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CONSTRUCTION PERMIT NUMBER CPPR-109
VOGTLE ELECTRIC GENERATING PLANT UNIT 2
STEAM GENERATOR SNUBBER REDUCTION AND
AUXILIARY LINE PIPE BREAK ELIMINATION PROGRAM

References:

1) NRC Letter to GPC, E. G. Adensam to D. O. Foster, February 5, 1985

2) GPC Letter to NRC, D. O. Foster to H. Denton, October 25, 1983

3) "Modification of General Design Criterion 4 Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures", Federal Register/Vol. 51, Number 70/ April 11, 1986/Rules and Regulations

4) NUREG-1061, Report of the U.S. NRC Piping Review Committee, Volume 3, Evaluation of Potential for Pipe Breaks

5) Modification of General Criterion 4 Requirements for Protection Against Dynamic Effects of Postulated Pipe Ruptures, Federal Register Vol. 51, No. 141, 7/23/86

6) NRC Letter to GPC, T. M. Novak to D. O. Foster, 6/28/84

7) NUREG/CR-3660 UCID-19988, Volume 3, February 1985, "Probability of Pipe Failure in the Reactor Coolant Loops of Westinghouse PWR Plants", Volume 3, "Guillotine Break Indirectly Induced by Earthquakes", Lawrence Livermore National Laboratory

Gentlemen:

A telephone conference on March 19, 1987, and a meeting on April 17, 1987, were held between the U.S. NRC, Georgia Power Company (GPC) and Westinghouse representatives regarding the steam generator snubber reduction and auxiliary line pipe break elimination program for Vogtle Nuclear Generating Plant Unit 2. The purpose of the April 17, 1987, meeting and this submittal



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is to provide a technical basis for the steam generator snubber reduction and the elimination of postulated pipe ruptures in large diameter ASME Class 1 piping systems, the associated dynamic effects and the required hardware to mitigate those dynamic effects, as well as to discuss GPC's proposed action plans and schedules. The proposed program consists of two parts; the reactor coolant loop and support structural reanalysis for the redesigned steam generator upper support and the application of leak-before-break (LBB) technology and advanced engineering methods in pipe break elimination for the pressurizer surge line, two residual heat removal (RHR) suction lines and four accumulator injection lines. All of these seven lines are 10 inch in diameter and larger. Our objective is to realize significant reductions in radiation exposure to plant operating and maintenance staff as well as reduction in engineering and construction costs. Benefits of the program are summarized below and described in more details in Appendices A and B:

- ALARA Improvements
- Increased accessibility
- Improvement in plant reliability
- Reduction in plant maintenance costs
- Reduction in engineering and construction costs

The criteria and methodology of steam generator snubber reduction and the LBB application to large diameter stainless steel piping are described in Appendices C and D.

GPC was granted an exemption from a portion of General Design Criterion 4 of Appendix A to 10CFR50 regarding the need to analyze large primary coolant loop pipe ruptures as the structural design basis for Vogtle Electric Generating Plant, Units 1 and 2 (Reference 1). GPC stated in the submittal (Reference 2) that the design of the reactor coolant system supports would remain unchanged. Subsequent to the Reference 1 exemption from GDC 4 design requirements, a limited scope GDC 4 rule change for the reactor coolant loop was completed and published in Federal Register, Volume 5, Number 70, April 11, 1986, Rules and Regulations, pp. 12502 - 12505 (Reference 3). This new criteria permits redesign of PWR primary coolant loop heavy component supports to reflect the exclusion of dynamic effects resulting from postulated pipe ruptures in primary coolant loops. GPC intends to use this new criteria to improve the reliability and availability of the NSSS by redesigning the steam generator upper support of Vogtle Unit 2, that is, reducing the number of hydraulic snubbers from five to two for each of the four steam generators. Similar programs were implemented for Surry and Crystal River Nuclear Units and the program for North Anna is currently being reviewed by the NRC.

GPC intends to seek a partial exemption from the requirements of General Design Criterion 4 of Appendix A to 10CFR50 for Vogtle Unit 2. The leak-before-break (LBB) approach will be applied to the pressurizer surge line, accumulator injection and residual heat removal (RHR) suction piping. This will eliminate the postulated pipe ruptures, the associated dynamic effects, and the required hardware to mitigate these dynamic effects. This program is consistent with the NRC Staff Piping Review Committee's recommendations in NUREG 1061, Volume 3 (Reference 4), and those programs which formed the bases for previous exemptions authorized for numerous plants for reactor coolant system main loop piping. In addition, the NRC Staff has approved a limited scope revision to GDC-4 which permits the application of LBB technology to the primary coolant loop piping to eliminate the need to design components for the dynamic effects of high energy pipe breaks. A draft of a broad scope revision to GDC-4 (Reference 5) has been issued for comment which extends the application of this technology to other high energy piping systems. Industry experience indicates that the LBB approach is acceptable for the size and material (stainless steel) of the piping systems included in this program. NRC review and approval for the application of LBB approach to Beaver Valley Unit 2 and South Texas Unit 2 are currently under final stage of review. The U.S. NRC is considering issuance of a schedular exemption from a portion of the requirements of GDC 4 to Beaver Valley Unit 2. The scope, criteria and methodology to be implemented on the Vogtle Unit 2 application are similar to those of the South Texas and Beaver Valley projects.

The two proposed programs have to be coupled since the results of the steam generator snubber reduction program will become part of the input to the LBB application to the selected Class 1 lines. Therefore, GPC will need the approval on both programs from the NRC on the schedule presented in Table 1 of this submittal. The new support configuration and the support stiffness must be determined to reflect the revised support design. The current reactor coolant loop model will then be updated to the new support configuration. The updated loop model will be used in performing the structural analysis of the reactor coolant loop and equipment support structures including effects from large auxiliary line pipe ruptures that could be eliminated by leak-before-break technology. The results of this structural analysis will be used as the basis for the ASME Code qualification for the reactor coolant loop piping and the supports. The conclusions from the previous LBB analysis for the loop will be reverified for the loads derived from the structural reanalysis. The equipment nozzles and support pads will also be shown acceptable under the loads derived from the reanalysis. A new reactor coolant loop model will be developed for the analysis of the auxiliary lines attached to the loop. This loop model will be coupled with the auxiliary piping model for the dynamic seismic analysis applicable to each auxiliary line. Upon verification of piping, supports and interface loading acceptability, the loads derived from the auxiliary line stress analysis will be used as the load inputs for the auxiliary line leak-before-break evaluation. From this process, it is clear that the steam generator support configuration must first be determined before the reactor coolant loop or the auxiliary line structural analysis can be initiated to provide the input data for the LBB evaluation. This forms the basis to submit these two programs together. GPC is proceeding with the analysis for both programs anticipating that both programs will be approved by the NRC. Each of the programs is discussed in more detail below.

