VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

July 25, 1997

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555-0001

Serial No.	97-214A
NL&OS / MWH:	R7
Docket Nos.	50-280, 50-281 50-338, 50-339
License Nos.	DPR-32, DPR-37 NPF-4, NPF-7

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS 1 AND 2
NORTH ANNA POWER STATION UNITS 1 AND 2
DEGRADATION OF CONTROL ROD DRIVE MECHANISM NOZZLE AND OTHER VESSEL CLOSURE HEAD PENETRATIONS

On April 1, 1997, the Nuclear Regulatory Commission issued NRC Generic Letter 97-01, "Degradation Of Control Rod Drive Mechanism Nozzle And Other Vessel Closure Head Penetrations." The generic letter requested that licensees provide the following information: (1) control rod drive mechanism (CRDM) nozzle and other vessel head penetration (VHP) inspection activities including, (a) a description of all inspections and results performed to the date of the generic letter, (b) the scope and schedule if a plan has been developed to periodically perform inspections, (c) the analysis that supports why no augmented inspection is necessary if a plan has not been developed to perform periodic inspections, (d) the analysis that supports the selected course of action for inspections, and (2) a description of any resin bead intrusions that have exceeded the current EPRI PWR Primary Water Chemistry Guidelines.

The generic letter requests two written responses. First, within 30 days of the issuance of the generic letter, a written response is required indicating: (1) whether or not the requested information will be submitted and (2) whether or not the requested information will be submitted within the requested time period. Second, within 120 days of the issuance of the generic letter, a written report is to be submitted providing the requested information describing the CRDM nozzle and other VHP inspection activities and describing any resin bead intrusions.

On April 28, 1997 Virginia Electric and Power Company (Virginia Power) provided a written response (Serial No. 97-214) to confirm that the requested information identified in the generic letter would be described in a written report. The purpose of this letter is to provide a written summary which describes the CRDM nozzle and other VHP inspection activities and describing any resin bead intrusions.

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A written summary of the requested information identified in the generic letter is provided in the attachment. Virginia Power is participating in the Westinghouse Owners' Group Materials Subcommittee which is developing an integrated reactor vessel head penetration inspection program. It should be noted that North Anna Unit 1 has performed a volumetric examination of the twenty (20) outermost head penetrations in February 1996 and no cracked penetrations were noted. An augmented inspection program (i.e., visual inspection of accessible areas on the top of the reactor vessel head with insulation in place) for boric acid deposits has been formulated at North Anna to monitor the CRDMs reactor vessel head penetrations. This same inspection activity is in the process of being implemented at Surry. No leaks associated with cracked penetrations have been found. Virginia Power has also completed a data review for potential resin intrusion for North Anna and Surry Power Stations and no indications of significant resin in-leakage were noted.

Should you have any questions, please contact us.

Very truly yours,

James P. O'Hanlon

James P. O'Hanlon
Senior Vice President - Nuclear

Attachment w/ enclosures

Commitments contained in this letter:

North Anna Units 1 and 2:

An augmented inspection program (i.e., visual inspection of accessible areas on the top of the reactor vessel head with insulation in place) for boric acid deposits will be performed during scheduled refueling outages.

Surry Unit 1:

An augmented inspection program (i.e., visual inspection of accessible areas on the top of the reactor vessel head with insulation in place) for boric acid deposits will be conducted each refueling outage commencing with the fall 1998 Surry Unit 1 refueling outage.

Surry Unit 2:

An augmented inspection program (i.e., visual inspection of accessible areas on the top of the reactor vessel head with insulation in place) for boric acid deposits will be conducted each refueling outage commencing with the fall 1997 Surry Unit 2 refueling outage.

cc: U. S. Nuclear Regulatory Commission
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Surry Power Station

Mr. M. J. Morgan
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North Anna Power Station

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DEGRADATION OF CONTROL ROD DRIVE MECHANISM NOZZLE AND OTHER VESSEL CLOSURE HEAD PENTRATIONS

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Introduction:

Generic Letter (GL) 97-01, Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations, was issued to request licensees to describe their program for insuring the timely inspection of Pressurized Water Reactor (PWR) control rod drive mechanism (CRDM) and other closure head penetrations for cracks/ degradation. This response provides information pertaining to Virginia Electric and Power Company's (Virginia Power's) nuclear units and the information requested by the GL. Virginia Power's nuclear units are: North Anna Power Station Units 1 and 2 and Surry Power Station Units 1 and 2.

Prior to the issuance of the GL, Virginia Power has worked with the Westinghouse Owners Group (WOG), the Electric Power Research Institute (EPRI) and the Nuclear Energy Institute (NEI) and has participated on the NEI Alloy 600 Task Team in order to understand the operational experience, as well as to identify technical issues and solutions surrounding this issue. One of these tasks was the development of safety evaluations that characterized the initiation of primary water stress corrosion cracks (PWSCC), subsequent crack propagation and the possible consequences of a through-wall crack. The initial WOG's safety evaluation is:

- WCAP-13565, *Alloy 600 Reactor Vessel Adaptor Tube Cracking Safety Evaluation*, issued March 1993 (Reference 1)

The NRC has formally reviewed WCAP-13565 and has issued a safety evaluation report (SER) to NEI on November 19, 1993 (Reference 4). The SER concurred with Westinghouse's conclusion that CRDM cracking was not an immediate safety issue. The WOG's safety evaluation and the SER establish the basis for the continued operation of North Anna Units 1 and 2 and Surry Units 1 and 2.

Additional safety evaluations were performed for Virginia Power in order to strengthen the basis for continued operation for North Anna and Surry. They include:

- WCAP-14219, *RV Closure Head Penetration- Supplemental Assessment of NRC SER Issues*, issued March 1995 (Reference 2)
  (Summary: WCAP-14219 addresses safety issues raised by the NRC concerning lack of fusion that was encountered in certain CRDM partial penetration

attachment welds at Ringhals 2. The analysis showed the maximum observed area of lack of fusion is well below allowable limits for structural integrity for all WOG plants.)

- WCAP-14552, *Structural Integrity Evaluation of Reactor Vessel Upper Head Penetrations to Support Continued Operations: North Anna and Surry Units*, issued January 1996 (Reference 3)
  (Summary: WCAP-14552 is a plant specific analysis of PWSCC cracking at North Anna and Surry. The analysis shows that PWSCC cracks at those units will never cause leaks. The driving force for PWSCC propagation drops to zero before through wall cracks can extend above the CRDM partial penetration weld which is the pressure boundary.)

Response to Requested GL 97-01 Information Item 1.1:

"1.1 A description of all inspections of CRDM nozzle and other VHP's performed to the date of this generic letter, including the results of these inspections."

Response:

PWSCC-induced leaks from cracked control rod drive mechanism (CRDM) penetrations are not expected at North Anna Units 1 and 2 and Surry Units 1 and 2 (see Reference 3). Even so, Virginia Power conducts the following inspection activities to detect cracked CRDM penetrations. First, ASME Section XI system leak tests on reactor vessel pressure retaining boundaries are performed during refueling outages (RFO). Second, ASME Section XI nondestructive examinations of welds on 10% of the peripheral CRDM housings are conducted three times over a ten year period. ASME Section XI visual examination of the partial penetration welds on 25% of the CRDM nozzles are performed every ten (10) year period. Third, during operation, reactor coolant inventory is closely monitored for unidentified leaks. Fourth, routine maintenance and outage activities associated with the reactor vessel upper head before, during, and after its removal during every refueling outage are performed.

None of these activities have identified leaking CRDM penetrations due to PWSCC for any Virginia Power Nuclear Unit.

In addition to the above, the following inspections were conducted and are planned:

North Anna Unit 1:

In February 1996, North Anna Unit 1 performed a volumetric (eddy current) examination of the outermost 20 penetrations, consisting of 4 thermocouple locations and 16 thermal sleeved penetrations. These 20 outermost penetrations were deemed the most crack susceptible of the 65 head penetrations in Unit 1 by Westinghouse's analysis. The inspection scope was scheduled to be expanded, if cracks were found in any of those 20 penetrations. However, no cracked penetrations were identified. The remaining forty-five (45) penetrations were judged to be less crack-prone penetrations and were not tested. Details of the inspection results can be found in Westinghouse Report EP-GDA-96-001, *North Anna Unit 1 Reactor Vessel Head Penetration Inspection and Replication*, issued March 1996 (Reference 5). In addition, replications of nine of those 20 penetrations were performed, as documented in WCAP -14626, *Microstructural and PWSCC Assessment of North Anna Unit 1 Alloy 600 R. V. Head Penetrations by Field Replication*, issued April 1996 (Reference 6). These microstructure replications were performed to update the WOG cracking susceptibility model.

After Unit 1 completed one additional fuel cycle, an augmented inspection program (i.e., visual inspection of accessible areas on the top of the reactor vessel head with insulation in place) for boric acid deposits was performed in the spring of 1997 during its Refueling Outage (RFO). No evidence of leaking penetrations due to PWSCC was found. An augmented inspection program will be conducted during scheduled refueling outages.

North Anna Unit 2:

An augmented inspection program (i.e., visual inspection of accessible areas on the top of the reactor vessel head with insulation in place) for boric acid deposits was performed in the fall of 1996. No evidence of leaking penetrations due to PWSCC was found. An augmented inspection program will be conducted during scheduled refueling outages.

Surry Unit 1:

In the fall 1995 Unit 1 RFO, a partial visual inspection of the upper head with insulation in place was performed for boric acid deposits. No evidence of boric acid deposits which could be an indication of cracked penetrations were found. An augmented inspection program (i.e., visual inspection of accessible areas on the top of the reactor vessel head) for boric acid deposits will be conducted each refueling outage commencing with the fall 1998 Unit 1 RFO.

Surry Unit 2:

Starting with the fall 1997 Unit 2 RFO, an augmented inspection program (i.e., visual inspection of accessible areas on the top of the reactor vessel head with insulation in place) for boric acid deposits will be performed each refueling outage.

Response to Requested Information GL 97-01 Item 1.2 through 1.4:

"1.2 *If a plan has been developed to periodically inspect the CRDM nozzle and other VHP's:*

a. *Provide the schedule for first, and subsequent inspections of the CRDM nozzles and other VHPs, including the technical basis for this schedule.*
b. *Provide the scope for the CRDM nozzle and other VHP inspections, including the total number of penetrations (and how many will be inspected), which penetrations have thermal sleeves, which are spares, and which are instrument or other penetrations.*

1.3 *If a plan has not been developed to periodically inspect the CRDM nozzle and other VHPs, provide the analysis that supports why no augmented inspection is necessary.*

1.4 *In light of the degradation of CRDM nozzles and other VHPs described above, provide the analysis that supports the selected course of action as listed in either 1.2 or 1.3 above. In particular, provide a description of all relevant data and/or tests used to develop crack initiation and crack growth models, the methods and data used to validate these models, the plant-specific inputs to these models, and how these models substantiate the susceptibility evaluation. Also, if an integrated industry inspection program is being relied on, provide a detailed description of this program."*

Response:

To predict the probability of crack initiation and of propagation through a penetration, Westinghouse has developed a structural reliability model utilizing Monte-Carlo simulation methods. (Reference 8, WCAP-14901, *Background and Methodology for Evaluation of Reactor Vessel Closure Head Penetration Integrity for the Westinghouse Owners Group*). Input parameters to the probabilistic model include: hours of operation, penetration setup angle (SA), operating temperature, yield strength (YS) and grain boundary carbide coverage (%). Enclosures 1 through 4 contain the input values for the North Anna and the Surry analyses. Virginia Power's assessment of the recalculated WOG probability analysis is still in progress. The results of the North Anna and Surry probability analyses will be included in the WOG integrated inspection program.

Virginia Power intends to participate in the Westinghouse Owners Group Reactor Pressure Vessel head penetration integrated inspection program. The WOG CRDM inspection program is under development and is expected to be finalized and provided to the NRC Staff by the end of 1997. The objectives of this integrated inspection program are: to inspect representative plants (which envelope the other WOG plants), to share inspection results with other Owners Groups (OGs), and to update the various OGs cracking susceptibility models. This integrated program intends to include the results of volumetric inspections of head penetration heats that have been performed and additional volumetric inspections that will be performed.

In addition to the WOG integrated inspection program, all three PWR owners groups, the Electric Power Research Institute, and the Nuclear Energy Institute [NEI] are cooperatively working together on an industry integrated inspection program. NEI has the lead responsibility for the industry integrated inspection program. All five industry bodies are compiling information on the estimated operating time from January 1, 1997, needed to initiate and propagate a crack 75% through wall in a vessel penetration. This information will be used to assess if an adequate number of U.S. plants have or are planning to inspect. This evaluation is expected to be issued by the end of 1997. It is our understanding that NEI will provide the results of the evaluation to the NRC.

There are no immediate plans to re-inspect by volumetric exam North Anna Unit 1 or to inspect by volumetric exam North Anna Unit 2 for the following reasons: (1) North Anna Units 1 and 2 share identical penetration material heats, (2) plant specific analyses performed by Westinghouse (Reference 3) using the methodology described in Reference 8 indicates that cracking is more likely for Unit 1 (which exhibited no cracked penetrations) than for Unit 2, and (3) the plant specific analyses described in Reference 3 predicts that a leak will not occur in the most crack susceptible outer two rows of penetrations even if a crack initiates. According to the analysis described in Reference 3, a through wall crack will stop growing before it can propagate above the penetration attachment weld [i.e., pressure boundary] and cause a leak.

Based on the plant-specific analysis performed (Reference 3), there are no immediate plans to inspect by volumetric exam Surry Units 1 and 2. The plant specific analysis predicts a leak will not occur in the most crack-prone outer two rows of penetrations even if a crack initiates. According to the analysis, a through-wall crack will stop growing before it can propagate above the penetration attachment weld (the pressure boundary) and cause a leak.

The CRDM cracking issue was confirmed by the NRC to be a long term concern rather than an immediate safety issue (Reference 4). Moreover, plant-specific calculations show PWSCC is not likely to lead to leaking CRDMs at either North Anna or Surry (Reference 3). Thus, the safety significance of this issue appears to be relatively small at present for North Anna and Surry. Therefore, the benefits of volumetric examinations do not appear to be commensurate with the cost. Scheduling of any future volumetric inspections will depend on the updated predictions of the revised WOG crack susceptibility model for North Anna and Surry and on the results of the industry integrated inspection program.

