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January 25, 1982

Mr. Darrell G. Eisenhut, Director
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Dresden Station Units 2 and 3
Quad Cities Station Units 1 and 2
Safety Concerns Associated with
Pipe Breaks in the BWR Scram
System, NUREG 0803
NRC Docket Nos. 50-237/249 and
50-254/265

References (a): D. G. Eisenhut letter to All BWR Licensees Dated August 31, 1981.

(b): T. J. Rausch letter to D. G. Eisenhut dated January 19, 1982.

Dear Mr. Eisenhut:

Enclosed is the Commonwealth Edison plant specific response for Dresden Units 2 and 3 and Quad Cities Units 1 and 2 requested in the Reference (a) transmittal of NUREG 0803, "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping". This response addresses the three general areas of concern outlined in section 5 of NUREG 0803.

To the best of my knowledge and belief, the statements contained herein and in the attachment are true and correct. In some respects these statements are not based on my personal knowledge but upon information furnished by other Commonwealth Edison employees and consultants. Such information has been reviewed in accordance with Company practice and I believe it to be reliable.

Please address any questions that you may have concerning this matter to this office.

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One (1) signed original and fifty-nine (59) copies of this transmittal are provided for your use.

Very truly yours,

E. Douglas Swartz
for Thomas J. Rausch
Nuclear Licensing Administrator

cc: Region III Inspector - Dresden
Region III Inspector - Quad Cities

SUBSCRIBED and SWORN to
before me this 25th day
of January, 1982

Rosalie A. Pienta
Notary Public

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3309N

Plant Specific Response To

NUREG-0803

"Generic Safety Evaluation Report

Regarding Integrity of BWR

Scram System Piping"

Dresden Station, Units 2 & 3

and

Quad Cities Station, Units 1 & 2

Piping Integrity

Inspections have been made of the scram discharge volume (SDV) piping and supports installed at Dresden Station, Units 2 & 3 and Quad Cities Station, Units 1 & 2. These inspections have verified the integrity of the SDV piping and supports systems.

Inspections and as-built drawings of the SDV piping were performed on both units at each station in response to I.E. Bulletin 80-17, item A1. The as-built inspections performed at each station verified that all SDV piping was positively vented and pitched for proper drainage. Copies of all as-built drawings were submitted to the Nuclear Regulatory Commission (NRC) in August, 1980.

A seismic operability analysis was performed by EDS Nuclear Inc. for the SDV header support systems installed at each station in response to I.E. Bulletin 79-14. As a result of the analysis, a discrepancy was discovered on the SDV header support system installed at Dresden Station Unit 2. This discrepancy was immediately rectified and the support system now meets the NRC approved seismic operability criteria. The analyses performed on the SDV header support systems installed at Quad Cities Staton, Units 1 & 2 and Dresden Station, Unit 3 have shown that the existing support systems meet the NRC approved operability criteria. The SDV header support systems will conform to original FSAR design criteria upon completion of IE Bulletin 79-14 work.

Initial installation of the SDV piping systems found at each station included radiographs of butt welds and hydrostatic testing of each system. Documentation does exist in each station central file that all welding was performed by qualified welders, butt welds were radiographed, and that each system was hydrostatically tested to 1 1/2 times system operating pressure.

The SDV systems at both Dresden and Quad Cities Stations are classified as ISI Class 2 piping. All ISI inspections to be performed on these systems comply with those requirements outlined by the ASME Code, Section XI, subsection IWC. The ISI programs at each station are currently under review by the NRC. Recent inspections of the SDV piping at Dresden Station, Units 2 & 3 have shown that no threaded joints exist on any piping components for these systems. A threaded joint inspection was not requested for Quad Cities Station, Units 1 & 2.

Personnel at each station have reviewed all procedures applicable to maintenance of the HCU's on the SDV systems. Each station review has verified that all HCU's requiring maintenance are hydraulically isolated and electrically disarmed before any maintenance proceeds and that all maintenance must be complete before the HCU is placed back into service. In general, HCU's are only repaired when necessary with the exception of the scram pilot valves. One third of all scram pilot valves on each unit are rebuilt every operting cycle at Dresden Station and one quarter of the scram pilot valves on each unit are rebuilt every operating cycle at Quad Cities Station.

In addition to the seismic operability analysis and the as-built piping inspections that have been performed, an evaluation of the existing SDV system design at each station has been completed. It has been determined that modifications to the system design at each station will have to be made in order to satisfy the long term design criteria outlined in the NRC Generic Safety Evaluation Report (SER) regarding BWR scram discharge systems, dated December 1, 1980. The modified scram discharge volumes will incorporate dual instrument volumes with diverse and redundant instrumentation. Each SDV header will include a separate instrument volume hydraulically coupled to each header piping. Independent and dedicated vent and drain lines will be installed on each SDV header with double isolation valves on each vent and drain line to ensure primary containment isolation.

All socket, fillet, and butt welds on new SDV piping components will be examined in accordance with ANSI B31.1, paragraph 127.4.2e and paragraphs 136.5.3.d and e. All SDV header piping will be seismically supported and hydrostatically tested to 1 1/2 times the system design pressure.