Steam Generator Snubber Reduction Program

The support redesign will involve the removal of three of the five hydraulic snubbers from each of the four steam generator supports in Vogtle Unit 2. The current support design includes snubbers which were previously required to mitigate the consequences of pipe ruptures that are no longer postulated. Redesigned support configurations can have fewer snubbers because of the new pipe rupture criteria. Elimination of large breaks in the primary coolant loop piping using leak-before-break analysis and elimination of arbitrary intermediate breaks (Reference 6) in the main steam line in the vicinity of the steam generator main steam nozzle, significantly reduces the loads on the NSSS equipment thereby requiring fewer numbers of snubbers. The seismic qualification of the redesigned supports is performed using analysis methods and Code allowable stresses defined in FSAR to ensure adequate plant safety. The faulted condition stress evaluation of the reactor coolant loop piping and primary equipment supports will include the SRSS combination of SSE with each of the five postulated pipe ruptures: a) accumulator injection, b) pressurizer surge, c) RHR suction, d) mainsteam, and e) feedwater. The inclusion of loads from pipe ruptures a, b and c above introduce additional conservatism to ensure the structural integrity of the primary coolant system against faulted condition loads, since these breaks are to be eliminated by the application of LBB technology.

The revised design for the steam generator supports will adequately consider all remaining design basis loads as specified in the FSAR. With this modification, the reactor coolant system equipment, piping and supports will have acceptable margins of safety under all licensed conditions. Furthermore, current basis of approval for the elimination of postulated pipe ruptures in the reactor coolant loop will be reverified with results of the reanalysis. The accident mitigation features (e.g., emergency core cooling system, containment) of the plant are not affected by the proposed program. Therefore, operation of the facility in accordance with this program would not involve a significant increase in the probability of an accident previously evaluated, nor create the possibility of a new or different kind of accident. In addition, a sufficient margin of safety will be maintained for the primary coolant loop piping and support structures under all applicable loading conditions.

An independent review of the design and construction practices used in Westinghouse PWR plants by Lawrence Livermore National Laboratory (Reference 7) has provided assurance that there are no deficiencies in the Westinghouse RCL design or construction which will significantly affect the probability of double ended guillotine break. The modeling techniques used by Westinghouse for Vogtle Unit 2 are similar to those used for many other plants. The reliability of these techniques is assured by the Westinghouse design control process and comparison of results to other computer programs, including STARDYNE, a public domain code and ME101, a Bechtel Engineering Corporation program. The comparison with STARDYNE was based on the reactor coolant loop analysis results obtained on Surry Nuclear Units by Westinghouse and an independent Architect/Engineer firm and found in good agreement. For this application, both Westinghouse and the independent Architect/Engineer had portions of the responsibility in the design and analysis of the primary coolant loop and support structures. However, for Vogtle Nuclear Generating plant, Westinghouse has the total responsibility of the reactor coolant loop piping and support system qualification using FSAR criteria and techniques.

The installation of the steam generator upper support will be based on Westinghouse design drawings and engineering approved construction tolerances. Quality Control inspection will be performed to verify that the installation is in accordance with the design drawings and applicable tolerances. This process will assure that the final as-built configuration will be enveloped by the engineering design and analysis.

Based on these considerations and the criteria and methodology of the program implementation described in Appendix C, GPC will perform the required structural analysis to demonstrate the adequacy of the redesigned steam generator upper support. The results of this analysis and a request for NRC's approval, will be submitted by the schedule defined in Table 1 of this letter.

Auxiliary Line Pipe Break Elimination Program

GPC will apply the LBB technology on the seven ASME Class 1 lines listed below. All of these lines are connected to the primary coolant loop and are located inside the containment building.

- Pressurizer surge line (16"/14") - Loop 4
- RHR suction lines from hot legs (12") - Loops 1 and 4
- Accumulator injection lines (10") - Loops 1, 2, 3 and 4

The technical bases, which will be provided in support of our program for eliminating postulated circumferential and longitudinal pipe breaks and their associated dynamic effects, include the following:

- Demonstration of LBB at all critical locations based on Vogtle leak detection systems satisfying requirements of Regulatory Guide 1.45.
- Verification of no susceptibility to failure from effects of:

  - o Corrosion
  - o Thermal and vibration induced piping fatigue
  - o Water/steam hammer

Elimination of these postulated breaks would have the following effects on the Vogtle Unit 2 design:

- Eliminate the need to install associated pipe break restraints and jet impingement shields.

- Eliminate the need to consider associated dynamic effects and loading conditions including jet impingement loads and subcompartment pressurization loads.

- Eliminate the need to consider the blowdown loads in the broken lines, as well as connecting lines and adjacent components attached to the primary coolant loop.

This program will not alter previous commitments in the following areas:

- Emergency Core Cooling Systems (ECCS) design bases
- Containment design bases
- Equipment qualification design bases
- Engineered Safety Features Systems response

GPC will submit a request for exemption to eliminate the need to install the associated pipe whip restraints and jet impingement shields and to eliminate the need to design for the dynamic effects associated with these breaks. The criteria and methodology for implementing this program are described in Appendix D. Technical reports, summarizing the results of the LBB analysis of Vogtle Unit 2 pressurizer surge, accumulator injection and RHR suction lines will be prepared to demonstrate the compliance to the NRC draft acceptance criteria contained in Reference 5. This exemption request does not affect the containment pressure boundary, the emergency core cooling system, environmental qualification design bases or engineered safety features system response.

