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July 22, 2003

U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Document Control Desk

Subject: Oconee Nuclear Station
Docket Numbers 50-269, 270, and 287
Supplement to License Amendment Request associated with the Passive Low Pressure Injection Cross Connect Modification
Technical Specification Change (TSC) Number 2003-02

In a submittal dated March 20, 2003 Duke proposed to amend Appendix A, Technical Specifications, for Facility Operating Licenses DPR-38, DPR-47 and DPR-55 for Oconee Nuclear Station, Units 1, 2, and 3 to support installation of a passive LPI Cross Connect inside containment. The proposed License Amendment Request (LAR) revises the licensing basis associated with a selected portion of the Core Flood (CF) and Low Pressure Injection (LPI)/Decay Heat Removal (DHR) piping to allow the exclusion of dynamic effects associated with postulated pipe rupture of that piping by application of leak-before-break (LBB) technology for Oconee Unit 1. The proposed LAR also revises the licensing basis for selected portions of the LPI/DHR piping to adopt Standard Review Plan (SRP), Section 3.6.2, Branch Technical Position (BTP) MEB 3-1 design requirements. The proposed LAR adds Technical Specification (TS) requirements for the passive LPI cross connect and eliminates Technical Specification requirements associated with the capability to cross connect the trains outside containment by manual operator action.

On April 23, May 9, May 15, May 20, June 10 and June 19, 2003, Duke received additional questions from the NRC

A001

related to the LPI Cross Connect LAR. A common understanding of the questions and required responses were obtained by electronic mail and telephone conversations. Attachment 1 documents Duke's response to the additional questions.

Attachment 2 provides a corrected UFSAR retyped page. One of the UFSAR retyped pages in Duke's March 20, 2003, submittal had inadvertently designated two valves as Unit 1 valves implying that Standard Review Plan 3.6.2 BTP MEB 3-1 guidelines only applied to Unit 1 rather than generically applying to Units 1, 2, & 3. As indicated, these guidelines will apply to each Unit after implementation of the modification on the respective unit.

Pursuant to 10 CFR 50.91, a copy of this proposed license amendment is being sent to the State of South Carolina.

A 90-day implementation period for the Technical Specification change is requested. If there are any questions regarding this submittal, please contact Boyd Shingleton at (864) 885-4716.

Very truly yours,

R. A. Jones, Vice President
Oconee Nuclear Site

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R. A. Jones, being duly sworn, states that he is Vice President, Oconee Nuclear Site, Duke Energy Corporation, that he is authorized on the part of said Company to sign and file with the U. S. Nuclear Regulatory Commission this revision to the Renewed Facility Operating License Nos. DPR-38, DPR-47, DPR-55; and that all the statements and matters set forth herein are true and correct to the best of his knowledge.

R. A. Jones, Vice President
Oconee Nuclear Site

Subscribed and sworn to before me this 22nd day of July, 2003

Notary Public

My Commission Expires:

6/12/2003

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ATTACHMENT 1

RAI's FOR OCONEE AMENDMENT REQUEST

Attachment 1
RAI's for Oconee Amendment Request

RAI-1 Attachment 3 of the March 20, 2003, submittal contains a proposed revision to UFSAR Section 3.6.1.2.1 which adopts Standard Review Plan Section 3.6.2 Branch Technical Position (BTP) MEB 3-1 for the treatment of pipe breaks for the Core Flood (CF)/Low Pressure Injection (LPI) system inside containment. Provide a comparison of the current pipe break requirements at Oconee with the requirements in BTP MEB 3-1.

RESPONSE Various small break loss of coolant accidents (SBLOCA) are described in the Oconee USFAR Section 15.14.4.3. The description includes a .44 ft^2 Core Flood line break. Each Core Flood train, including piping to the associated Core Flood Tanks, and the Reactor Vessel / Core Flood nozzles are located in separate cavities on opposite sides of the Reactor Building. Thus previous to the Cross Connect Modification, a break in one Core Flood train could not interact dynamically with the other train. The remaining intact train was credited with replenishing inventory. The design basis was then that the two redundant Core Flood trains were sufficiently separated to prevent interaction.

The new cross connection structurally links the two redundant Core Flood trains. Since the previous design basis did not specify the location of the breaks, a strategically located break in one train could structurally affect the other train. This was recognized and formed the rationale for requesting the use of MEB 3-1 to eliminate break locations based on stress levels in the Core Flood system.

RAI-2 Attachment 5 of the March 20, 2003, submittal provides the technical justification for adopting BTP MEB 3-1 provisions for postulating pipe breaks for the CF/LPI system. The submittal indicates that piping upstream of valves LP-47 and LP-48 qualify as moderate energy lines per MEB 3-1, B.2.e, footnote 5. MEB 3-1, B.2.e allows postulation of leakage cracks instead of pipe breaks in the piping of systems that qualify as high energy fluid systems for only a short operational period, but qualify as moderate energy fluid systems for the major operational period. Describe the operating conditions under which these sections of piping qualify as high energy.

RESPONSE The piping upstream of LP-47 and LP-48 is only postulated to be used at high energy conditions during decay removal operation of the Low Pressure Injection (LPI) system during cooldown and heatup of the Reactor Coolant System (RCS). The Low Pressure Injection system is aligned to the RCS at or below 315 psig for Unit 1 and 2 and 310 psig for Unit 3. LPI is aligned in the decay heat removal mode when RCS temperature is less than 246°F.

From operating data, the LPI system operates in high energy conditions for approximately 45 hours during plant shutdown. The LPI pumps are started in a controlled manner with the LPI pump discharge valves closed and LPI pump suction pressure at ~305 psig. The LPI pumps at no flow conditions typically operate below 190 psid developed head. Therefore, the pressure in piping downstream of the LPI pumps and upstream of LP-12/-14 (i.e. upstream of LP-47/-48) during initial LPI pump start is ~495 psig (305+190). The LPI pump discharge valve is almost immediately opened to supply DHR flow which corresponds to an operating point on the pump below 160 psid developed head. Thus, during the majority of this shutdown sequence, the piping downstream of the LPI pumps and upstream of LP-47/-48 experiences ~465 psig (305+160). From actual data, the suction temperature typically ranges from 220°F to 240°F. The duration of the shutdown evolution involving LPI system high energy conditions is affected by operation of the Reactor Coolant Pumps (RCPs) during chemistry cleanup (i.e. crudburst). Once these chemistry actions are completed, the RCPs are secured and the RCS pressure and temperature are reduced to levels below which the LPI system would experience high energy conditions (i.e. LPI pressure ~225 psig and temperature ~85°F) until startup of the unit.

During the startup of the unit, RCS pressure must be increased (by increasing temperature) for acceptable Reactor Coolant Pump operation. From operating data, the LPI system experiences high energy conditions for approximately 35 hours during startup. The LPI system is operated to RCS pressures (i.e. LPI pump suction pressure) of approximately 305 psig. The LPI pumps are operated at flow conditions which correspond to an operating point below 160 psid. Therefore, the piping downstream of the LPI pumps and upstream of LP-47/-48 experiences ~465 psig (305+160). From actual data, the suction temperature typically ranges from 180°F to 200°F prior to the LPI system being secured during startup.

Therefore, the piping downstream of the LPI pumps and upstream of LP-47/-48 experiences pressures between 465-495 psig and temperatures between 180-240°F during startup and shutdown of the unit for refueling outages. From historic data the LPI system experiences high energy conditions a total of approximately 80 hours during a refueling outage (i.e. every 18 months).

RAI-3 Attachment 5 of the March 20, 2003, submittal indicates that the piping between valves LP-176 and LP-48 and between valves LP-177 and LP-47 and the crossover piping is classified as high energy. Provide a comparison of the highest calculated stresses in these piping segments with the criteria for postulating pipe breaks and pipe cracks specified in BTP MEB 3-1.