Mitigation Capability and Environmental Qualifications

Having analyzed the accident scenario postulated in NUREG-0803, it is the position of Commonwealth Edison that the probability of an SDV pipe rupture resulting in a loss of coolant accident is of such small magnitude that the event is beyond the range of a credible occurrence. This position is based on the results of two independent studies that have been performed on SDV piping systems and the low probability that all scram outlet valves will fail simultaneously with the pipe break.

The first of these studies was performed by the General Electric Company and documented in report NEDO-24342. In this generic evaluation report regarding BWR scram system pipe breaks, the probability of an SDV pipe break per challenge was calculated to be less than 5×10^{-6} per reactor year.

In addition to the General Electric report, a plant specific pipe fracture analysis was performed on the SDV system piping installed at LaSalle Station, Unit 1. The analysis was performed by SAI-San Jose under the direction of D.O. Harris. A copy of the report is found in Appendix A. As a result of this study, the probability of an SDV pipe break was conservatively calculated to be 7×10^{-6} per reactor year. The SDV system piping at LaSalle Station is similar to the SDV system piping found at Dresden and Quad Cities Stations.

The SDV pipe break scenario can only result in a loss of coolant accident when the postulated pipe break occurs simultaneously with a failure of all scram outlet valves to close. Conservatively assuming that the probability of all scram outlet valves failing to close is 1×10^{-1} per reactor year, the combined events result in a probability of 10^{-7} per reactor year or less. This places the frequency of occurrence beyond the range which need to be taken into account in the design of a nuclear facility.

The results of the two independent studies combined with the positive results of the hydrostatic tests performed on the SDV piping at each station, and the low probability of all scram outlet valves failing to close, all provide sufficient evidence that the probability of an SDV pipe break resulting in fuel failure at Dresden and Quad Cities Stations is of such small magnitude that it does not merit further review. For this reason, no changes to the existing station emergency procedures will be made nor will any additional environmental qualification criteria be established for equipment located in the reactor building.

Reactor Water Specific Activity Limits

The technical specification limits on the reactor coolant system radioactivity concentrations in water are 20 uCi/ml of total iodine for Dresden Station, Units 2 & 3 and 5 uCi of I-131 dose equivalent per gram of water at Quad Cities Station, Units 1 & 2. A survey was conducted to find the high and low radioactivity concentrations measured for each unit since January 1, 1980. The average concentrations measured during the month of December, 1981 were also calculated. The results are shown in Table 1. As can be seen, the radioactivity concentrations measured are well below each station technical specification limit. Also, the reactor coolant radioactivity concentrations at each station have not exceeded the standard technical specification limit of .2 uCi/ml I-131 dose equivalent in the last 2 years.

Because of the low probability of an SDV pipe break and the fact that reactor coolant radioactivity concentrations at both Dresden and Quad Cities Stations have been well below standard technical specification limits for the last two years, no changes to the technical specifications will be proposed. The current monthly reactor coolant activity averages and the previous 2 year operating histories for each station provide sufficient evidence that the probability of operating each unit at coolant activity levels in excess of .2 uCi/ml I-131 dose equivalent will be less than 10^{-3} per reactor year. The technical specifications for Dresden Station, Unit 2 are currently under SEP review. Technical specification changes may be proposed as a result of this review.

	Tech. Spec. Limit	2-Year Coolant Activity Low	2-Year Coolant Activity High	Coolant Activity Average For December, 1981
Dresden Unit 2	20 uCi/ml total iodine	9.71×10^{-3} uCi/ml total iodine	9.71×10^{-2} uCi/ml total iodine	1.62×10^{-2} uCi/ml total iodine
Dresden Unit 3	20 uCi/ml total iodine	5.73×10^{-1} uCi/ml total iodine	1.07 uCi/ml total iodine	6.35×10^{-1} uCi/ml total iodine
Quad Cities Unit 1	5 uCi/ml I-131 equivalent	2.16×10^{-4} uCi/ml I-131 equivalent	7.64×10^{-2} uCi/ml I-131 equivalent	2.78×10^{-2} uCi/ml I-131 equivalent
Quad Cities Unit 2	5 uCi/ml I-131 equivalent	2.30×10^{-4} uCi/ml I-131 equivalent	1.79×10^{-1} uCi/ml I-131 equivalent	2.02×10^{-2} uCi/ml I-131 equivalent

Table 2

Reactor Coolant Radioactivity Concentrations

Note: The 2-year coolant activity high of 9.71×10^{-2} uCi/ml total iodine for Dresden, Unit 2 was equal to 5.26×10^{-3} uCi/ml I-131 dose equivalent.

The 2-year coolant activity high of 1.07 uCi/ml total iodine for Dresden, Unit 3 was equal to 4.84×10^{-2} uCi/ml I-131 dose equivalent.

APPENDIX A

PROBABILISTIC FRACTURE MECHANICS EVALUATION OF SCRAM SYSTEMS PIPING

FRACTURE MECHANICS ANALYSIS OF SCRAM PIPING RELIABILITY

D.O. Harris
SAI — San Jose
January 11, 1982

INTRODUCTION

In order to address concerns regarding the integrity of BWR scram system piping (1,2), and to assess the consequences of failure of such piping, it is necessary to estimate the probability of failure of various line sizes. This has been accomplished in a preliminary manner (3) using procedures developed for the reactor safety study (4). However, such procedures do not take into account specifics of the pipe design and operation that are known to influence the piping integrity. Such factors include operating stress levels, number of stress cycles, and frequency of inspection and proof testing. Additionally, no estimates are made of the reliability of pipes of diameter less than 2 inches. Much of the scram piping is 3/4 inch diameter. In order to estimate the piping reliability for a variety of pipe sizes and to account for specific operating conditions of the piping system, an analysis was performed to estimate the failure probability of the scram piping at La Salle Unit 1. The procedures employed and results obtained will be presented in the following sections.