Schedule of Implementation

The design and analysis efforts required to justify the implementation of the steam generator snubber reduction and the auxiliary line pipe break elimination program are in progress. A schedule to provide information to the NRC of significant analysis completion dates is included in the attached Table 1 to this transmittal. We look forward to meeting with NRC staff to discuss this program further.

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TABLE 1
SCHEDULE FOR IMPLEMENTATION OF
STEAM GENERATOR SNUBBER REDUCTION AND
AUXILIARY LINE PIPE BREAK ELIMINATION PROGRAM

| | | |
|---|---|---|
| o | GPC to meet with NRC to discuss the program | April 17,1987 |
| o | GPC letter to NRC on Technical Description and schedule for Steam Generator Snubber reduction and LBB application to Auxiliary Lines | May 1, 1987 |
| o | NRC letter of concurrence on Technical Description and schedule outlined in GPC letter | June, 1987 |
| o | Presentation to NRC on results of the RCL reanalysis and steam generator upper support redesign | June, 1987 |
| o | Presentation to NRC on Surge Line LBB results | June, 1987 |
| o | Submittal of surge line LBB WCAP report for NRC review and technical approval | July, 1987 |
| o | Submittal of FSAR changes to NRC on steam generator upper support redesign | July, 1987 |
| o | NRC technical approval on surge line LBB application | Aug., 1987 |
| o | Presentation to NRC on RHR and Accumulator Line LBB results | Aug., 1987 |
| o | NRC approval on steam generator upper support redesign | Sept., 1987 |
| o | Submittal of RHR and accumulator line WCAP report for NRC review and approval | Oct., 1987 |
| o | Request for exemption for surge, RHR and accumulator lines | Oct., 1987 |
| o | NRC approval for exemption request on surge, RHR and accumulator line LBB application | Dec., 1987 |

## APPENDIX A

## STEAM GENERATOR SNUBBER REDUCTION COST BENEFITS

The cost benefits to be realized from the reduction of the steam generator snubbers center primarily on the ALARA improvements, maintenance savings and material savings. The program will reduce the number of snubbers from five to two for each steam generator. This will provide a total reduction of twelve (12) large bore snubbers which would have to be installed otherwise.

Significant reduction in man-rem exposure will be realized due to the reduction of the number of steam generator snubbers. This reduction would come from spending fewer hours in high radiation areas for inspection, maintenance and testing activities. Conservatively, it is estimated that 50 man-rem could be saved from the reduction of inspection, preventive maintenance, and testing requirements for the eliminated snubbers. This estimate is based on the planned refueling outages for Vogtle Unit 2 under 18-month fuel cycles.

Steam generator snubbers must be periodically inspected, functionally tested, and maintained to ensure operability. A visual inspection is conducted for all snubbers each refueling outage; a minimum of 10% of the snubbers are functionally tested each outage. In addition, certain maintenance activities at 5 year and 10 year intervals are recommended by the snubber manufacturer. The costs associated with these activities vary significantly depending on the test facility, the test sampling plan, and the number of snubbers which fail to meet the acceptance criteria. For the purposes of benefit analysis, it is assumed that 12 fewer snubbers require periodic maintenance and inspection each outage, and 2 fewer snubbers require functional testing each outage. The total savings for the life of the plant are estimated to be $700,000. This estimate does not include the man-rem savings associated with the eliminated snubbers.

The material saving for the eliminated snubbers is conservatively estimated at $40,000 for each snubber. The associated construction savings will include the reduction of a six-person crew for four hours to install each large bore snubber. The total material and installation savings are estimated at $500,000.

The overall plant availability will be improved if steam generator snubbers are reduced from the existing plant design due to the reduced inspection, maintenance and testing requirement and thus reduces plant down time. Fewer snubbers will increase accessibility at the steam generator upper support area for plant normal operational and maintenance activities. The benefits for Vogtle Unit 2 steam generator snubber reduction program are summarized in Table A-1.

TABLE A-1

SUMMARY OF BENEFITS FROM THE
STEAM GENERATOR SNUBBER REDUCTION

| | CATEGORY | BENEFITS |
|---|---|---|
| 1. | ALARA Improvement | 50 man-rems |
| 2. | Maintenance and functional testing savings | $700,000 |
| 3. | Material and installation savings | $500,000 |
| 4. | Improvement in plant reliability | Increased accessibility |

## APPENDIX B

## LEAK-BEFORE-BREAK COST BENEFITS

The cost benefits to be realized from the elimination of the postulated pipe break locations on the pressurizer surge line, the two (2) RHR suction lines and the four (4) accumulator injection lines center primarily on the elimination of the associated pipe whip restraints and the structural evaluation for the dynamic loads. The application of the leak-before-break criteria to the large diameter auxiliary lines will eliminate twenty-three (23) break locations and fourteen (14) pipe break restraints. Significant operational benefits will be realized over the 40 year life of the plant. The cost savings for Vogtle Unit 2 are summarized in the attached Table B-1.

A significant reduction in man-rem exposure can be realized through fewer hours spent in radiation areas. It is estimated that a reduction of 2 man-rem will be actualized over the plant lifetime for each of the 14 pipe break restraints eliminated due to the improved accessibility for piping inspections.

Repair, maintenance and inspection of components within the vicinity will be more effective if the pipe break restraints are eliminated. It is estimated that at least three hours of operation and maintenance time will be saved during each outage for every restraint eliminated.

The access during plant operation for inservice inspection activities will be improved due to the reduction of congestion created by the pipe break restraints and the supporting structural steel. In addition, access to welds can be improved and the need for restraint removal during weld inspection can be reduced.

The material required for these pipe break restraints has partially been procured but the installation of these restraints has not yet been initiated. The average construction effort for each of the fourteen pipe break restraints is estimated at 600 hours. Additional savings will be realized during and after the hot functional testing for the measurement and shimming of the gap at each restraint. This is necessary to avoid stresses due to restraint of thermal expansion if the pipe break restraints would come into contact with the pipes. The elimination of the postulated pipe breaks would also reduce the effort involving the verification of gaps during each plant outage.