According to fabrication records (Reference 9 and 10), the only other reactor vessel penetrations made of Inconel Alloy 600 (besides CRDMs) which traverses through North Anna's and Surry's reactor vessel heads are vent pipes (upper head) and instrument tubes (lower head). The residual stress in the vent pipes is minimal compared to CRDMs since the vent pipes are located near the top of the head. As a consequence, the driving force for PWSCC is minimal. The operating temperature of the instrument tubes is greater than 50 degrees lower (Reference 9 and 10) than the operating temperature of the CRDMs. The lower operating temperature will reduce the calculated crack initiation time by a factor of approximately 10, as compared to that of the CRDMs. As a result, the probability of PWSCC occurring in the instrument tubes is not significant. Since cracking of the vent pipes and instrument tubes are bound by cracking in CRDMs, no volumetric inspections of those locations are planned at North Anna or Surry.

Response to Requested Information GL 97-01 Item 2:

"2.0 Provide a description of any resin intrusions, as described in IN 96-11, that have exceeded the current EPRI PWR Primary Water Chemistry Guidelines recommendations for primary water sulfate levels, including the following information:

2.1 Were the intrusions cation, anion, or mixed bed?
2.2 What were the durations of the intrusions?
2.3 Does the plant's RCS water chemistry Technical Specifications follow the EPRI guidelines?
2.4 Identify any RCS chemistry excursions that exceed the plant administration limits for the following species: sulfates, chlorides or fluorides, oxygen, boron, and lithium.
2.5 Identify any conductivity excursions which may indicative of resin intrusions. Provide a technical assessment of each excursion and any follow-up actions.
2.6 Provide an assessment of the potential for any of these intrusions to result in a significant increase in the probability for IGA of VHP's and any associated plan for inspections."

<u>Response</u>:

North Anna and Surry Power Stations have reviewed the plant historical records to determine if any incident of resin ingress similar to those which occurred in 1980 and

1981 at the Jose Cabrera (Zorita) plant has occurred. This data search is structured to identify all resin intrusion events into the primary coolant system that were of a magnitude greater than 1 ft^3 (30 liters). The threshold of 1 ft^3 was chosen as a conservative lower bound since it represents less than 15% of the estimated volume of resin released into the reactor coolant system during the two events at Jose Cabrera.

For the period of plant operation prior to initiation of routine analysis for sulfate in reactor coolant, the data search was based on a review of the plant's reactor coolant chemistry records relative to specific conductance of the reactor coolant. An elevation of a 28 μS/cm increment in specific conductance was the value used as an indicator of cation resin ingress equivalent to a volume of 1 ft^3.

Routine analysis for sulfate in the reactor coolant system was performed for plant operation from January 20, 1989 (Surry) and January 3, 1990 (North Anna) to present. A sulfate concentration in the range of 15-17 ppm peak concentration was used as the indicator of cation resin ingress. This concentration is approximately equivalent to a volume of 1 ft^3 cation resin.

Had either specific conductivity or sulfate concentration increases indicated resin ingress to the magnitude of the threshold quantity identified above, additional data evaluation was performed to look for a corresponding depression in pH or elevation in lithium as corroborating information of the incident. In the case of the use of sulfate data as the indicator, specific conductance would also have been included as confirmatory data had a significant in-leakage event been identified.

The results of the data review as described in the above paragraphs showed *no* indication of significant resin in-leakage at North Anna or Surry Power Stations.

It is considered unnecessary to review the plant records for boron, chlorides, fluorides, and oxygen since these species are not viewed as valid indicators of cation resin ingress and degradation within the primary coolant system of a PWR.

North Anna and Surry Power Stations have followed the EPRI PWR Primary Water Chemistry Guidelines since they were issued and have implemented revisions when issued. The following exceptions to the guidelines exist:

- Virginia Power monitors RCS pH and Specific Conductivity three times per week, versus the EPRI recommendation of seven times per week

- Virginia Power monitors RCS suspended solids once per quarter, versus the EPRI recommended monitoring frequency of once per week

Although monitoring of the RCS pH and specific conductivity differ from the EPRI Guidelines, sulfate analysis is performed more frequently than the EPRI Guidelines and would detect minor resin intrusions. In addition to the quarterly monitoring of the RCS suspended solids, reactor coolant filters are checked at least once per day to identify changes in filter loading which would indicate resin intrusion.

REFERENCES:

1. WCAP-13565, *Alloy 600 Reactor Vessel Adaptor Tube Cracking Safety Evaluation*, issued March 1993 [Proprietary and Non-Proprietary].
2. WCAP-14219, *RV Closure Head Penetration- Supplemental Assessment of NRC SER Issues*, issued March 1995 [Proprietary and Non-Proprietary].
3. WCAP-14552, *Structural Integrity Evaluation of Reactor Vessel Upper Head Penetrations to Support Continued Operations: North Anna and Surry Units,* issued January 1996 [Proprietary].
4. NRC letter from William T. Russell to William Rasin of Nuclear Management and Resources Council (NUMARC), now NEI, dated November 19, 1993
5. Westinghouse Report EP-GDA-96-001, *North Anna Unit 1 Reactor Vessel Head Penetration Inspection and Replication*, issued March 1996 [Proprietary].
6. WCAP -14626, *Microstructural and PWSCC Assessment of North Anna Unit 1 Alloy 600 R. V. Head Penetrations by Field Replication*, issued April 1996 [Proprietary].
7. Virginia Power Station Administrative Procedure VPAP-1103, *ASME Section XI Visual Examination Program (VT-1, 2, and 3)*
8. WCAP-14901, *Background and Methodology for Evaluation of Reactor Vessel Closure Head Penetration Integrity for the Westinghouse Owners Group*, issued July 1997 [Non-Proprietary - See Enclosure 5].
9. Westinghouse letter reports MSE-MNA-368 & MSE-MNA-389, issued October 1994 (These plant specific reports summarize the reactor vessel application of Alloy 600, excluding CRDMs for North Anna Units 1 & 2 and Surry Units 1 & 2).

10. Westinghouse report MED-PCE-9799, "Report on Reactor Vessel Applications of Inconel 600," issued November 9, 1990.

Enclosure 1

Virginia Power Response to NRC Generic Letter 97-01
Degradation of Control Rod Drive Mechanism Nozzle and
Other Vessel Closure Head Penetrations

Serial No.: 97-214A

TABLE 5-2
SURRY UNIT 1
INPUT VALUES FOR PROBABILISTIC ANALYSIS

Case	Pen. No.	Temp.	SA	Y.S. (ksi)	GBC (%)
1	62 thru 65	597.8°F	42.6	48.5	22.9
2	66, 67, 69	for 39,900 hrs	42.6	46.0	43.4
3	68		42.6	39.0	41.8
4	58 thru 61	604.8°F	40.0	48.5	22.9
5	51, 53, 55, 57	for 87,300 hrs	38.6	48.5	22.9
6	48, 49		37.3	47.5	53.5
7	46, 47	597.8°F	37.3	40.5	43.1
8	45	thereafter	33.1	47.5	53.5
9	44		33.1	32.5	7.1
10	40, 41		33.1	40.5	43.1
11	42, 43		33.1	60.0	44.8
12	38, 39		33.1	46.5	51.0
13	37		28.6	39.0	41.8
14	35, 36		28.6	32.5	7.1
15	30, 31		28.6	43.0	71.6
16	32 thru 34		28.6	46.5	51.0
17	29		27.0	43.0	71.6
18	26 thru 28		27.0	58.0	59.0
19	22, 23		25.0	40.5	43.1
20	24, 25		25.4	46.5	51.0
21	14 thru 21		19.8	43.0	71.6
22	12, 13		17.6	40.5	43.1
23	10, 11		17.6	58.0	59.3
24	6 thru 9		12.4	58.0	59.3
25	2 thru 5		8.7	39.0	41.8
26	1		0	40.5	43.1

Notes: Pen. = penetration; SA = setup angle of penetration (degrees); Y.S. = yield strength; GBC = grain boundary coverage of carbides.

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Enclosure 2

Virginia Power Response to NRC Generic Letter 97-01
Degradation of Control Rod Drive Mechanism Nozzle and
Other Vessel Closure Head Penetrations

Serial No.: 97-214A

TABLE 5-2
SURRY UNIT 2
INPUT VALUES FOR PROBABILISTIC ANALYSIS
(WORST CASE)

Case	Pen. No.	Temp.	SA	Y.S. (ksi)	GBC (%)
1	62 thru 69	597.8°F	42.6	32.7	3.8
2	58 thru 61	for 32,900 hrs	40.0	32.7	3.8
3	51,53,55,57		38.6	32.7	3.8
4	46 thru 49	604.8°F	37.3	32.7	3.8
5	38 thru 45	for 92,500 hrs	33.1	32.7	3.8
6	30 thru 37		28.6	32.7	3.8
7	26 thru 28		27.0	32.7	3.8
8	29	597.8°F	27.0	38.1	75.1
9	22 thru 25	thereafter	25.4	32.7	3.8
10	14 thru 21		19.8	32.7	3.8
11	10 thru 13		17.6	32.7	3.8
12	6 thru 9		12.4	32.7	3.8
13	2 thru 5		8.7	32.7	3.8
14	1		0	32.7	3.8

Notes: Pen. = penetration; SA = setup angle of penetration (degrees); Y.S. = yield strength; GBC = grain boundary coverage of carbides.

Enclosure 3

Virginia Power Response to NRC Generic Letter 97-01
Degradation of Control Rod Drive Mechanism Nozzle and
Other Vessel Closure Head Penetrations

Serial No.: 97-214A

TABLE 5-2
NORTH ANNA UNIT 1
INPUT VALUES FOR PROBABILISTIC ANALYSIS
(WORST CASE)

Case	Pen. No.	Temp.	SA	Y.S. (ksi)	GBC (%)
1	63, 65, 66, 68	600.1°F	42.6	49.8	60.4
2	62	for 24,900 hrs	42.6	49.8	7.0
3	67		42.6	49.8	3.5
4	64		42.6	49.8	2.5
5	69		42.6	49.8	2.0
6	59 thru 61		40.0	51.2	58.1
7	58		40.0	51.2	1.0
8	50, 51, 53, 54, 56, 57		38.6	51.2	58.1
9	55		38.6	51.2	32.2
10	52		38.6	51.2	3.0
11	46, 48	607.1°F	37.3	51.2	58.1
12	47	for 60,300 hrs	37.3	51.2	39.0
13	49		37.3	51.2	36.0
14	38 thru 45		33.1	51.2	58.1
15	30 thru 37		28.6	51.2	58.1
16	26 thru 29	600.1°F	27.0	46.1	69.3
17	22 thru 25	thereafter	25.4	51.2	58.1
18	15, 17, 19, 21		19.8	41.4	43.7
19	10 thru 13		17.6	41.4	43.7
20	6 thru 9		12.4	41.4	43.7
21	2 thru 5		8.7	41.4	43.7
22	1		0	41.4	43.7

Notes: Pen. = penetration; SA = setup angle of penetration (degrees); Y.S. = yield strength; GBC = grain boundary coverage of carbides.

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Enclosure 4

Virginia Power Response to NRC Generic Letter 97-01
Degradation of Control Rod Drive Mechanism Nozzle and
Other Vessel Closure Head Penetrations

Serial No.: 97-214A

TABLE 5-2
NORTH ANNA UNIT 2
INPUT VALUES FOR PROBABILISTIC ANALYSIS
(WORST CASE)

Case	Pen. No.	Temp.	SA	Y.S. (ksi)	GBC (%)
1	62 thru 69	600.1°F	42.6	49.8	60.4
2	58 thru 61	for 17,300 hrs	40.0	51.2	58.1
3	50 thru 57		38.6	51.2	58.1
4	46 thru 49	607.1°F	37.3	51.2	58.1
5	38 thru 45	for 41,000 hrs	33.1	51.2	58.1
6	30 thru 37		28.6	51.2	58.1
7	26. thru 29	600.1°F	27.0	46.1	69.3
8	22 thru 25	thereafter	25.4	51.2	58.1
9	15, 17, 19, 21		19.8	41.4	43.7
10	13		17.6	42.2	81.5
11	10 thru 12		17.6	41.4	43.7
12	6 thru 9		12.4	41.4	43.7
13	2 thru 5		8.7	41.4	43.7
14	1		0	41.4	43.7

Notes: Pen. = penetration; SA = setup angle of penetration (degrees); Y.S. = yield strength; GBC = grain boundary coverage of carbides.

Enclosure 5

Virginia Power Response to NRC Generic Letter 97-01
Degradation of Control Rod Drive Mechanism Nozzle and
Other Vessel Closure Head Penetrations

Serial No.: 97-214A

WCAP-14901

REVISION 0

BACKGROUND AND METHODOLOGY FOR EVALUATION OF REACTOR VESSEL CLOSURE HEAD PENETRATION INTEGRITY FOR THE WESTINGHOUSE OWNERS GROUP

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WCAP-14901

Background and Methodology for Evaluation of Reactor Vessel Closure Head Penetration Integrity for the Westinghouse Owners Group

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July 1997

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EXECUTIVE SUMMARY

This report is intended for use in response to NRC Generic Letter 97-01. Cracking in Alloy 600 reactor vessel head penetrations is a relatively new issue to the nuclear industry. The issue was first brought to the world's attention in 1991 when, after 10 years of operation, a leak was detected during a hydrotest of the reactor coolant system at the Bugey Unit 3 power plant in France. Since then a significant number of studies and research programs have been funded by the industry to determine the causes of the problem and develop strategies for repair and management.

Through these programs and subsequent studies it was concluded that reactor pressure vessel head CRDM penetration cracking at Bugey Unit 3 is induced by is a thermally activated stress corrosion mechanism operative in primary water environments, more commonly known as primary water stress corrosion cracking (PWSCC). Based on conservative evaluation results, the NRC and industry concluded that PWSCC cracks were most likely to initiate from the inside surface of the penetrations, in the axial orientation, and would take at least six years to propagate through the wall under the typical plant operating conditions. Fracture mechanics evaluations have determined that the crack is non-critical until its axial length reaches 8.5 inches to 20 inches, depending on plant design. Therefore this issue is an economic one, and does not constitute a serious challenge to plant safety.