Basically, the methodology assumes that piping failures occur due to the growth of crack-like defects introduced into welds during fabrication of the pipe. These initial defects are considered to be randomly distributed in both the number of defects and their size. The as-fabricated defect distribution is altered by pre-service inspection according to detection probabilities associated with the inspection procedures. The post-inspection defect distribution then serves as initial conditions for fracture mechanics calculations of crack growth that occurs as a result of service conditions. The probability of failure at a given weld location at a given time is equal to the probability of a crack larger than the critical crack size existing at that location and time.

Such procedures are generally referred to as "probabilistic fracture mechanics" and have been widely applied to nuclear reactor pressure vessels and piping. Reference 5 provides a comprehensive review of work in this area and also serves as an example of the current state-of-the-art. References 6 - 8 provide additional discussions in this area. Figure 1 schematically shows the various steps involved in the analysis.

The scram piping under consideration is seamless, and all welds are therefore circumferential. Interior surface part-circumferential cracks, such as shown schematically in Figure 2, are therefore the crack geometry of most concern. Complex calculations of crack growth can be performed for such cracks (5,8). The calculations can be greatly simplified if it is assumed that the crack is very much longer than it is deep ($b/a \gg 1$). In such a case the crack becomes one-dimensional and the analysis is greatly simplified. Reference 6 provides an example of a 1-D approach, which will be used in the analysis of the scram piping system.

Proof testing can be very beneficial in increasing piping reliability (5,9), because the fact that a weld joint survived a proof test indicates that no cracks larger than the critical size were present during the proof. This allows the crack-size distribution to be truncated at the critical size corresponding to the proof conditions. This will be discussed in more detail later, and can have a significant impact on the calculated failure probabilities.

REVIEW OF PIPING INPUTS

Pipe failures (leaks or complete pipe severances) that can produce appreciable leak rates and can not be isolated by valves are of concern. Leak rates due to failure of a 3/4 inch scram discharge line between the hydraulic control unit (HCU) and reactor pressure vessel will be limited to the leak rate past the control rod seal. Such failures are therefore not of concern. However, failure of such a line between the HCU and header can result in higher leak rates that are limited only by the 3/4 inch pipe diameter. Therefore, attention will be concentrated on all lines downstream of the HCU, including lines out to the first isolation valve. This results in pipe lines out to the valve in the 2 inch drain line (10) and the valve in the 1 inch header vent (11) being considered. Table 1 summarizes the piping systems considered and their corresponding sizes and wall thicknesses. Also included is the piping material and the number of welds in each line. These numbers were estimated either directly from piping drawings that indicate weld locations, or from piping layouts with a bend (elbow) comprising 2 welds, a valve 2 welds and a "tee" three welds. This will tend to overestimate the number of welds, because some lines are bent rather than having welded fittings. This procedure is therefore somewhat conservative.

The stresses in the various piping systems listed in Table 1 are required as inputs to the fracture mechanics analysis. The scram piping is subjected to a number of transient types. Only normal operating stresses will be considered here, and only the stresses due to pressure, (σ_p), deadweight, (σ_{DW}) and restraint of thermal expansion (σ_{TE}) will be considered. Maximum stress levels or loads were obtained from various references with results summarized in Table 2. The axial component of the stress due to internal pressure (σ_p) was calculated from the following expression

$$\sigma_p = \frac{p\,(ID)}{4h} \qquad (1)$$

where h is the pipe wall thickness and p is the reactor operating pressure of 1250 psi.

The deadweight (σ_{DW}) and restraint of thermal expansion stress (σ_{TE}) were evaluated from the corresponding moments by use of the following expression

$$\sigma_{(DW\ or\ TE)} = \frac{M_{(DW\ or\ TE)}\ OD/2}{I} \qquad (2)$$

The following stress components are also of interest

$$\text{load controlled stress} = \sigma_{LC} = \sigma_{DW} + \sigma_p \qquad (3)$$

$$\text{cyclic stress} = \Delta\sigma = \sigma_p + \sigma_{TE} \qquad (4)$$

$$\text{max. proof stress} = \sigma_{prf} = 1.25\ \sigma_p + \sigma_{DW} \qquad (5)$$

The 1.25 coefficient in equation 5 is due to the proof pressure being 1.25 times the operating pressure (reference 2, page 3-2).

The number of times during the plant lifetime that the pipes are subjected to the stresses shown in Table 2 is also required for the fracture mechanics analysis. This number of stress cycles will equal the number of times the reactor is scrammed. Reference 1 (page A-1) suggests a rate of 2 per year, which would result in 80 cycles during a 40 year plant lifetime. However, this estimate may be somewhat optimistic. Reference 3 (page A-2) suggests a rate of 6.1/yr. or 244 per plant lifetime, and reference 14 suggests 320 per plant lifetime. Values of 200 and 320 will be considered, with the latter value serving as an upper bound.