The elimination of jet impingement loads associated with each break will reduce the required engineering effort for performing hazard study during the plant design stage. The engineering savings vary from break to break. It is conservatively estimated that 50 hours of engineering effort will be saved for each of the 23 postulated breaks.

In addition to the above benefits, the plant construction schedule could be improved by not installing the pipe break restraints. It would also simplify the initial startup phase because the need for hold points to inspect clearances between piping and pipe break restraints could be eliminated.

TABLE B-1
SUMMARY OF BENEFITS FROM THE
LEAK-BEFORE-BREAK APPLICATION TO SEVEN
AUXILIARY LINES ON VOGTLE UNIT 2

| | CATEGORY | BENEFITS |
|---|---|---|
| 1. | Reduction in Man-Rem Exposure | 28 Man-Rem |
| 2. | Elimination of Pipe Break Restraint Installation and Gap Shimming | \$280,000 |
| 3. | Reduction of effort in hazard study | \$ 62,000 |
| 4. | Improvement for operations, inspection and maintenance during outages | \$ 40,000 |
| 5. | Improvement in construction and startup schedule | Reduction in construction efforts for installation and verification of pipe break restraints. |

## APPENDIX C
## CRITERIA AND METHODOLOGY FOR STEAM GENERATOR SNUBBER REDUCTION

### INTRODUCTION

The current design basis for Vogtle Unit 2 is based on the elimination of the dynamic loadings associated with intermediate pipe break locations in the mainsteam line and all pipe breaks in the large diameter primary loop piping. This design basis permits improvements in plant reliability and man-rem reductions by reducing the number of steam generator upper lateral support snubbers. This appendix describes the configuration of the steam generator upper supports and shows the planned snubber reduction. The criteria and methodology that are used to demonstrate the structural adequacy of the piping and support system are presented.

### STEAM GENERATOR UPPER SUPPORT CONFIGURATION

The upper steam generator support consists of an octagonal ring girder placed around the generator shell. The girder is hung from the steam generator trunions by four tie rods. These tie rods support the dead weight of the ring and aid in the vertical positioning of the girder. Laterally, the girder is connected to five hydraulic snubbers placed parallel to the hot leg on the reactor side of the steam generator. These snubbers, along with a strut behind the steam generator and parallel to the hot leg, restrain the steam generator for motions and loadings along the hot leg. Restraint of motions and loadings normal to the hot leg is provided by two additional struts that bear against the ring girder. These struts are attached to the secondary shield wall with embedded anchor bolt assemblies. Loads are transferred from the steam generator shell to the ring girder by means of curved bearing plates welded to the ring girder. This upper support system allows unrestrained thermal loop expansion to the final hot operating position. At this position, each strut to ring girder bearing surface is shimmed to provide proper contact, thus providing restraint to the steam generator in the operating position. A sketch of the steam generator upper supports is shown in Figure C-1. The redesigned snubber arrangement is shown in Figure C-2. The steam generator and its combined support system are shown in Figure C-3.

Detailed descriptions of the remaining reactor coolant loop equipment supports (SG lower, RPV, and RCP supports) can be found in the VEGP FSAR Section 5.4.14.

### STRUCTURAL QUALIFICATION OF PIPING AND SUPPORTS

The structural analysis interfaces and responsibilities of the engineering organizations are summarized in Figure C-4. These interfaces assure that the appropriate input boundary conditions are used in the loop piping/support and auxiliary piping/support models. In addition, the results of the piping models are properly used in the detailed evaluations of the piping and supports, as well as the connecting nozzles and concrete embedments. The updated loop piping model, which incorporates the modified steam generator upper lateral support snubbers, is used in the auxiliary Class 1, mainsteam and feedwater piping models. These auxiliary piping models are then used to determine the loading input for the leak-before-break evaluation.

Westinghouse topical reports WCAP-10551 (Class 2) and 10552 (Class 3), have provided a substantial and adequate technical basis for limiting postulated design basis flaws in the Vogtle Unit 2 stainless steel primary coolant loop piping. The analyses have demonstrated that the probability of rupturing such piping is extremely low under design basis conditions. These WCAP's have documented the plant specific fracture mechanics study in demonstrating the leak-before-break capability. With the redesigned steam generator support configuration, revised loads (forces and moments) in the primary coolant loop piping will be generated and reverified to demonstrate leak-before-break.

The loop piping model consists of mass and stiffness representations for the loop and the reactor vessel. Each loop includes the primary loop piping, steam generator and reactor coolant pump. The primary equipment supports are represented by stiffness matrices. The seismic analysis is performed by the envelope response spectra method with damping values of 2 and 4 percent for OBE and SSE, respectively. The analysis is performed with the WESTDYN computer program. This program has been used for many other plants, is verified for this application and a controlled version is maintained by Westinghouse.

The stress criteria for the loop piping and supports are presented in FSAR Sections 3.9.B.3 and 3.9.N.1. The faulted condition includes the SRSS combination of SSE with each of five postulated pipe ruptures: accumulator, surge, RHR suction, mainsteam and feedwater. The allowable stresses are obtained from ASME Section III Code Subsections NB and NF.

The loop piping and support stress evaluation for the revised steam generator support design is currently in progress. The faulted condition margins for the existing steam generator upper support are summarized in Table C-1. The existing margins indicate that acceptable results will be obtained for the revised support design.

TABLE C-1

STEAM GENERATOR UPPER SUPPORT MEMBER STRESSES [c]

| Member | Member Loads and Stresses (Percent of Allowable/Loading Condition) Units (kips) | | | |
|---|---|---|---|---|
| | Upset | | Faulted | |
| | Load | Percent Stressed | Load | Percent Stressed |
| US-1, US-2 US-3 (Bumpers) | 868 | 39[b] | 1721 | 58[b] |
| Girder | 868 | 33 | 1721 | 58 |
| US-4 (Snubber)[a] | 868 | 39 | 1721 | 34 |

NOTES:

a) The snubbers were qualified by a 450 kip upset capacity and a 1000 kips faulted capacity, per snubber, in accordance with the snubber stress report [1].

b) Includes effect of small bore support attached to bumper

c) These stresses are based on current snubber configuration at the steam generator upper support for Unit #1.