External circumferential cracking is less probable. It may occur only in the presence of an above the weld through-wall crack, with active leakage. Assuming coolant is present on the outer diameter of the penetration, one conservative analysis estimated that it would take more than 90 years before penetration failure would occur. In the presence of reactor coolant, corrosion wastage of the alloy steel RV head is possible. Conservative evaluations estimate that it would take longer than six years after a through-wall crack occurs before the code structural integrity margin for the RV head would be impacted by corrosion. It was concluded that periodic visual inspection of the RV head in accordance with Generic Letter 88-05 is adequate to maintain plant safety, and sufficient to detect leakage prior to significant penetration cracking and vessel head corrosion.

Worldwide, approximately 5,200 Alloy 600 RV head penetrations have been inspected since the first cracking was observed in 1991. Approximately 2 percent of these penetrations are reported to be cracked. Most of the cracks were observed in French RV head penetrations. If the French inspection records are removed from the inspected population, the percentage of head penetrations with indications is only about 0.5 percent. Only one plant worldwide has experienced PWSCC head penetration through-wall leakage, and this was from a single penetration.

Specialized NDE methods have been developed and verified using mock-ups to ensure accurate inspections. Flaws were introduced into the mock-up penetrations by artificial means. The ability of these NDE methods to detect and size the potential PWSCC indications in the vessel head penetrations was demonstrated. Flaw acceptance criteria were established by the industry, and approved by the NRC staff.

The Westinghouse Owners Group has developed methods to evaluate the PWSCC susceptibility and the probability of a penetration initiating a crack, or a leak, as a function of

plant operation time. This information has been used to evaluate the need for inspection of the reactor vessel head penetrations or other appropriate actions.

Through participation in WOG and U.S. industry programs the Westinghouse plant owners have taken a proactive approach to address the cracking issue in RV head penetrations. This approach is based on the conclusion that the issue is not an immediate safety concern, because (1) the PWSCC process is slow; (2) the allowable or critical flaw size is large; (3) leak-before-break (LBB) will occur to allow safe shutdown of a plant and (4) at least six additional years of operation with a penetration leak is required before ASME Code structural margins are challenged.

In addition to the material contained in this report, detailed integrity assessments have been completed for all Westinghouse plants, and these results are being incorporated into an integrated response to the Generic Letter 97-01, which is being prepared in cooperation with the Nuclear Energy Institute. This response will be transmitted to the NRC by the end of 1997.

1.0 INTRODUCTION

1.1 SUMMARY OF THE SAFETY EVALUATIONS

The purpose of this section is to review the significance of cracking in pressurized water reactor (PWR) vessel head penetrations and to describe the management of the issue in response to the recently released NRC Generic Letter 97-01. This report covers the following areas: worldwide PWSCC history in head penetrations; safety evaluation conclusions reached by WOG and industry and approved by the NRC relative to PWSCC; and a number of supporting tasks performed by Westinghouse for the WOG concerning this issue. The latest findings on this subject are summarized, along with response to specific questions in Generic Letter 97-01.

In February of 1993, Westinghouse and the Westinghouse Owners Group performed an assessment of the continued safe operation of Westinghouse designed NSSS plants in light of the cracking that had been reported in French supplied and operated plant reactor vessel head penetrations.

Westinghouse reviewed the available metallographic and fractographic data from the French plant and concurred with the EdF conclusion that the mechanism of degradation of the Bugey 3 reactor vessel penetration was due to primary water stress corrosion cracking.

The Westinghouse safety evaluation [1] provided the following elements:

1. *A summary of the vessel head penetration stress analyses that focuses on the nature and orientation of cracking that may occur in the Alloy 600 penetration material.* The Westinghouse evaluation concluded that the penetration residual stress induced by welding into the reactor vessel head was the initiating source promoting crack initiation and growth in a susceptible microstructure.

2. *A summary of the crack propagation analysis along with the basis of the prediction methodology.* As indicated in Section 2 of this report, continued crack growth testing has confirmed the initial expectations. The analysis also predicted that cracking would be axial and any cracks formed would be limited in extent by the penetration stress field distribution. The crack lengths predicted were found to be much smaller than the length of cracking required for any instability. The existence of circumerential cracking is unlikely due to the nature of stress distribution in the penetrations (i.e., hoop stress dominates the stress field).

3. *A description of an assessment of the Westinghouse Owners Group vessels with respect to crack indications reported at Ringhals, Beznau, and various EdF plants.* Important parameters applicable for crack initiation (i.e., time, temperature, stress, and material) were compared to those of Ringhals, Beznau and EdF plants. A comparison of susceptibility predictions suggested that the WOG vessels were generally less susceptible than Ringhals. However, several vessels were found to be more susceptible. Since this initial evaluation, three of these vessels were inspected for penetration cracking. One vessel head was found with cracking in a single penetration and no cracking was found in the penetrations of the other two plants. The level and depth of cracking was found to be covered by the Westinghouse Safety Evaluation.

4. *A penetration leakage assessment summarizing leak rate vs. crack size.* Expectations from this evaluation were that (a) leakage would be detected well before cracks extended to their critical flaw size (through-wall, and 8.5-20 inches long) and (b) Boron deposits would be significant enough from small flaws to be readily visible during a Generic Letter 88-05 walkdown.

5. *A vessel head wastage and structural evaluation.* The evaluation showed that the loss of approximately 1.0 in3 of vessel head material per year could be expected if cracks initiated and propagated through wall, however, vessel structural margins would be maintained for at least six additional years following the through wall leak.

1.2 HISTORICAL BACKGROUND

In 1991, during a hydrotest of the reactor coolant system at the Bugey Unit 3 power plant in France, a leak from the reactor vessel head was detected by acoustic monitoring [2]. Subsequent investigation, by visual examination and destructive testing, revealed that the leak came from a through wall flaw in one of the head penetrations. Further inspections on this and many other plants in France led to the discovery of flaws in the head penetrations of several plants. Examinations confirmed that the problem was directly related to Primary Water Stress Corrosion Cracking (PWSCC).

EdF conducted additional CRDM (Control Rod Drive Mechanism) penetration inspections at its nuclear plants, using eddy current techniques for indication detection and ultrasonic methods for defect size determination. Inspection results and metallurgical examinations confirmed PWSCC in CRDM penetrations at several other EdF plants. This was a concern to the French regulatory authorities as well as to the other PWR owners and regulatory authorities around the world.

These incidents are similar in nature to what occurred to other Alloy 600 tubular parts used in the Reactor Coolant System (RCS). Over the past few years, cracks in Alloy 600 pressurizer heater sleeve penetrations and instrumentation nozzles [3, 4] have been reported at non-Westinghouse supplied domestic and French PWR plants. In February 1990 the USNRC issued information Notice 90-10 on this issue [5]. The Notice informed PWR utilities of a number of incidences of PWSCC of Alloy 600 in applications other than steam generator tubing and suggested that utilities review their Alloy 600 applications and implement an augmented inspection program as necessary. In 1990, EPRI issued a report [4] which suggested that utilities should identify locations where Alloy 600 is used on the primary side, review the material and fabrication records to assess material susceptibility to PWSCC in terms of microstructure, stress, and environment, and implement an inspection program to detect leakage or cracking with the view of replacing susceptible components, as appropriate.

The Westinghouse Owners Group (WOG) and Westinghouse initiated and helped to lead a joint industry owners group under NUMARC, now the Nuclear Energy Institute (NEI), beginning in 1992. The group consists of all owners of Pressurizer Water Reactors in the USA along with EPRI. This group shared technical information and developed consistent safety evaluations and evaluation procedures for flaws that may be found during inspections. The group also worked with EPRI to develop inspection performance demonstrations for the head penetration inspections. The group demonstrated to the US Nuclear Regulatory Commission that cracking on the head penetrations was not an immediate safety issue. The NRC concurred with the Westinghouse conclusion, stating that vessel head penetration cracking is not an immediate safety issue [5].

1.3 INSPECTIONS PERFORMED TO DATE

In 1994, two WOG/Westinghouse PWR plants in the US (Point Beach Unit 1 and D. C. Cook Unit 2) voluntarily performed inspections of the CRDM penetrations. The results showed that there were no indications found in Point Beach Unit 1. Three indications were found in a single penetration at D.C. Cook Unit 2. These were significant cracks but considerably smaller than the NRC approved acceptance limit.

In Spring of 1996, D. C. Cook Unit 2 re-inspected some of their penetrations that had been previously inspected and confirmed the same indications reported earlier. No new indications were found and the existing indication was successfully repaired. Meanwhile, North Anna Unit 1 inspected 20 out of the total complement of 65 penetrations. No indications were found.

A large number of inspections have been performed on Westinghouse supplied reactor vessel head penetrations throughout the world, and this section will document those inspections, and the findings to date.

ASME Code Section XI inspections (VT-3) have been performed for a number years on the head penetration to reactor vessel partial penetration weld, and the weld between the head penetration tube and the control rod drive mechanism (CRDM). While these inspections do not cover the Alloy 600 inside diameter surface region of the head penetration directly, they do provide surveillance information on the head penetration region, and must be performed on every penetration once every ten years. To date no indications have been reported.

A second series of inspections which have been carried out regularly since 1988 involves visual surveillance of the head for boron deposits which would be evidence of leaks, following NRC Generic Letter 88-05. Some boron deposits have been found by this surveillance, but the sources of the leakage were not from cracked head penetrations. Generally these leaks have been associated with mechanical seals or canopy seals on the vessel head.

Westinghouse supplied NSSS plants in Spain, Sweden, Switzerland, Belgium, Brazil, and Korea have conducted NDE inspections on Reactor Vessel Head Penetrations. By the beginning of 1996, some 5200 penetrations had been inspected worldwide. The results are summarized in Table 1-1. On average, indications were found in approximately 2% of the penetrations that were inspected. Based on Table 1-1, it appears that the rate of indications at U.S. plants is significantly less than that of the French plants. The operating time for the plants of US manufacture where the inspections have been performed has in most cases been much longer than for the French plants. Of all these inspections, only one penetration was found to have through-wall cracking: the Bugey plant where cracking was first identified.

It will be of interest to examine the history of inspections of the plants of Westinghouse design worldwide, as well as the plants of Westinghouse design with US fabrication. A relatively large number of these plants have been inspected, and very few indications have been found. Outside of France, a total of 39 plants of Westinghouse design have been inspected. Of approximately 1900 penetrations inspected, only 10 were reported to be cracked, amounting to a less than 0.6 percentage. Of the 39 plants, 9 were manufactured in the USA, and for these plants approximately 310 penetrations were inspected with only one reported to be cracked. Thus, for Westinghouse plants manufactured in the USA, only 0.3 percent of the penetrations have been found to be cracked.

Root cause evaluations concluded that the cracks were caused by PWSCC of the Alloy 600 material. Electricite de France (EdF) and Westinghouse concluded that the following factors contributed to the Bugey Unit 3 PWSCC.

- Susceptible microstructure produced during manufacturing
- Surface finish on the inside diameter surface of the penetration
- Stresses induced during welding, which caused ovalization of the penetration

TABLE 1-1
WORLDWIDE VESSEL HEAD PENETRATION PWSCC INSPECTION RESULTS*

Country	Number of Plants Inspected	Total No. of Penetrations in the plants	Number of Penetrations Inspected	Penetrations With Indications	Rate of Indication Detected**
France	47	3225	3213	105	3.3%
Sweden	3	195	190	7	3.7%
Switzerland	2	72	72	2	2.8%
Japan	17	960	834	0	0
Belgium	7	435	435	0	0
Spain	5	325	102	0	0
Brazil	1	40	40	0	0
South Africa	1	63	63	0	0
South Korea	1	65	65	0	0
United States	5	314	217	1***	0.5%
Total:	89	5694	5231	115	2.0%

* Based on data available as of January 1996 (Europe) and July 1996 (U.S.).

** Ratio of number of penetrations with indications detected to number of penetrations inspected.

*** Oconee indications were not counted as cracks, because they had no measurable depth. Eddy current reinspection after one cycle did not indicate any growth

1.4 WOG AND NUCLEAR INDUSTRY PROGRAMS SUMMARY

A number of WOG programs were initiated to investigate the reactor vessel head penetration PWSCC issue. The key programs are summarized in Table 1-2. Additionally, selected utility programs have been responsible for the resolution of IGA due to sulfur species, and penetration attachment weld cracking. Domestically, the Babcock and Wilcox Owners Group (BWOG), Combustion Engineering Owners Group (CEOG), Westinghouse Owners Group (WOG) and the Electric Power Research Institute (EPRI) agreed to combine their efforts as part of the Nuclear Energy Institute's (NEI) Alloy 600 CRDM Head Penetration Cracking Task Force. The purpose of the task force was to evaluate the issue and to recommend appropriate generic actions. Through this effort, the Owners Groups (OGs) and EPRI have conducted the following tasks:

- Performed safety analyses of vessel head penetration cracking
- Standardized flaw evaluation methods
- Developed flaw acceptance criteria
- Developed inspection methodologies to size indications in head penetrations
- Evaluated remedial measures and created probabilistic and economic decision making tools
- Evaluated leakage effects on the vessel head low alloy steel shell

In addition, WOG has developed penetration repair techniques, plant inspection guidelines, and evaluated available leakage detection devices.

The NRC has evaluated the safety analyses and concluded that PWSCC of Alloy 600 head penetration is not an immediate safety concern [6].

Under the programs, research on PWSCC was conducted domestically and overseas, for example, as shown in Refs. 3, 7, 8, 9 and 10. The studies focused on material aspects and mechanics. Material aspects, thermomechanical processing effects, material properties, residual stresses, and microstructure were studied. A model of PWSCC susceptibility and cracking probability was developed [10].