FRACTURE MECHANICS MODEL INPUTS

The inputs to a simplified one-dimensional crack fracture mechanics model will be summarized in this section. These inputs will be combined with information provided in the previous section to perform the fracture mechanics analysis presented in the next section.

Failure Criterion: A failure criterion is required in order to define the critical crack size for the various piping systems considered. The pipes are fabricated from SA106B carbon steel, except for the 304 stainless steel scram discharge lines (See Table 1). Both of these materials are tough and ductile, and will not fail in a brittle manner. Likewise they will not fail due to a tearing instability. Reference 5 provides a detailed discussion on this topic for 304 SS. However, a catastrophic failure can occur when a crack of sufficient size (or area) exists to reduce the remaining cross-sectional area of the pipe to the point when it is not sufficient to sustain the load controlled component of the applied stress. Hence, a net section stress failure criterion is applicable (12). This criterion was applied to 304 SS reactor piping in reference 5 and to SA106B carbon steel reactor piping in reference 3 and can be expressed as

$$(A_p - A_{cr})\,\sigma_{flo} = A_p\,\sigma_{LC} \qquad (6)$$

σ_{flo} is the critical net section stress, which is equal to (yield strength + tensile strength)/2 (12). A_p is the cross-sectional area of the pipe and A_{cr} is the critical crack area. A value of σ_{flo} of 45 ksi will be used for both materials (3,5).

If the crack is taken to be one-dimensional, it can be conservatively assumed to be complete circumferential with a depth a. Equation 6 then reduces to the following expression (6)

$$a_c = h\,(1 - \sigma_{LC}/\sigma_{flo}) \tag{7}$$

Subcritical Crack Growth Characteristics: Subcritical crack growth in reactor piping can occur due to stress corrosion cracking (SCC), fatigue crack growth or environmentally enhanced fatigue crack growth. SCC has not been observed in carbon steel lines, but has been observed in sensitized welds of 304 stainless steel (15). However, SCC is a time dependent process, and the scram discharge line is under load only for a short period (~200 hrs., see reference 2). Therefore SCC is ruled out as a significant contributor to crack growth. Environmentally enhanced fatigue crack growth then remains as the dominant contributor to subcritical crack growth. The following relations provide conservative estimates for the materials under consideration in an operating reactor environment

304 SS (ref. 16) $$\frac{da}{dn} = 10^{-9}\,(\Delta K)^4 \tag{8}$$

SA106B (ref. 17) $$\frac{da}{dn} = 1.68\text{x}10^{-9}\,(\Delta K)^{2.37} \tag{9}$$

ΔK - cyclic stress intensity factor, ksi - $\text{in}^{\frac{1}{2}}$

da/dn - crack growth rate, inches/cycle

These relations are conservative and are applicable to high mean stresses. In fact, information in reference 5 suggests that, for stainless steel, the probability that the coefficient in equation 8 exceeds the value of 10^{-9} is $3\text{x}10^{-6}$. Thus, the use of equation 8 is indeed very conservative.

Stress Intensity Factors: In keeping with the conservative use of a failure criterion for complete circumferential cracks to represent the behavior of part-circumferential cracks, the stress intensity factor relation for complete circumferential cracks subjected to uniform stress will be employed here. This relation (which is applicable for pipes with ID/H = 10) is available from reference 18 and is as follows

$$\frac{K}{\sigma a^{\frac{1}{2}}} = \frac{2+C_1\alpha+C_2\alpha^2+C_3\alpha^3+C_4\alpha^4}{(1-\alpha)^{\frac{1}{2}}} \tag{10}$$

$\alpha = a/h$

$C_1 = -1.00250$

$C_2 = 4.79463$

$C_3 = -6.21135$

$C_4 = 1.79864$

Initial Crack Distribution: The initial crack distribution consists of two components: (i) the probability of a crack existing at the weld location, and (ii) the size distribution of cracks given that a crack is present. Following the approach of reference 5, cracks will be assumed to be Poisson distributed with a crack existence frequency per unit volume of $10^{-4}/in^3$. This value is denoted as p_V^*. The probability of having a crack in a weld of volume V, which is denoted as p*, is then given as

$$p^* = 1 - e^{-V\, p_V^*} \tag{11}$$

The volume of weld, V, is taken to include a distance h on each side of the weld, and is given by

$$V \sim \pi(ID)\, h\, (2h) = 2\pi\, (ID)h^2 \tag{12}$$

The size distribution of cracks, given that a crack is present, is denoted as p_{cond}. The complementary cumulative conditional crack depth distribution will be assumed to be exponential with a parameter, λ, of 0.246 inch. This value was used in the Marshall report (7) and was employed for the marginal distribution of crack depths in reference 5. The crack size distribution must be adjusted to account for the impossibility of having a crack deeper than the wall thickness h. The following expression is obtained (5)

$$P_{cond}\,(a > x) = \begin{cases} \dfrac{e^{-x/\lambda} - e^{-h/\lambda}}{1 - e^{-h/\lambda}} & 0 \geq x \geq h \\ 0 & \text{otherwise} \end{cases} \tag{13}$$

$$\lambda = 0.246 \text{ inch}$$

This is considered to be the initial as-fabricated crack depth distribution. This distribution should be conservative for the relatively thin wall pipes under consideration, because it was estimated for reactor pressure vessels — which are much thicker.