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[1] Paul-Monroe Hydraulics Inc., Report A-690623, Revision 0, entitled "Multiplant II 1000 kip Snubber Stress Report".

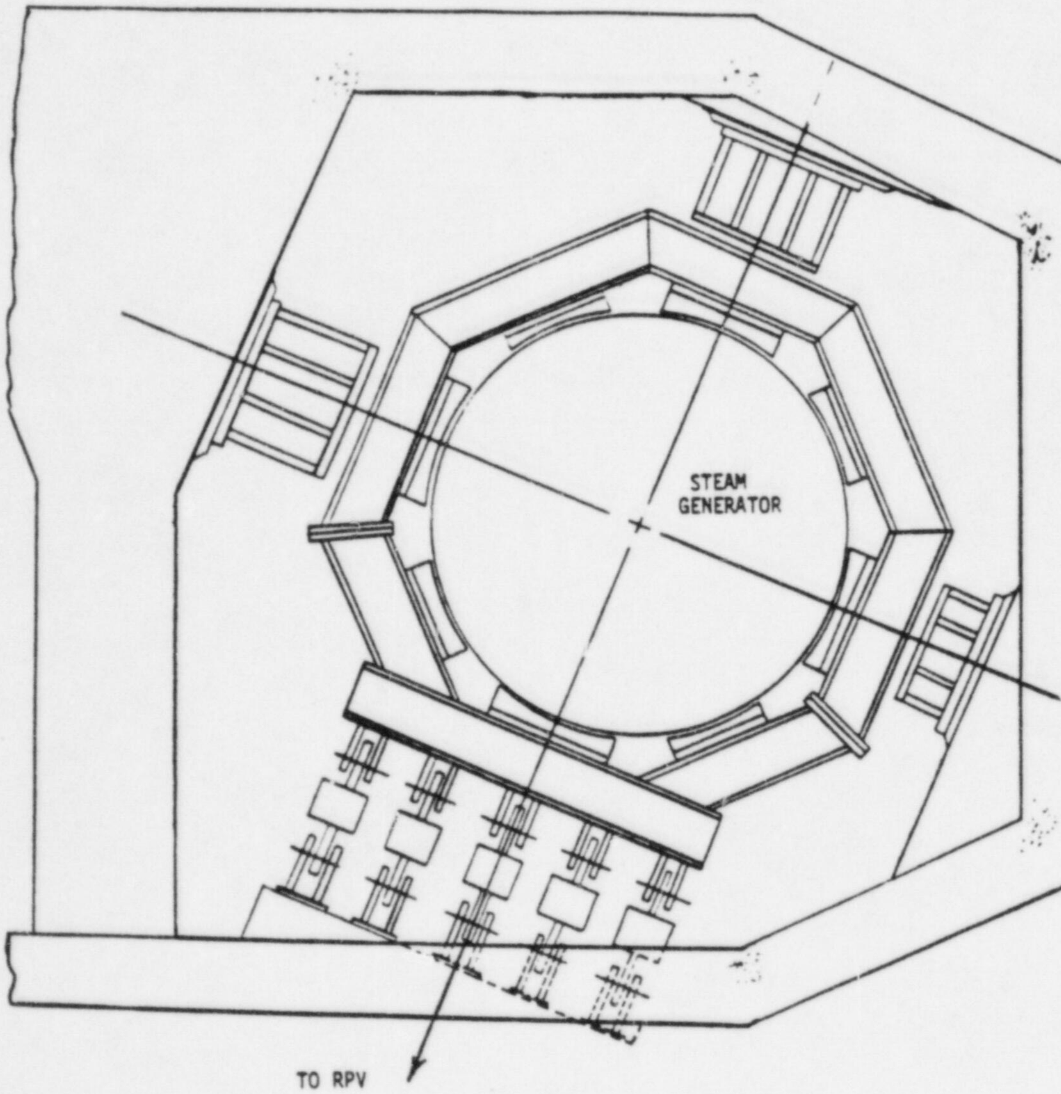



