ENCLOSURE 2

TOPICAL REPORT EVALUATION

Report Title and Number:

1. Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Throughwall Crack, WCAP 9558, Rev. 2, Westinghouse Class 2 Proprietary, May, 1982.

2. Tensile and Toughness Properties of Primary Piping Weld Metal For Use In Mechanistic Fracture Evaluation, WCAP 9787, Westinghouse Class 2 Proprietary, May, 1982.

3. Westinghouse Response to Questions and Comments Raised by Members of ACRS Subcommittee on Metal Components During the Westinghouse Presentation on September 25, 1981, Letter Report NS-EPR-2519, E. P. Rahe to Darrell G. Eisenhut, November 10, 1981.

1.0 Background

In 1975, the NRC staff was informed of some newly defined asymmetric loads that result by postulating rapid-opening double-ended ruptures of PWR primary piping. The asymmetric loads produced by the postulated breaks result from the theoretically calculated pressure imbalance, both internal and external to the primary system. The internal asymmetric loads result from a rapid decompression that causes large transient pressure differentials across the core barrel and fuel assembly. The external asymmetric loads result from the rapid pressurization of annulus regions, such as the annulus between the reactor vessel and the shield wall, and cause large transient pressure differentials to act on the vessel. These large postulated loads are a consequence of the rapid-opening break at the most adverse location in the piping system.

The staff requested, in June 1976, that the owners of operating PWRs evaluate their primary systems for these asymmetric loads. Most owners formed owners groups under their respective NSSS vendors to respond to the staff request. The Babcock and Wilcox (B&W) and Combustion Engineering (CE) owners groups each submitted a probability study, prepared by Science Applications Inc., and the Westinghouse owners submitted a proposal for augmented inservice inspection. The staff reviewed these submittals and concluded at that time that neither approach was acceptable for resolving this problem. In general, the staff concluded that the existing data base was not adequate to support the conclusions of the probability study and that the state-of-the-art for inservice inspection alone was not acceptable for this purpose.

8308250623

XA

The staff formalized these conclusions in a letter to the owners of all operating PWRs in January 1978. This letter also reiterated our desire to have the PWR owners evaluate their plants for asymmetric loads. Plant analyses for asymmetric loads were submitted to the staff for review in March and July 1980. The results of these plant analyses indicated that some plants would require extensive modifications if the rapid-opening double-ended break is required as a design basis postulation.

Subsequent to the 1975 postulation of the relatively large theoretical loads resulting from the assumed pipe rupture, it has been found that the loads decrease significantly when more realistic break sizes and break opening times are assumed. In fact, if these mechanistic assumptions are validated, the resulting loads are well within the capabilities of the PWR primary coolant systems and their respective support structures to accommodate them.

Also, in the interim, the technology regarding the potential rupture of relatively tough piping such as is used in PWR primary coolant systems, has advanced significantly. Thus, a much better understanding of the behavior of flawed piping under normal and even excessive loads now exists. The NRC staff utilized these technological developments in its review. Tests of deliberately cracked pipes in addition to theoretical fracture mechanics analyses indicate that the probability of a full double-ended rupture of tough piping in a typical PWR primary coolant system is vanishingly small. The subject of PWR pipe cracking is discussed in NUREG-0691 and other references listed in Section 6 of this evaluation.

In parallel with the performance of plant analyses for asymmetric loads, some owners, anticipating potential modifications resulting from the double-ended rupture assumption, engaged Westinghouse to perform a mechanistic fracture evaluation to demonstrate that an assumed double-ended rupture is not a credible design basis event for PWR primary piping. Upon completion of this evaluation, Westinghouse, on the owners group behalf, submitted to the staff for review the topical report, "Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Throughwall Crack," WCAP 9558, Rev. 2. In response to questions raised by the staff, a second report, "Tensile and Toughness Properties of Primary Piping Weld Metal For Use In Mechanistic Fracture Evaluation," WCAP 9787, was also submitted by Westinghouse for our review. In addition, in the third report listed above, Westinghouse submitted responses to questions and comments of the ACRS Subcommittee on Metal Components during the Westinghouse presentation on September 25, 1981.

2.0 Scope and Summary of Review

The analyses contained in WCAP 9558, Revision 2, were performed to demonstrate, on a deterministic basis, that the potential for a double-ended failure of the stainless steel primary piping for the facilities identified by the Westinghouse Owners Group was low enough so that a double-ended break need not be considered a design basis for defining structural loads for resolution of Unresolved Safety Issue (USI) A-2, "Asymmetric Blowdown Loads on Reactor Primary Coolant Systems." Consequently, the staff's review focuses only on the structural integrity of PWR main reactor coolant loop piping and the staff does not apply the analyses to other issues such as containment design, release of radioactive materials, or ECCS design.

Our evaluation includes definition of general criteria that can be used to evaluate the integrity of piping with large postulated loads and cracks. However, because application of the safety criteria requires system specific input that would vary significantly in LWR piping systems and because there can be significant differences in pipe loads and materials at various other nuclear facilities, our review and conclusions again apply only to the asymmetric LOCA loads issue for the plants named in WCAP 9558, Rev. 2.

Based on our review and evaluation, we have concluded that sufficient technical information has been presented to demonstrate that large margins against unstable crack extension exist for stainless steel PWR primary piping postulated to have large flaws and subjected to postulated safe shutdown earthquake (SSE) and other plant loadings. However, several plants in the owners group previously have not performed seismic analyses to define the SSE loading. These analyses are now being conducted for two domestic facilities as part of the Systematic Evaluation Program. Until the analyses are completed, we will be unable to make a final decision on the affected facilities. For the remaining facilities included in the Westinghouse Owners Group, the safety margins indicate that the potential for failure is low enough so that full double-ended breaks need not be postulated as a design basis for defining structural loads. Also, because the safety margins are large, we tentatively conclude that the facilities not having seismic analyses are conditionally acceptable provided that the seismic analyses confirm that SSE loadings are less than the maximum acceptable levels identified later in this safety evaluation.

The remainder of this safety evaluation includes a summary of the topical reports, our evaluation of the reports, and the bases for our conclusions and recommendations.

3.0 Summary of Topical Reports

The information contained in topical reports WCAP 9558, Rev. 2, and WCAP 9787 included a definition of the plant-specific primary piping loadings; analyses to define the potential for fracture from ductile rupture and unstable flaw extension; materials tests to define the material tensile and toughness properties; and predictions of leak rate from flaws that are postulated to exist in PWR primary system piping. The essential aspects of these areas are summarized below.

3.1 Loads

Reactor coolant pressure boundary (RCPB) piping is required to function under loads resulting from normal as well as abnormal plant conditions. Loads acting on the RCPB piping during various plant conditions include the weight of the piping and its contents, system pressure, restraint of thermal expansion, operating transients in addition to startup and shutdown, and postulated seismic events. In the design of this piping, the limiting loading combination must be determined. The operating facilities that have been evaluated as part of the Westinghouse Owners Group are shown in Table 1.

Based on the loads reported by Westinghouse, bounding loads were defined to envelope the plant-specific loads; these bounding loads were used in the fracture mechanics analyses that were performed to determine the potential for flaw-induced fracture of the primary system piping.

3.2 Fracture Mechanics Analysis

An elastic-plastic fracture mechanics analysis was performed to demonstrate that large margins against double-ended pipe break would be maintained for PWR stainless steel primary piping that contains a large postulated crack and is subjected to large postulated loadings. Key tasks in the analyses were to determine (1) if the postulated flaw would grow larger on the application of the load, and (2) if any additional crack growth that might occur would be stable and not result in a complete circumferential break. The analysis was performed using axial and bending loads that are upper bounds of the loads associated with the facilities identified in Table 1. For analytical purposes,

TABLE 1

Operating Facilities**

Included in Westinghouse A-2 Owners Group

Haddam Neck*
D. C. Cook No. 1 & 2
R. E. Ginna
Point Beach No. 1 & 2
H. R. Robinson
San Onofre No. 1
Surry No. 1 & 2
Turkey Point No. 3 & 4
Yankee Rowe *
Zion No. 1 & 2
Fort Calhoun

*Seismic requirements did not exist for these plants.

**The Owners Group list of operating facilities included a foreign facility, Ringhals No. 2 over which the NRC has no regulatory authority. Thus, we made no formal judgments regarding this facility.

a throughwall crack, seven inches in length around the circumference, was postulated to exist in the pipe at the section where the bounding bending moments and axial forces occur. This flaw is sufficiently large so that it would be very unlikely to exist undetected during normal operation. (As discussed in NUREG-0691 (Ref. 8), no PWR primary coolant system degradation has been detected to date.)

The fracture mechanics analysis required determination of a numerical value for a parameter that represents the potential for the growth, or extension, of a crack in a pipe that is subjected to specific system loads. This parameter is called the J integral (Ref. 1) and is denoted as J. The J integral is typically employed in fracture evaluations where the section containing the flaw undergoes some plastic deformation due to the loading. Extension or growth of an existing flaw occurs when the value of J reaches a critical value called J initiation, which normally is denoted as J_{Ic}.

When extension of the existing crack is predicted, it is necessary to evaluate this extension and determine if it occurs in a stable manner or if the crack will extend in an uncontrolled manner and result in a doubled-ended break. The NRC staff requires that predicted crack extension be evaluated to assess stability. To comply with this requirement, the Owners Group evaluated the predicted crack extension using the tearing stability concept and the tearing modulus stability criterion (Ref. 2). The tearing stability concept is used when the mechanism for flaw extension is ductile tearing. This mechanism can be expected to prevail for the primary piping materials in the Owners Group's facilities which are discussed further in the following sections. The tearing modulus is the parameter used to measure the stability of crack extension and is denoted as T. Tearing modulus is defined as

$$T = \frac{dJ}{da} \frac{E}{\sigma_o^2} \tag{1}$$

where $\frac{dJ}{da}$ indicates the increment of J needed to produce a specified increment of crack extension at any given load and crack state,

E is the material elastic modulus, and

σ_o is the material flow stress defined as one half the sum of the material yield and ultimate strengths

To determine the margin against fracture, the values of J and T are first calculated for the structure using the applied loads and specified crack geometry. The values obtained from the structural analysis create the potential for fracture and are denoted as J applied, or J_{app}, and T applied or T_{app}. The resistance of the structure to fracture is determined experimentally from materials test data that show the relationship between J and crack extension. This relationship is called the J resistance, or J-R, curve. From this curve the material tearing modulus, or the resistance to unstable crack extension, is obtained and is denoted as T_{mat}. At any specified J level greater than J_{Ic}, stable crack extension will occur when

$$T_{mat} > T_{app}$$

The amount by which T_{mat} exceeds T_{app} is a measure of the margin against unstable crack extension or, in this case, the margin against a double-ended break upon application of the loading to the flawed pipe.

