms indicative of a displaced Jet Pump Mixer. If a Jet Pump failure is confirmed the unit will be shutdown in accordance with GP-3, "Normal Plant Shutdown" per Technical Specification requirements. The analysis assures structural integrity and existing procedures will monitor safety performance and reliability of the Jet Pumps. Therefore, there is no increase in probability of occurrence of a malfunction of equipment Important to Safety for the specified operating conditions.

B-2 May the proposed activity or discovered condition increase the consequences of a malfunction of equipment Important to Safety than previously evaluated in the SAR?

No. The crack sizes identified in the Jet Pump Thermal Sleeve with conservative crack growth assumed through the specified operating conditions are bounded by the allowable flaw size evaluation performed by GE. Therefore, margin exists in the remaining ligament to assure structural integrity and Jet Pump operability through the specified operating conditions. No onsite or offsite radiological conditions assumed in the SAR will be affected.

Since the structural integrity of the Jet Pumps is assured, there are no increases to the consequences of a malfunction of equipment Important to Safety currently evaluated in the SAR.

B-3 May the proposed activity or discovered condition create the possibility of a different type of malfunction of equipment Important to Safety than any previously described in the SAR?

No. The GE evaluation supports the operability and the structural integrity of the Jet Pumps in terms of this equipment's (Important to Safety) ability to mitigate the consequences of an accident, as described above. Additionally, the Jet Pumps are not accident initiators and no new accident initiators will be created by operating with the evaluated cracks in the Jet Pump Thermal sleeves. No new failure modes of safety related system, structures, and components, initiation of a new limiting transient, or new sequence of events that can lead to a radiological release are created.

Since the structural integrity of the Jet Pump has been assured and there are no new failure modes introduced, there are no new or different types of malfunctions of equipment Important to Safety created, other than those currently presented in the SAR.

C-1 Does the proposed activity or discovered condition reduce the margin of safety as defined in the basis for any Technical Specification?

No. There are no specific margins associated with the structural integrity of the Jet Pumps as defined in the SAR or the Technical Specifications. However, the analysis described in Section II of this document establishes the Jet Pump will maintain its structural integrity with a Safety Factor greater than 2.77. This exceeds the minimum Safety Factor of 2.25 (normal/upset conditions) applied to other Vessel Internals outlined in UFSAR Table C.5.5.

Jet Pump operability will be monitored in accordance with Technical Specification Surveillance Requirements . Continued operability will assure that Jet Pumps will be able to perform the passive safety function of maintaining 2/3 core coverage and provide a LPCI flow path post-LOCA.

Leakage through the cracks during LPCI injection is calculated to be approximately 150 GPM for the inservice loop. This leakage is bounded by the allowable design leakage documented in the UFSAR Section 3.3.5.2.1.

The accuracy of the APRM flow-biased setpoints are not impacted. These setpoints do not have an associated margin of safety since they are not credited in any accident analyses.

Since the core flow measurement accuracy and uncertainty are unaffected, the licensing basis for the Safety Limit MCPR is unaffected, and there is no reduction in the margin of safety as described in the SAR.

Based on the above discussion the margin of safety as defined in the basis of the Technical Specifications have not been reduced.

D-1 Does this activity as proposed involve an Unreviewed Safety Question?

No. Based on the response for Sections IV parts A through C of this Safety Evaluation, continued operation of the subject Jet Pumps with the identified cracks, is acceptable and does not constitute an Unreviewed Safety Question.

E-1 Is a change to the UFSAR necessary?

Yes. The disposition of this Safety Evaluation documents that the subject Jet Pumps will continue to function as described in the UFSAR. The change will revise the identified leakage from the core inner volume during LPCI injection as documented in UFSAR Section 3.3.5.2.1. Documenting the cracks found in the Jet Pump thermal sleeves is beyond the level of detail described in the UFSAR.

E-2 Is a change to any other SAR document necessary? No.