FIGURE C-1 : CURRENT STEAM GENERATOR UPPER SUPPORT SNUBBER ARRANGEMENT

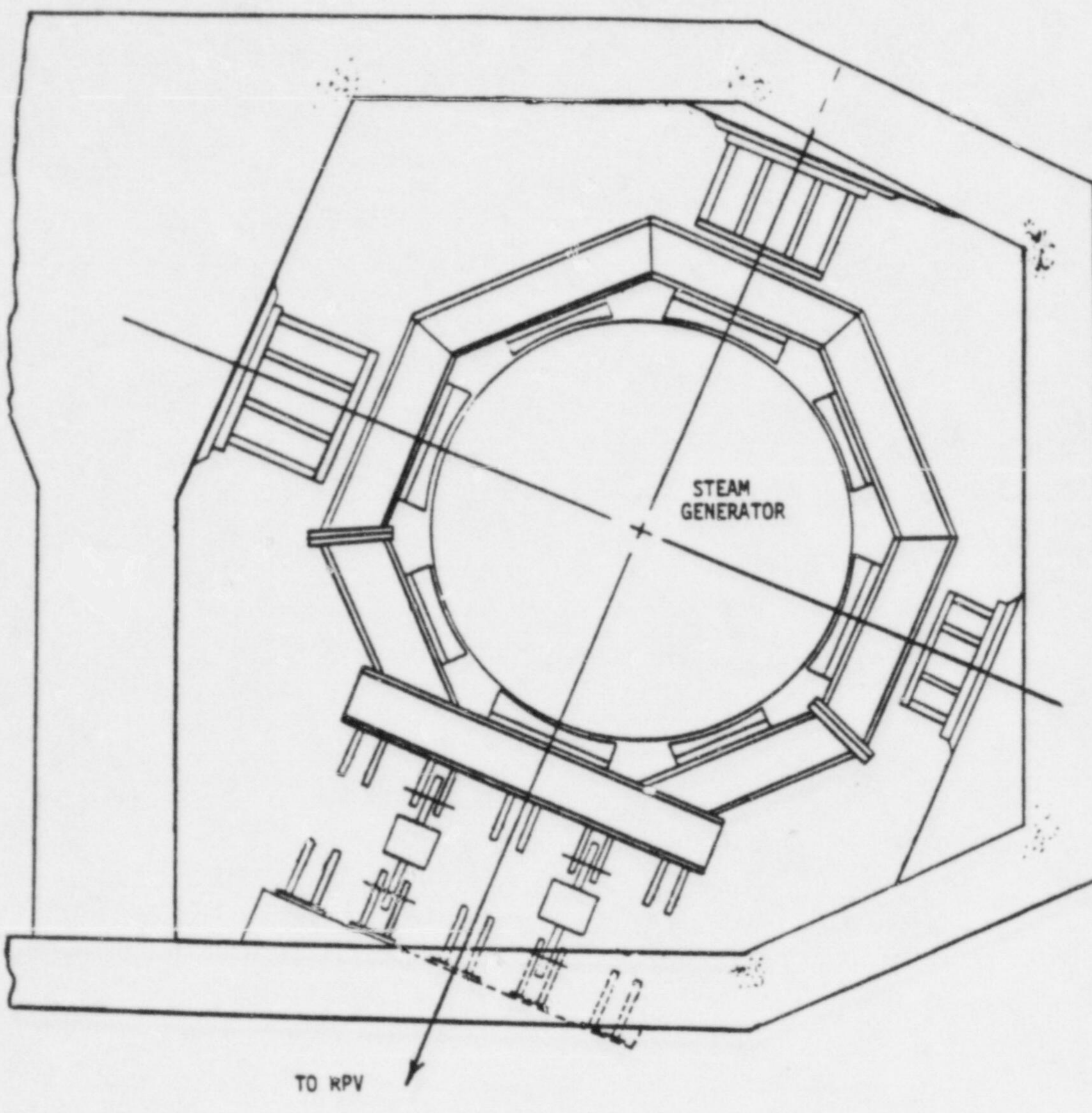



FIGURE C-2: REDESIGNED STEAM GENERATOR UPPER SUPPORT SNUBBER ARRANGEMENT

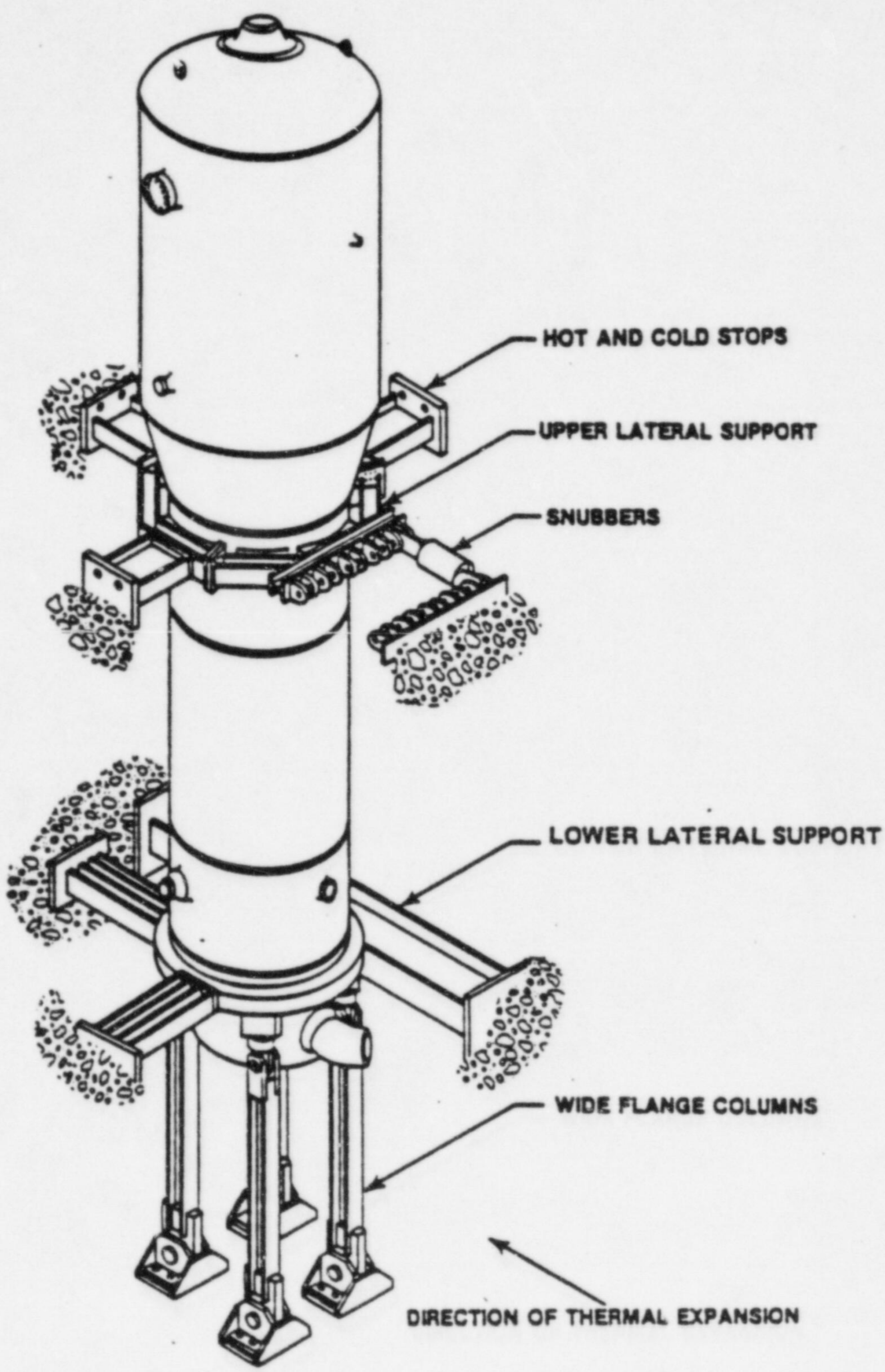
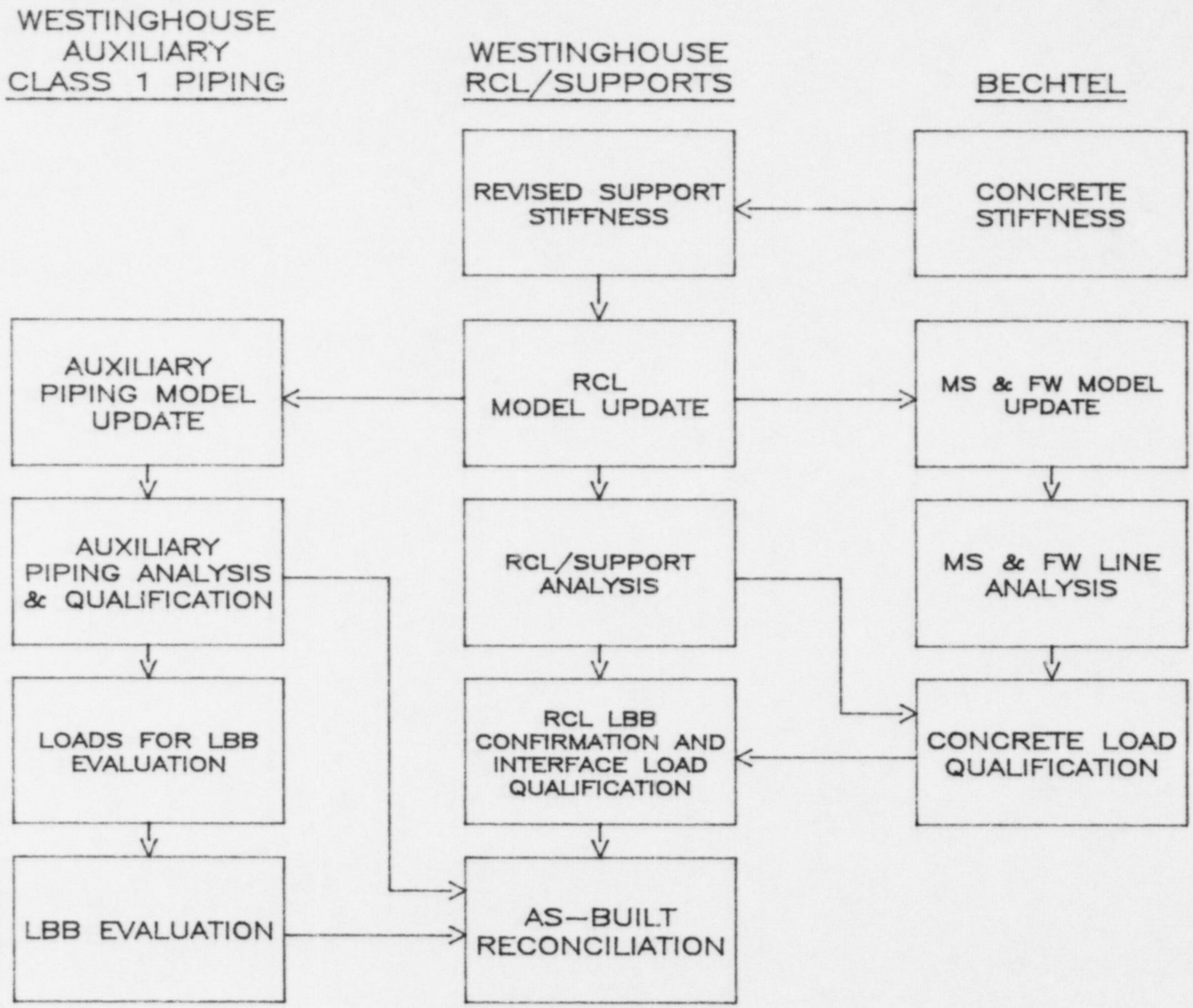