Topical report WCAP 9558 contains the results of the analyses performed to determine J_{app} and T_{app}. The value of J_{app} was determined from an elastic-plastic analysis using a finite element computer code. The analysis was based on the bounding load conditions, the postulated seven-inch circumferential throughwall crack, and a lower bound material stress-strain curve obtained at 600°F. The value of T_{app} was obtained using previously developed analytical methods contained in Reference 3.

The material J-R curves used to determine if crack growth would occur under the postulated loading and flaw conditions and to define values of T_{mat} are defined in WCAP 9558 for base metal and in WCAP 9787 for weld metal. The carbon steel safe-end is discussed in the Westinghouse response to ACRS questions (Subject Document No. 3). A summary of the scope of the materials testing follows.

3.3 Materials Testing Program

Base metals representative of those in plants included in the Westinghouse Owners Group were selected for testing. All plants in the Westinghouse Owners Group have wrought stainless steel primary coolant piping except one, which has centrifugally cast stainless steel piping.

Westinghouse selected three heats of cast and three heats of wrought stainless steel for testing. Westinghouse also conducted tests of weld metals to demonstrate that the tensile and fracture toughness properties of the weld metal are comparable to those determined for the base metal in the primary piping system.

A survey of quality assurance files was conducted to identify the primary piping welds in each of the plants in the Owners Group and to define the details of each weld, such as the welding process, electrode size and material, thermal treatment, and other pertinent information. Based on the survey results, a matrix of representative welding parameters was established and a set of six representative welds was fabricated using typical 2.5-inch-thick base plate. The welds were then radiographically examined and heat treated where applicable. Compact tension and tensile specimens were machined from each weld and tested.

Tensile tests were conducted at 600°F using conventional and dynamic loading rates for five of the six heats of base materials. The sixth heat of base material was tested at conventional loading rates only. Weld metal tensile specimens were tested at conventional loading rates for each weld. Dynamic loading rate tests were not conducted for the weld specimen.

J-resistance (J-R) curves to measure material fracture resistance were generated by multiple specimen testing at 600°F using compact tension specimens at conventional and dynamic loading rates for five of the six heats of base metal. J-resistance curves for the sixth heat of base metal and the weld m.terials were generated at 600°F using conventional rates only. The conventional load rate testing and J calculations were performed in accordance with the procedures presented in Reference 4. To perform the dynamic toughness test, Westinghouse used a procedure to stop the tests at predetermined displacements, thus allowing development of a J-resistance curve from multiple-specimen dynamic testing.

A minimum of five specimens were tested at conventional and dynamic loading rates for each of the base metal heats. The base metal specimens were machined from pipe sections and oriented so that the crack would grow in the circumferential direction of the pipe. Westinghouse estimated J_{Ic} and T_{mat} values for each of the heats of materials tested.

The values of J_{Ic} and T_{mat} were estimated from the slopes of the best-fit straight line through the data points for each base metal heat. T_{mat} was then adjusted to account for the nonlinear effects of crack extension using a variation of the incremental correction scheme suggested by Ernst, et al. (Ref. 5). For the fast rate tests, the data points exhibited a large amount of scatter and, in some cases, there were not enough data points to estimate J_{Ic} or T_{mat}. A minimum of three specimens were tested for each weld metal using the same test procedure that was used for the base metal testing. All of the weld metal data points fell within the scatter band of the base metal data points except those for the welds with Inconel filler metal. The data points for the Inconel weld indicated much higher toughness than any of the other base or weld metals. Because of the small number of data points, Westinghouse made no attempt at estimating J_{Ic} or dJ/da values for the weld metals; however, the weld metal data points were fitted with straight lines to demonstrate trends comparable to the base metal.

3.4 Leak Rate Calculations

To comply with the NRC criteria specified in Section 4.1 for defining postulated flaw size, calculations were performed to define the relationship between leak rate and crack opening area. The leak rate calculations were performed to show that a postulated throughwall crack was large enough to produce leaks that could be detected at normal operating conditions by leakage detection devices normally used to detect primary system leakage.

The leak rate calculations were performed using the method developed by Fauske (Ref. 6) for two-phase choked flow; this method was augmented to include frictional effects of the crack surface. An iterative computational scheme was used such that at a given crack opening area and flow rate the sum of the momentum pressure drop (Ref. 6) and the frictional pressure drop was equal to the pressure drop from the primary system pressure to atmospheric (i.e., 2250 - 14.7 psia).

To calculate the frictional pressure drop, the relative surface roughness was estimated from fatigue-cracked stainless steel specimens. The leak rate calculations were performed for a 7-inch-long circumferential throughwall crack at 2250 psi pressure; for conservatism, the bending stress was assumed to be equal to zero for this analysis. The leak rate calculated was approximately 10 gpm.

Although leak rate calculations, especially for small cracks, are subject to uncertainties, the leak rate calculation scheme was correlated with previously generated laboratory data (Ref. 7) and compared with service data from leakage previously detected in the PWR feedwater lines at D. C. Cook and the BWR recirculation line at Duane Arnold. In spite of the uncertainties, the calculated leak rate is sufficiently large so as to have a high probability of detection during normal operation. Further discussion of the leak rate analyses is presented in the Westinghouse response to ACRS questions, the third report listed on page one of this evaluation.

4.0 Evaluation

4.1 NRC Evaluation Criteria

The evaluation of the integrity of PWR primary system piping is based on the margin against ductile rupture and resistance to fracture for a postulated throughwall flaw and loading conditions. To determine the potential for flaw-induced fracture, the staff required the use of analysis methods that (1) included an explicit crack tip parameter, (2) predicted the potential for growth of an existing crack, and (3) determined if any predicted crack extension would occur in a stable manner. These requirements, coupled with the fact that crack extension in ductile piping material likely will result from ductile tearing, led the staff to use the J integral based tearing stability concept as the basis for our evaluation. The tearing stability concept and the associated tearing modulus stability criterion (Ref. 2) have been evaluated previously by the staff and found acceptable for use in the evaluation of LWR piping.

The specific criteria used with the tearing stability analysis to evaluate the integrity of PWR primary system piping and determine if adequate margins against flaw-induced failure and pipe rupture are maintained include the following:

4.1.1 Loading - The loading consists of the static loads (pressure, deadweight and thermal) and the loads associated with safe shutdown earthquake (SSE) conditions.

4.1.2 Postulated Flaw Size - A large circumferential throughwall flaw is postulated to exist in the pipe wall. The circumferential length of the postulated throughwall flaw is to be the larger of either (1) twice the wall thickness or (2) the flaw length that corresponds to a calculated leak rate of 10 gallons per minute (gpm) at normal operating conditions.

Although this safety evaluation has been written exclusively for the primary system piping at the PWR facilities listed in Table 1, cracking potential in LWR piping is system specific and some additional comments are appropriate concerning the generic application of the assumed flaw sizes used in the piping analyses. References 8 and 9 indicate that piping systems other than PWR primary systems have some service history of observed cracking. For these systems, consideration should be given to assuming flaw sizes and shapes different from those specified for the PWR primary system depending on the history of observed service cracking, the potential for cracking, and leak detection capabilities. Specific details of LWR piping systems that are subject to cracking, the mechanism for cracking, the nature of the crack sizes and shapes for these systems, and the effectiveness of flaw and leakage detection methods are presented in References 8 and 9.

The NRC staff concludes that the above evaluation criteria are sufficient to demonstrate the integrity of PWR primary coolant system piping and that, if met, a full double-ended pipe break need not be considered as a design basis to resolve generic safety issue A-2, "Asymmetric Blowdown Loads on PWR Primary System." As noted in Footnote 1 to Appendix A of CFR Part 50, further details relating to the type, size, and orientation of postulated breaks in specific components of the reactor coolant pressure boundary are under development. We do not anticipate that the final criteria will differ significantly from those stated above. Studies and pipe rupture tests have shown that loads far in excess of those specified above still would not result in a double-ended pipe rupture. (These loads might result, for instance, if all the snubbers restraining the steam generators were postulated to fail simultaneously. The staff believes this assumption to be unrealistic and, if utilized, would depend upon further characterization of material and piping behavior for larger crack extensions.) Other abnormal conditions which might affect the evaluation criteria such as waterhammer, stress corrosion cracking or unanticipated cyclic stresses need not be considered for PWR primary coolant main loop piping.

We have reviewed the information provided by Westinghouse relative to the carbon steel safe-ends at the reactor vessel and conclude that our criteria also can apply to this piping-to-vessel interface.

4.1.3 Materials Fracture Toughness

Material resistance to fracture should be based on a reasonable estimate of lower bound properties as measured by the materials resistance (J-R) curve. The lower bound material fracture resistance should be obtained from either archival material of the specific heat of the piping material under evaluation or from at least three heats of material having the same material specification, and thermal and fabrication histories. Both base and weld metal should be tested using a sufficient number of samples to accurately characterize the material J-R curve. To ensure that adequate margins against unstable crack extension exists, the NRC staff concludes that the condition $T_{mat} \geq 3T_{app}$ should be satisfied at the applied J level.