RESPONSE The maximum calculated stress due to dead weight, longitudinal pressure, operational basis earthquake, and thermal expansion is 14,898 psi.

The material at this location is ASME SA-376 TP304. Per USAS B31.1, 1967 Edition, the value of S_h at the operating temperature of 125°F is approximately 18,200 psi and the value of S_a is approximately 27,987 psi. The crack threshold would then be 18,475 psi. The calculated stress of 14,898 psi is less than the crack threshold, and thus no cracks or breaks are postulated in this segment of piping.

RAI-4 Attachment 5 of the March 20, 2003, submittal indicates that the stress analysis model of the piping system includes piping upstream and downstream of valves LP-47 and LP-48. The submittal also indicates that valves LP-47 and LP-48 form the boundary between the high and moderate energy portions of the piping. The submittal cites footnote 3 of MEB 3-1 as justification for not considering the valves terminal ends for the purpose of postulating breaks. However, footnote 3 of MEB 3-1 contains the following statement: "In piping runs which are maintained pressurized during normal plant conditions for only a portion of the run (i.e., up to the first normally closed valve) a terminal end of such runs is the piping connection to this closed valve." On the basis of the previous quote from footnote 3, provide additional justification why valves LP-47 and LP-48 should not be considered as terminal ends for the purpose of postulating pipe breaks.

RESPONSE BTP MEB 3-1, B.1.c.1)(a) footnote 3 defines the meaning of a terminal end, and states:

"Extremities of piping runs that connect to structures, components (e.g., vessels, pumps, valves), or pipe anchors that act as rigid constraints to piping motion and thermal expansion. A branch connection to a main piping run is a terminal end of the branch run, except where the branch run is classified as part of a main run in the stress analysis and is shown to have a significant effect on the main run behavior. In piping runs which are maintained pressurized during normal plant conditions for only a portion of the run (i.e., up to the first normally closed valve), a terminal end of such runs is the piping connection to this closed valve."

The passage of interest is the statement that the branch side of a connection is a terminal end unless it is classified as part of the main run in the stress analysis and is shown to have a significant effect on the main run behavior. Applying that rationale to a closed valve that represents the boundary between the high and moderate energy portions of a piping system would lead one to conclude that if such a valve were classified as part of the main run in the stress analysis and shown to have a significant effect on the main run behavior, then the valve would not represent a terminal end. In this instance the valves LP-47 and LP-48 are a part of the main run in the stress analysis and the valves do have a significant effect on the main run behavior. The appropriate design parameters are applied such that the lower pressure is applied to the moderate energy portion, and the higher pressure is applied to the high energy portion. The valves are not independently supported. When these facts are considered, Duke concludes that these valves do not represent a terminal end.

Other licensees have reached the same conclusion. Two examples are included below:

Florida Power Corporation (now Progress Energy) submitted a revised pipe rupture analysis criteria for Crystal River Unit 3 by letter dated March 31, 1989 and later revised by letter dated December 18, 1989. Page 7 of the pipe rupture analysis criteria report defines a terminal end as: "Extremities of piping runs that connect structures, large components (e.g., vessels, pumps) or pipe anchors that act as essentially rigid constraints to piping thermal expansion including rotational movement from static or dynamic loading. In line fittings such as valves, adequately modeled and not anchored in the piping stress analysis, are not terminal ends." NRC accepted the new licensing basis by letter dated April 11, 1990.

In Tennessee Valley Authority's Watts Bar FSAR 3.6.A.2, "Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping," Subsection 3.6.A.2.1.2.3, "High/Moderate Energy Interfaces," reads as follows: "Line supported valves sometimes form the interface between high energy lines and moderate energy lines. In this case, the fixity as implied in the word, 'terminal,' does not exist at the line supported valve. This condition is treated as if there were no terminal (end)."

RAI-5 The March 20, 2003, submittal indicates that rupture restraints will be installed at Oconee Unit 1 to protect against postulated breaks at the CF reactor vessel nozzle. Describe the criteria used to design the rupture restraints, including the criteria used to develop the break loads.

RESPONSE The High Energy Line Break Loads were developed per the methods of ANSI/ANS 58.2 (1988). Alternate checking criteria were taken from NUREG/CR-2913 (1983). The rupture restraints for each Core Flood / RV nozzle consist of a primary restraint and several secondary restraints. The primary restraint was designed to absorb the principal lateral rupture load of 235.3 kips. The secondary restraints were designed for stability of the piping system in the aftermath of the rupture. All restraints were designed to faulted allowables for the various structural steel components, anchor bolts and structural welds. For vendor supplied items, ASME Level D allowables were used for qualification.

The NRC mentioned on June 9, 2003, that Duke referenced ANSI/ANS 58.2 (1988) for break loads. The NRC reviewer was only interested in the pipe reaction forces used as input to the whip restraint design and indicated that it would be better if Duke could make a simple statement that the pipe reaction loads were developed in accordance with SRP 3.6.2.

Since Oconee is not a SRP plant, the resulting pipe reaction forces were developed based on ANSI/ANS 58.2 (1988) methods. However, Duke performed a supplemental review of the SRP 3.6.2 requirements in paragraph III.2.c and confirmed that the methods used from ANSI/ANS 58.2 (1988) meet the requirements of paragraph III.2.c of SRP 3.6.2 (July 1981).

RAI-6 Figure 3-1 is meant to show the Ramberg-Osgood stress-strain equation fit to the experimental data of Reference 11, but it shows, instead, the fit of a J-R curve. Provide a revised Figure 3-1.

RESPONSE The LBB topical report has been revised to include the revised figure (See attached Figure 1).

RAI-7 The Ramberg-Osgood parameters for the same material from different sources may be quite different. For instance, you reported (α, n) of (8.0, 3.5) for 304 stainless steel (SS) base metal and (α, n) of (0.565, 8.28) for 316 stainless steel base metal at 550°F. However, the Ramberg-Osgood parameters are reported to be (5.98, 4.29) for 304 SS base material in EPRI NP-3596-SR, Revision 1, "PICEP: Pipe Crack Evaluation Program (Revision 1)," and (6.9, 4.8 or 5.8, 3.6) for 316 SS base material in an NRC safety evaluation, "Staff Review of the Submittal by Rochester Gas and Electric Company to Apply Leak-Before-Break Status to Portions (of) the R.E. Ginna Nuclear Power Plant Residual Heat Removal System Piping [TAC NO.: MA0389]," dated February 4, 1999. Assess the impact to margins on flaw size due to uncertainty of (α, n) at such a magnitude. The assessment should include all materials in Table 7-1 of your submittal.

RESPONSE The Ramberg-Osgood (R-O) parameters listed in Tables 3-5 and 3-6 of the LBB topical report are the values used in the flaw stability analysis. It should be noted that the leak rate analysis, summarized in the topical report, utilized α, and n values of 3.46 and 5.68, respectively for the 304 stainless steel (SS) material. For the 316 SS, because the plasticity term in the crack opening displacement (COD) calculations was determined to be negligible (0.0000 in), the R-O parameters for 304 SS given above were also used for the 316 SS material for simplicity. The impact to margins on flaw sizes, as a result of considering various R-O parameters in the RAI, were determined by initially evaluating for the 10 gpm leakage crack sizes associated with each of the applicable sets of R-O parameters. This helps determine the sensitivity of the leakage crack size predictions to the selection of the R-O parameters, at piping locations 1 and 2, as discussed below.

Since the piping material at piping location 1 is 304 SS, only the R-O parameters for 304 SS (5.98, 4.29) as given in the RAI were utilized. For this location, the 10 gpm predicted leakage crack size remained unchanged (from the value reported in the topical report) at 6.43 inches.