FIGURE C-3: STEAM GENERATOR SUPPORT SYSTEM

FIGURE C-4

# OVERALL FLOW OF ANALYTICAL ACTIVITIES



## APPENDIX D

## CRITERIA AND METHODOLOGY FOR LEAK-BEFORE-BREAK EVALUATION

The recommendations and criteria proposed in NUREG 1061 Volume 3 (D-1) will be used in this evaluation. These criteria are identical to those accepted in the limited scope modification of GDC-4. These criteria and resulting steps of the evaluation procedure are summarized below:

1) The sections of the following piping systems which will be shown to meet the LBB criteria are selected to justify the elimination of pipe ruptures that would be required based on the terminal end and high stress criteria in the VEGP FSAR:

   a) Surgeline

   The entire length of the 14 and 16 inch diameter surge line is classified as a high energy piping system. The loads for the LBB evaluation include all loads at each circumferential weld joint from anchor at loop nozzle to anchor at pressurizer nozzle. The materials for the LBB evaluation include base metal, weldments, and nozzle safe-ends.

   b) Accumulator Line

   The entire length of the 10-inch diameter schedule 140 and schedule 40S accumulator line is classified as a high energy piping system. The pipe ruptures being eliminated are in the schedule 140 portion of the line from the loop nozzle to the schedule change beyond the third valve. The pipe ruptures in the schedule 40S portion are not being eliminated. The loads for the LBB evaluation include all loads at each circumferential weld from the cold leg nozzle to the accumulator nozzle. The materials for the LBB evaluation include base metal, weldments, and nozzle safe-ends.

   c) Residual Heat Removal (RHR) Line

   The 12-inch diameter RHR line is classified as a high energy line from the loop nozzle to the first weld to the second valve. Pipe breaks are not postulated in the remaining portion of the line from the second valve to the containment anchor. The loads for the LBB evaluation include all loads at each circumferential weld from the loop nozzle to the containment penetration anchor. The materials for the LBB evaluation include base metal, weldments, and nozzle safe-ends.

2) The applied loads will be calculated and will include the static forces and moments resulting from normal operation (pressure, deadweight, and thermal expansion) and the forces and moments associated with the Safe Shutdown Earthquake including the effect of anchor motion. These forces and moments and the base metal and weld metal properties will be used to define the governing location in the pipe run for fracture mechanics evaluation.