4.1.4 Applicability of Analytical Method

The J-integral and tearing modulus computational methods have certain limits of applicability that are associated with the assumptions and conditions from which they were derived. Generally the limitations are derived from certain stress-strain requirements near the crack tip. These requirements translate into restrictions on structural size and material strength and toughness related parameters and are expressed as (see Refs. 10 and 11)

$$b > 25 \frac{J}{\sigma_o} \quad (2)$$

and

$$\omega = \frac{dJ}{da} \frac{b}{J} >> 1 \quad (3)$$

where b = characteristic structural dimension, in this instance pipe wall thickness;

σ_o = material flow stress;

and $\frac{dJ}{da}$ = slope of the J-R curve at any given value of J.

When satisfied, the conditions specified by equations (2) and (3) are sufficient to ensure that the J-integral and tearing modulus computational methods can be applied in a rigorous manner and that the results are acceptable for engineering application. The requirement in equation (3) that $\omega >> 1$ is somewhat indefinite. Generally, a range of ω between 5 and 10 satisfies this requirement mathematically and is the range used to perform this evaluation. While these requirements are used here, they are not necessary conditions. Less restrictive values (lower values of b and ω) also may be sufficient but will have to be demonstrated to be so by additional data. These data are not now available for the piping materials considered in this investigation.

4.1.5 Net Section Plasticity

The ASME Code specifies margins for pipe stress relative to material yield and ultimate strengths at faulted loading conditions. Because very large flaws may significantly reduce the net load carrying section of the piping, analyses should be performed to demonstrate that the code limits for faulted conditions are not exceeded for the uncracked section of the flawed piping. Flawed piping having net section stresses that satisfy the code limits for faulted conditions are acceptable. When net section stresses do not meet the code limits, additional analyses or action will be required on a case-by-case basis to ensure that there are adequate margins against net section plastic failure.

4.2 Evaluation Results

4.2.1 Loads

The loads used to perform the fracture mechanics analyses for the primary piping include:

axial tension: 1800 KIPS (includes 2250 psi pressure load), and

bending moment: 45,600 in-KIPS.

These loads were derived by "enveloping" the loads obtained from the analyses of record for the highest stressed reactor vessel nozzle/pipe junction of each plant in the Owners Group.

With the exception of several plants indicated in Table 1, the enveloping loads include those from deadweight, thermal, pressure, and safe shutdown earthquake (SSE) conditions. The static loads (pressure, deadweight, and thermal) were combined algebraically and then summed absolutely with the SSE loads.

The exceptions noted in Table 1 reported axial loads and bending moments that are comprised of only normal operating loads (i.e., thermal, deadweight, and internal pressure) and did not include loads associated with the SSE, the major contributor to the bending moment. Our evaluation is predicated on inclusion of the SSE loadings. However, Connecticut Yankee and Yankee Rowe are being evaluated as part of the Systematic Evaluation Program (SEP) and are committed to perform seismic analyses of their RCPB, safe shutdown systems, and engineered safety features using site-specific spectra that will be available in the near future. The completion of such analyses is scheduled for 1983. Confirmation of the margins against unstable crack extension under SSE loading will await the seismic analysis of the RCPB main loop piping for these two facilities.

The development of the enveloping loads, including the analytical models, assumptions, and computer codes, were reviewed and approved by the staff during the licensing process for each Owners Group plant and were not reviewed again as part of this effort. We find that these loads, therefore, are upper bound loads and are acceptable for application in the fracture mechanics evaluation of the RCPB piping.

4.2.2 Materials Properties

Tensile Tests - Tensile tests were conducted at conventional and fast loading rates for the base metals and at conventional loading rates for the weld metals. These tests are relatively straightforward and unambiguous. A comparison of the results from the conventional and fast loading rate tests indicated increased yield and ultimate strengths and decreased percentage in elongation at faster loading rates. Except for the weld with the Inconel filler metal, the yield and ultimate tensile strengths for the weld materials were comparable to those for the base metal. The Inconel weld demonstrated a comparable yield but higher ultimate strength than the base metals. With the exception of the Inconel weld, the percent elongations reported for the weld materials were significantly less than those for the base materials, indicating lower relative ductility for the weldments.

The tensile properties for the actual base metals in the plants and the test program materials were compatable, indicating that the test materials were representative of the in-plant materials. Similarly, the Westinghouse survey of weld materials and techniques was comprehensive and the weld specimens fabricated for testing should be representative of welds in the plants.

Fracture Toughness Testing - Currently, neither an NRC nor a national standard exists for establishing J_{Ic} or J-resistance curves, therefore various methods are employed by different laboratories. All fracture toughness testing in the Westinghouse program was performed using the multiple compact tension specimen procedure outlined in Reference 4.

This procedure is the basis for the proposed J_{Ic} test procedure currently being considered by ASTM Committee E-24 and is generally considered acceptable for determining J_{Ic}. The proposed test procedure recommends calculations for determining J-Integral values and several criteria for ensuring valid J_{Ic} determination. These criteria include considerations of specimen size and data evaluation.

J-Integral Formulation - The expression used by Westinghouse for calculating J for the compact tension specimens has been shown to overestimate the value of J because the experimental data are not corrected for the nonlinear effects of crack growth and plasticity. The effect of this overestimate is to increase calculated values of T_{mat}. In order to account for these effects, Westinghouse applied a correction scheme based on work by Ernst, et al. (Ref. 5). The NRC has reviewed this scheme and found it to be acceptable.

Specimen Size and Geometry - Equations 2 and 3 in Section 4.1.4 specify certain limitations to the applicability of the J-Integral and tearing instability analysis techniques. Because of the high toughness of the heats sampled, not all of the tests satisfied both of these criteria. However, a lower bound J-R curve, discussed later in this section, was developed for the purpose of this evaluation. This lower bound curve typically meets the requirements of equations 2 and 3 over most of the range of analysis. The exception is for higher levels of J where the specimen dimensions were not adequate as specified by equation 2. However, the specimen thickness of 1.65 inches to 2 inches for the base metals and 2.5 inches for the weld metals approximate the actual thickness of the primary coolant piping (2.5 inches). This similarity in thickness simulates the restraint condition in the neighborhood of a crack so that the piping toughness can be represented by the materials test data.

Side grooving of specimens is a related subject of interest. Side grooving increases the degree of triaxiality in the crack tip stress field and has been shown to result in straighter crack fronts during crack extension. Side grooves are desirable when J-resistance curves are developed using the single specimen unloading compliance test or when the data are applied in the evaluation of heavy section structures such as pressure vessels. However, since the specimen dimensions used in these tests approximate the full thickness of the pipes, we conclude that the J-resistance curves developed from specimens without side grooves are acceptable.

Dynamic Tests - The proposed testing procedure used by Westinghouse is intended for quasi-static testing rates. Dynamic toughness tests that were conducted in the Westinghouse program have not previously been performed. Although a full understanding of dynamic fracture toughness in the elastic-plastic regime currently is not available, the significant result of the dynamic tests was that the materials consistently demonstrated greater resistance to crack initiation (higher J_{Ic}) at faster loading rates. However, it is noted that two heats of wrought stainless steel exhibited lower estimated T_{mat} values at the faster loading rates.

Based on our review of the materials test data, we conclude that the proposed J-resistance curve test procedure referenced in the subject documents is acceptable for determining J_{Ic} and T_{mat}. Although the tests conducted did not strictly conform to the criteria recommended in Reference 4, the test specimens and procedures are judged to realistically represent the performance of the actual piping systems. In general, the reported ranges of J_{Ic} and T_{mat} values are acceptable as representative of the structures and materials under consideration.

To perform a generic analysis and account for variations in material behavior, the staff used the data supplied by the Owners Group to define lower bound J-R curves for the piping materials. The data indicated that two lower bound curves were warranted. One lower bound curve was constructed by a composite of the wrought and weld data while the second lower bound curve was defined for the cast material. These two lower bound curves were then used with the analyses described in the next section to evaluate the margin against unstable crack extension for wrought and cast stainless steel piping.

4.2.3 Fracture Mechanics Evaluation

We have reviewed the elastic-plastic fracture mechanics analyses that were submitted by the Owners Group. Our review included independent calculations that were performed to evaluate the acceptability of the Owners Group's conclusions.

To demonstrate that the postulated throughwall flaw would not sustain unstable crack extension during the postulated loading, finite element calculations first were performed by the Owners Group to determine J_{app} as a function of applied bending moment with a constant axial force equal to the bounding value of 1800 kips. The relationship between J_{app} and bending moment provided a convenient means to associate the potential for crack extension with the individual plants listed in Table 1.

We have performed independent calculations to verify the relationship between J_{app} applied bending moment. Our calculations are approximate and are based on elastic methods corrected for plasticity associated with the loading and the presence of the postulated flaw. While our confirmatory calculations are approximations, they do demonstrate that the Owners Group calculations are accurate at lower loads where elastic or small-scale yielding conditions prevail and are conservative at larger loads where plastic deformation occurs. Further, the Owners Group elastic-plastic analysis is conservative because the analysis was performed essentially for a section of pipe as a free body with applied end loads equal to the bounding loads. This is the limiting (conservative) condition relative to system compliance; a pipe in a real system would be in a less compliant situation and would have lower potential for unstable crack extension.

Based on the J_{app} values calculated for the Owners Group by Westinghouse and the lower bound J-R curves defined by the staff from the Owners Group materials data, we find that 7 of the 11 United States facilities listed in Table 1 have sufficient postulated loads to cause extension of the postulated 7-inch-long circumferential throughwall flaw. The loads at the remaining facilities are not high enough to produce extension of the postulated flaw.

Of the seven facilities where crack extension was predicted, one has cast stainless steel piping. Because of the differences in toughness and tensile properties between the wrought, weld, and cast materials, it was necessary to construct two distinct J-R curves. One curve was constructed from cast material while the second was constructed from a composite of the weld and wrought data.