SAR Document Review

- Unit 3 Technical Specifications 2.0, 3.2, 3.3.1, 3.4.1, 3.4.2., 3.5.1.
- Unit 3 Core Operating Limits Report.
- Unit 3 Technical Specifications Bases B2.0, B3.2, B3.3.1, B3.4.1, B3.4.2, B3.5.1.
- Unit 3 Technical Requirements Manual 3.10, B3.10
- UFSAR Sections 1.6.2.11, 3.3, 4.2, 4.3, 4.8, 6.4, 6.5, 7.5, 7.7, 7.8, Chapter 14, Appendices A, C, I, J, and Figure 4.2.2.
- Safety Evaluation Report by the Directorate of Licensing U. S. Atomic Energy Commission in the matter of Philadelphia Electric Company Peach Bottom Atomic Power Station Units 2 and 3, August 11, 1972.
- Safety Evaluation Report for the General Electric Company Topical Report Qualification of the One Dimensional Core Transient Model for Boiling Water Reactors, June 1980.
- Safety Evaluation Report by the Office of Nuclear Reactor Regulation supporting Amendments Nos. 65 and 64 to Facility License No. DPR-44 and DPR-56, March 26, 1980.
- Safety Evaluation Report by the Office of Nuclear Reactor Regulation supporting Inspection and Repair or Reactor Coolant System Piping, Recirculation Safe Ends and Core Spray Spargers, Peach Bottom Atomic Power Station Unit 3, March 20, 1986.
- Safety Evaluation Report by the Office of Nuclear Reactor Regulation supporting Amendments Nos. 125 and 128 to Facility License No. DPR-44 and DPR-56, September 24, 1987.

- Safety Evaluation Report for Topical Report PECO FMS-0004, Methods of Performing BWR System Transient Analysis", November 23, 1988.

V. References

1. GE letter dated 10/29/97, "Unit 3 Jet Pump Riser Cracking Evaluations".
2. GE letter Keck to Oliver dated 10/28/97 "Jet Pump Riser Inspections".
3. EPRI letter Selby to Hinkle dated 10/27/97 "Review of Riser VT/UT Inspections".
4. NEDC-32163P Class III, January 1993, "Peach Bottom Atomic Power Station Units 2 and 3 SAFER/GESTR-LOCA Loss of Coolant Accident Analysis".
5. BWR Jet Pump Assembly Inspection and Flaw Evaluation Guideline, BWRVIP-41, October 1997
6. ASME Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components", 1980 including addenda through Winter 1981
7. ASME Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components", 1989, Appendix C
8. Original Weld Installation details for Jet Pump assembly (microfilm tape PB-166).
9. GE SIL No. 605, Revision 1, "Jet Pump Riser Pipe Cracking", February 25, 1997 and BWRVIP letter dated 1/31/97 (letter no. 97-139)
10. BWRVIP-28, "Assessment of BWR Jet Pump Riser Elbow to Thermal Sleeve Weld Cracking", December 1996.
11. DBD P-T-18, "Reactor Vessel Internals"
12. DBD P-T-12, " Design Basis Accidents, Transients, and Events".
13. ST-O-02F-550-3, Rev. 12, "Jet Pump Operability".
14. ST-O-02F-560-3, Rev. 0, "Daily Jet Pump Operability".
15. ST-I-60A-220-3, Rev. 7, "Drive Flow-Core Flow Correlation Check".
16. ON-100, Rev. 3, "Failure of a Jet Pump".
17. Mod 1536 CBI Contract No. 873001 Drawings 55-62, N2 Inlet Safe End Replacement (M-1-E-446)
18. GE SIL 330 Supplement 2, "GE BWR/6 Jet Pump Inlet Mixer Ejection", October 27, 1993
19. Specification M-733, Rev. 3, "Inservice Inspection Program".
20. A/R A1117310, "Implementation of Unit 3 Recirculation Limits"