Finite element analyses were performed to determine stresses in the penetrations. The finite element analyses performed included simulation of the whole spectrum of the mechanical fabrication sequences experienced by the RV head penetrations, such as the welding process, hydrotest, straightening and service loads. The finite element simulations allowed the determination of the applied as well as the residual stresses in the penetrations under any given specific geometrical, material, welding, temperature, and loading conditions. Based on the stress data, PWSCC initiation, crack propagation, and final failure were then evaluated. The analysis also furnished results for the time period required for the PWSCC to penetrate through the wall thickness of the penetration and the critical crack size above which instability would occur. Initial crack growth behavior was assumed to be represented by the model developed by P. Scott [11].

Confirmatory crack growth laboratory testing was immediately begun to verify that this initial assessment was correct. The integrity model was structured to be applicable to all penetrations regardless of product form or vessel fabricator. Subsequent testing to obtain comparison data in this area was initiated in 1996. The crack growth test results and preliminary crack initiation test results are discussed in Sections 2 and 3.

TABLE 1-2 SUMMARY OF KEY TASKS PERFORMED BY WOG		
Item	**Task Description**	**Status**
1	Root Cause of Cracking	C
2	Key Material & Operation Parameters	C
3	Elastic Finite Element Analysis: Residual/Operational	C
4	Elastic/Plastic Finite Element Analysis: Residual/Operational; 3 Locations	C
5	Crack Propagation/Acceptable Flaw Size Analysis	C
6	Penetration Leakage & Vessel Head Wastage Assessment	C
7	Safety Evaluation	C
8	Plant Screening/Susceptibility Criteria	C
9	Material Microstructure Characteristics	C
10	Leakage Detection Methods Survey	C
11	Evaluation of PWSCC Mitigation Methods	O
12	Grinding Effect on Residual Stresses	C
13	Development/Evaluation of Repaired Configurations	C
14	OD Crack Assessment	C
15	Crack Growth Data and Testing	O
16	Inspection Timing and Economic Decision Tools	C
17	Penetration Attachment Weld Safety Evaluation Report	C
18	Crack Initiation Characterization Studies	O
19	Residual Stress Measurements	C
20	Development of PWSCC Susceptibility Ranking Models	C

Key: C = Complete O = Ongoing.

2.0 DEVELOPMENT OF A CRACK GROWTH RATE MODEL FOR ALLOY 600 HEAD PENETRATIONS

Crack growth rate testing has been underway since 1992 to characterize the behavior of head penetration materials. The "modified Scott model," as described below was initially used for safety evaluation calculations in the NRC submittals made in 1992 and 1993. The goal of this section of the report is to review the applicability of that model in light of the past five years of testing, during which over forty specimens have been tested representing 15 heats Alloy 600 of material. The original basis of the model will be reviewed, followed by all the available laboratory results, and finally a treatment of the available field results.

The effort to develop a reliable crack growth rate prediction model for Alloy 600 began in the Spring of 1992, when the Westinghouse, Combustion Engineering, and Babcock and Wilcox Owners Groups were developing a safety case to support continued operation of plants. At the time there was no available crack growth rate data for head penetration materials, and only a few publications existed on growth rates of Alloy 600 in any product form.

The best available publication was found to be that of Peter Scott of Framatome, who had developed a growth rate model for PWR steam generator materials [11]. His model was based on a study of results obtained by McIlree and Smialowska [12] who had tested short steam generator tubes which had been flattened into thin compact specimens. His model is shown in Figure 2-1. Upon study of his paper there were several ambiguities, and several phone conversations were held to clarify his conclusions. These discussions indicated that Reference 11 contains an error, in that no correction for cold work was applied to the McIlree/Smialowska data. The revision of the Peter Scott model is presented below.

An equation was fitted to the data of Reference 12 for the results obtained in water chemistries that fell within the standard specification for PWR primary coolant. Results for chemistries outside the specification were not used. The following equation was fitted to the data for a temperature of 330°C:

$$\frac{da}{dt} = 2.8 \times 10^{-11} (K-9)^{1.16} \text{ m/sec}$$

where K is in MPa[m]$^{0.5}$. This equation implies a threshold for cracking susceptibility, K_{Iscc} = 9 MPa[m]$^{0.5}$. Correction factors for other temperatures are shown in Table 2-1.

The next step described by Scott [11] in his paper was to correct these results for the effects of cold work. Based on work by Cassagne and Gelpi [13], he concluded that dividing the above equation by a factor of 10 would be appropriate to account for the effects of cold work. This step was inadvertently omitted from Scott's paper, even though it was discussed. The revised crack growth model for 330°C then becomes:

$$\frac{da}{dt} = 2.8 \times 10^{-12} (K-9)^{1.16} \text{ m/sec}$$

This equation was verified by Scott in a phone call in July 1992.

Scott further corrected this model for the effects of temperature, but his correction was not used in the model employed. Instead, an independent temperature correction was developed based on service experience. This correction uses an activation energy of 32.4 kCal/mole, which gives a smaller temperature correction than that used by Scott (44 kcal/mole), and will be discussed in more detail below.

Scott's crack growth model for 330°C was independently obtained by B. Woodman of ABB-CE [14], who went back to the original data base, and had a smaller correction for cold work. His equation was of a slightly different form:

$$\frac{da}{dt} = 0.2 \exp\left[A + B \ln\{\ln (K-C)\}\right]$$

Where A = -25.942

B = 3.595

C = the threshold for cracking

This equation is nearly identical with Peter Scott's original model uncorrected for cold work. This work provided an independent verification of Scott's work. A further verification of the modified Scott model used here was provided by some operational crack growth rates collected by Hunt, et al [15].

The final verification of Peter Scott's model will come from actual data from head penetration materials in service, as will be discussed in detail below. To date 15 heats have been tested in carefully controlled PWR environment. One heat did not crack, and of the fourteen heats where cracking was observed, the growth rates observed in twelve were bounded by the Scott model. Two heats cracked at a faster growth rate, and the explanation for this behavior is being investigated.

A compilation was made of the laboratory data obtained to date in the Westinghouse laboratory tests at 325°C, and the results are in Figure 2-3. Notice that much of the data is far below the Scott model, and a few data points are above the model. These results represent 14 heats of head penetration materials.

The effect of temperature on crack growth rate was first studied by compiling all the available crack growth rate data, for both laboratory and field cracking of Alloy 600. This information is summarized in Figure 2-2, where the open symbols are for steam generator tube materials, and the solid symbols are for head penetration materials. The results are presented in a simple format, with crack growth plotted as a function of temperature. The effect of stress intensity factor variation has been ignored in this presentation, and this doubtless adds to the scatter in the data. The remarkable result is a consistent temperature effect over a temperature range from 288°C to 370°C, more than covering the temperature range of PWR plant operation although there is a wide scatter band in the figure. The work done originally in 1992 results in a calculated activation energy of 32.4 Kcal/mole, which has been used to adjust the base crack growth law to account for different operating temperatures.

A series of crack growth tests is in progress under carefully controlled conditions to study the temperature effect for head penetration materials, and the results obtained to-date are shown in Figure 2-2. Sufficient results are available to report preliminary findings. The tests were performed with an applied stress intensity factor of 23 Ksi $\sqrt{in}$ (25.3 MPa[m]$^{0.5}$), periodic unload/reload parameters of a hold time of one hour and a water chemistry of 1200 ppm B + 2 ppm Li + 25 cc/kg H_2. The results are consistent with the previous steam generator and head penetration material work. In the case of heat 69, the three results in the middle of the temperature range, 309°C, 327°C and 341°C have the same trend as the scatter band, almost exactly, while the high temperature and low temperature results are both lower than would be predicted by the activation energy, as shown in Figure 2-2. The results for heat 20 show a similar behavior, with the results at 325°C and 340°C also within the scatter band and nearly parallel to the heat 69 specimens, but at a lower crack growth rate, as shown in Figure 2-2.

The effects of several different water chemistries have been investigated in a closely controlled series of tests, on two different heats of archive material. Results showed that there is no measurable effect of Boron and Lithium on crack growth.

The key test of the laboratory crack growth data is its comparison to field data. Crack growth from actual head penetrations has been plotted on Figure 2-2 as solid points. The solid circles are from Swedish and French plants and the solid stars are from a US plant.

Figure 2-4 shows a summary of the inservice cracking experience in the head penetrations of French plants, prepared by Amzallag [16], compared with the Westinghouse laboratory data, corrected for temperature. This figure shows excellent agreement between lab and field data, further supporting the applicability of the lab data.

Therefore it can be seen that the laboratory data is well represented by the Scott model corrected for temperature using an activation energy of 32.4 kcal/mole. Also the laboratory results are consistent with the crack growth rates measured on actual installed penetrations. Therefore the use of the modified Scott model in the safety evaluations and other evaluations of head penetration integrity is still justifiable, in light of both laboratory and field data obtained to date.

TABLE 2-1
TEMPERATURE CORRECTION FACTORS FOR CRACK GROWTH: ALLOY 600

Temperature	Correction Factor (CF)	Coefficient (Co)
330C	1.0	2.8×10^{-12}
325	0.798	2.23×10^{-12}
320	0.634	1.78×10^{-12}
310	0.396	1.11×10^{-12}
300	0.243	7.14×10^{-13}
290	0.147	4.12×10^{-13}

$$\frac{da}{dt} = Co\,(K-9)^{1.16}\ \mathrm{m/s}$$

where K is in MPa[m]$^{0.5}$

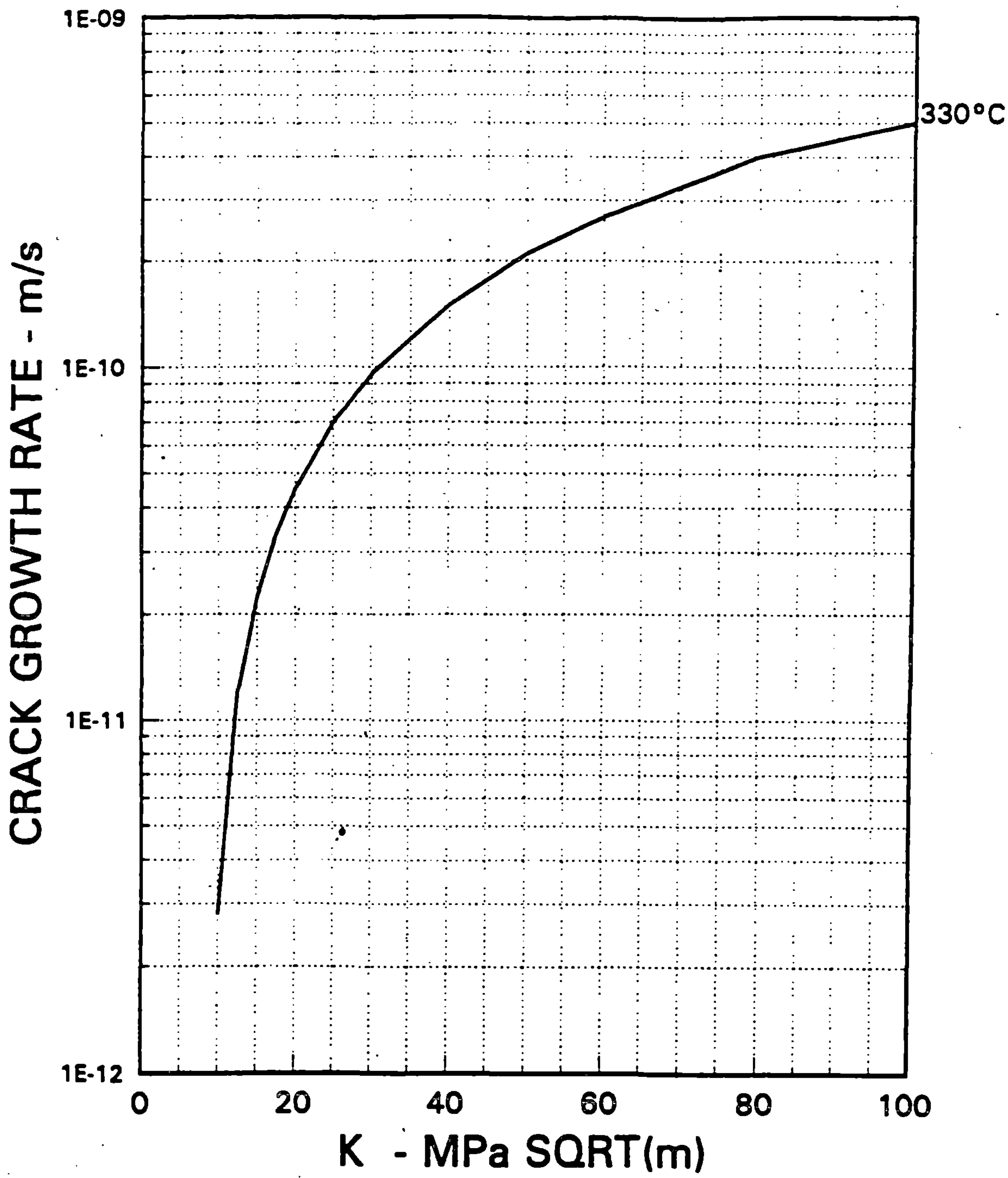

Figure 2-1 Scott Model for PWSCC of Alloy 600 at 330°C, as modified from Reference 11