To determine if the crack extension predicted for the seven facilities would be stable, the Owners Group was required to determine the applied tearing modulus, T_{app}. The value of T_{app} was calculated using the methods described in Reference 3. We have performed independent calculations to verify the Owners Group T_{app} calculations using the same methods employed in our J_{app} computations. Again, our results indicate that the Owners Group calculations are conservative. Based on the calculated values of T_{app} and the values of T_{mat} obtained from the J-R curve, we find that large margins against unstable crack extension exist for the seven facilities with predicted crack extension for the postulated flaw sizes and bending loads.

We also have reviewed the method of analyses that have been performed to estimate the leak rate from the postulated flaw size for normal operating conditions. These calculations were performed to satisfy a staff requirement that leak detection capability be included, at least qualitatively, in the piping analyses. Based on our review of the leak rate calculations, we conclude that the calculations presented by the Owners Group represent the state-of-the-art and can be used to qualitatively establish the leak rate for compliance with current staff criteria. The leak rate has been determined to be approximately 10 gpm at normal operating conditions and represents, within reasonable limits of accuracy, detectable leakage rates at operating facilities with their available leakage detection systems or devices. For the purposes of this evaluation, there is no need to backfit Regulatory Guide 1.45 to require seismic qualification since such leakage occurs during normal operating conditions.

Based on our review, we have determined that all the facilities listed in Table 1. with the exception of the two facilities without seismic analyses, satisfy the acceptance criteria defined in Section 4.1. Compliance with the acceptance criteria in Section 4.1 ensures that a large margin against unstable crack extension exists and that the potential for double-ended pipe break is sufficiently low to preclude using it as a design basis for defining structural loads at the facilities listed in Table 1. In addition, the facilities that do not have seismic analyses are found to be conditionally acceptable until the seismic analyses are completed and the loads are defined. Our conditional acceptance is based on: (1) our estimate that the seismic loads are not likely to be higher than those listed for the other facilities in Table 1, (2) the wide margin against unstable fracture that exists at the maximum moments reported by Westinghouse, and (3) the low probility that large loadings will occur prior to completing the seismic analyses.

Based on our review of the analyses and materials data, we conclude that the remaining facilities will satisfy all the criteria in Section 4.1 provided that the bending moment in the welded/wrought piping at these facilities does not exceed 42,000 in-kips. If the seismic analyses indicate bending moments in excess of 42,000 in-kips at these two facilities, additional analyses, materials tests, or remedial measures will be necessary to justify these larger values. It is noted that the 42,000 in-kip limit applies only to welded/wrought piping material; a somewhat lower limit would apply for cast material because of the differences in the lower bound J-R curves. However, the facility having the cast material is acceptable and this note is only intended to caution against the generic use of the 42,000 in-kip limit.

The magnitude of the 42,000 in-kip limit on bending load was determined by finding the largest moment that would satisfy the evaluation criteria specified in Sections 4.1.3 and 4.1.4 for margin on tearing modulus and size requirements, respectively.

At the 42,000 in-kip load, the margin on tearing modulus is satisfied and the value of ω for the test specimens and the primary piping is within the specified range of 5 to 10; however, the value of b for the base metal test specimens is about 30% less than that indicated in equation 2. The lower b value is not a limiting factor in this analysis, however, because as Section 4.2.2 discusses, the specimen thickness is representative of the pipe wall thickness. In addition, the influence of the restriction on size is less than indicated because of the conservatism in the J-integral calculations due to use of a limiting compliance condition.

The values of b and ω chosen by the staff for our evaluation criteria are sufficient conditions and are believed conservative; however, a quantitative estimate of the degree of conservatism cannot be defined without additional experimental data. It is likely that experimental data will show that lower values of ω and b (and higher allowable moment) could be allowed. Experiments now being conducted or planned by the Office of Research, NRC, and industry organizations such as EPRI should help to clarify this matter in the future. These additional data are not necessary to complete this review; however, these additional data will be useful for other studies or for further evaluation of this issue if the bending moments for the remaining facilities are found to exceed 42,000 in-kips.

As indicated in Section 4.1, the staff's evaluation criteria are designed to ensure that adequate margins exist against both unstable flaw extension and net section plasticity of the uncracked pipe section. Both conditions are evaluated because either may be associated with pipe failure depending on the specific pipe load, material, flaw, and system constraint conditions.

Because there may be significant variations or uncertainties associated with these variables, the staff criteria do not attempt to relate margin to actual failure point but is based on maintaining an established margin relative to a combination of conservative bounds for the variables. The margins against actual failure from unstable crack extension are particularly difficult to assess accurately by analysis because the tough materials used in LWR primary piping typically produce data that fail to satisfy the size restrictions of equations (2) and (3) at the very high J levels where failure would be expected to occur.

The 42,000 in-kip limit established by the staff for welded/wrought stainless steel primary PWR piping in Table 1 facilities provides a significant margin against pipe failure. The staff also has reviewed the Owners Group's elastic-plastic analysis and data to provide additional information relative to margin against failure. Based on this review, we conclude that, for the conditions evaluated in this application, the limiting condition is associated with net section plasticity rather than unstable crack extension and that the margin against net section plastic failure is approximately 2.3 relative to the 42,000 in-kip limit and the postulated 7.5-inch circumferential throughwall flaw. This margin also can be translated into an estimate of margin on flaw size of about 5, i.e. the throughwall flaw size corresponding to net section plastic failure at 42,000 in-kips would be about 38 inches long or 140 degrees around the circumference.

5.0 Conclusions and Recommendations

1. Based on our review and evaluation of the analyses submitted for the facilities listed in Table 1, we conclude that the Owners Group has shown that large margins against unstable crack extension exist for stainless steel PWR primary piping postulated to have large flaws and subjected to postulated SSE and other plant loadings. The analytical conditions and margins against unstable crack extension satisfy the criteria established by the staff to ensure that the potential for failure is low so that full double-ended breaks need not be postulated as a design basis for defining structural loads on or within the reactor vessel. Based on compliance with the staff acceptance criteria, we conclude that full double-ended pipe break need not be considered as a design basis to resolve generic safety issue A-2, "Asymmetric Blowdown Loads on PWR Primary System," for the operating facilities identified in Table 1.

2. Seismic analyses are now being performed for the two domestic facilities listed in Table 1; the reactor primary piping at these facilities are conditionally acceptable and full double-ended breaks need not be postulated for resolution of generic issue A-2 provided that the seismic analyses confirm that the maximum bending moments do not exceed 42,000 * in-kips for the highest stressed vessel nozzle/pipe junction.

3. The criteria used to ensure that adequate margins against double-ended break includes the potential to tolerate large throughwall flaws without unstable crack extension so that leakage detection systems can detect leaks in a timely manner during normal operating conditions. To ensure that adequate leak detection capability is in place, the following guidance should be satisfied for the facilities listed in Table 1:

   Leakage detection systems should be sufficient to provide adequate margin to detect the leakage from the postulated circumferential throughwall flaw utilizing the guidance of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," with the exception that the seismic qualification of the airborne particulate radiation monitor is not necessary.

*For all the facilities listed in Table 1, the actual moment is less than 42,000 in-kips and the J_{app} is less than J_{mat} for each facility.

4. The additional information provided by Westinghouse in response to ACRS questions does not alter our conclusions.

6.0 References

1. Rice, J. R. in Fracture Vol. 2, Academic Press. New York, 1968

2. Paris, P. C., et al., "A Treatment of the Subject of Tearing Instability," U.S. Nuclear Regulatory Commission Report NUREG-0311, August 1977.

3. Tada, H., et al., "Stability Analysis of Circumferential Cracks in Reactor Piping Systems," U.S. Nuclear Regulatory Commission Report NUREG/CR-0838, June 1979.

4. Clarke, G. A., et al., "A Procedure for the Determination of Ductile Fracture Toughness Values Using J Integral Techniques," Journal of Testing and Evaluation, JETVA, Vol. 7, No. 1, January 1979.

5. Ernst, H. A., et al., "Estimations on J Integral and Tearing Modulus T from Single Specimen Test Record," presented at the 13th Material Symposium on Fracture Mechanics, Philadelphia, PA, June 1980.

6. Fauske, H. K., "Critical Two-Phase, Steam Water Flows," Proceeding of the Heat Transfer and Fluid Mechanics Institute, Stanford, California, Stanford University Press, 1961.

7. Agostinelli, A. and Salemann, V., "Prediction of Flashing Water Flow Through Five Annular Clearances," Trans. ASME, July 1958, pp. 1138-1142.

8. U. S. Nuclear Regulatory Commission, "Investigation and Evaluation of Cracking Incidents in Piping in Pressurized Water Reactors," USNRC Report NUREG-0691, September 1980.

9. U. S. Nuclear Regulatory Commission, "Investigation and Evaluation of Stress-Corrosion Cracking in Piping of Light Water Reactor Plants," USNRC Report NUREG-0531, February 1979.

10. Begley, J. A. and Landes, J. D., in Fracture Analysis, ASTM STP 560, American Society for Testing and Materials, 1974, pp. 170-186.

11. Hutchinson, J. W. and Paris, P. C., "Stability Analysis of J-Controlled Crack Growth," Elastic-Plastic Fracture, ASTM STP 668, American Society for Testing and Materials, 1979, pp. 37-64.

ENCLOSURE 3

BACKGROUND INFORMATION FOR CRGR REVIEW

The following information is provided in the format specified in NRR Office Letter No. 39, Revision 1 - "NRR Procedures for Control and Review of Generic Requirements," December 15, 1982. For each item of information, the request is stated and is followed by the response.

1. The proposed generic requirement as it is proposed to be sent out to licensees.

   We propose to issue the topical report evaluation via the letters which are attached to this CRGR review package as Enclosure 1.

2. Draft staff papers or other underlying staff documents supporting the requirements.

   The supporting information for the topical report evaluation is provided in the enclosed program package. Copies of any references listed will be made available upon request.

3. A brief description of each of the steps anticipated that licensees must carry out in order to complete the requirements; e.g.,

3.1. Are there separate short-term and long-term requirements?

There are no "short term" actions that will be implemented on an interim basis until superseded by "long term" actions.