21. GE letter dated 11/10/97, "Unit 3 Jet Pump Riser Cracking Evaluation of Alternative Operation".
22. GE letter dated 11/11/97, "Jet Pump Riser Elbow Crack Single Loop Operation Evaluation".
23. RT-R-02F-900-3, "Recirc Flow Monitoring for Jet Pump Riser Indications"
24. GE letter dated 03/06/98, "Unit 3 Jet Pump Riser Cracking Evaluation, Given Operation Through 3/5/98"

Approvals

Prepared by Karen Tom Date 3/6/98
(PB Design Engineering, Civil/Structural)

Peer Review Paul R. Ra Date 3/6/98
(PB Design Engineering, Modifications)

Approval MR Delowery Date 3/6/98
(PB Manager, Design Engineering)

GE Nuclear Energy

General Electric Company
175 Curtner Avenue, San Jose, CA 95125

GENE-B1301869-121-9
March 6, 1998

Mr. Albert Piha
Peach Bottom Atomic Power Station
(717) 456-3332

Subject: **Unit 3 Jet Pump Riser Cracking Evaluation, Given Operation Through 3/5/98**

Reference: [1] Letter GENE-B1301869-121-8, dated 12/22/97, TA Caine to A Piha, same subject, operation through 12/11/97.

Dear Albert,

This letter documents the verified results of an evaluation of potential jet pump riser cracking for a period of Unit 3 operation beyond 3/6/98, taking into consideration actual operation through 0600 hours on 3/5/98.

Two operating scenarios were considered, per Mike Delowery's instructions:

1. Assuming that Unit 3 operates at 91% drive flow to 2300 hours on 3/6/98, how many hours at 80% drive flow can Unit 3 operate while maintaining the original analysis margin.

2. Assuming that Unit 3 operates to 2300 hours on 3/20/98, what is the maximum number of hours the plant can operate at 91% drive flow, with the balance at 80% drive flow, while maintaining the original analysis margin.

The calculations for actual operation and the two operating scenarios are summarized below:

Conditions	**Analyzed Time (hrs)[*]**	**Predicted Crack Length (in.)**
Actuals to 12/11/97 14:00	970 (includes 70 at 91%) [1]	13.06 to 13.28
Actuals to 3/5/98 06:00	2010 (includes 480 at 91%)	13.28 to 14.37
<u>Case 1:</u> 91% drive flow (31.5 Mlb/hr)	40	14.37 to 14.48
80% drive flow (27.7 Mlb/hr)	800	14.48 to 14.85
<u>Case 2:</u> 91% drive flow (31.5 Mlb/hr)	130	14.37 to 14.73
80% drive flow (27.7 Mlb/hr)	250	14.73 to 14.85

These scenarios maintain the safety margin that was the basis for the original analysis last year, which established operating criteria of 80% drive flow for 6000 hours.

The analysis results of crack length as a function of operating hours, considering the drive flows and times in Table 1, are shown in Figure 1 (Case 1) and Figure 2 (Case 2) for jet pump 13/14, which has the limiting crack. The predicted crack length at the end of the operating periods evaluated is 14.85 inches for both cases.

Analysis tasks and evidence of verification are contained in design record file B13-01921. If you have any questions on the evaluation, please call me at the number below.

Regards,

Tom Caine

TA Caine, Manager
Structural Mechanics and Materials
(408) 925-4047

Verified:

HS Mehta, Technical Leader
Structural Mechanics and Materials

cc: K Faynshtein, GE
HS Mehta, GE
D Robare, GE

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[*] Given the precision of the methodology, it is acceptable for the actual operating hours to be within ±5 hours of the analyzed value.