3) The material data will include identification of materials used for base metal, weld metal and safe-ends. The Vogtle Unit 2 specific mechanical properties (yield strength and ultimate tensile strength) for the base metal and weld metal will be reviewed from the material certification records to characterize the material. Justification will be provided to support validity of the strength and toughness (J-R curve) data used for the fracture mechanics evaluations.

   It has been found for cast stainless steels that the chrome enriched ferrite of the two-phase alloy becomes hardened and embrittled when thermally aged. This results in significant loss of toughness over a long period of time. Such long-term thermal aging effect will be included in the material characterization where applicable.

4) Surface flaws will be postulated at governing location with a size that is permitted by the acceptance criteria of Section XI of the ASME Boiler and Pressure Vessel Code. The flaws will be subjected to the design transient loadings (service level A&B) to determine the crack growth during service. The aspect ratio for the postulated surface flaw in the fatigue crack growth evaluation will be 6. The aspect ratio will be kept constant throughout the analysis. The final crack size will be shown to be less than 60% of wall thickness as per the NRC staff's present position.

5) A through-wall crack will be postulated at the governing location (determined from criteria 1 and 2 above). The size of the crack will be large enough so that the leakage will be detectable using the installed leak detection equipment when the pipe is subjected to normal operating loads. NUREG 1061 Volume 3, recommends a margin of a factor of 10 between the magnitude of calculated leakage and the capability of the leak detection system. This crack size is referred to as the "leakage size" crack.

6) It will be demonstrated that the postulated leakage size crack is stable under normal plus SSE loads. The margin, in terms of applied loads, would be determined by crack stability analysis, i.e. that the leakage size crack will not experience unstable crack propogation even if larger loads (larger than design loads) are applied. This analysis will demonstrate that unstable crack propagation resulting in double-ended guillotine pipe break will not occur. In addition, using normal plus SSE loads, a margin of at least 2 between the leakage size crack and the critical size crack will be demonstrated.

7) For the piping run/system under evaluation, all pertinent information will be provided which demonstrates that the degradation or failure of the piping resulting from corrosion, erosion-corrosion, stress-correosion, fatigue, water hammer or other environmental conditions is not likely.

   Relevant operating history from PWR plants in operation will be cited, which will include information such as system operational procedures, system or component modifications, water chemistry parameters, limits and control, resistance of material to various forms of stress corrosion, and performance under cyclic loadings.

## Analytical Methods for Stability of Through-Wall Cracks in Pipes

### Global Failure Mechanism

The ability of a flawed pipe to withstand stresses resulting from applied loads is typically determined by the material's strength provided that the piping material has a high fracture toughness and therefore is not sensitive to the presence of a crack. One method for predicting the failure of ductile material is the plastic instability. This method is based on traditional plastic limit load concept, but accounting for strain hardening and taking into account the presence of a flaw. The flawed pipe is predicted to fail when the remaining section reaches a stress level at which a plastic hinge is formed. The stress level at which this occurs is termed as the flow stress. The flow stress is generally taken as the average of the yield and ultimate tensile strength of the material at the temperature of interest. This methodology has been shown to be applicable to ductile piping through a large number of experiments and will be used here to predict the critical flaw size in the piping system. A detailed discussion of the method (and its limitations) is provided in NUREG 1061, Volume 3.

### Local Failure Mechanism

The local mechanism of failure is primarily dominated by the crack tip behavior in terms of crack-tip blunting, initiation, extension and finally crack instability. The material properties and geometry of the pipe, flaw size, shape and loadings are parameters used in the evaluation of local failure.

The stability will be assumed if the crack does not initiate at all. It has been demonstrated that the initiation toughness, measured in terms of $J_{Ic}$ from a J-integral resistance curve, is a material parameter defining the crack initiation. If, for a given load, the calculated J-integral value is shown to be less than $J_{Ic}$ of the material, then the crack will not initiate.

If the initiation criterion is not met, one can calculate the tearing modulus as defined by the following relation:

$$T_{app} = \frac{dJ}{da} \frac{E}{\sigma_f^2}$$

where $T_{app}$ = applied tearing modulus

$E$ = modulus of elasticity

$\sigma_f$ = flow stress = $(\sigma_y + \sigma_u)/2$

$a$ = crack length

$\sigma_y$, $\sigma_u$ = yield and ultimate strength of the material respectively.

In summary, the local crack stability will be established by the two-step criteria:

$J < J_{Ic}$, or

$T_{app} < T_{mat}$, if $J \geq J_{Ic}$

## Leak Rate Predictions

For postulated through-wall flaws the crack opening area resulting from the application of normal operating loads will be calculated. The crack opening area can be obtained either from the nodal displacements using the finite element method or by using the method described in reference D-2. The latter method is used for lower stress levels in which case linear elastic fracture mechanics is applicable for analyzing the pipes with hypothezied flaws. Once the crack opening area is calculated, the leak rate will be predicted as described below.

Generally, the flow of hot pressurized water through an opening to a lower back pressure causes flashing which can result in choking. For long channels where the ratio of the channel length, L, to hydraulic diameter, $D_H$, ($L/D_H$) is greater than 40, both choking and frictional effects must be considered. In this situation the flow can be described as being single phase through the channel until the local pressure equals the saturation pressure of the fluid. At this point, the flow begins to flash and choking occurs. Pressure losses due to momentum changes will dominate for $L/D_H<40$. However, for large $L/D_H$ values, friction pressure drop will become important and must be considered along with the momentum losses due to flashing. The basic method used in the leak rate calculations is the method developed by Fauske (Ref. D-3) for the two phase choked flow, and then adding to it the additional frictional pressure loss upstream of the choked exit plane. The calculated leak rate will be compared with the plant leak detection capability to assure detection.

## References

(D-1) NUREG 1061, Volume 3, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee -- Evaluation of Potential for Pipe Breaks", U.S. NRC, November 1984

(D-2) Tada, H., "The Effects of Shell Corrections on Stress Intensity Factors and the Crack Opening Area of Circumferential and a Logitudinal Through-Crack in the Pipe", Section II-1, NUREG/CR-3464, September 1983.

(D-3) Fauske, H.K., "Critical Two-Phase, Steam Water Flows", Proceedings of the Heat Transfer and Fluid Mechanics Institute, Stanford University Press 1961.