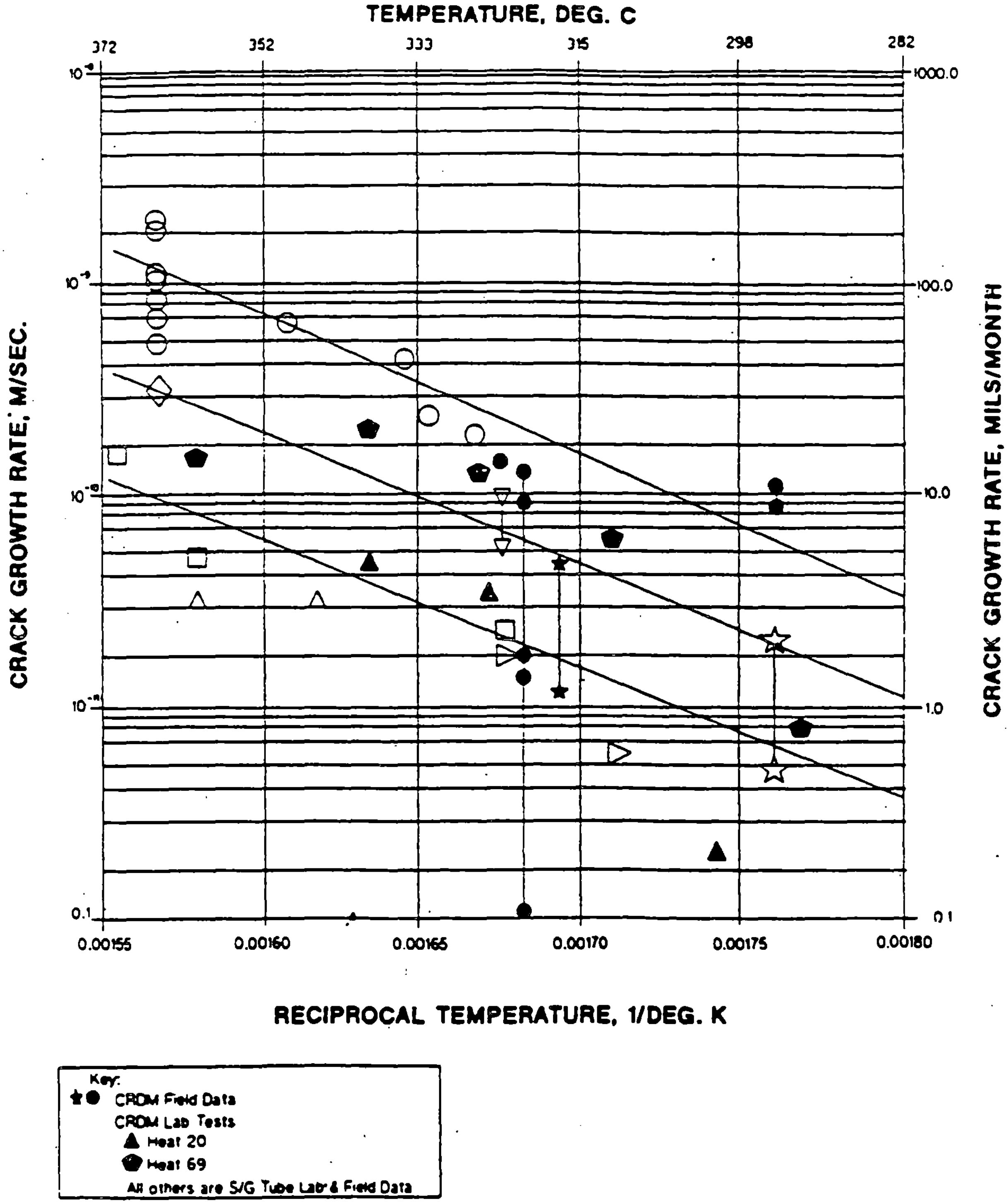

Figure 2-2 Comparison of Temperature Effects Results with Other Laboratory and Field Data

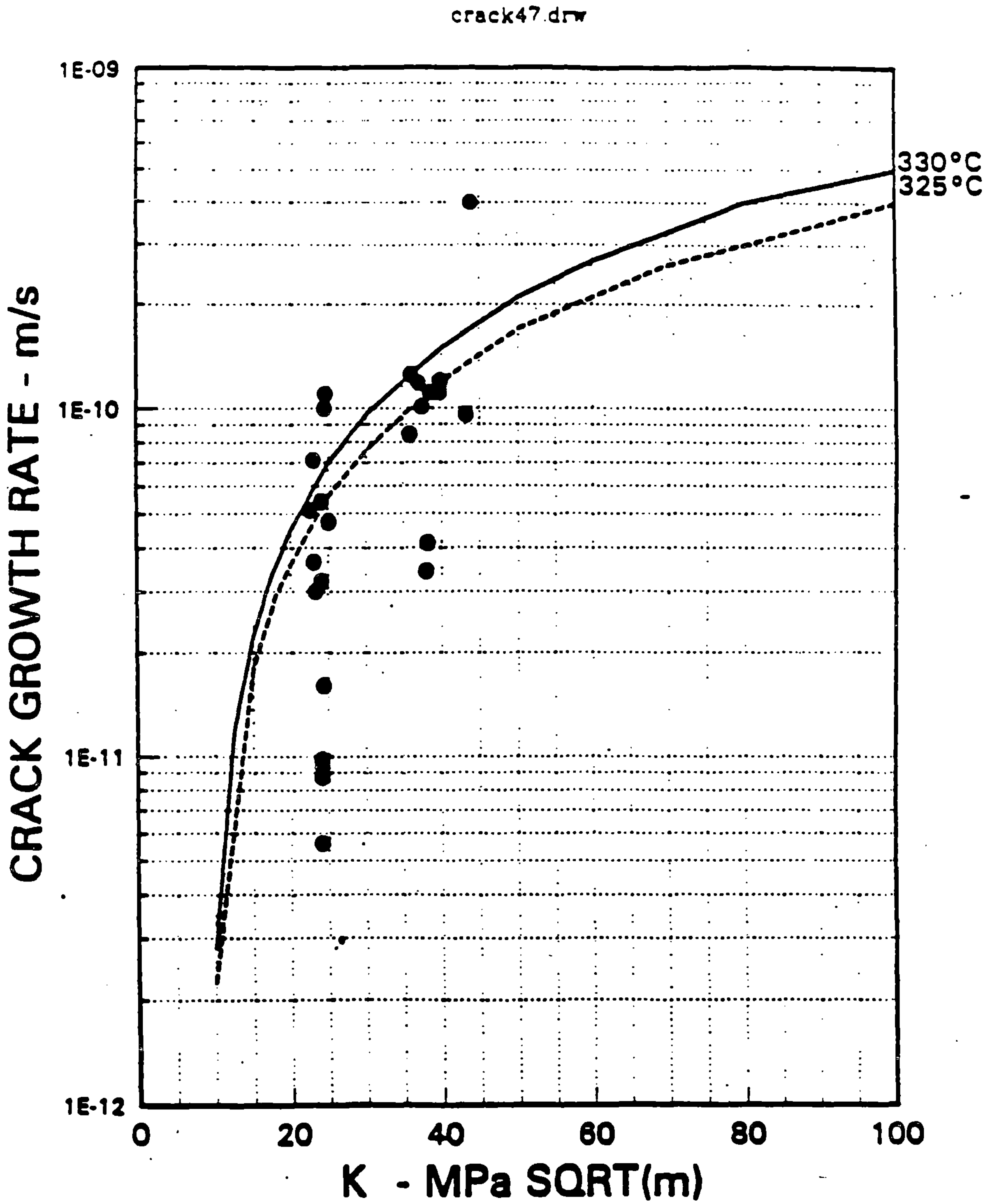

Figure 2-3 **Summary of Available Westinghouse Laboratory Data for Alloy 600 Head Penetrations at 325°C**

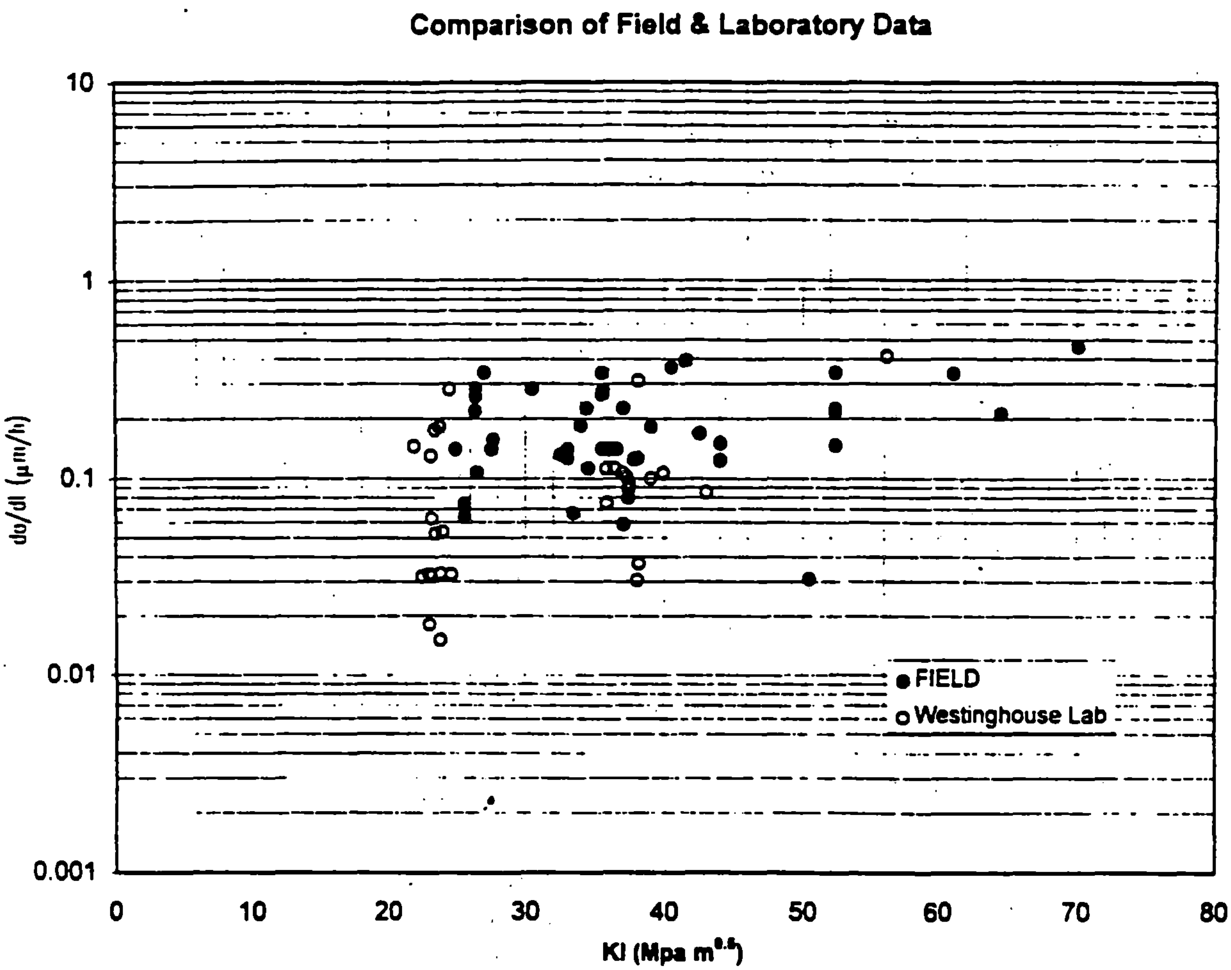

Figure 2-4 Comparison of French Field Data and Westinghouse Laboratory Data (W results reduced to 290°C using Q = 130 KJ/mole) [16]

3.0 WESTINGHOUSE CRACK INITIATION MODEL DEVELOPMENT AND CRACK INITIATION TESTING

3.1 CRACK INITIATION MODEL

Westinghouse advanced an Alloy 600 PWSCC initiation model for primary components in Pressurized Water Reactors [10]. Briefly, the model incorporates three contributing factors for the prediction of crack initiation time; namely, material condition, stress, and temperature. These are discussed below.

Material Condition and Microstructure

As reported by several authors [17, 18, 19, 20, and 21], the Alloy 600 microstructure is a function of the thermomechanical history of the material heat as well as its carbon content. Alloy 600 material heats subjected to mill annealing at low temperatures, i.e., 926°C or less, exhibit a fine grained microstructure with heavy transgranular carbide precipitation and little or no carbides precipitate on the grain boundaries. Such a microstructure is reported to be more susceptible to PWSCC. On the other hand, a high temperature mill-anneal (>1000°C) tends to put more carbon into solution, increases grain size, produces grain boundary chromium carbide precipitation and renders the material more resistant to resist PWSCC. Norring, et. al. [22], did not find a correlation between the total content of carbon and the crack initiation time, but they observed good correlation between the amount of grain boundary carbides and crack initiation time. The fact that grain boundary precipitation is beneficial to PWSCC has been reported by many researchers [23]. Norring, et. al., [22], showed that the crack initiation time varied directly (linearly) with grain boundary carbides. Their data suggested that when the grain boundary carbide coverage is increased by a factor of 3, the crack initiation time also increased by a similar factor (from 4,000 hours to 12,000 hours). Bandy and Van Rooyen [24], pointed out that in addition to grain boundary carbide coverage, other features relating to processing history variables such as carbon concentration gradients, substructural features, grain size distribution, cold work, intragranular carbide distribution and the grain boundary segregates all play an important role in the cracking behavior of the Alloy 600 material.

When considering the influence of microstructure on the PWSCC susceptibility for the purpose of the current evaluation, to enable comparison of heats fabricated at different vendor shops, the thermomechanical processing history effect is separated from the grain boundary carbide coverage effects. In general, the influence of the grain boundary carbides is known and the coverage (G) can be easily measured directly from the microstructure. The influence of other structural features due to processing history cannot be assessed directly. These processing effects are represented in the current treatment by a single parameter (A) characteristic of the fabrication shop (vendor). This approach provides a means of comparing the PWSCC susceptibilities of Alloy 600 material heats from different vendor shops although they may contain similar grain boundary carbide contents.

Influence of Stress

Steady state tensile stress in the component, either due to residual and/or applied loads, has a strong influence on the PWSCC.

Bandy and Van Rooyen [24], reported that the time to failure varied inversely as the fourth power of applied stress in both annealed and coldworked specimens. They also reported data to support that coldwork reduces the resistance to PWSCC. The effective stress at a given Alloy 600 location is a function of the fabrication steps and their sequence, the yield stress of the material, and the service stress. In general, the local residual stresses resulting from fabrication can play a more significant role than the service stresses themselves.

Temperature Effects

Several investigators [17, 24], examined the role of temperature on PWSCC. It is well established from these results that the crack initiation time decreases exponentially with temperature and that they are related through an Arrhenius equation expressed as a function of the activation energy of the process. The experimental results confirm that Alloy 600 PWSCC is a thermally activated process and the activation energy for the process varies approximately between 50 to 55 kcal per mole. An activation energy value of 55 kcal/mole is consistently applied throughout the current assessments, for crack initiation. A different value, 32.4 applies for crack growth as was discussed in Section 2.