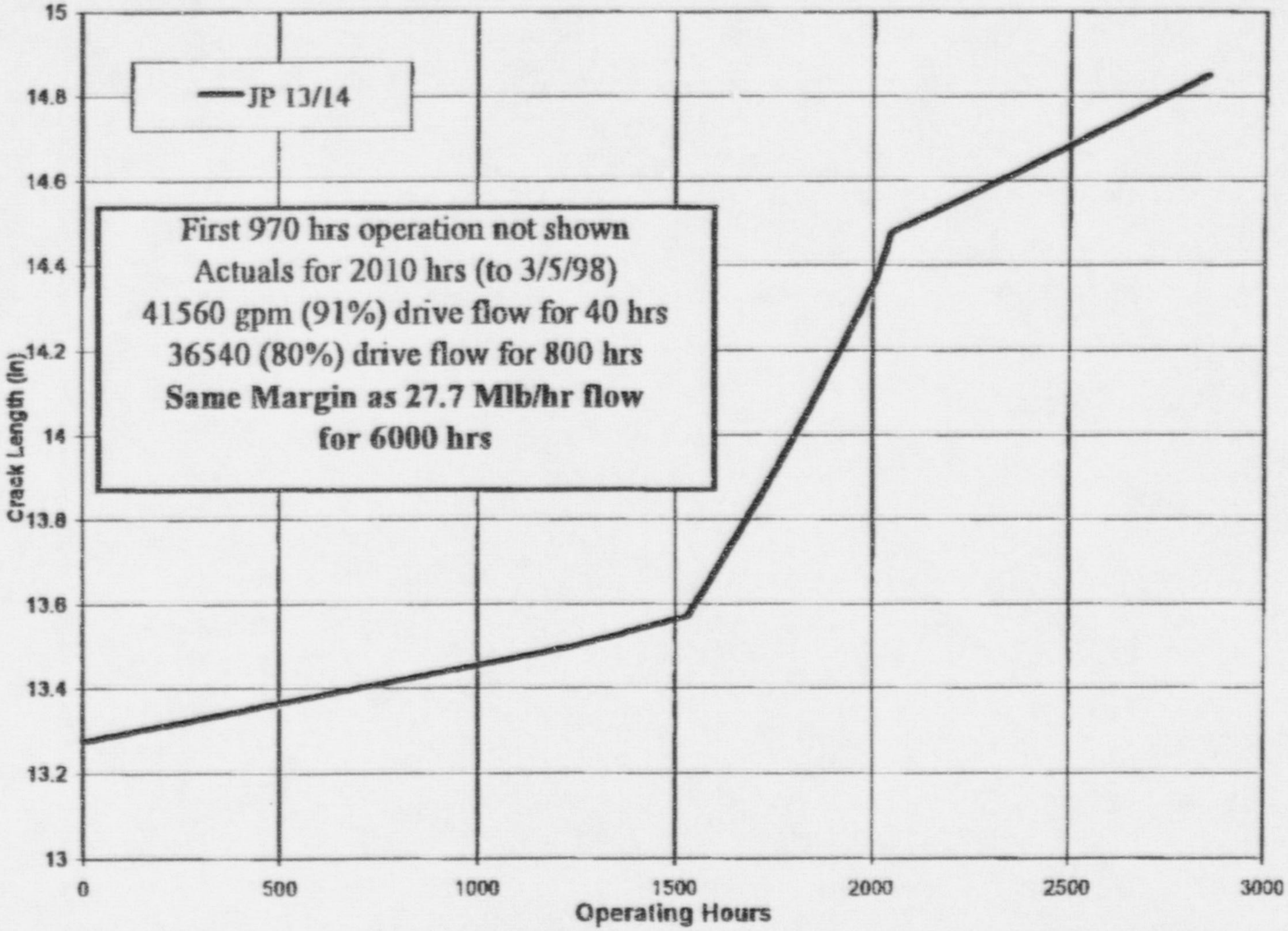

Figure 1. Predicted Crack Length for JP 13/14 - Case 1