3.2. Is it the definitive, comprehensive position on the subject or is it the first of a series of requirements to be issued in the future?

The staff's position on the subject is clearly described in the topical report evaluation, Enclosure 2 to this review package. It represents the definitive, comprehensive position for implementation of multiplant issue D-10, (USI: A-2) for the licensee's facilities identified in Table 1 of Enclosure 2.

3.3. How does this requirement affect other requirements? Does this requirement mean that other items or systems or prior analyses need to be reassessed?

The staff's position constitutes a relaxation of the requirements stemming from NUREG-0609 in terms of postulating double-ended breaks in PWR primary coolant main loop piping and any subsequent need for installation of piping restraints. No other requirements are affected (e.g., design bases for containment or ECCS due to a LOCA). If any identified licensee chooses to submit an exemption request, (i.e., to GDC 4 of 10 CFR Part 50) one of the conditions specified for justification will require a reassessment of the unit's leakage detection capability. The other condition relates to seismic reanalysis of the RCPB for two plants in the Systematic Evaluation Program. These reanalyses are in progress.

If other PWR licensees or applicants choose to submit deterministic fracture mechanics evaluations of their main loop RCS piping which are comparable to that of the Westinghouse topical reports, the staff will evaluate these with regard to justification for exemptions to the regulations.

3.4. Is it only computation? Or does it require or may it entail engineering design of a new system or modification of any existing systems?

There is no requirement for engineering design of a new system or identified modification of any existing systems. Refer to Enclosure 4 to this package as it relates to leakage detection systems.

3.5. What plant conditions are needed to install, conduct preoperational tests and declare operable?

Licensees who elect to upgrade their leakage detection capability may need to recalibrate existing instrumentation. This task plus operational tests and declaration of operability could be performed during a planned outage.

3.6. Is plant shutdown necessary? How long?

No unscheduled plant shutdowns are necessary.

3.7. Does design need NRC approval?

All requests for exemptions to the regulations will be reviewed and evaluated by the NR staff.

3.8. Does it require new equipment? Is it available for purchase in sufficient quantity by all affected licensees or must such equipment be designed? What is the lead time for availability?

There is no identified need for new equipment.

3.9. May it be used upon installation or does it need staff approval before use? Does it need tech. spec. changes before use?

Exemption requests which involve upgrades to leakage detection systems may require technical specification changes (e.g., surveillance requirements).

4. Identification of the category of reactors to which the generic requirement is to apply (that is, whether it is to apply to new plants only, new OLs only, OLs after a certain date, OLs before a certain date, all OLs, all plants under construction, all plants, all water reactors, all PWRs only, some vendor types, some vintage types such as BWR 6 and 4, jet pump and nonjet pump plants, etc.)

The operating PWRs addressed by the topical report evaluation, Enclosure 2 to this CRGR review package, are identified in Table 1 of that evaluation as listed below:

Haddam Neck
D. C. Cook No. 1 & 2
R. E. Ginna
Point Beach No. 1 & 2
H. B. Robinson
San Onofre No. 1
Surry No. 1 & 2
Turkey Point No. 3 & 4
Yankee Rowe
Zion No. 1 & 2
Fort Calhoun

5. For each such category of reactor, the following information should be provided:

5.1. A value impact analysis prepared in accordance with Office Letter No. 16.

Refer to Enclosure 4 of this CRGR review package.

5.1.1. A risk reduction assessment performed using a data base and methodology commonly accepted within NRC.

Refer to Enclosure 4 of this CRGR review package.

5.1.2. An assessment of costs to NRC, and assessment of cost to licensees, including resulting occupational dose increase or decrease, added plant and operational complexity, and total financial costs.

Refer to Enclosure 4 of this CRGR review package.

5.2. Other information:

5.2.1. Consistent with the first two items above, provide the basis for requiring or permitting implementation by a given date or on a particular schedule.

The staff is proposing an implementation plan and schedule consisting of the following elements. Letters to Westinghouse and to the eleven licensees previously identified will inform them of the results of the staff's topical report evaluation and transmit a copy of that evaluation. As specified in these letters and in the topical report evaluation, Enclosures 1 and 2, respectively, justification for an exemption to the pertinent regulations is contingent upon satisfying the staff's leakage detection criteria and in addition, for two licensees under the Systematic Evaluation Program, the results of seismic reanalyses. Exemption requests will be treated as routine licensing actions.

5.2.2. Other acceptable implementation schedules and the basis therefor. This should include sufficient information to demonstrate that the schedules are realistic and provide sufficient time for in-depth engineering, evaluation, design procurement, installation, testing, development of operating procedures, and training of operators.

The proposed implementation and scheduling plan is believed to be the most practical and expeditious approach.

5.2.3. Schedule for staff actions involved in completion of requirement (based on hypothesized effective date of approval).

Exemption requests submitted as a result of CRGR approval of this package will be treated as routine licensing actions. The NRC project manager for each plant, on the basis of knowledge of the overall work effort at a plant and on the basis of guidance received from NRC management, shall reach agreement on schedules which optimize use of NRC and utility resources.

5.2.4. The Value-Impact assessment in Enclosure 4 presents the values of the safety benefit and cost factors considered in prioritization of generic safety issues. The assessment results demonstrate that the safety benefit to cost ratio is very low, 4-5 man-rem ÷ \$50 million considering only the public risk reduction and increased industry and NRC implementation costs if plant modifications were to be required to mitigate consequences of asymmetric pressure loads resulting from a possible primary system double-ended guillotine pipe break. The assessment also presents additional significant results that should be considered in determining a priority ranking. Exempting the plants from the above modifications would (a) avoid about 11000 man-rem in industry installation occupational exposure; (b) avoid about \$60 million in power replacement costs due to extended plant outages; (c) avoid about \$400 thousand in NRC review costs; however, (d) such an exemption could increase public and onsite property damage costs by about \$40 thousand in present dollars discounted at 10%. On balance, the dose and cost net benefits indicate that the proposed exemption basis should be recommended.

Considering the Value-Impact assessment results above, the piping fracture mechanics evaluation presented in Enclosure 2 and the fact that this generic issue has been identified since 1975, implementing the proposed action and resolving this issue should be given MEDIUM priority.

5.2.5. For proposed requirements involving reports and/or record keeping, an assessment of whether such reporting or record keeping is the best means of implementation and the appropriate degree of formality and detail to be imposed.

The submittal by licensees of an exemption request to NRR is the most expedient method, both for NRC and for licensees, of acquiring the information for justification of NRC approval. There are no additional recurring reporting requirements associated with this issue. The requirements for reporting already contained in section 6.9 of the Standard Technical Specifications and the requirements of 10 CFR Part 50, Appendix B for record keeping will, of course, continue to apply to safety-related activities falling within the scope of STS 6.9 and Appendix B.

5.2.6. To the extent that the category contains plants of different types or vintages, the items listed above shall be provided for each type and vintage, or justification shall be provided demonstrating that the analysis of each item is valid for all types and vintages covered.

The responses to parts 4 and 5 are believed to adequately address this issue.

6. Each proposed requirement shall contain the sponsoring Office's position as to whether the requirement implements existing regulations or goes beyond them.

The staff's position justifies an exemption for the identified licensees, to General Design Criterion 4, "Environmental and Missile Design Bases", in the context of the definition of a LOCA which includes a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant systems (Appendix A to 10 CFR Part 50). The staff's position is within General Design Criterion 30, "Quality of Reactor Coolant Pressure Boundary" as it relates to the detection, identification, and monitoring of the source of reactor coolant leakage.

7. The proposed method of implementation along with the concurrence (and any comments) of OELD on the method proposed.

   Refer to item 5.2.1. OELD has not identified objections to this proposed implementation method.

8. Regulatory analysis sufficient to address the Paperwork Reduction Act, the Regulatory Flexibility Act and Executive Order 12291.

   An OMB clearance package is not required since the staff's position does not impose requirements or request information [e.g., 50:54(f)]. The transmittal of the staff's topical report evaluation stipulates the conditions which the staff considers justification for an exemption request at the option of the identified licensee.

ENCLOSURE 4

Regulatory Analysis of Mechanistic Fracture Evaluation of Reactor Coolant Piping A-2 Westinghouse Owner Group Plants

ENCLOSURE 4

Regulatory Analysis of Mechanistic
Fracture Evaluation of Reactor Coolant Piping
A-2 Westinghouse Owner Group Plants

1. Statement of the Problem

The problem of asymmetric blowdown loads on PWR primary systems results from postulated rapid-opening, double-ended guillotine breaks (DEGB) at specific locations of reactor coolant piping. These locations include the reactor pressure vessel (RPV) nozzle-pipe interface in the annulus (reactor cavity) between the RPV and the shield wall plus other selected break locations external to the reactor cavity. These postulated ruptures could cause pressure imbalance loads both internal and external to the primary system which could damage primary system equipment supports, core cooling equipment or core internals and thus contribute to core melt frequency.

This generic PWR issue, initially identified to the staff in 1975, was designated Unresolved Safety Issue (USI) A-2 and is described in detail in NUREG-0609 which provides a pressure load analysis method acceptable to the staff.

The plants to which this analysis applies are the A-2 Westinghouse Owner Group plants identified in Enclosure 2.

2. Objective

The objective of this proposed action is to demonstrate that deterministic fracture mechanics analysis which meets the criteria evaluated in Enclosure 2 is an acceptable alternative to (a) postulating a DEGB, (b) analyzing the structural loads, and (c) installing plant modifications to mitigate the consequences in order to resolve issue A-2. Demonstrating by acceptable fracture mechanics analysis that there is a large margin against unstable extension of a crack in such piping, (leak before break) contingent upon satisfying the staff's leak detection criteria, will establish a technical justification for the identified plants to be exempted from General Design Criterion 4 in regard to the associated definition of a LOCA. Section 4 below provides a Value-Impact assessment of this alternate method for resolving issue A-2 for these plants.