Detection Probability: The as-fabricated crack depth distribution is modified by the detection probability of the pre-service inspection employed. The detection probability will be conservatively taken to be zero. This is equilivalent to not considering the pre-service examination, and simplifies the following analysis.

FRACTURE MECHANICS ANALYSIS AND RELIABILITY RESULTS

The input components necessary to perform the fracture mechanics analysis of piping reliability have now been presented. This section will present the procedures involved and the results obtained.

Effect of Proof Testing: The proof testing performed on the piping can have a strong influence on the calculated reliability. This is because the fact that the piping has survived the proof means that no cracks larger than the critical size corresponding to the proof conditions (a_p) existed at the time of the proof test — otherwise the pipe would have failed during the test. Hence, the crack size distribution can be truncated at a_p, which can have a marked effect on the calculated reliability. The following modification of equation 13 is then applicable

$$P_{cond}\,(a>x) = \begin{cases} \dfrac{e^{-x/\lambda} - e^{-a_p/\lambda}}{1 - e^{-h/\lambda}} & 0 \le x \le a_p \\ 0 & \text{otherwise} \end{cases} \qquad (14)$$

The value of a_p for the piping systems considered is obtainable from equation 7 with σ_{LC} taken equal to σ_{prf} (eq. 5) which is provided in Table 2. Such values will be presented along with other relevant crack sizes discussed in the next section.

Subcritical Crack Growth Calculations: The size distribution of cracks remaining after the proof test, as given in equation 14, will change during operation of the plant due to the cyclic stresses imposed. Hence, the probability of pipe failure will be time-dependent and equal to the probability of a crack larger than the critical size existing at a given time. The critical crack size is obtainable from equation 7 in conjunction with information from Table 2.

An alternative viewpoint is to consider "tolerable initial crack sizes". For a given pipe weld and time, t, this is the crack size at t = 0 that would just grow to critical size in t. Denoting this as a_{tol} (t), the probability of failure within time t is then equal to the probability of having a crack larger than a_{tol} (t) at t = 0. Hence, once a_{tol} (t) is known (which is strictly a fracture mechanics calculation) the conditional cumulative failure probability is given by

$$P_{f(cond)}(t) = P[a > a_{tol}(t)]$$
$$= \begin{cases} \dfrac{e^{-a_{tol}(t)/\lambda} - e^{-a_p/\lambda}}{1 - e^{-h/\lambda}} & 0 \leq a_{tol}(t) \leq a_p \\ 0 & \text{otherwise} \end{cases} \qquad (15)$$

The value of $a_{tol}(t)$ at t = 0 is a_c(by definition). The value of a_p will be less than a_c, because the proof stress is higher than the load controlled stress during normal operation. Therefore, the failure probability will be zero for the time it would take a crack to grow from a_p to a_c.

The value of $a_{tol}(t)$ can be calculated on a cycle-by-cycle basis by the following procedure.

$$a_{tol}(0) = a_c$$

$$a_{tol}(1\text{ cycle}) = a_c - \frac{da}{dn}\Big|_{a=a_c} = a_c - C[\Delta K|_{a=a_c}]^m$$

$$a_{tol}(2\text{ cycle}) = a_{tol}(1\text{ cycle}) - C[\Delta K|_{a_{tol}(1\text{ cycle})}]^m$$

$$\vdots \qquad (16)$$

etc.

The appropriate values of C and m follow from equations 8 and 9. K for a given σ and a is obtained by use of equation 10 with $\Delta\sigma$ obtainable from Table 2.

A conservative bound on a_{tol} after n_T cycles is obtained by assuming that da/dn remains constant at the value corresponding to a_c. This provides the following expression

$$a_{tol}^{LB}(n_T) = a_c - n_T \quad C[\Delta K|_{a=a_c}]^m \qquad (17)$$

This simple estimating procedure provides a screening of the piping systems considered. If a_{tol}^{LB} (n_T) (for n_T = total life) is greater than a_p, then failure of the piping system will not occur during n_T cycles. This allows several of the piping systems to be immediately eliminated from consideration — as shown in Table 3, which also includes other relevant fracture mechanics information. In cases where the piping system can not be immediately removed from consideration, the tolerable initial crack depth for n_T = 200 and 320 cycles can be calculated by the more involved procedure given in equation 16. The results of such calculations are also shown in Table 3 which reveals that most of the piping systems can be eliminated from consideration due to the influence of the proof test that was mentioned above. Only the scram discharge, header, and instrumentation lines remain to be considered.

Failure Probabilities: The cumulative conditional probability of failure within n_T cycles (plant lifetime) can be calculated from the a_{tol} (n_T) results in Table 3 along with a_p and equation 15. Table 4 summarizes the results of such calculations along with other closely related information. The conditional probability of failure of a given weld can be converted to a non-conditional value by multiplying by the probability of having a crack present in the weld (p*)

$$P_f(t) = p^* P_{f(cond)}(t) \qquad (18)$$

The probability of failure in the system of L welds is then obtained by conservatively assuming that all welds in the system have the same failure probabilities but are independent of one another. The following expression provides the cumulative system failure probability within time t

$$P_{f(sys)}(t) = 1 - [1-P_f(t)]^L = 1 - [1 - p^* P_{f(cond)}(t)]^L$$

$$\sim p^* L \, P_{f(cond)}(t) \qquad (19)$$

The average failure rate during the period t is obtained as

$$\overline{p}_f = P_{f(sys)}(t)/t \qquad (20)$$

The time t for the 200 (or 320) scrams is taken to be estimated plant lifetime of 40 years.