3.2 THE WESTINGHOUSE CRACK INITIATION MODEL

Consistent with the contributing factors discussed above, the crack initiation time (t_i) or the rate of crack initiation ($1/t_i$) is proportional:

$$1/t_i \ \alpha \ (\text{Stress})^n$$

$$\alpha \ e^{-Q/RT}$$

$$\alpha \text{ inverse of the grain boundary carbide coverage factor, } (1/G)$$

so that $$1/t_i \ \alpha \ \frac{\sigma^n \ e^{-Q/RT}}{G}$$

Since the nature of the vendor thermomechanical processing is also a significant contributing factor, one can say that for a given fabrication process

$$1/t_i = A \ \frac{\sigma^n \ e^{-Q/RT}}{G} \tag{3-1}$$

The proportionality constant "A" can be chosen to represent the processing conditions representative of a given manufacturing process or manufacturer, and could include parameters such as yield strength as part of the expression.

"A" can be assessed for a given heat by substituting the parameters of a service component with a known cracking history for the heat of material. "A" will then represent the processing condition (or the vendor) by the definition we have just established.

The parameters in the above rate equation (3-1) are described below:

A is a constant, relating to the processing, and fabrication conditions of the material

G is the grain boundary carbide coverage factor

σ is the effective tensile stress (resulting from applied and residual stresses)

n is the stress exponent having a value ranging from 3.5 to 4.5 for Alloy 600 in primary water

Q is the activation energy for the crack initiation process and has an approximate value of 55 kcal/mole

R is the gas constant (1.987 cal/mole degrees K)

T is the absolute temperature in degrees K, and

t_i is the time to initiate cracking.

3.3 CRACK INITIATION TESTING

Westinghouse currently has an ongoing autoclave test program to establish the PWSCC crack initiation behavior of archive Alloy 600 RV head penetration material heats from a variety of fabricators representative of microstructures of RV head penetrations that are currently in service. The objectives of the Program are:

- **To determine the effect of penetration microstructure and material type (vendor) on the relative susceptibility to cracking.**
- **To define a material index (A) to assist in plant maintenance planning.**

The program is sponsored by EPRI and the CE, W, and B&W owners groups. The accelerated testing is conducted under dense steam with hydrogen at 400°C and utilizes full size ring samples fabricated from RV head penetration tubing from different vendor shops. A listing of vendor shops representing the ring samples employed in the testing is provided in Table 3-1.

To provide reference benchmarking, samples from steam generator rolled transition tubing and Alloy 690 penetration material are also included in the test matrix. Penetration material specimens with known crack growth behavior measurements from previous test programs are included for comparison with other data.

This environment has been shown to provide adequate acceleration (up to 500x) to provide results within the test period. This will be verified using the specimens from heats that have been tested previously. Test samples under the doped steam test will be inspected at 25, 50, 100, 200, 400, 800, 1400 and 2000 hours. Inspection will include visual, metallographic and destructive examinations.

The ID surfaces of the ring samples are strained by controlled cyclic ovalization to simulate the residual hoop stresses in the plant. The stresses are quantified based on the ovalization. The final cycle of ovalization is calibrated to induce a 2mm difference in measured inside diameter. This corresponds to the upper 95% of the measured ovality in the outermost penetrations in service. The cyclic straining procedure of the full ring samples is illustrated by the loading curve shown in Figure 3-1.

The testing is conducted under two phases. The first phase involves a cumulative exposure of up to 800 hours in six exposure intervals. Periodic inspections are performed at 25, 50, 100, 200, 400 and 800 cumulative hours of exposure. The second phase testing involves the exposure of specimens for a cumulative exposure of up to 2000 hours with an interim inspection at 1400 hours. Currently, with the Phase I testing completed, the preliminary test results indicate clear trends in the initiation behavior. Out of the six heats of material tested, two of the heats consistently showed higher susceptibility to cracking; the worst heat being the heat that also showed the highest crack growth rate under the crack growth test program discussed in Section 2. Further useful trends in cracking behavior are expected at the end of the 2000 hours exposure. The overall results of the program are expected to provide useful information for plant maintenance planning.

TABLE 3-1
MATERIAL HEATS EMPLOYED IN THE
ALLOY 600 RVHP CRACK INITIATION TESTS

S No.	Heat No.	Supplier	Fabricator	As Pred. Size
1	93510	B&W	B&W	½" (6 pcs)
2	93510-R	B&W	B&W	½" (6 pcs)
3	91069	B&W	B&W	½" (6 pcs)
4	93511	B&W	B&W	½" (6 pcs)
5	WF675	B&W	Creusot Loire	3-5/8" (1 pc)
6	WF151	Sizewell	Creusot Loire	3-½" (1 pc)
7	M-7817-1 (EO-6943#2)	CE	Standard Steel	4-1/8" (1 pc)
8	R13-4 (NX64209)	CE	Huntington	4-1/8" (1 pc)
9	NX8101-75		Huntington	6" (1 pc)
10	NX34C3-68		Huntington	6" (1 pc)
11	R177	Vattenfall	Sanvik	6" (1 pc)

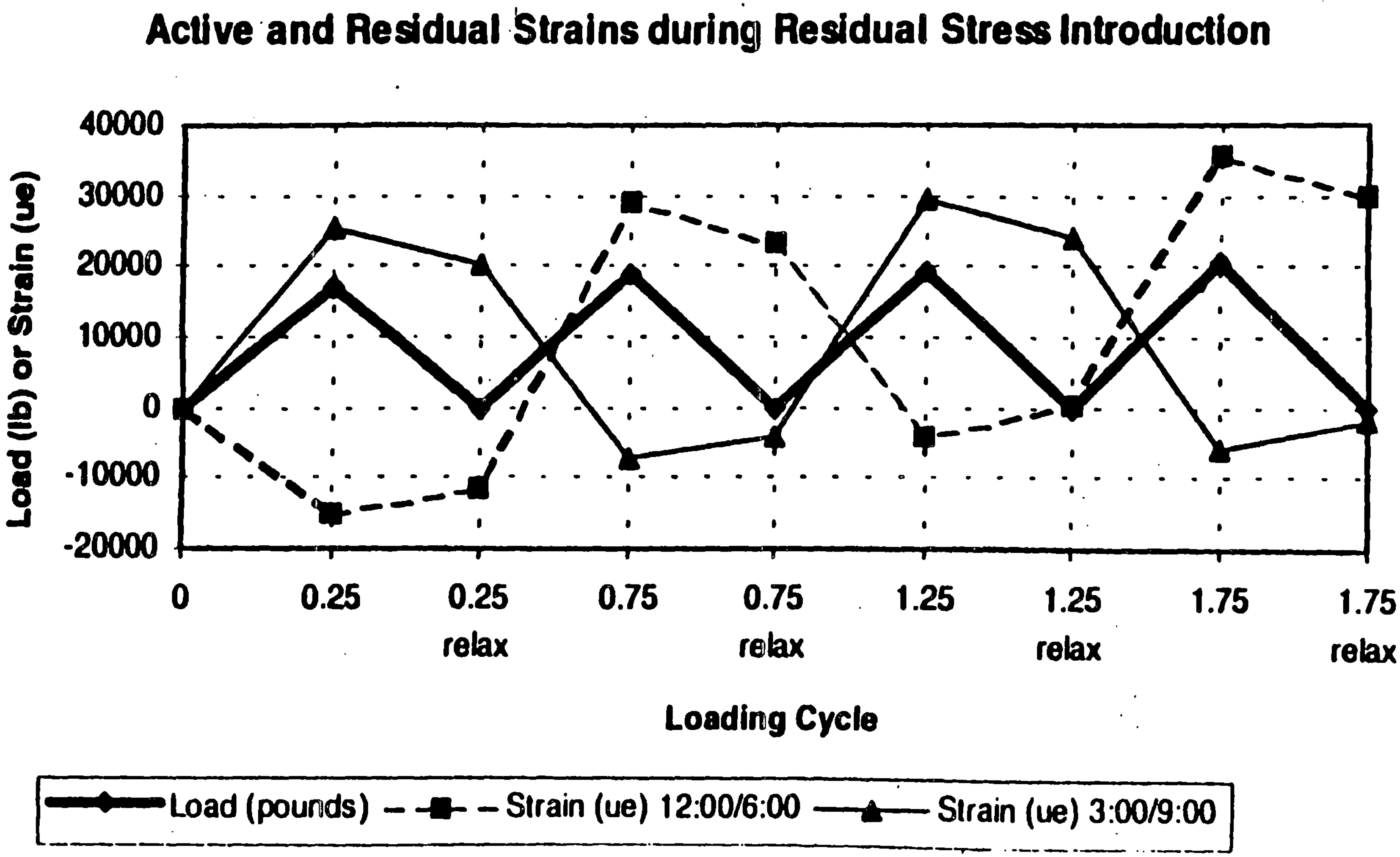

Figure 3-1 Residual Stresses and Strains Induced During Controlled Cyclic Ovalization of RV Penetration Ring Samples

4.0 TECHNICAL DESCRIPTION OF PROBABILISTIC MODELS

To calculate the probability of failure of the Alloy 600 vessel head penetration as a function of operating time t, $Pr(t \le t_f)$, structural reliability models were used with Monte-Carlo simulation methods. This section describes these structural reliability models and their basis for the primary failure mode of crack initiation and growth due to primary water stress corrosion cracking (PWSCC). The models used for the evaluation of head penetration nozzles are based upon the probabilistic and economic decision tools developed previously for the Westinghouse Owners Group (WOG). The capabilities of this software have already been verified in the following ways:

1. Calculated stresses compare well with measured stresses (see Figure 4-1),

2. Crack growth rates agree with measured field data (see Figures 2-3 and 2-4).

Recent improvements have also been made to the model in order to maximize its use for individual plant predictions. Among the changes were:

1. The model accepts measured microstructure (replication) and also has the capability to ignore its effects, if desired.

2. The relationship of initiation time to material microstructural effects and yield strength has been improved to more closely match the observations from the recent inspection at North Anna Unit 1,

3. Statistically based Bayesean updating of probabilities due to initial inspection results has been added (e.g. the lack of any indications at any given plant),

4. The uncertainty on crack growth rate after initiation has been updated to reflect the findings observed in the recent Westinghouse test data and the recent in-reactor measurement data to be published by EdF [16] (see Figure 4-2),

5. All models have been independently reviewed by APTECH Engineering (Begley and Woodman)[25], and an improved model was developed for the effect of monotonic yield strength on time to initiation, and

6. A wide range (both high and low values) of calculated probabilities are consistent with actual plant observations as discussed below.

The most important parameter for estimating the failure probability is the time to failure, t_f in hours. It is defined as follows:

$$t_f = t_i + (a_f - a_0) / da/dt \qquad (4\text{-}1)$$

where:

t_i	=	time to initiation in hours,
a_f	=	failure crack depth in inches,

a_0 = crack depth at initiation in inches and
da/dt = crack growth rate in inch/hour.

In equation (4-1), both the crack depths at failure and initiation may be specified as a fraction of the penetration wall thickness, (w). The failure depth a_f depends upon the failure mode being calculated. Since the failure mode of concern is axial cracks in the penetration that are deeper than the structural limit of 75% of the penetration wall thickness (w), it would be specified as:

$$a_f = 0.75\ w \tag{4-2}$$

The time to PWSCC crack initiation, t_i in hours, consistent with the previous equation 3.1 by RAO [3] and is defined by:

$$t_i = \frac{C_1 + (1 + C_2 P_{GBC})}{\sigma^{n_1} \cdot S_y^{n_2}} \exp\left(\frac{Q_1}{RT}\right) \tag{4-3}$$

C_1 = a log-normal distribution on the initiation coefficient, which was based upon the data of Hall and others [26] for forged Alloy 600 pressurizer nozzles, with only the uncertainty based upon the data of Gold and others [27],

C_2 = coefficient for the effect of grain boundary carbide coverage, which is based upon the data of Norring and others [22],

σ = the maximum residual and operating stress level derived from the detailed elastic-plastic finite-element analysis from the WOG study of Ball and others [28] as shown in Figure 4-1, with its normally distributed uncertainty being derived from the variation in ovality from Duran and others [29] (see Figure 4-3), which is a trigonometric function of the penetration diameter and setup angle (local angle between the head and longitudinal axis of penetration).

S_y = yield strength of the penetration material,

$n_1 n_2$ = exponents on stress and yield strength, respectively ($n_1 = 4$, $n_2 = 2.5$)

Q_1 = the activation energy for crack initiation, which is normally distributed,

R = universal gas constant, and

T = the penetration absolute temperature, which is uniformly distributed based upon the calculated variation of the nominal head operating temperature.

Equation 4-3 is equivalent to the initiation equation by Rao [3] as listed in Section 3.2, where $G/A = C_1 + (1 + C_2\, P_{GBC})/S_y^{n_2}$.

Either data from field replication [30] or the correlation model by RAO [31] can be used to determine the percent grain boundary carbide coverage, P_{GBC} in equation (4-3). The model [31] is a statistical correlation of measured values with the following materials certification parameters:

- Carbon content,
- Nickel content,
- Manganese content,
- Ultimate tensile strength and
- Yield strength.

The uncertainty on this model, which is as shown in Figure 4-4, applies equally well to both the predicted and measured values.