3. Alternative

The major alternative to the proposed action would be to require each operating PWR to add piping restraints to prevent postulated large pipe ruptures from resulting in full double ended pipe break area, thus reducing the blowdown asymmetric pressure loads and the need to modify equipment supports to withstand those loads as determined in plant specific analysis reported in WCAP-9628 and WCAP-9748, "Westinghouse Owners Group Asymmetric LOCA Loads Evaluation" (Evaluation of DEGB outside and inside the reactor cavity respectively).

4. Consequences

A. Costs and Benefits

I. Introduction

A detailed Value-Impact (V-I) assessment of the proposed alternate resolution of issue A-2 for the 16 Westinghouse A-2 Owners Group plants has been completed by PNL and is attached to this enclosure. The V-I assessment uses methods and data suggested in the February 1983 draft of proposed Handbook for Value-Impact Assessment (PNL4646) and in NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development." The nominal estimate results, major assumptions, uncertainties, and conclusions of the assessment are discussed in Sections II, III, and IV below. The results of the upper and lower estimates are included in the table in Section IV below.

II. Values-Public Risk and Occupational Exposure

A. Results

The estimated reduction in public risk for installing additional pipe restraints and modifying equipment supports as necessary to mitigate or withstand asymmetric pressure blowdown loads is very small, only about 3½ man-rem total for the nominal case for all 16 plants considered. Similarly, the reduction in occupational exposure associated with accident avoidance due to modifying the plants is estimated to total less than 1 man-rem. These small changes result from the estimated small reduction in core-melt frequency of $1 \times 10^{-}$[illegible] events/reactor-year that would result from modifying the plants. However, the occupational exposure estimated for installing and maintaining the plant modifications would increase by 11,000 man-rem. Consequently, the savings in occupational exposure by not requiring the plant modifications far exceed the potentially small increase in public risk and avoided accident exposure associated with requiring the modifications.

B. Major Assumptions

The above estimated changes in public risk and accident avoided occupational exposure were obtained by examining WASH-1400 accident sequences leading to core melt from reactor pressure vessel (RPV) rupture and large LOCA's in conjunction with the major assumptions identified below.

1. If a DEGB occurs inside the reactor cavity, it could displace the RPV, possibly rupturing it or other piping, or disrupt core geometry which could lead directly to core melt in accident sequences analagous to those for RPV rupture in WASH-1400.

2. A DEGB in the primary system outside the reactor cavity could lead to core melt through the additional risk contribution from subsequent safety system failures, such as ECCS, induced by previously unanalyzed asymmetric pressure loads on equipment or from core geometry disruptions. It was assumed that failure of safety systems independent of asymmetric pressure loading is already accounted for in the plant design.

3. Three sources of data were used to develop estimates of DEGB frequencies for large primary system piping used in the analysis. These frequency estimates range from an upper estimate of 10^{-5} breaks per reactor year down to a lower estimate of 7×10^{-12} breaks in a reactor lifetime.

   The upper estimate of 10^{-5}/reactor-year is based on a paper on nuclear and non-nuclear pipe reliability data in IAEA-SM-218/11, dated October 1977 by S. H. Bush which indicates a range of 10^{-4} to 10^{-6} per reactor-year. Additional data in the paper indicates that 10^{-5} may be 100 times too high for the pipe size being considered in issue A-2.

   An intermediate or nominal estimate of 4×10^{-7} per reactor-year for primary system piping outside the reactor cavity and 9×10^{-8}/reactor-year for piping inside the reactor cavity are based on Report SAI-001-PA dated June 1976 prepared by Science Applications Inc. which modeled crack propagation in piping subject to fatigue stresses. These values represent an average over a 40-year plant life for a two loop plant and conservatively ignore in-service inspection as a method to discover and repair cracks prior to unstable propagation.

The lower estimate is based on NUREG/CR-2189, Vol 1, dated September 1981 prepared by LLL. The report uses simulation techniques to model crack propagation in primary system piping due to thermal, pressure, seismic and other cyclic stresses. The report indicates that the probability of a leak is several orders of magnitude more likely than a direct* seismically induced DEGB which is estimated to have a probability of 7×10^{-12} over a plant lifetime. For this analysis the lower estimate of 7×10^{-12} is considered essentially zero.

It is acknowledged that both the upper and nominal estimate DEGB frequencies used in this analysis are less than the WASH-1400 large LOCA median frequency of 1×10^{-4}/reactor-year. However, the upper estimate of 10^{-5}/reactor-year is consistent with WASH-1400 median assessment pipe section rupture data. A review of the 16 plants under consideration indicates there are an

---

*Later work (to be published) by LLL indicates that an indirect seismically induced DEGB (e.g., earthquake-induced failure of a polar crane or heavy component support-steam generator or RC pump) is more probable ranging from 10^{-5} to 10^{-10}/reactor-year with a median of 10^{-7}/reactor-year for plants east of the Rockies. Since the nominal DEGB frequency obtained from the IAEA paper approximates the median indirect DEGB frequency, the direct DEGB estimate of 7×10^{-12} over a plant lifetime was used for the lowewr estimate.

average of 10.3 sections of primary system piping per reactor. Multiplying this value by 8.8×10^{-7} rupture/section-year for large (>3") pipe obtained from Table II 2-1 results in an estimate of 9×10^{-6} rupture/reactor-year. The following table identifies several factors associated with issue A-2 compared to the data base used for WASH 1400 that support use of a lower pipe break frequency:

Factor	W A-2 Plants	WASH-1400 Large LOCA
Pipe size	>30" diameter	> 6" diameter
Pipe material	Austenitic stainless steel	Carbon steel and stainless steel
System and Class of pipe	Only Class I primary system pipe with nuclear grade QA and ISI	Miscellaneous primary and secondary system piping of various classifications
Type of failure	Double-ended guillotine (DEG) break only	Circumferential and longitudinal breaks, large cracks
Failure location	Selected primary system break locations	Random system break locations
Leak detection system (LDS)	LDS capability to detect leak in a timely manner to maintain large margin against unstable crack extension	No requirement or provision for leak detection

4. Public dose estimates for the release categories were derived using the CRAC-2 code and assuming the quantities

of radioactive isotopes as used in WASH-1400, the meteorology at a typical midwestern site (Byron-Braidwood), a uniform population density of 340 people per square-mile (which is an average of all U.S. nuclear power plant sites) and no evacuation of population. They are based on a 50-mile release radius model.

5. The change in occupational exposure associated with accident avoidance assumes 20,000 man-rem/core melt to clean up the plant and recover from the accident as indicated in NUREG/CR-2800, Appendix D.

6. The estimated occupational exposure associated with installing and maintaining plant modifications considers the plants into two groups. One group of three plants requires extensive modifications according to Westinghouse A-2 Owners Group asymmetric load analysis (WCAP 9628). The modifications consisted of added RPV nozzle-pipe restraints and substantial modification of all steam generator and pump supports. The occupational exposures for these modifications were based on an estimate of 2600 man-rem submitted by San Onofre 1 for modifying three loops. The load analysis for the remaining 13 plants indicates less required plant modification consisting primarily of RPV nozzle-pipe restraints with minor modification of steam generator and/or pump supports for some of the plants. Recalibration of the leak detection systems to assure leak detection capability is assumed to be required at 14 of the 16 plants and would incur about 200 man-rem total.

III. Impacts - Industry/NRC Costs - Property Damage

A. Results

The estimated industry costs to install plant modifications to withstand asymmetric pressure loads is about $50 million. It is, also estimated that power replacement costs would be an additional $60 million since the plant modifications would be extensive and involve working in areas with limited equipment access and significant radiation levels so that the work would probably extend plant outages beyond normal planned shutdowns. Also, it is estimated that maintenance and inspection of the modifications for the remaining life of all the plants would cost $650K to $1 million in present dollars based on discounting at 10% and 5% respectively. The cost for recalibrating leak detection systems is estimated at about $350K. The above costs do not include the industry costs expended to date to perform asymmetric pressure load analysis and fracture mechanics analysis. These analyses costs are considered small compared to the plant modification and power replacement cost indicated above.

It is estimated that it would cost NRC about $800K in staff review effort if plant modifications to withstand asymmetric pressure loads were to be installed. If they are not installed and this cost is saved, then it is estimated that NRC cost would be $400K to review leak detection system calibration work and plant technical specification revisions Exempting the plants from installing modifications would result in a net saving of $400K in NRC costs.

It is estimated that installing plant modifications to withstand asymmetric pressure loads would avoid public property damage costs due to an accident by $24K to $38K total in present dollar for all the plants based on a discounting at 10% and 5% respectively. Similarly the avoided onsite property damage cost avoided is estimated at $15K to $29K in present dollars.

Considering the impacts identified above, it is apparent that the industry and NRC costs savings by not requiring the plant modifications far exceed the small increases in public and onsite property damage costs due to a potential accident.

B. Major Assumptions

1. The costs for installing the plant modifications were determined by separating the plants into two groups. The cost for the first group of three plants which require extensive modifications used an estimate submitted by San Onofre Unit 1 which was prorated to the other two plants based on the number of primary loops in each plant. The costs for the remaining 13 plants which would require less modification are derived from Report UCRL-15340 "Costs and Safety Margin of the Effects of Design for Combination of Large LOCA and SSE Loads," and from industry estimates including informal estimates from DC Cook. The estimates were adjusted to 1982 dollars.

2. The cost estimates for public and onsite property damage due to an accident were calculated by multiplying the change in core melt frequency by a generic property damage estimate. This damage estimate was obtained by using the methods and data in NUREG/CR2723, "Estimates of the Financial Consequences of Nuclear Power Reactor Accidents." Public risk upper and lower bound variations are related to Indian Point 2 and Palo Verde values calculated from NUREG/CR 2723.

3. Power replacements costs were based on an assumed $300K per plant outage day.

IV. Conclusions

The results of the Value-Impact assessment are summarized in the table below. In the table, values are those factors relating directly to the NRC role in regulating plant safety, such as reduced public risk or reduced occupational exposure, and are indicated as positive when the results of the proposed action improve plant safety. Impacts are defined as the costs incurred as a result of the proposed action and indicated as positive when the resulting costs are increased.