For piping location 2, the R-O parameters for both the 304 SS (5.98, 4.29) and 316 SS materials (6.9, 4.8 or 5.8, 3.6) were considered to conservatively bound the 316 SS material at the location. As depicted in Figure 2, the predicted 10 gpm leakage crack sizes ranged from 6.83 inches to 6.99 inches. Therefore, the maximum leakage crack size of 6.99 inches (obtained using α, and n values of 6.9 and 4.8, respectively) is conservatively considered in this assessment.

The flaw stability analysis was subsequently performed considering all the R-O parameters given in RAI item 2, for 304 SS and 316 SS base metals. The flaw stability analysis for the SMAW weld was evaluated using the NRC proposed equation (given in RAI, item 8) that considers the effect of thermal aging of SS weld materials. The results of this re-assessment considering the sensitivity of the R-O parameters and the thermal aging for the SMAW weld on the circumferential flaws are shown in Table 1. These results are based on using the Zahoor's modified EPRI/GE methodology, which is the same methodology used in the topical report. The results of the sensitivity analysis can therefore be compared against the results given in Table 7-1 of the topical report. It should be noted that the J_{IC} value for GTAW weld as given in Table 7-1 has a typographical error. The value should be 3.2 kips/in and not 2.3 kips/in. The revised Table 7-1 is included in this RAI response.

The results show that the margin on flaw size at piping location 1 actually increases from 2.5 to 3.0. This may be attributed to the conservative approach of using $J_{upper\ limit}$ instead of $J_{instability}$ (See Table 1). At piping location 2, the margin on flaw size for the SMAW weld decreased (as expected) slightly from 2.8 to 2.7 while the margin on flaw size for the 316 SS base metal increased from 2.8 to 3.0.

RAI-8 In recent NRC staff's evaluation of plant-specific LBB applications, the staff have considered the effect of thermal aging of SS weld materials on J-R curves by using the following equation:

$$J(kJ/m^2) = 73.4 + 83.5\ \Delta a\ (mm)^{0.643}.$$

This position is based on the NUREG/CR-6428 finding, which indicates no difference between SAW and SMAW J-R curves. Please recalculate the margin on flaw size for the SMAW weld using the above equation.

RESPONSE As stated in response to RAI item 7, the margin on flaw size for the SMAW weld was re-calculated using the NRC provided equation that considers the effects of thermal aging for SS weld materials. The margin on flaw size for the SMAW weld decreased from 2.8 to 2.7.

RAI-9 The J-T analysis has become a little complex by your consideration of an additional parameter $J_{upper\ limit}$. Explain the need to have $J_{upper\ limit}$ included in your stability analysis.

RESPONSE In the J-T Analysis summarized in the topical report, Framatome ANP considered an additional limit on J, called $J_{upper\ limit}$, which is less than $J_{instability}$ in a J-T analysis, as an added conservatism. $J_{upper\ limit}$ is based on the J at maximum crack extension from an actual compact test specimen. This approach is eliminated when performing the new J-T analysis whose results are reported in Tables 1 and 2 and Figures 3 through 7.

RAI-10 Provide qualitative justification to demonstrate that transients such as heatups and cooldowns won't be felt at Location 2 so that a fatigue crack growth analysis is not necessary.

RESPONSE A detailed stress analysis (ASME Section III) of the core flood nozzle was reviewed to determine the effects of the transients at location 2. The only transient that is considered to be of any significance at the end of the nozzle (considered representative to piping location 2) is the check valve test transient performed during normal cooldown which has 240 cycles associated with it. The stress analysis showed that the cumulative usage factor at this location due to this transient is essentially 0.0. It is therefore concluded that this location experiences negligible pressure and thermal transient stresses so that an explicit fatigue crack growth analysis is not necessary.

RAI-11 This report proposes to use Zahoor's version of the GE/EPRI J-Integral estimation scheme in the flaw stability analysis for circumferential flaws. This approach has no theoretical basis as indicated in a discussion of this scheme in NUREG/CR-4878. Please revise your analysis by using the original GE/EPRI J-Integral estimation scheme and report the revised margins on flaw size.

RESPONSE The flaw stability analysis summarized in the topical report utilized Zahoor's modified version of the GE/EPRI J-Integral estimation scheme where the reference stress (σ_o) was assumed to be the yield stress of the material. A correction to Zahoor's modified equation is given in NUREG/CR-4878 to give unique results for unique stress-strain curve.

However, as noted in that report, the corrected equation simplifies to Zahoor's modified equation when the reference stress is chosen to be the yield stress. This is the case, for the flaw stability analysis contained in the topical report. The circumferential through wall pipe fracture data (including 304 SS data ranging from 2 to 16-inch pipe diameter) from various Industry sources were compared against the predictions using Zahoor's modified equation in EPRI NP-4883M report.

As stated in Section 2-2.2 of the EPRI report, "σ_o was selected as the 0.2% offset yield strength, but deviations of a few percent on this value also were accepted if the accuracy of the overall (R-O stress-strain curve) fit was improved." Table 2-2 of the EPRI report shows that the correction to Zahoor's modified equation provides results that are in excellent agreement (when the reference stress is selected to be at or very near the yield stress of the material) with independent finite element results or inferred initiation J values from fracture toughness tests.

However, an evaluation was performed using the original GE/EPRI J-Integral estimation scheme as recommended by the RAI. It is worth noting that when comparing the experimental loads (for crack initiation and maximum loads) to the predicted loads using various analytical methods the original GE/EPRI method was the most conservative (as stated in NUREG/CR-4878). The re-evaluation (using the original GE/EPRI method) considered all the recommended R-O parameters (sensitivity analysis), the J-R curve equation that accounts for thermal aging of stainless steel welds and the results based on $J_{instability}$ point in a J-T analysis per RAI items 7, 8, and 9, respectively. The revised margins on flaw sizes using the original GE/EPRI are given in Table 2 with the J-T analysis results illustrated in Figures 3 through 7. The tabulated results can be compared against the results of Zahoor's modified GE/EPRI method reported in Table 1.

It is noted that the predicted $J_{applied}$ is greater when utilizing the original GE/EPRI method while the $J_{instability}$ values are very comparable for both methods. As a result, the critical flaw sizes are lower when utilizing the original GE/EPRI method.

Using the original GE/EPRI method, the margins on flaw size at piping locations 1 and 2 are determined to be 2.6 and 2.2 (for the 304 SS material), respectively (see Table 2). At these piping locations, the margins, when using Zahoor's modified GE/EPRI method are 3.0 and 2.6, respectively (see Table 1). As expected, the original GE/EPRI method predicts more conservative results. However, based on this sensitivity study, it is concluded that the margins on flaw size remain greater than the required margin of 2.0 when using either method.

RAI-12 Due to the recent V.C. Summer event of primary water stress corrosion cracking (PWSCC) in the primary loop bimetallic weld, the staff has a general concern regarding PWSCC and other unidentified degradation mechanisms on proposed LBB piping. As a result, the staff requested recent LBB applicants to perform a sensitivity study using a crack morphology (surface roughness and number of turns) characteristic of transgranular stress corrosion cracks (TGSCCs). Information contained in NRC NUREG/CR-6443, "Deterministic and Probabilistic Evaluations for Uncertainty in Pipe Fracture Parameters in Leak-Before-Break and In-Service Flaw Evaluations," may be useful. Please perform this analysis for the SMAW weld at location 2 which has a margin of 2.4 on flaw sizes. The staff understands that using the suggested TGSCC crack morphology will reduce the margin significantly. The purpose is to know how much margin (10 for leakage, and 2 for flaw sizes) that the piping still has should a TGSCC occur.

RESPONSE Duke is currently performing the requested sensitivity study. However, this task, which is being performed by Framatome ANP, is not complete. Duke will provide the results of this study to NRC by August 6, 2003.