The hours at temperature per operating cycle (year), which is normally distributed, is used to check if crack initiation has occurred. Once the crack has initiated, it is assumed to have a depth of a_0 and its growth rate, da/dt, is calculated by the Peter Scott model, which matches the latest Westinghouse and EdF data and the previous data given in the WOG report on the industry Alloy 600 PWSCC growth rate testing results [32], as discussed in Section 2. The crack growth model is:

$$\frac{da}{dt} = C_3(K_I - K_{TH})^{1.16} \exp\left(\frac{Q_2}{RT}\right) \tag{4-4}$$

C_3 = a log-normally distributed crack growth rate coefficient (see Figure 4-2),

K_I = the stress intensity factor conservatively calculated assuming a constant stress through the penetration wall for an axial flaw at the inside surface with a length 6 times its depth using the following form of the Raju and Newman equations [33]:

$$K_I = 0.982 + 1.006\ (a\ / w)^2 s(\pi\ a)^{0.5} \tag{4-5}$$

Q_2 = activation energy for PWSCC crack growth, which is also normally distributed, and

K_{TH} = threshold stress intensity factor for crack growth

The probability of failure of the Alloy 600 vessel head penetration as a function of operating time t, $Pr(t \leq t_i)$, is calculated directly for each set of input values using Monte-Carlo simulation. Monte Carlo simulation is an analytical method that provides a histogram of failures with time in a given number of trials (simulated life tests). The area under the simulated histogram increases with time due to PWSCC. The ratio of this area to the total number of trials is approximately equal to the probability of failure at any given time. In each trial, the values of the specified set of random variables is selected according to the specified distribution. A mechanistic analysis is performed using these values to calculate if the penetration will fail at any time during its lifetime (e.g. 60 years). This process is repeated many times (e.g. 6000) until a sufficient number of failures is achieved (e.g. 10 per year) to define a meaningful histogram, which is an approximation of the lower tail of the true statistical distribution in time to failure (see Figure 4-5). The shape of the distribution depends upon the input median values and specified distributions of the random variables. It is not forced to be an assumed type of distribution (e.g. Weibull) as is done for other non-mechanistic probabilistic methods. For the worst penetration in one plant, the mean time to failure was greater than 160 years but its uncertainty was so large that the normalized area under the histogram (estimated probability) at 60 years was 8 percent.

To apply the Monte Carlo simulation method for vessel head penetration nozzle (VHPN) failure, the existing PROF (probability of failure) object library in the Westinghouse Structural Reliability and Risk Assessment (SRRA) software system was combined with the PWSCC structural reliability models described previously. This system provides standard input and output, including plotting, and probabilistic analysis capabilities (e.g. random number generation, importance sampling). The result was program VHPNPROF for calculation of head penetration failure probability with time.

As reported previously [34], the Westinghouse SRRA Software System has been verified by hand calculation for simple models and alternative methods for more complex models. Recently the application of this same Westinghouse SRRA methodology to the WOG sponsored pilot program for piping risk based inspection has been extensively reviewed and verified by the ASME Research Task Force on RBI Guidelines [35] and other independent NRC contractors. Table 4-1 provides a summary of the wide range of parameters that were considered in this comprehensive benchmarking study that compared the Westinghouse calculated probabilities from the analysis (labeled SRRA) with those from the pc-PRAISE program [36]. As shown in Figure 4-6, the comparison of calculated probabilities after 40 years of operation is excellent for both small and large leaks and full breaks, including those reduced due to taking credit for leak detection.

In addition, the VHPNPROF Program calculated probabilities of getting a given crack depth due to PWSCC were compared for four plants where sufficient head penetration information and inspection results were available. The four plants are identified in Table 4-2 along with the values of the key input parameters and calculated failure probabilities. Table 4-2 also shows the agreement between the latest available inspection results and VHPNPROF predicted failure trends due to PWSCC.

TABLE 4-1
PARAMETERS USED FOR THE pc PRAISE BENCHMARKING STUDY

Type of Parameter	Low Value	High Value
Pipe Material	Ferritic	Stainless Steel
Pipe Geometry	6.625" O.D. 0.562" Wall	29.0" O.D. 2.5" Wall
Failure Modes	Small Leak, Through-Wall Crack	Full Break, Unstable Fracture
Last Pass Weld Inspection	No X-Ray	Radiographic
Pressure Loading	1000 psi	2235 psi
Low-Cycle Loading	25 ksi Range 10 cycles/year	50 ksi Range 20 cycles/year
High-Cycle* Loading	1 ksi Range 0.1 cycles/min.	20 ksi Range 1.0 cycles/sec.
Design Limiting Stress	15 ksi	30 ksi
Disabling Leak Rate	50 gpm	500 gpm
Detectable Leak Rate	None	3 gpm

* Note: Mechanical Vibration (low value of stress range and high value of frequency) for small pipe, Thermal Fatigue (high value of stress range and low value of frequency) for large pipe.

TABLE 4-2
COMPARISON OF VHPNPROF CALCULATED PROBABILITIES WITH PLANT OBSERVATIONS

Parameters	Almaraz 1	D. C. Cook 2	Ringhals 2	North Anna 1
Hours of Operation	85,400	87,000	108,400	91,000
Setup Angle (°)	42.6	50.5	38.6	*
Temperature (°F)	604.3	598.5	605.6	600.0
Yield Strength (ksi)	37.5	58	51.2	51.2
Percent GBC	57.0	44.3	3.0	2.0
Flaw Depth/Wall	0.10	0.43	0.25	0.10
Initiation Probability	1.1%	41.4%	37.6%	15.3%
Failure Probability**	1.1%	38.1%	34.6%	15.3%
Penetrations With Reported Indications from ISI	0	1	3 (2 with scratches)	0

* Calculations performed at an equivalent setup angle for the 2nd highest stress location that could be inspected.

** Defined here as the probability of reaching the specified flaw depth for the individual penetration.

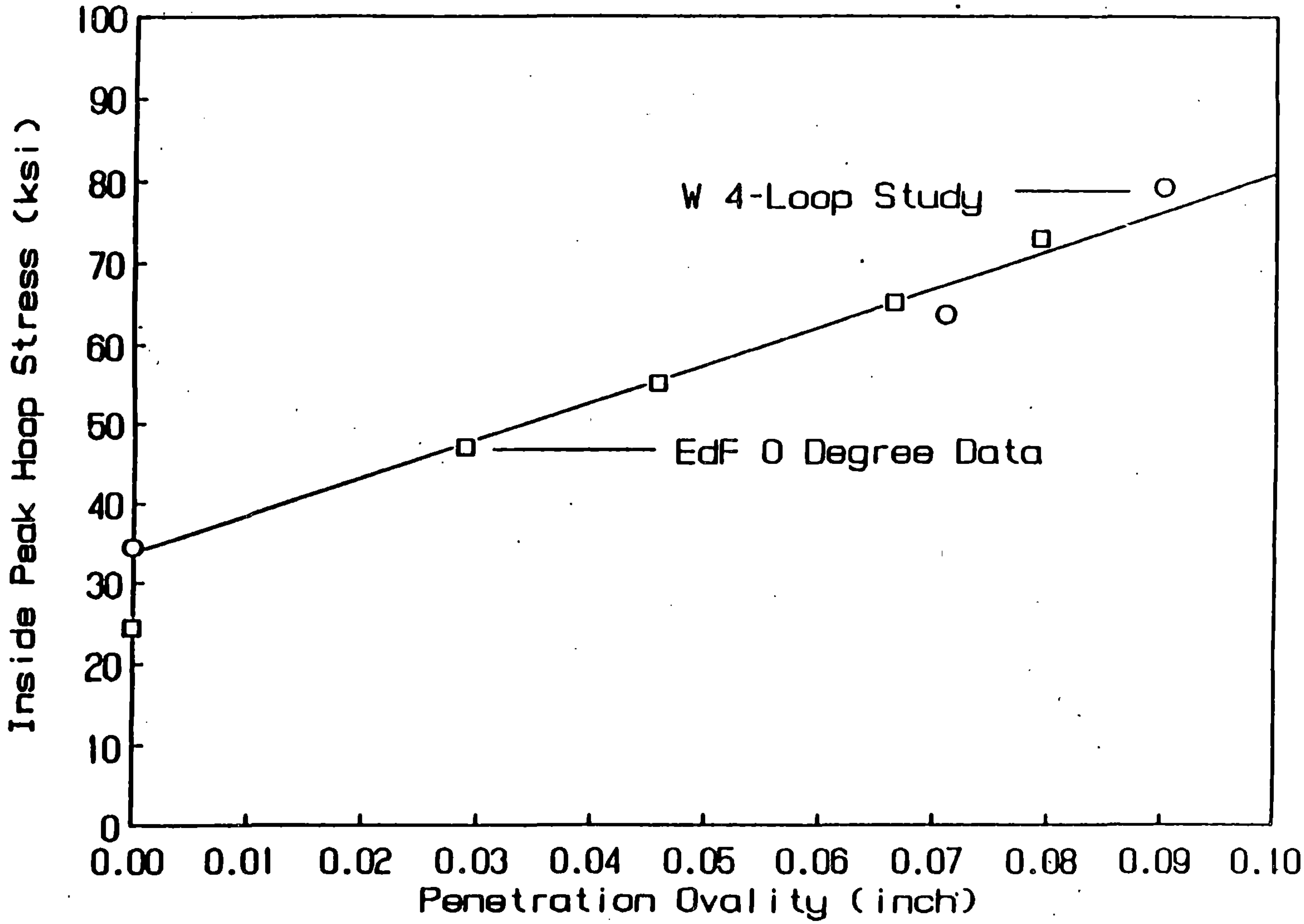

Figure 4-1 Vessel Head Penetration Stresses from WOG 4-Loop Plant Study [28]