From the table, the main conclusion to be made is that the dose and cost net benefits indicate that not requiring installation of plant modifications to mitigate consequences of asymmetric pressure loads resulting from a possible primary system DEG pipebreak would result in very little increase in public risk and accident avoided occupational exposure (less than 5 man-rem) and would avoid significant plant installation occupational exposure (11,000 man-rem) and industry and NRC costs ($110 million - including $60 million power replacement cost). Three additional observations are worth noting:

a) the uncertainty bounds show net positive benefits for either dose or cost. The upperbound is very positive.

b) This assessment does not address costs of core or core support modifications. Adding these costs would increase the avoided cost.

c) The cost results are not sensitive to discount rates used in this assessment.

The detailed PNL Value-Impact assessment is attached to this enclosure.

LEAK BEFORE BREAK VALUE-IMPACT SUMMARY - TOTAL FOR 16 PLANTS

	Dose (man-rem)			Cost ($)		
Factors	Nominal Estimate	Lower Estimate	Upper Estimate	Nominal Estimate	Lower Estimate	Upper Estimate
Values (man-rem)						
Public Health	-3.4	0	-37	-	-	-
Occupational Exposure (Accidental)	-0.8	0	-30	-	-	-
Occupational Exposure (Operational)	$+1.1\times10^4$	+3500	$+3.2\times10^4$	-	-	-
Values Subtotal	$+1.1\times10^4$	+3500	$+3.2\times10^4$	-	-	-
Impacts ($)						
Industry Implementation Cost[a]	-	-	-	-50×10^6	-25×10^6	-75×10^6
Industry Operating Cost	-	-	-	-6.5×10^5	-3.3×10^5	-9.8×10^5
NRC Development and Implementation Cost[b]	-	-	-	-4.0×10^5	-2.0×10^5	-6.0×10^5
Power Replacement Cost	-	-	-	-60×10^6	-30×10^6	-90×10^6
Public Property	-	-	-	$+2.4\times10^4$	0	$+2.6\times10^6$
Onsite Property	-	-	-	$+1.5\times10^4$	0	$+4.6\times10^5$
Impact Subtotal	-	-	-	-110×10^6	-55×10^6	-165×10^6

(a) Does not include industry costs expended to date to prepare plant asymmetric pressure load analyses and pipe fracture mechanics analysis.

(b) Does not include NRC cost expended to date to develop issue (NUREG-0609) and to evaluate Westinghouse pipe fracture mechanics analysis.

B. Impact on Other Requirements

The impact of the proposed action on other requirements is discussed in Section 3.3 of Enclosure 3.

C. Constraints

Constraints affecting the implementation of the proposed action are discussed in Sections 3.5 thru 3.9 and 5.2.1, 5.2.2, and 5.2.3 of Enclosure 3.

5. Decision Rationale

The evaluation in Enclosure 2 demonstrates that for the A-2 Westinghouse Owner Group Plants there is a large margin against unstable crack extension for stainless steel PWR large primary system piping postulated to have large flaws and subjected to postulated SSE and other plant loads. Having leak detection capability in each of the plants comparable to the guidelines of Regulatory Guide 1.45 (except for seismic I Category air particle radiation monitoring system) assures detecting leaks from throughwall pipe cracks in a timely manner under normal operating conditions; thus maintaining the large margin against unstable crack extension.

Also, the Value-Impact assessment summarized above indicates that there are definite dose and cost net benefits in not requiring installation of plant modifications to mitigate consequences of a possible primary system piping DEG break.

6. Implementation

The steps and schedule for implementation of the proposed action are discussed in Sections 3.5 thru 3.9 and 5.2.1, 5.2.2, 5.2.3 of Enclosure 3.

Attachment to Enclosure 4

LEAK BEFORE BREAK VALUE-IMPACT ANALYSIS

1. INTRODUCTION

This report presents a value-impact assessment of the consequences of exempting Westinghouse A-2 Owners Group plants from having to install modifications to mitigate asymmetric blowdown loads in the primary system. This assessment uses methods suggested in the Handbook for Value-Impact Assessment (Heaberlin et al. 1983) and data developed for safety issue prioritization (Andrews et al. 1983). The assessment relies heavily upon existing industry and NRC reports generated for Generic Task Action Plan (GTAP) A-2, Asymmetric Blowdown Loads on PWR Primary Systems (Hosford 1981).

The proposed action will efficiently allocate public resources in the generation of electric power and avoid occupational dose with only small increments to public risk. Modification of plant designs to accommodate asymmetric loads in primary systems of selected Westinghouse plants would incur large costs and significant occupational doses for insignificant gains to public safety.

Generic Safety Issue A-2 deals with safety concerns following a postulated major double-ended pipe break in the primary system. Previously unanalyzed loads on primary system components have the potential to alter primary system configurations or damage core cooling equipment and contribute to core melt accidents. For postulated pipe breaks in the cold leg, asymmetric pressure changes could take place in the annulus between the core barrel and the RPV. Decompression could take place on the side of the reactor pressure vessel (RPV) annulus nearest the pipe break before the pressure on the opposite side of the RPV changed. This momentary differential pressure across the core barrel induces lateral loads both on the core barrel it lf and on the reactor vessel. Vertical loads are also applied to the core intern ls and to the vessel because of the vertical flow resistance through the core ; l asymmetric axial decompression of the vessel. For breaks in RPV nozzles, the annulus between the reactor and biological shield wall could become asymmetrically pressurized, resulting in additional horizontal and vertical external loads on the reactor vessel. In addition, the reactor vessel is loaded simultaneously by the effects of strain-energy release and blowdown thrust at the pipe break. For breaks at reactor vessel outlets, the same type of loadings could occur, but the internal loads would be predominantly vertical because of the more-rapid decompression of the upper plenum. Similar asymmetric forces could also be generated by postulated pipe breaks located at the steam generator and reactor-coolant pump. The blowdown asymmetric pressure loads have been analyzed and reported in WCAP-9628 (Campbell et al. 1980) and WCAP-9748 (Campbell et al. 1979), "West ighouse Owners Group Asymmetric LOCA Loads Evaluation."

2.0 PROPOSED ACTION AND POTENTIAL ALTERNATIVES

It is proposed that Westinghouse A-2 Owner Group plants listed in Enclosure 2 be exempted from plant modifications to mitigate asymmetric blowdown loads to primary system components. This proposal is based on consideration of public risk, occupational dose and cost impacts. The alternative would be to require each operating PWR to add piping restraints and primary system component supports to withstand the blowdown asymmetric pressure loads.

Public risk reductions for installing/modifying equipment to mitigate asymmetric blowdown loads are small. Extensive analyses of pipe material properties and crack propagation by industry (WCAP-9558 and WCAP-9787, Campbell et al, 1982 and 1981) and the NRC indicate that catastrophic failures without through-the-wall cracks are extremely unlikely. It is proposed that these plants upgrade leak detection systems, as necessary, to provide adequate leak detection capabilities. This will allow cracks to be identified and repaired before they propagate to major failures. Plant modifications would increase occupational dose and inspection time for primary system components. The reduction in the frequency of core-melt accidents and avoidance of post-accident doses as a result of the plant modifications is not significant.

Cost impacts for equipment to mitigate asymmetric blowdown loads are plant dependent. In the worst case, they cost many millions of dollars, require replacement power purchases and are of questionable feasibility. Some plants considered can handle asymmetric loads with few changes. However, all plants will realize cost savings for the proposed action.

3.0 AFFECTED DECISION FACTORS

Decison Factors	Causes Quantified Change	Causes Unquantified[a] Change	No Change
Public Health	X		
Occupational Exposure (Accidental)	X		
Occupational Exposure (Routine)	X		
Public Property	X		
Onsite Property	X		
Regulatory Efficiency			X
Improvements in Knowledge			X
Industry Implementation Cost	X		
Industry Operation Cost	X		
NRC Development Cost	X		
NRC Implementation Cost	X		
NRC Operation Cost	X		

(a) In this context, "unquantified" means not readily estimated in dollars.

4.0 VALUE-IMPACT ASSESSMENT SUMMARY - Total for 16 Plants

Decision Factors	Nominal Estimate	Lower Estimate	Upper Estimate
Values[a] (man-rem)			
Public Health	-3.4	0	-37
Occupational Exposure (Accidental)	-0.8	0	-30
Occupational Exposure (Operational)	1.1E+4	3500	3.2E+4
Regulatory Efficiency	N/A		
Improvements in Knowledge	N/A		
Total Quantified Value	1.1E+4	3500	3.2E+4
Impacts[b] ($)			
Industry Implementation Cost[c]	-1.1E+8	-5.3E+7	-1.6E+8
Industry Operating Cost	-6.5E+5	-3.3E+5	-9.8E+5
NRC Development Cost[d]	0	0	0
NRC Implementation Cost	-4.0E+5	-2.0E+5	-6.0E+5
NRC Operation Cost	0	0	0
Public Property	2.4E+4	0	2.6E+6
Onsite Property	1.5E+4	0	4.6E+5
Total Quantified Impact	-1.1E+8	-5.3E+7	-1.6E+8

(a) A decision term s a value if it supports NRC goals. Principle among these goals is the regulation of safety.
(b) Impacts are defined as the costs incurred as a result of the proposed action. Negative impacts indicate cost savings.
(c) Does not include industry cost expended to date (fracture mechanics and plant asymmetric pressure load analyses). Replacement power costs of $60M are included.
(d) Does not include NRC costs to evaluate asymetric loads (Hosford 1981) or industry fracture mechanics (Campbell 1982).
N/A = Not Affected

5.0 UNQUANTIFIED RESIDUAL ASSESSMENT

There are no unquantified decision factors in the assessment of this action.

6.0 DEVELOPMENT OF QUALIFICATION

A. Public Health

A risk analysis was performed to assess the effects of exempting Westinghouse GTAP A-2 owner group plants from modifications to mitigate asymmetric blowdown loads on primary system components. This was accomplished by examining WASH-1400 accident sequences leading to core melt from vessel rupture and large LOCAs.