Table 4 presents the average system failure rates as estimated by the above procedures. These can be expressed as a function of pipe diameter as follows

line size, in.	average system failure rate, $Yr.^{-1}$
3/4 (scram disch. & side instr. line)	4.0×10^{-4}
2	3.0×10^{-7}
6	7.0×10^{-6}
>6	0

The values from Table 4 for $n_T = 320$ were conservatively used in the above tabulation. Combining the results for lines $\geq$ 2" results in

$$p_f (\geq 2") \sim 7 \times 10^{-6} \ Yr^{-1}$$

A pipe is considered to have failed when a through-wall crack exists. Hence, both leaks and double guillotine breaks are included in the above results. Estimates of leak rate probabilities can not be obtained from these results unless some assumption is made regarding the failure mode — such as the conservative assumption that all pipe failures are sudden and complete double guillotine failures. However, such an assumption may be overly conservative, expecially for the larger size lines. More complex and sophisticated analyses based on the procedures presented in reference 5 could be employed to discriminate between leaks and sudden complete severances. However, such refinements are felt to not be warranted at this time.

The above results are considered to be conservative for the following reasons:

- quantities of weld conservatively estimated, especially for the scram discharge line,
- influence of in-service inspection ignored,
- influence of in-service proof tests ignored, only the pre-service proof test was considered,
- stress intensity factors conservatively estimated assuming

all cracks to be very long relative to depth

- initial crack depth distribution for much thicker material utilized,
- upper bound estimates on fatigue crack growth characteristics employed,
- conservative estimate of flow stress used,
- all welds in a piping system assumed to have stresses equal to those for the highest stress joint in the system.

The failure probability for the scram system piping was estimated in NEDO-24342 (Appendix B, reference 3) to be

$$\bar{p}_f\ (\geq 2") = 3.0 \times 10^{-4}/\text{Yr}.$$

There is a 1 1/2 order of magnitude difference between this value, and the estimate derived above. The results from reference 3 were obtained by use of estimates from the reactor safety study (4) which were averaged out over many piping systems and plants. Hence, they do not reflect the proof testing and inspection schedules used in the scram piping. Additionally, the stresses summarized in Table 2 are far below code allowables, which would tend to reduce the failure probabilities relative to lines designed more closely to code limits. Another factor is the relatively small number of stress cycles imposed on the scram piping and the small time spent under load. Therefore, the values obtained above are felt to be reasonable and representative for the relatively low stressed limited length runs of piping employed in the scram system.

SUMMARY AND CONCLUSIONS

A fracture mechanics analysis of scram piping reliability was performed in order to assess concerns regarding the integrity of such piping under operating reactor conditions. The fracture mechanics analysis of tolerable crack-sizes was combined with estimates of the initial crack size distribution to obtain estimates of the probability of failure due to the subcritical and catastrophic growth of crack-like defects introduced during fabrication. This mode of failure is believed to be the dominant one for the pipes considered. Environmentally enhanced fatigue crack growth was considered to be the mode of subcritical crack growth. Stress corrosion cracking (SCC) was ruled out for the carbon steel lines because such a crack growth mechanism has not been observed in this material. Additionally, SCC was not considered for the 304 stainless steel lines because of the small time they spend under stress. The influence of a pre-service proof test was found to be large, and eliminated a number of the piping systems from consideration for failure. Fracture mechanics calculations of the remaining lines led to the following estimates for the average failure rate for different size lines in the scram piping.

3/4 inch	$\overline{p}_f = 4.0 \times 10^{-4}$/Yr
$\geq$ 2 inches	$\overline{p}_f = 7.0 \times 10^{-6}$/Yr

These values are believed to be conservative for reasons detailed in earlier sections. Comparison of the results for lines 2 inches in diameter and above with earlier estimates reveal the above value to be 1 1/2 orders of magnitude lower than the earlier values. This is felt to be reasonable because the values obtained herein reflect the beneficial influence of the proof testing employed in these pipes and their relatively benign stress history.