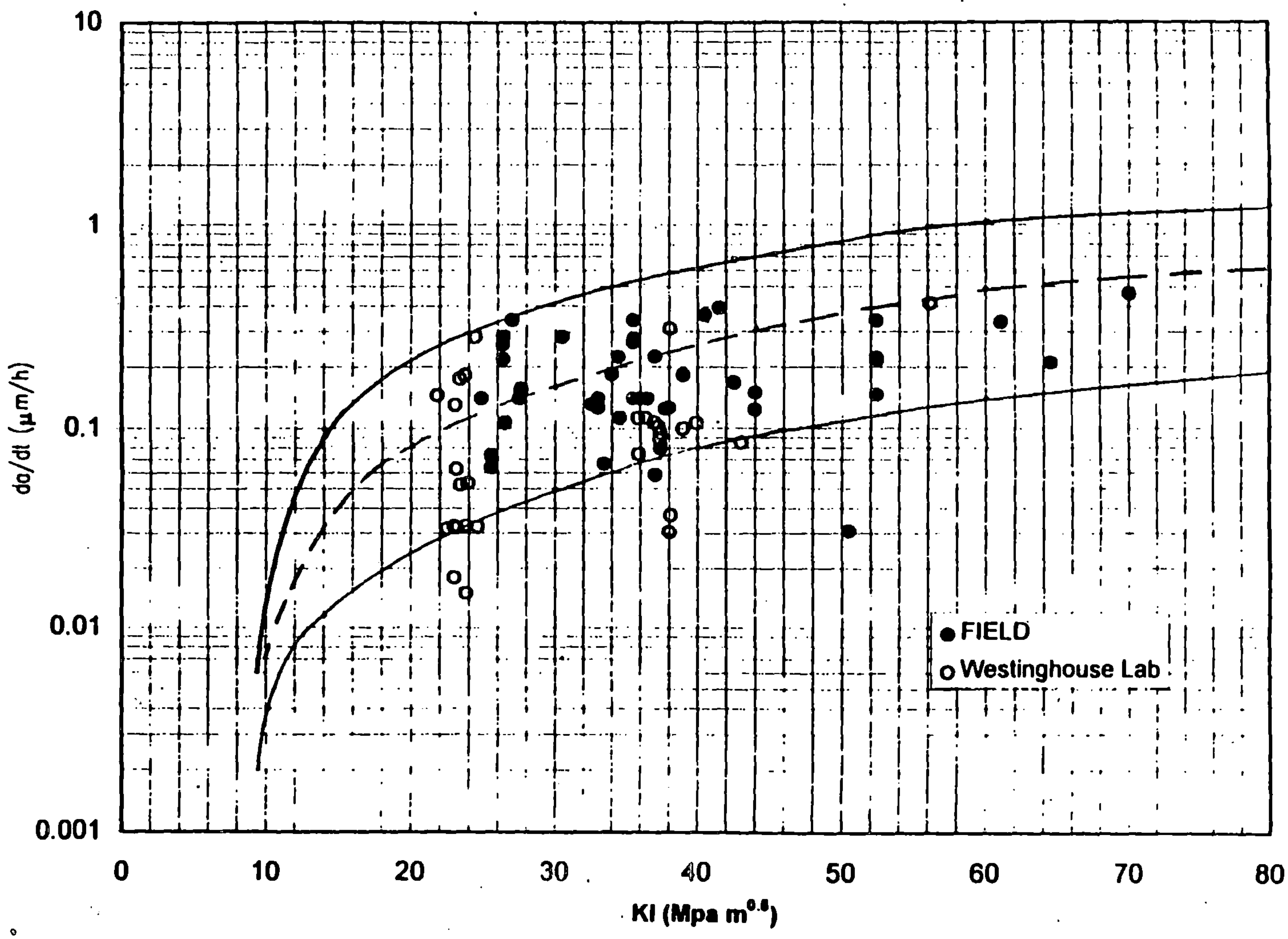

Figure 4-2 Comparison of Recent Alloy 600 Data with the Crack Growth Rate Model

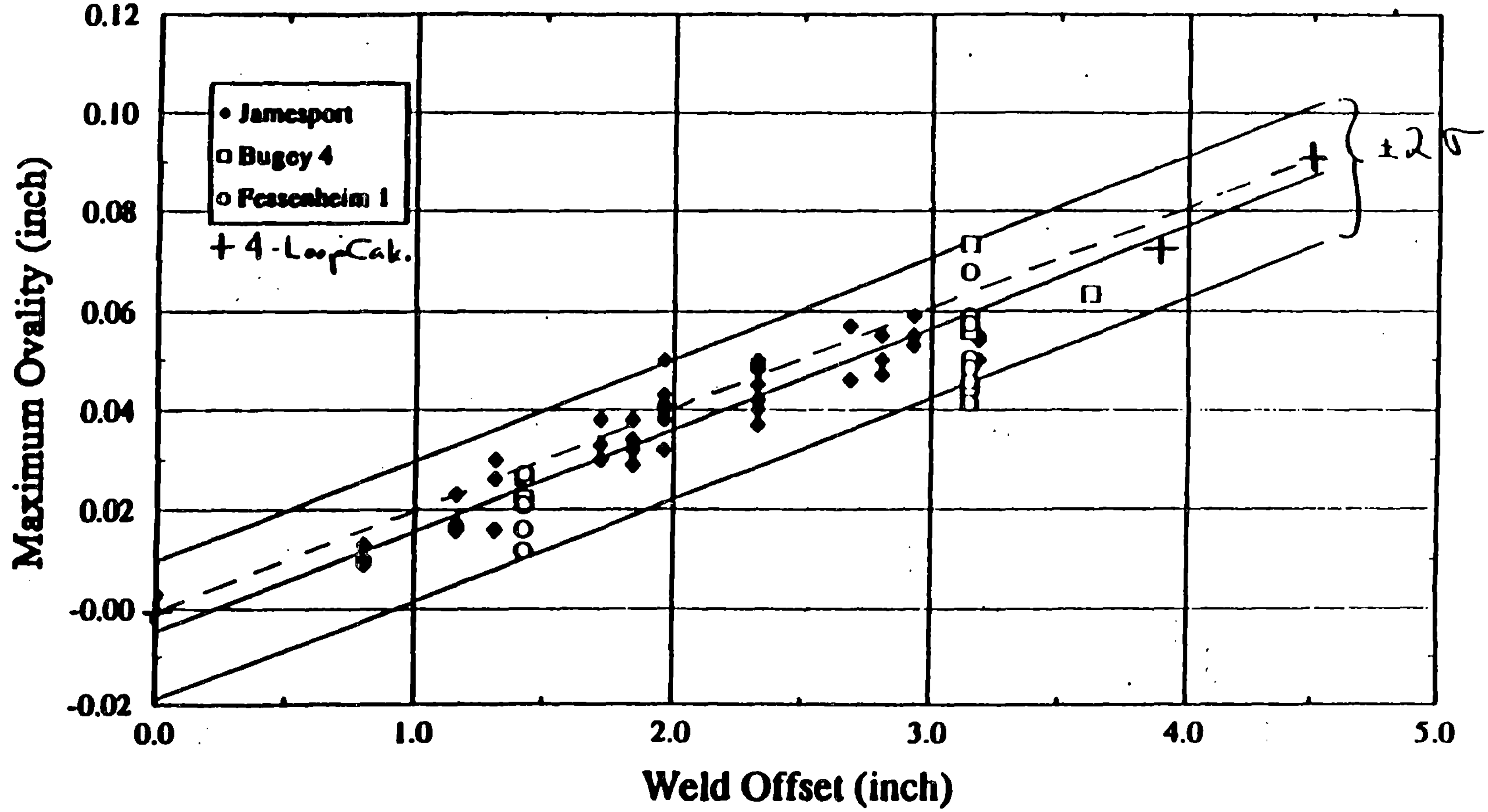

Figure 4-3 Measured Vessel Head Penetration Ovality Data and Regression Results [29]

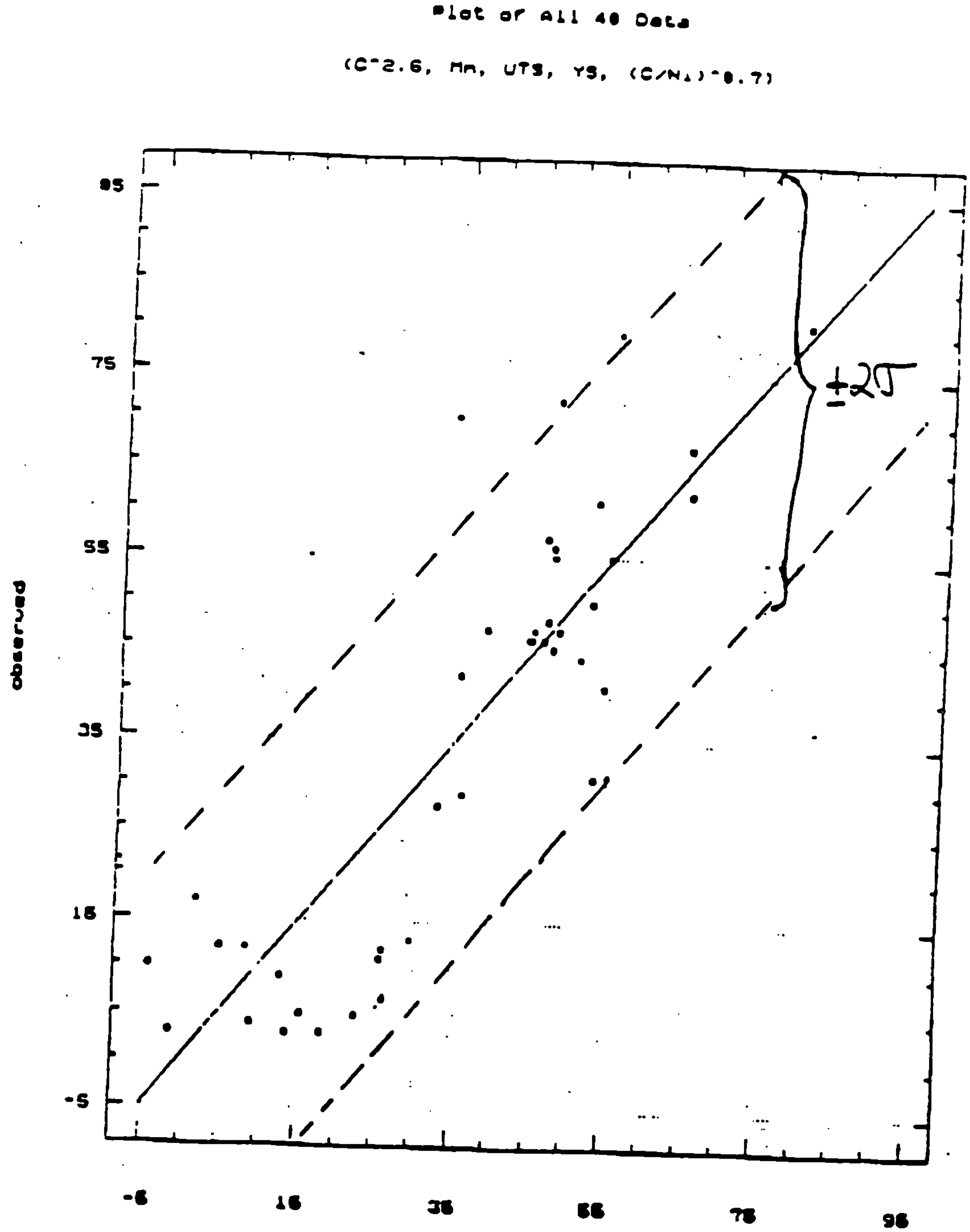

Figure 4-4 Curve Fit of Alloy 600 Grain Boundary Carbide Coverage Results [31]

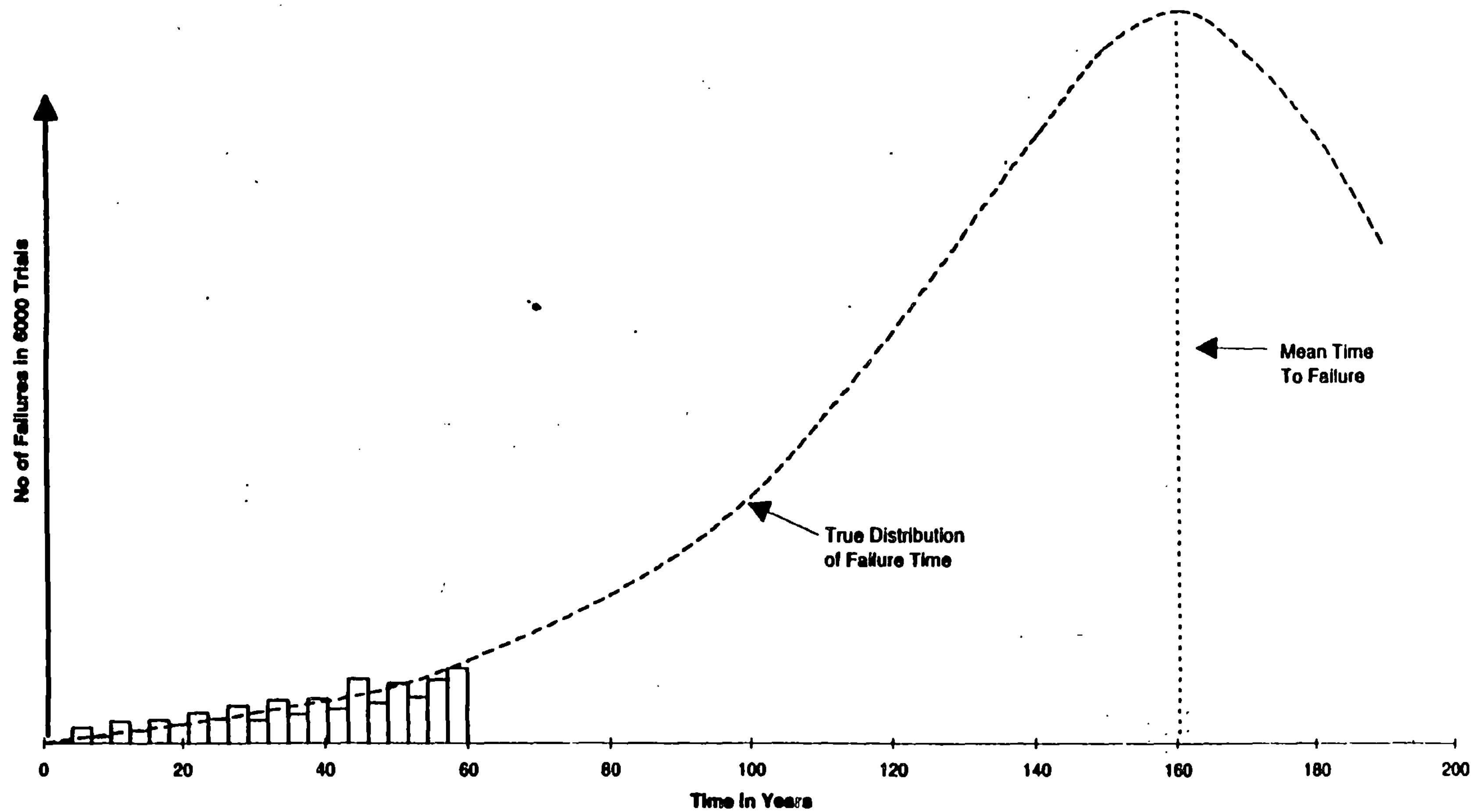

Figure 4-5 Histogram of Failures from Monte-Carlo Simulation

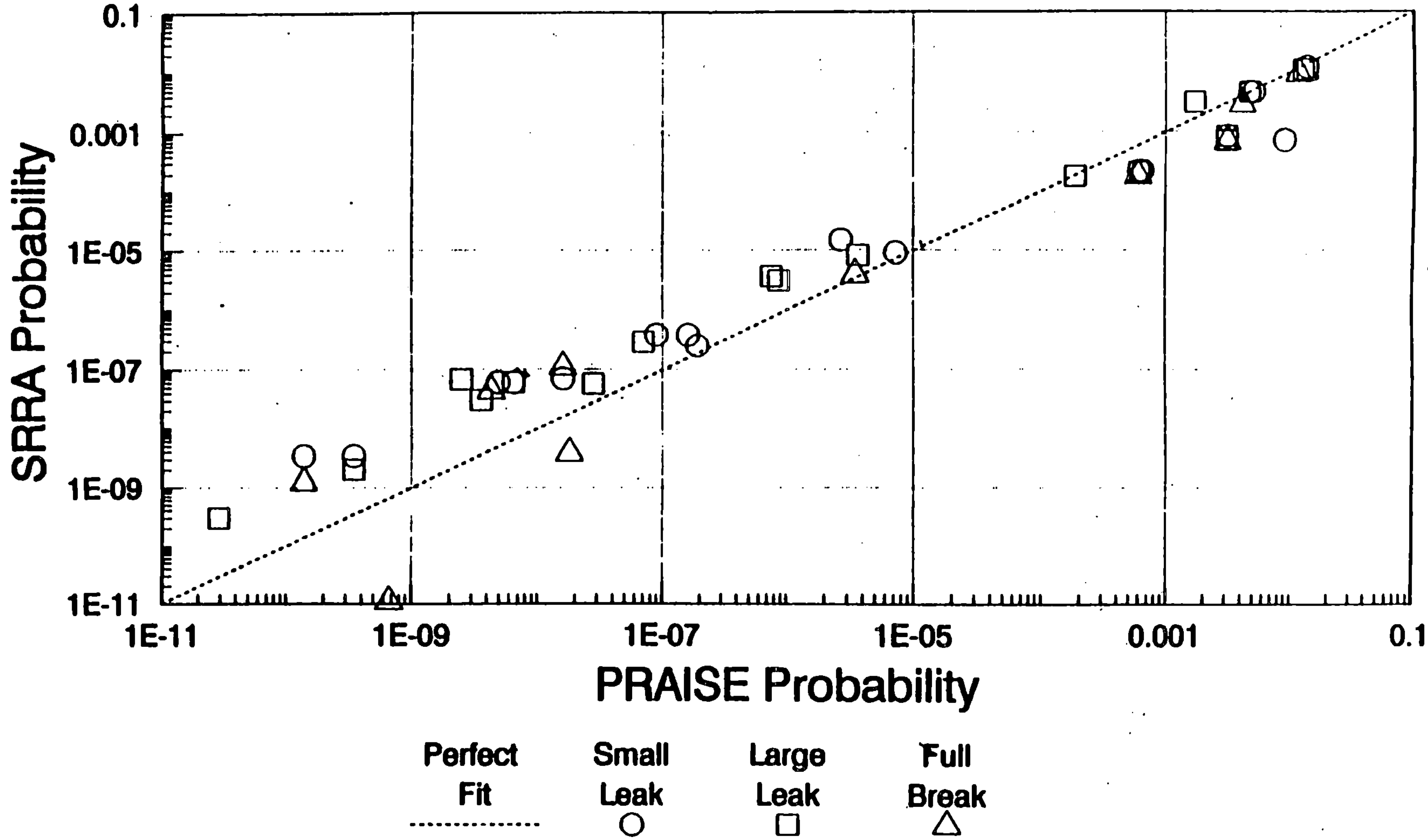

Figure 4-6 Comparison of Calculated Piping Probabilities

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