RAI-13 In Attachments 5 and 8 of your March 20, 2003 application, the capabilities of the Oconee reactor coolant system (RCS) leakage detection systems are discussed and it is stated that the systems used in the plant were reviewed to demonstrate that a 1 gallon per minute (gpm) leak can be identified within 1 hour. However, you have not provided the results of that review. Attachment 8 also states that the sensitivities of the detection system are consistent with Regulatory Guide 1.45 with respect to detecting a 1-gpm leak rate within 1 hour. Recent operating experience has shown that at some plants, the actual (or usable) sensitivities of the airborne radioactivity monitors (gaseous and particulate) are not nearly as high as they were when the plant was built. This is due to the existence of much lower activity levels in the RCS during normal operation than was assumed in the original design of the airborne radioactivity monitors.

Also, Oconee Technical Specification (TS) 3.4.15, RCS Leakage Detection Instrumentation, requires one containment normal sump level indicator, and one containment atmosphere radioactivity monitor (gaseous or particulate) to be operable. This implies that the gaseous and particulate monitors have very similar sensitivities since they are interchangeable. The recent operating experience discussed above also indicates that the gaseous monitors could be much less sensitive than the particulate monitors assuming the much lower activity levels in today's RCSs. In fact, among four units at two sites, the actual sensitivities of the gaseous monitors ranged from 200-800 hours to detect a 1-gpm leak rate.

Please provide the sensitivities of your airborne radioactivity monitors assuming the approximate RCS activity levels that usually exist during normal plant operation and discuss how both monitors can support your leak-before-break (LBB) evaluation. If one or the other, or both, cannot support the LBB evaluation, provide assurance that adequate leak detection capability will remain available.

RESPONSE The airborne radioactivity detector sensitivities (MDC) are as follows:

Gaseous (XE-133)	Particulate (CS-137)
5.5E-7 μCi/cc @ 2.5 mR/hr	7E-12 μCi/cc@ 2.5 mR/hr
9.8E-7 μCi/cc @ 5 mR/hr	1E-11 μCi/cc@ 5 mR/hr
9.8E-5 μCi/cc @ 500 mR/hr	7.8E-10 μCi/cc@ 500 mR/hr

Since Duke's May 1, 2003 meeting with NRC, Duke has performed a more thorough evaluation of airborne radioactivity monitor leak detection capability. This evaluation established with good confidence that the particulate monitor is capable of identifying a 1 gpm RCS leak in less than one hour. The most conservative particulate RCS radioactivity levels (Unit 3 RCS) during normal operations were assumed. We had previously indicated a capability of 1 gpm in 72 hours. However, this was based on a preliminary evaluation that extrapolated empirical data from a 1991 RCS leak. Several conservative assumptions were made during the evaluation involving empirical data, including the size of the 1991 RCS leak, the ratio of particulate concentrations, and the distribution of activity in containment. The evaluation also determined that the gaseous monitor is not capable of identifying 1 gpm in 1 hour. Gaseous monitor sensitivity compares to those NRC described for other plants in the RAI above (1 gpm in 200 - 800 hrs). Duke is evaluating the feasibility of a setpoint change that would improve the current gaseous monitor sensitivity by an order of magnitude.

The RB normal sump level monitor also monitors for RCS leak detection. This monitor is capable of identifying a 1 gpm RCS leak in less than 10 minutes. Duke also monitors RCS leakage by observing RCS makeup flow and Letdown Storage Tank Level. An RCS leakage calculation is performed every 24 hours.

Based on the leak detection methods and capability described above, Duke believes adequate leak detection capability is available to support this LBB application.

Figure 1. True Stress - True Strain for Type 304SS at 75°F

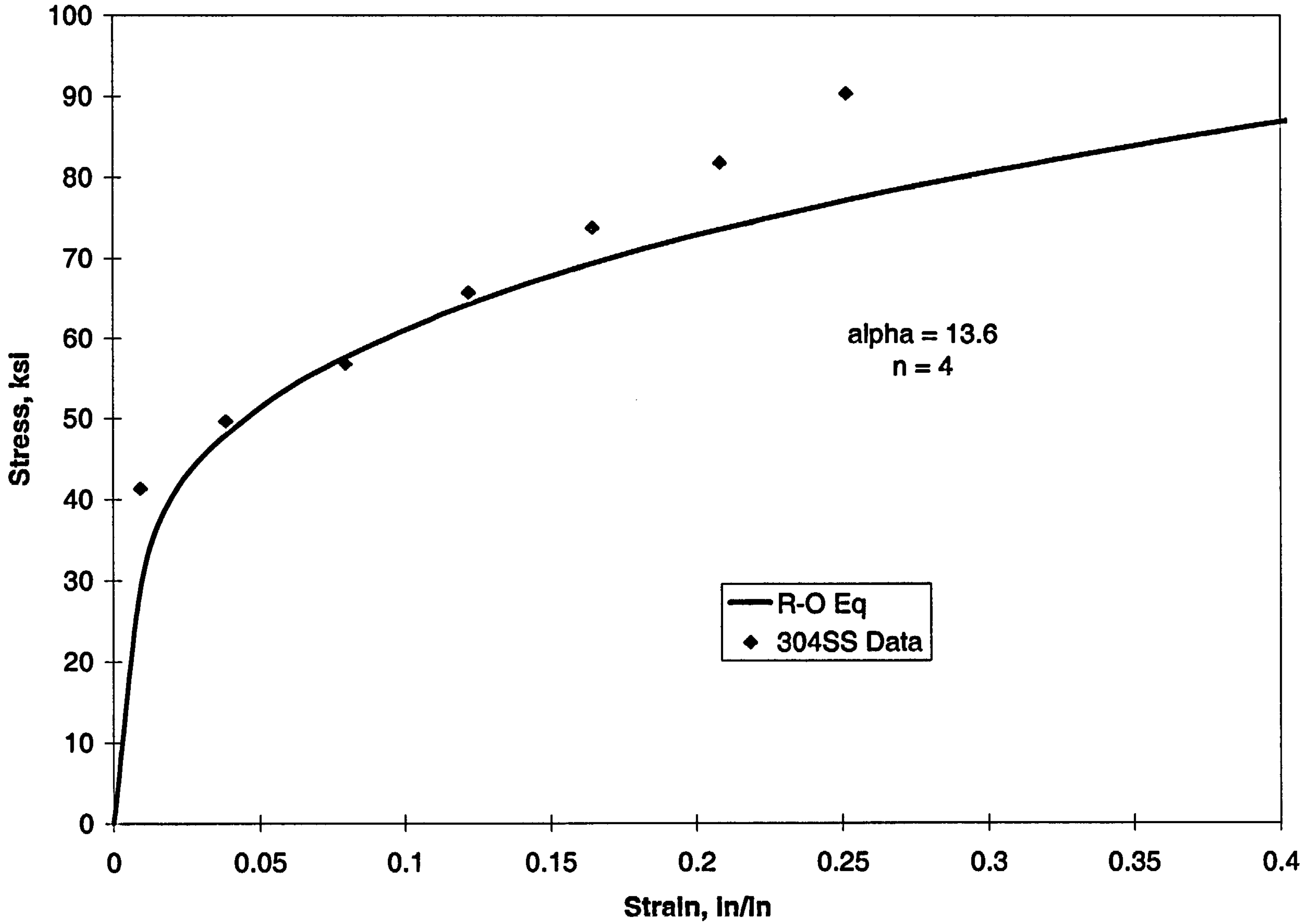


Figure 2. Leakage Size Cracks Considering Various Ramberg-Osgood Parameters at Piping Location 2