REFERENCES

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TABLE 1

SUMMARY OF PIPING SYSTEMS CONSIDERED

Line Name	Nom. Diam	Sched	ID, in	Wall Thickness, in	Number of Welds N(90°)	S(270°)	E	Matl.	Source, Ref.
Scram Discharge	3/4	80	0.742	0.154	8 x 185	8 x 185	--	304SS	1, p. 16
Header	8	80	7.625	0.500	37	40	--	SA106	12
Instrument Volume	12	80	11.376	0.687	2	2	--	SA106	13
Header Vent	1	80	0.957	0.179	70	52	70	SA106	11
Drain	2	160	1.689	0.343	7	32	--	SA106	10
<u>Instrumentation Lines</u>									
Top	2	160	1.689	0.343	11	11	--	SA106	13
Side	3/4	160	0.614	0.218	72	72	--	SA106	13
Sensors	1/2	160	0.466	0.187	25	25	--	SA106	13

TABLE 2

SUMMARY OF STRESSES IN PIPING SYSTEMS
(all stresses in ksi)

Line Name	σ_p	σ_{LC}	Sustained + TE	Source	σ_{TE}	σ_{prf}	$\Delta\sigma$
Scram Discharge	1.51	5.24[1]	28.88	11, Table 8-2	---	5.62	25.15
Header	4.77	6.33[1]	29.30	11, Table 9	---	7.52	27.74
Instrument Volume	5.18	∿5.18[2]	---	---	7.51[5]	6.48	12.69
Header Vent	1.67	6.28	18.58	11, Table 10	---	6.70	13.97
Drain	1.54	∿3.0 [3]	---	---	7.14[6]	3.39	8.68
<u>Instrumentation Lines</u>							
Top	1.54	5.30	22.98[4]	11, Table 10	---	5.69	19.22
Side	0.92	5.30[4]	22.98[4]	---	---	5.53	18.68
Sensors	0.78	0.78[2]	∿0.78	---	---	low	low

(1) Stress due to sustained load.
(2) Assumed $\sigma_{DW} \sim 0$.
(3) σ_{DW} assumed ∿1.5 ksi.
(4) Assumed same as "top".
(5) Calculated by assuming bending moment same as for header.
(6) Calculated from bending moment due to deadweight + thermal expansion 1 from Reference 10.

TABLE 3

SUMMARY OF CRACK SIZES

Line Name	a_c, in	a_p, in	$\Delta K\|_{a=a_c}$ ksi-in$^{\frac{1}{2}}$	a_{tol}^{LB} (n_T) n_T=200	320	eliminate from consideration[(1)] 200	320	a_{tol} (n_T) 200	320
Scram Discharge	0.1361	0.1348	45.4	< 0	< 0	No	No	0.0643	0.0558
Header	0.4297	0.4167	83.3	0.4177	0.4105	Yes	No	----	0.4131
Instrument Volume	0.6079	0.5881	48.6	0.6046	0.6026	Yes	Yes	----	----
Header Vent	0.1540	0.1523	25.2	0.1533	0.1529	Yes	Yes	----	----
Drain	0.3201	0.3172	24.6	0.3194	0.3190	Yes	Yes	----	----
Instrumentation Lines									
Top	0.3026	0.2996	51.5	0.2988	0.2965	No	No	(2)	(2)
Side	0.1923	0.1912	39.7	0.1902	0.1890	No	No	(2)	(2)
Sensor line eliminated due to very low stresses (see Table 2)									

(1) Eliminate from consideration if a_{tol}^{LB} (n_T) > a_p.
(2) Conservatively taken equal to lower bound values $\left[a_{tol}^{LB}\ (n_T)\right]$.

TABLE 4

SUMMARY OF RESULTS FOR FAILURE PROBABILITIES

Line Name	Weld Volume (eq. 12) in^3	p* (eq. 11)	Qty. of Welds (Table 1)	$P_{f(cond)}(n_T)$ (1)		System $P_f(n_T)$ (2)		$\overline{p}_f$, yr^{-1}	
				n_T=200	320	200	320	200	320
Scram Discharge	0.1106	1.11×10^{-5}	2960	0.412	0.471	0.0135	0.0154	3.36×10^{-4}	3.84×10^{-4}
Header	11.98	1.20×10^{-3}	77	---	3.12×10^{-3}	---	2.88×10^{-4}	---	7.21×10^{-6}
Top Instr. Line	1.248	1.25×10^{-4}	22	1.28×10^{-3}	4.99×10^{-3}	3.52×10^{-6}	1.37×10^{-5}	8.80×10^{-8}	3.43×10^{-7}
Side Instr. Line	0.1833	1.83×10^{-5}	144	3.52×10^{-3}	7.76×10^{-3}	9.28×10^{-6}	2.04×10^{-5}	2.32×10^{-7}	5.12×10^{-7}

(1) For single joint, from equation 15.
(2) From equation 19.