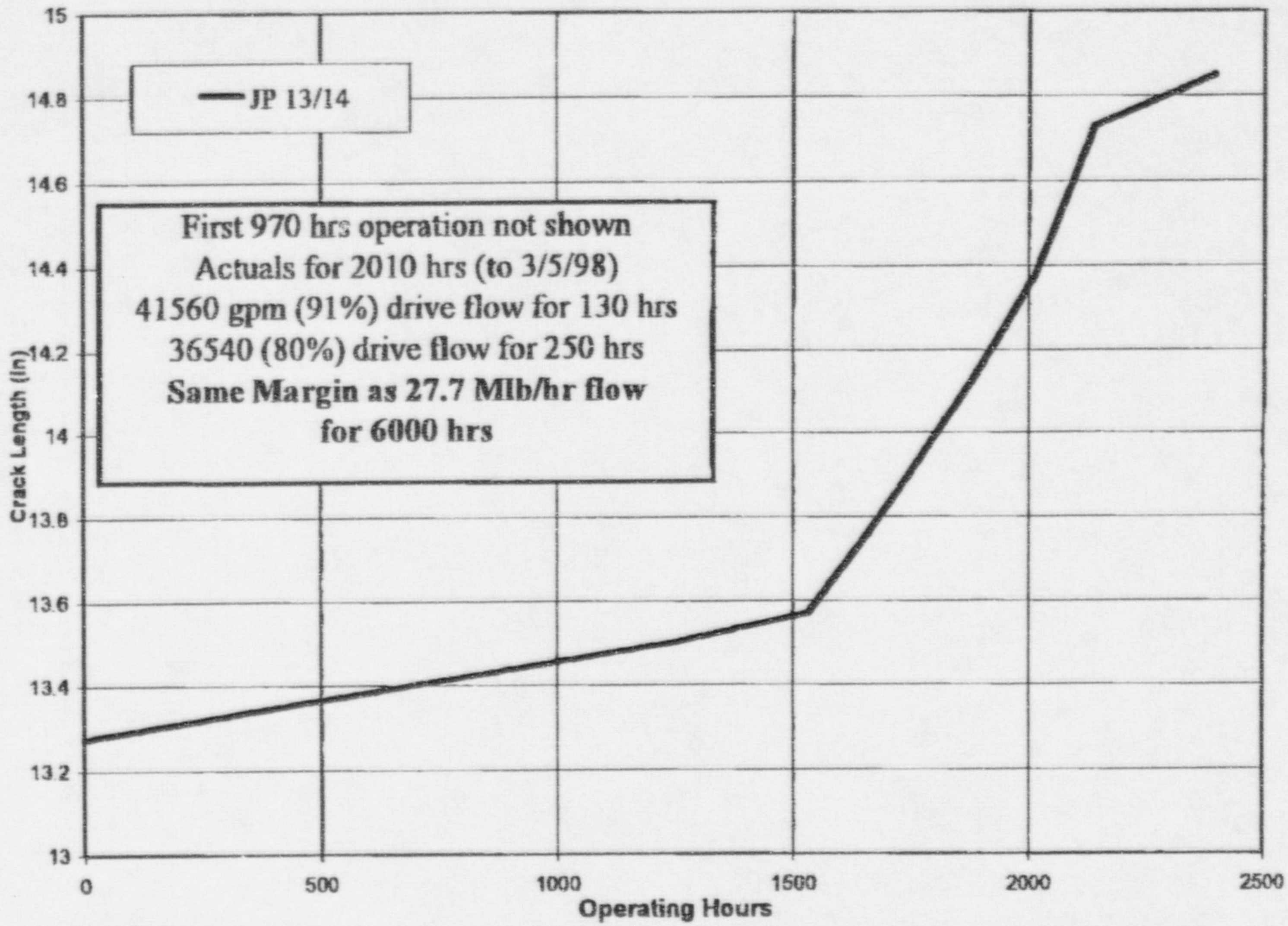

Figure 2. **Predicted Crack Length for JP 13/14 - Case 2**