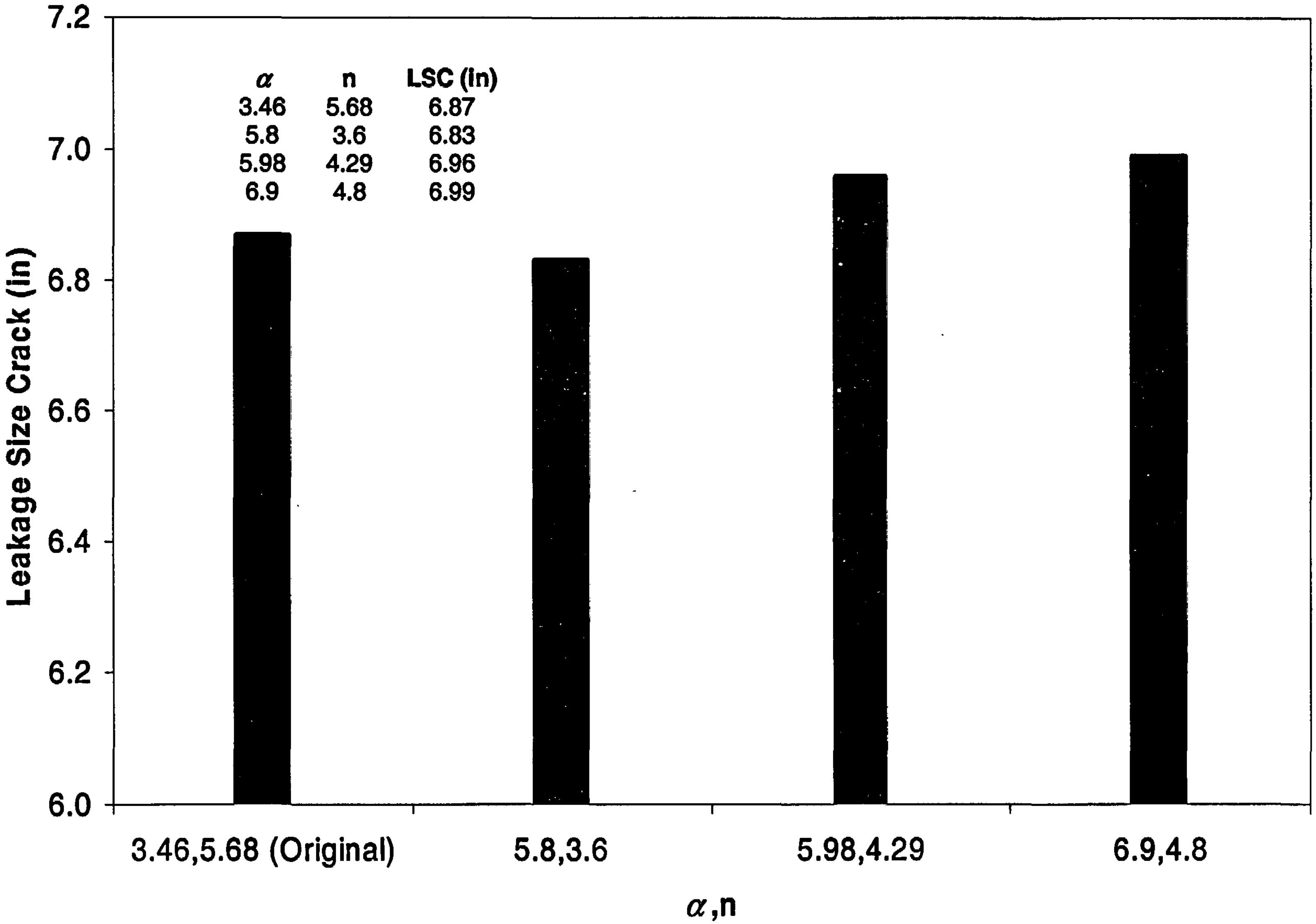


Table 7-1. Summary of Results for Circumferential Flaws

Piping Location[1]	Material[2]	Leakage Flaw Size[3], a (in)	$J_{applied}$[4] (kips/in)	Material J_{IC}[5] (kips/in)	Instability Criteria $J_{U.L}$[6] (kips/in)	Critical Flaw Size[7] a_c (in)	Margin on Flaw Size[8]
1	Type 304SS Base	3.21	2.834	5.04	10.	8.00	2.5
	GTAW Weld	3.21	1.203	3.2	10.	9.70	3.0
2	SMAW Weld	3.435	0.286	0.994	2.5	9.74	2.8
	Type 316SS Base	3.435	0.304	4.05	10.0	9.52	2.8
	Type 304SS Base	3.435	1.071	4.58	7.	8.87	2.6

1 Location 1 refers to core flood piping attached to bottom of the core flood tank nozzle.
Location 2 refers to reactor vessel core flood nozzle safe-end to pipe juncture.

2 General description of base metal or weld metal considered in the analysis

3 Corresponds to one-half of the leakage crack length as predicted by KRAKFLO in Reference 19.

4 Due to applied moment loading (using the absolute load combination method) with a factor of two on the leakage flaw size.

5 From deformation J-R curve for the material

6 A value less than $J_{instability}$. $J_{U.L}$ is based on a maximum crack extension from an actual compact test specimen data.

7 Maximum allowable flaw size that ensures stability of the flaw for the given applied loading.

8 Critical flaw size divided by the leakage flaw size.

Table 1. **Summary of Results (using Zahoor's Modified GE/EPRI Method) for Circumferential Flaws using NRC recommended Material Properties**

Piping Location[1]	Material[2]	Leakage Flaw Size[3], a (in)	$J_{applied}$[4] (kips/in)	Material J_{IC} (kips/in)	Instability Criteria $J_{inst.}$[6] (kips/in)	Critical Flaw Size[7] a_c (in)	Margin on Flaw Size[8]
1	Type 304SS Base	3.215	2.504	n/a	50.	9.72	3.0
	GTAW Weld	3.215	1.209	3.2[5]	40.	10.62	3.3
2	SMAW Weld	3.495	0.302	n/a	1.9	9.60	2.7
	Type 316SS Base	3.495	0.439	n/a	19.8	10.36	3.0
	Type 304SS Base	3.495	1.019	n/a	11.6	9.00	2.6

1 Location 1 refers to core flood piping attached to bottom of the core flood tank nozzle.
Location 2 refers to reactor vessel core flood nozzle safe-end to pipe juncture.

2 General description of base metal or weld metal considered in the analysis

3 Corresponds to one-half of the leakage crack length as predicted by KRAKFLO in Reference 19.

4 Due to applied moment loading (using the absolute load combination method) with a factor of two on the 10 gpm leakage flaw size.

5 From deformation J-R curve for GTAW weld in EPRI NP-4768.

6 $J_{instability}$ point derived from a J-T analysis.

7 Critical flaw size = Maximum allowable flaw size that ensures stability of the flaw for the given applied loading.

8 Critical flaw size divided by the 10 gpm leakage flaw size.

Table 2. Summary of Results (using Original GE/EPRI Method) for Circumferential Flaws using NRC recommended Material Properties

Piping Location[1]	Material[2]	Leakage Flaw Size[3], a (in)	$J_{applied}$[4] (kips/in)	Material J_{IC} (kips/in)	Instability Criteria $J_{inst.}$[6] (kips/in)	Critical Flaw Size[7] a_c (in)	Margin on Flaw Size[8]
1	Type 304SS Base	3.215	6.760	n/a	50.	8.51	2.6
	GTAW Weld	3.215	1.347	3.2[5]	40.	9.8	3.0
2	SMAW Weld	3.495	0.373	n/a	1.7	8.45	2.4
	Type 316SS Base	3.495	0.998	n/a	19.1	9.37	2.7
	Type 304SS Base	3.495	3.854	n/a	11.0	7.85	2.2

1 Location 1 refers to core flood piping attached to bottom of the core flood tank nozzle.
Location 2 refers to reactor vessel core flood nozzle safe-end to pipe juncture.