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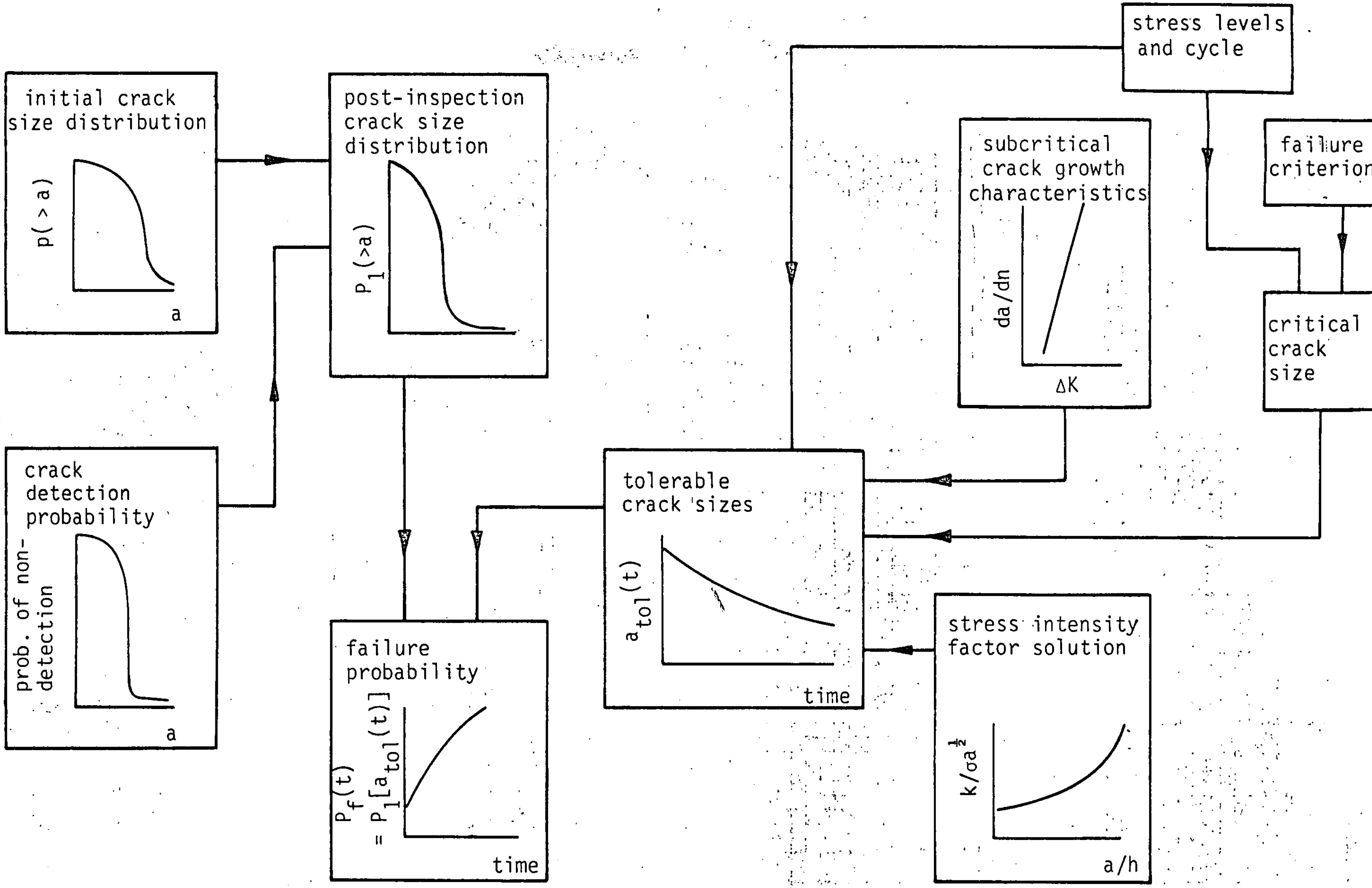

Figure 1: Schematic Representation of Various Components of Analysis of Probability of Failure of a Single Weld Joint.

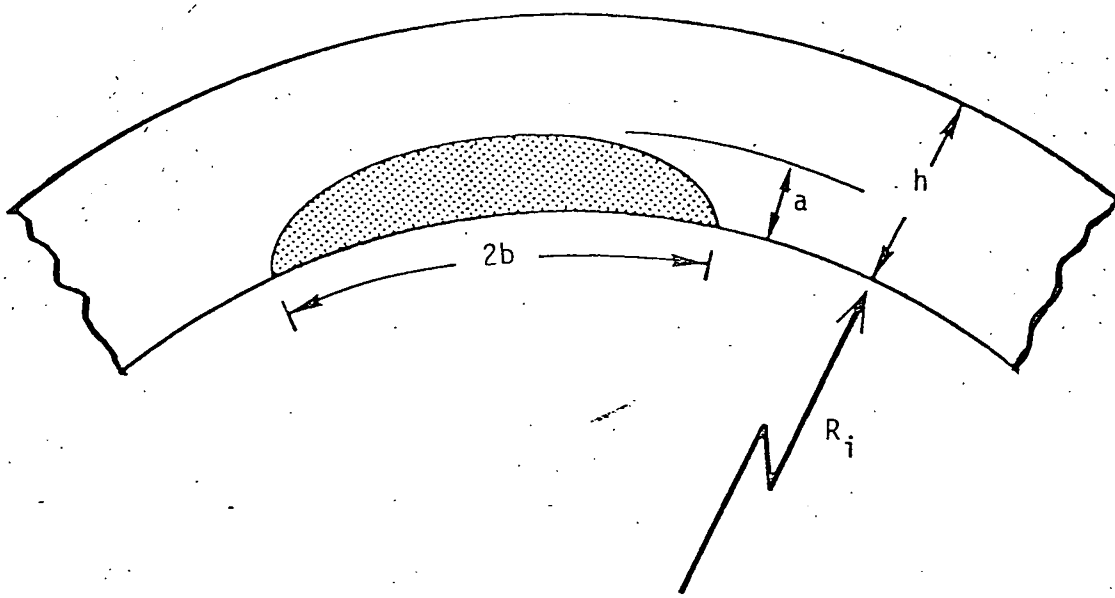

Figure 2 Geometry of Part-Circumferential Internal Surface Crack Considered in this Investigation.