2 General description of base metal or weld metal considered in the analysis

3 Corresponds to one-half of the leakage crack length as predicted by KRAKFLO in Reference 19.

4 Due to applied moment loading (using the absolute load combination method) with a factor of two on the 10 gpm leakage flaw size.

5 From deformation J-R curve for GTAW weld in EPRI NP-4768.

6 $J_{instability}$ point derived from a J-T analysis.

7 Critical flaw size = Maximum allowable flaw size that ensures stability of the flaw for the given applied loading.

8 Critical flaw size divided by the 10 gpm leakage flaw size

Figure 3. J versus T Diagram for 304SS Base Metal at LBB Location 1

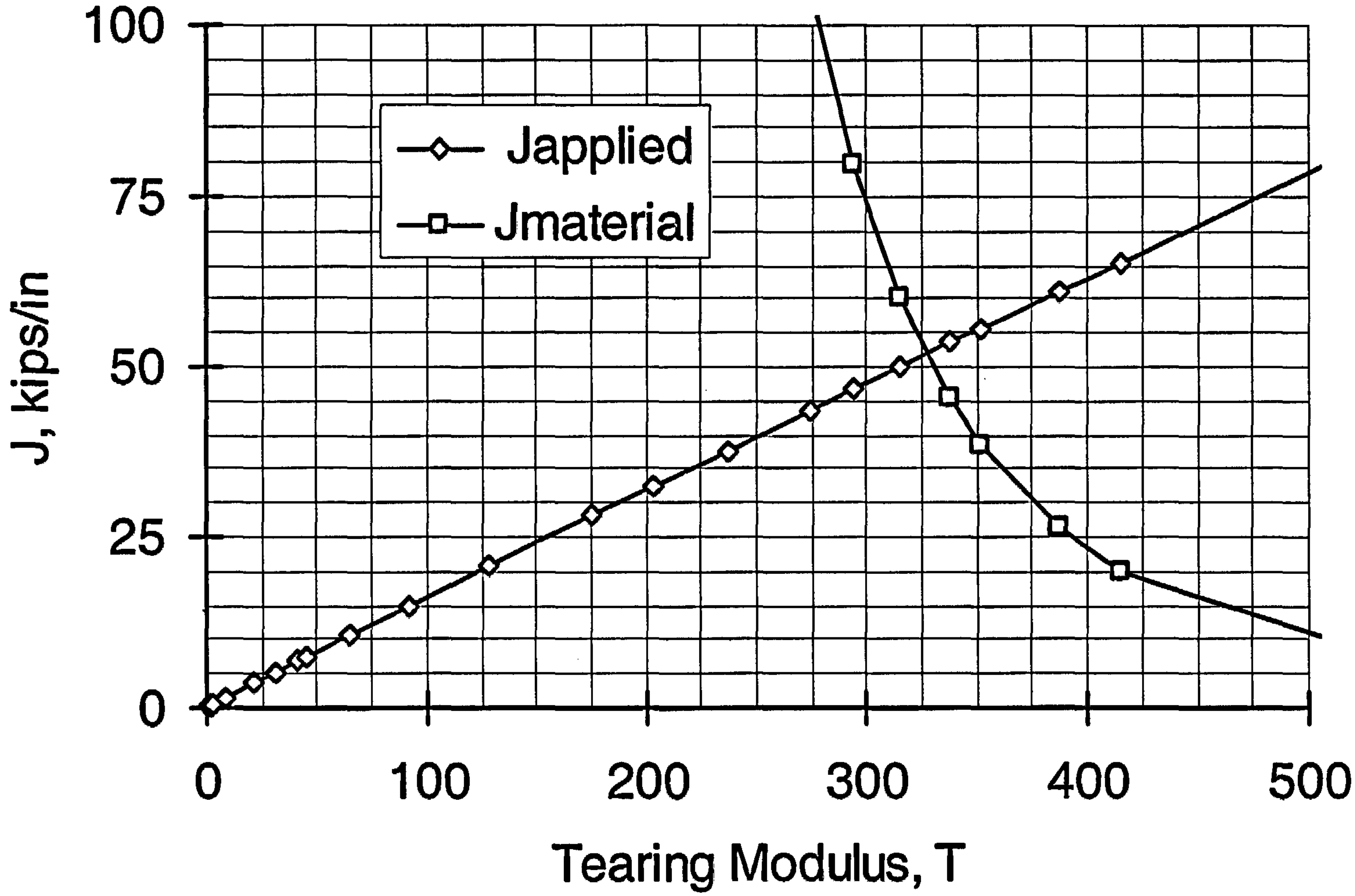


Figure 4. J versus T Diagram for TIG Weld Metal at LBB Location 1

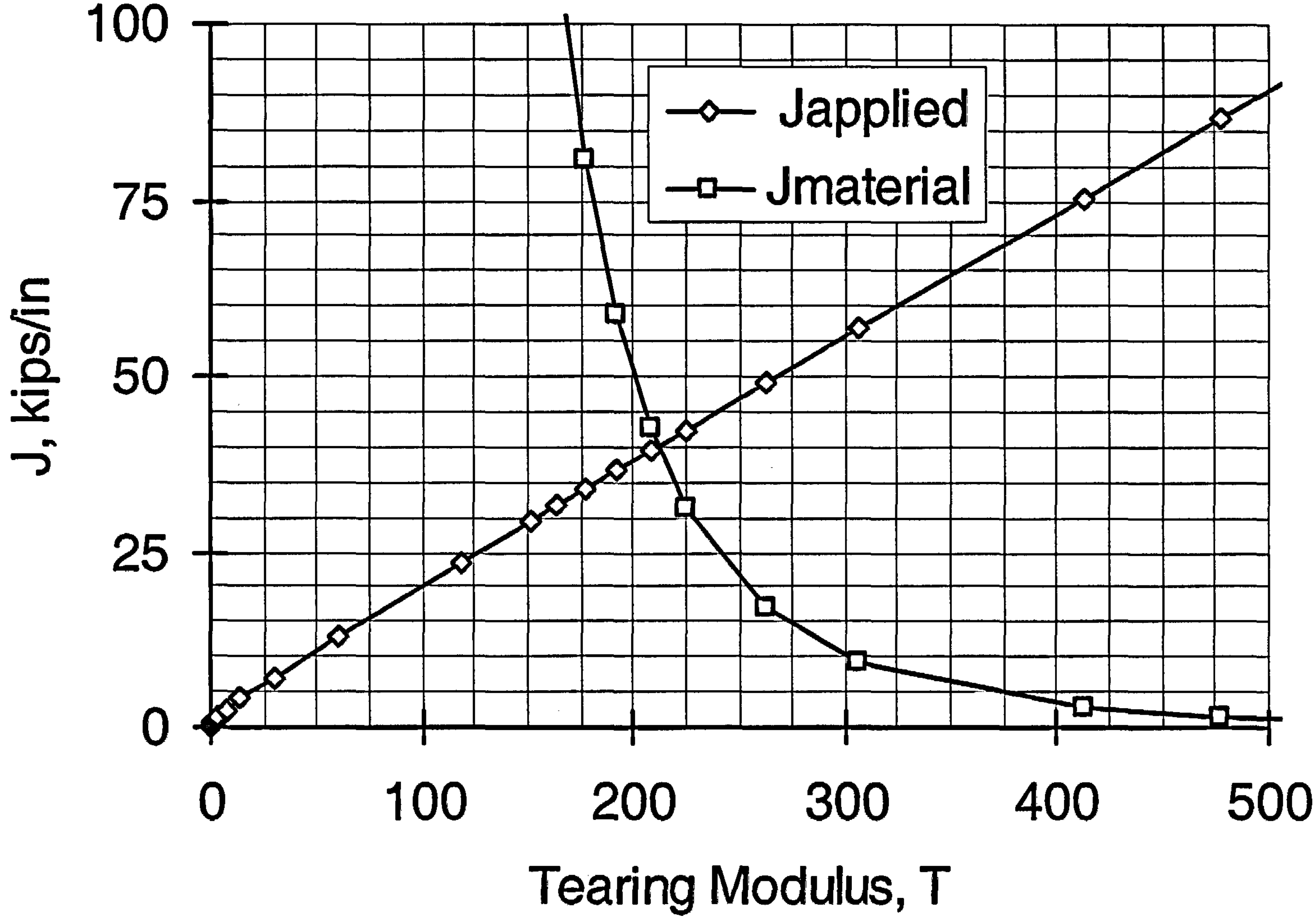


Figure 5. J versus T Diagram for SMAW Weld Metal at LBB Location 2

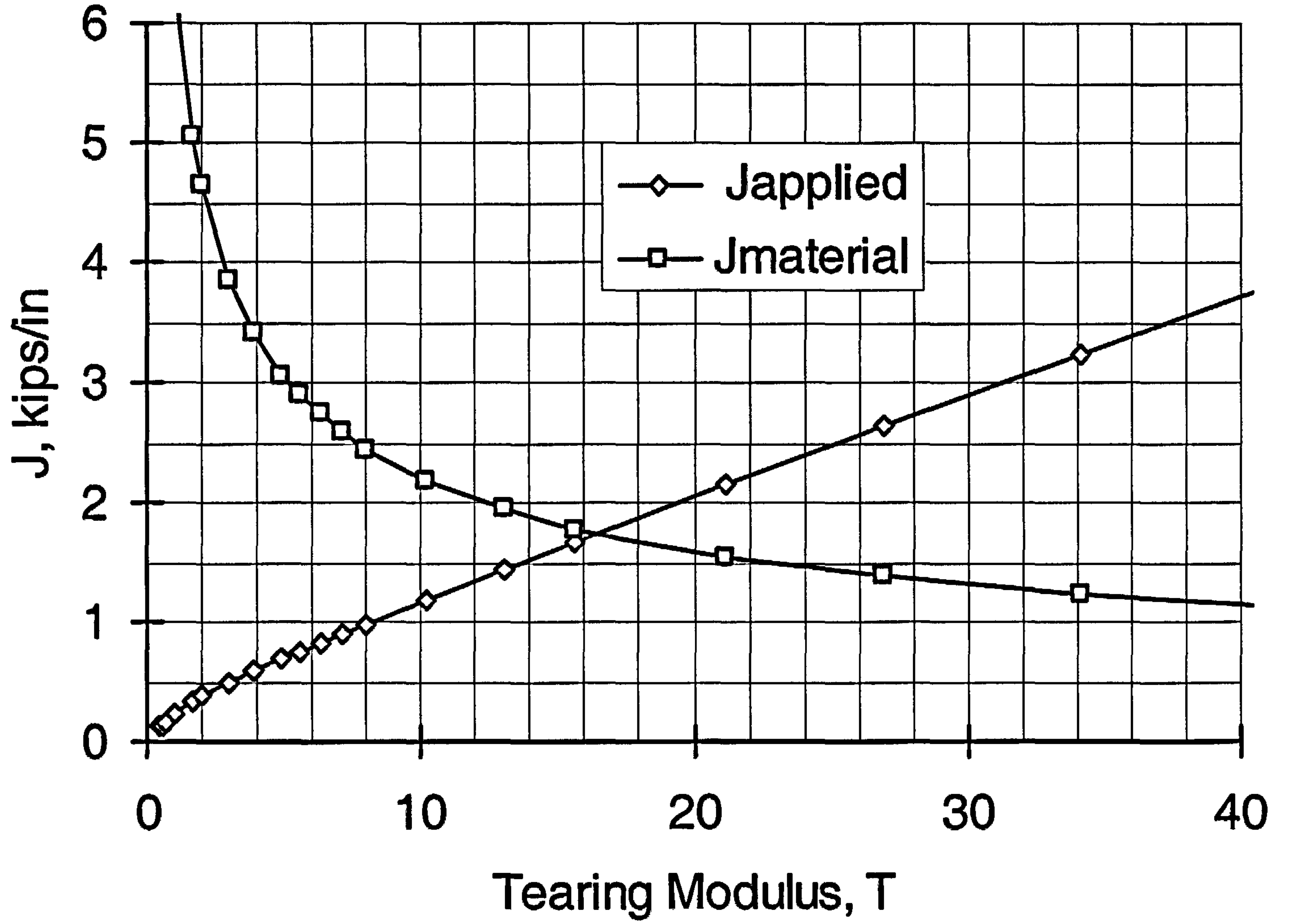


Figure 6. J versus T Diagram for 316SS Base Metal at LBB Location 2

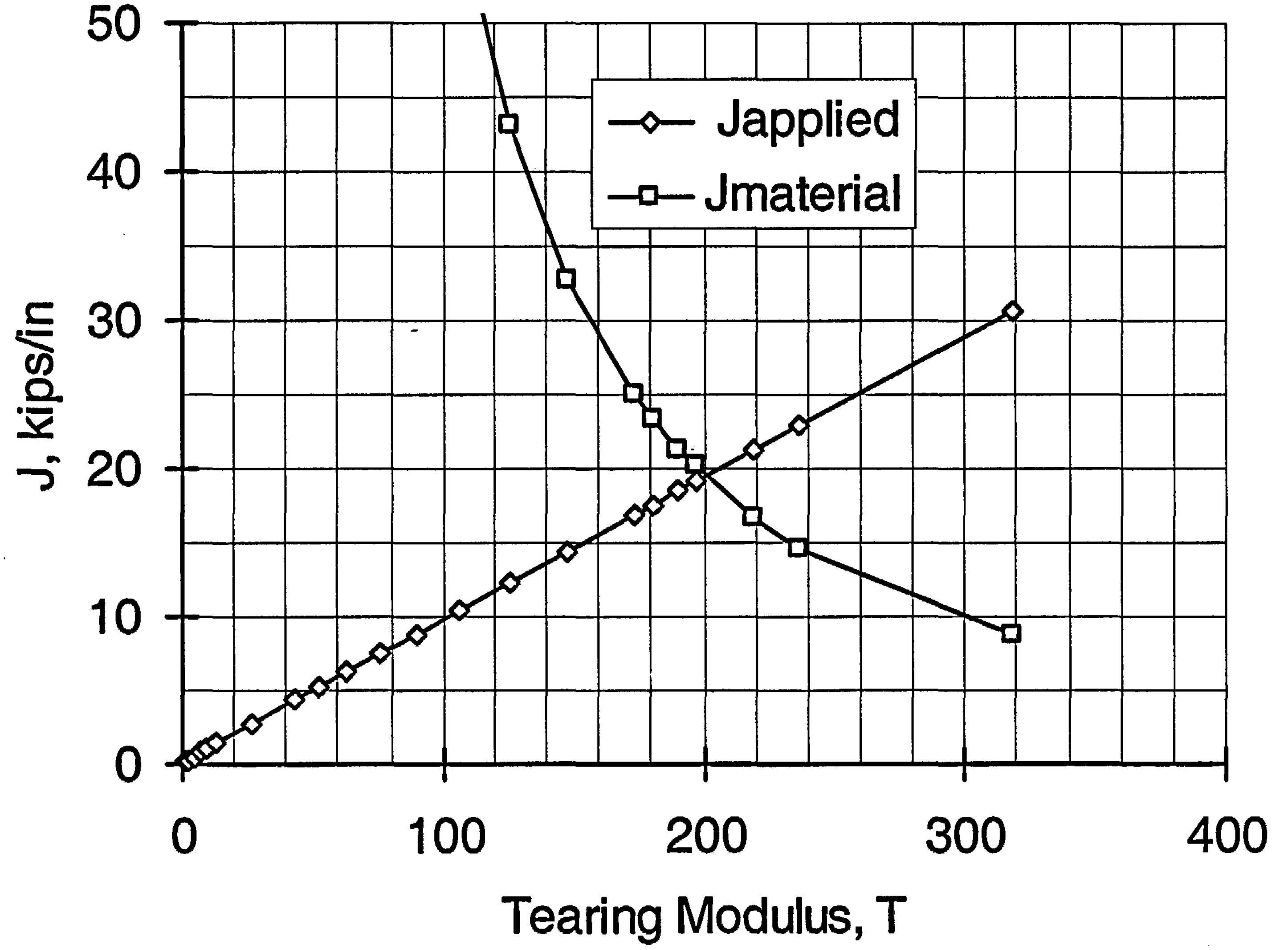
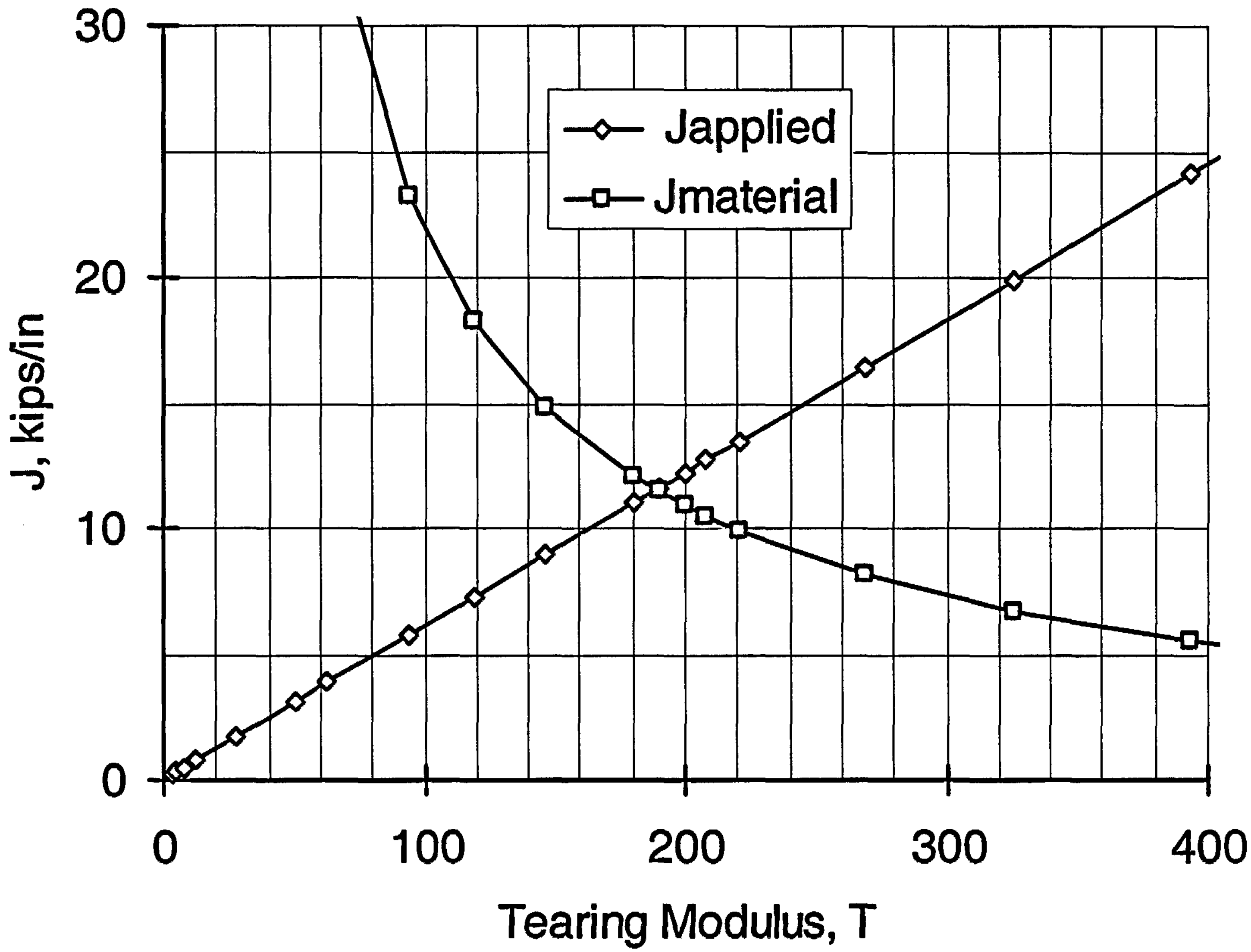


Figure 7. J versus T Daigram for 304SS Base Metal at LBB Location 2



ATTACHMENT 2

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3.6 Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping

3.6.1 Postulated Piping Failures in Fluid Systems Inside and Outside Containment

3.6.1.1 Design Bases

The basic design criteria for pipe whip protection is as follows:

1. All penetrations are designed to maintain containment integrity for any loss of coolant accident combination of containment pressures and temperatures.

2. All penetrations are designed to withstand line rupture forces and moments generated by their own rupture as based on their respective design pressures and temperatures.

3. All primary penetrations, and all secondary penetrations that would be damaged by a primary break, are designed to maintain containment integrity.

4. All secondary lines whose break could damage a primary line and also breach containment are designed to maintain containment integrity.

3.6.1.2 Description

The major components including reactor vessel, reactor coolant piping, reactor coolant pumps, steam generators, and the pressurizer are located within three shielded cubicles. Each of two cubicles contain one steam generator, two coolant pumps, and associated piping. One of the cubicles also contains the pressurizer. The reactor vessel is located within the third cubicle or primary shield. The reactor vessel head and control rod drives extend into the fuel transfer canal.

Openings are provided in the lower shield walls to provide vent area. Pipe lines carrying high pressure injection water are routed outside the shield walls entering only when connecting to the loop.

3.6.1.2.1 Core Flood / Low Pressure Injection System

After implementation of the passive Low Pressure Injection (LPI) cross connect modification on each Oconee Unit, the pipe rupture design basis of Core Flood (CF) / LPI system inside containment is based on the system function during full power operations. The CF section (defined as the "A" and "B" train piping downstream of LP-176 and LP-177 respectively) qualifies as high energy during full power operations. For this CF piping, up to but not including the CF / Reactor Vessel nozzles, Leak Before Break technology was employed to eliminate the dynamic effects associated with postulated breaks (Ref. USFAR Section 5.2.1.9). For the LPI section of the system (defined as the "A" and "B" train piping upstream of LP-176 and LP-177 to their respective Reactor Building penetrations, and including the cross connect piping between the "A" and "B" trains), USNRC Standard Review Plan Section 3.6.2 Branch Technical Position MEB 3-1 (Ref. 3) was used for treatment of postulated pipe ruptures.

3.6.1.3 Safety Evaluation

The analysis of effects resulting from postulated piping breaks outside containment is contained in Duke Power MDS Report No. OS-73.2, dated July 16, 1973 including supplement 2, dated March 12, 1974. An evaluation of potential non-safety grade control system interactions during design basis high energy line break accidents is contained in the Duke Power/Babcock and Wilcox Report dated October 5, 1979. After modifications were made to the EFW system so that makeup to the OTSGs could be assured postulating the HELB and a single failure, Report OS-73.2 credited secondary side cooling within 15 minutes and HPI