For this analysis, it was assumed that a double-ended guillotine (DEG) large LOCA can occur either inside or outside the reactor cavity. In addition to the "standard" stresses caused by a large LOCA (depressurization and loss of coolant inventory), the DEG break can have additional effects:

1. If the DEG break occurs inside the reactor cavity, it can cause an asymmetric blowdown which displaces the reactor vessel, possibly rupturing other pipes or the vessel itself.

2. If the DEG break occurs anywhere in the primary loop, it can cause an asymmetric blowdown which 1) displaces the core such that its geometry becomes uncoolable and/or 2) fails needed emergency core cooling system (ECCS) piping through dynamic blowdown forces.

Three sources of data were used to develop estimates of DEG break probabilities used in this analysis. These probability estimates range from an upper estimate of 1E-5 breaks per reactor year down to a lower estimate of 7E-12 breaks in a reactor lifetime.

The upper estimate is based on a study of nuclear and non-nuclear pipe reliability data (Bush 1977). This data indicates a range of 1E-4 to 1E-6 failures per reactor year. Failures considered include leaks, cracks, ruptures, disruptive and potentially disruptive. Bush indicates values of 1E-5 to 1E-6 are representative of disruptive failures. A value of 1E-5 was used in this analysis as an upper estimate. Additional data presented by Bush indicates that this value may be 100 times too high for the pipe sizes being considered in the proposed action.

An intermediate or nominal estimate is based on a study by SAI (Harris and Fullwood 1976) that modeled crack propagation in piping that is subject to fatigue stresses. While the study was done for Combustion Engineering plants, the approach and data are not plant specific. Conservatively ignoring in-service inspection as a method to discover and repair cracks prior to unstable propagation, SAI reports DEG break frequency estimates of 4E-7/py for the primary system and 9E-8/py in the reactor cavity averaged over a 40-year plant life for a two loop plant (Figure 23, Harris and Fullwood 1976).

The lower estimate of a LOCA was developed by Lawrence Livermore Laboratories (Lu et al. 1981) using simulation techniques to model direct effects on crack propagation in primary system piping due to thermal, pressure,seismic and other cyclic stresses. Indirect effects such as external mechanical damage were not included. Results indicate leaks are several orders of magnitude more likely than breaks and that breaks have a probability of 7E-12 over a plant lifetime. This value is essentially zero for risk calculation purposes, so no additional lower estimate calculations were performed.

It is acknowledged that both the upper and nominal estimate DEG break frequencies used in this analysis are less than the WASH-1400 large LOCA median frequency of 1E-4/reactor-yr. However, the upper estimate of 1E-5/reactor-year is consistent with WASH-1400 median assessment pipe section rupture data. A review of the 16 plants under consideration indicates there are an average of 10.3 sections of primary system piping/reactor. Multiplying this value by 8.8E-7 rupture/section-year for large (>3") pipe obtained from Table III 2-1 results in an estimate of 9E-6 ruptures/reactor-year. There are several additional factors associated with this issue compared to the data used for WASH-1400 that support use of a lower pipe break frequency. These factors are tabulated below:

Factor	Westinghouse A-2 Owners Group Plants	WASH-1400 Large LOCA
Pipe size	- >30 inches diameter	- >6 inches diameter
Pipe material	- austenitic stainless steel	- carbon steel and stainless steel
System and class of pipe	- only class I primary system pipe with nuclear grade QA and ISI	- miscellaneous primary and secondary system piping of varying classification
Type of failure	- double ended guillotine (DEG) break only	- circumferential and longitudinal breaks, large cracks
Failure location	- selected primary system break locations	- random system break locations
Leak detection system (LDS)	- LDS capability to detect leak in a timely manner to maintain large margin against unstable crack extension	- no requirement or provision for leak detection

It was assumed that asymmetric blowdown from a DEG large LOCA automatically causes core melt <u>only</u> if the LOCA occurs within the reactor cavity. Accident sequences analogous to those for reactor vessel rupture in WASH-1400 are assumed. These sequences are as follows (Table V.3-14, dominant only):

RC-α (PWR-1) with frequency = 2E-12/py
RC-γ (PWR-2) with frequency = 3E-11/py
RC-δ (PWR-2) with frequency = 1E-11/py
RC-δ (PWR-2) with frequency = 1E-12/py
R-α (PWR-3) with frequency = 1E-9/py
R-ϵ (PWR-7) with frequency = 1E-7/py

WASH-1400 assumes a vessel rupture frequency of 1E-7/py. Replacing this with 9E-8/py (the nominal estimate frequency of in-cavity asymmetric blowdown automatically causing a core melt in a way analogous to vessel rupture) results in the same previous sequence frequencies.

Dose estimates for the release categories were derived using the CRAC code and assuming the quantities of radioactive isotopes and guidelines used in WASH-1400, the meteorology at a typical midwestern site (Byron-Braidwood), a uniform population density of 340 people per square-mile (which is an average of all U.S. nuclear power plant sites) and no evacuation of population. They are based on a 50-mile release radius model.

The nominal estimate risk from the in-cavity DEG large LOCA in a two loop plant becomes:

$$\begin{aligned} \text{Risk} &= (2E\text{-}12/py)(5.4E{+}6 \text{ man-rem}) + (4E\text{-}11/py)(4.8E{+}6 \text{ man-rem}) + \\ &\quad (1E\text{-}9/py)(5.4E{+}6 \text{ man-rem}) + (1E\text{-}7/py)(2300 \text{ man-rem}) \\ &= 0.006 \text{ man-rem/py} \end{aligned}$$

It was assumed that asymmetric blowdown from a DEG large LOCA outside the reactor cavity does not automatically lead to a core-melt. Subsequent safety system failures would be needed to result in core-melt, although the potential for the DEG large LOCA to cause such failures directly (or displace the core such that its geometry becomes uncoolable) still exists.

Presumably, failure of safety systems independent of asymmetric loading are accounted for in the plant design. Since the DEG break is only part of the WASH-1400 large LOCA sequence, it was assumed that no risk is added by the break itself. Only safety system failures induced by unanticipated asymmetric loads on equipment or core geometry disruptions contribute to this issue.

To calculate the contribution to core melt from breaks outside the reactor cavity, a two-step analysis was followed. First, the contribution to core melt from DEG breaks outside the reactor cavity was calculated. Second, an additional fraction of this contribution, based on previous systems interaction analyses, was calculated to represent the risk contribution due to asymmetric blowdown. Only this fraction would be incurred for the proposed action since DEG breaks were previously considered in the plant design.

To estimate the risk contribution from DEG breaks outside the reactor cavity, accident sequences analogous to those for a large LOCA in WASH-1400 are assumed applicable. These sequences are as follows (Table V.3-14, dominant only):

AB-α	(PWR-1)	with frequency		= 1E-11/py
AF-α	(PWR-1)	"	"	= 1E-10/py
ACD-α	(PWR-1)	"	"	= 5E-11/py
AG-α	(PWR-1)	"	"	= 9E-11/py
AB-γ	(PWR-2)	"	"	= 1E-10/py
AB-δ	(PWR-2)	"	"	= 4E-11/py
AHF-γ	(PWR-2)	"	"	= 2E-11/py
AD-α	(PWR-3)	"	"	= 2E-8/py
AH-α	(PWR-3)	"	"	= 1E-8/py

AF-δ	(PWR-3)	"	"	= 1E-8/py
AG-δ	(PWR-3)	"	"	= 9E-9/py
ACD-β	(PWR-4)	"	"	= 1E-11/py
AD-β	(PWR-5)	"	"	= 4E-9/py
AH-β	(PWR-5)	"	"	= 3E-9/py
AB-ε	(PWR-6)	"	"	= 1E-9/py
AHF-ε	(PWR-6)	"	"	= 1E-10/py
ADF-ε	(PWR-6)	"	"	= 2E-10/py
AD-ε	(PWR-7)	"	"	= 2E-6/py
AH-ε	(PWR-7)	"	"	= 1E-6/py
TOTAL				3E-6/py

WASH-1400 assumes a median large LOCA frequency of 1E-4/py. Replacing this with 4.0E-7/py (the nominal estimate frequency of outside-of-cavity DEG large LOCAs) results in lowering the previous sequence frequencies by a factor of 250. The risk from the outside-of-cavity DEG large LOCA becomes (ignoring dependent failures):

$$\begin{aligned}\text{Risk} &= (1\text{E-}12/\text{py})(5.4\text{E+}6 \text{ man-rem}) + (6\text{E-}13/\text{py})(4.8\text{E+}6 \text{ man-rem}) + \\ &\quad (2\text{E-}10/\text{py})(5.4\text{E+}6 \text{ man-rem}) + (4\text{E-}14/\text{py})(2.7\text{E+}6 \text{ man-rem}) + \\ &\quad (2\text{E-}11/\text{py})(1.0\text{E+}6 \text{ man-rem}) + (5\text{E-}12/\text{py})(1.5\text{E+}5 \text{ man-rem}) + \\ &\quad (1.2\text{E-}8/\text{py})(2300 \text{ man-rem}) \\ &= 1\text{E-}3 \text{ man-rem/py}\end{aligned}$$

As assessed in the report for safety issue II.C.3 (Systems Interaction) in Supp. 1 to NUREG/CR-2800 (Andrews et al. 1983), systems interactions typically contribute 10% to total core-melt frequency (and risk), with a range of 1%-20%. The types of safety system failures which could be induced directly by adverse forces from a DEG large LOCA causing asymmetric blowdown are typical systems interactions

The Westinghouse GTAP A-2 owners group has provided analyses for ex-cavity breaks that indicate disruption of core geometry is unlikely to occur (Campbell 1980) for 13 out of 16 plants. However, to account for this possibility and that of asymmetric-blowdown-induced damage to safety equipment, the upper end of the range for systems interaction contribution (20%) is assumed applicable to estimate the risk from dependent failures resulting from outside-of-cavity asymmetric blowdown. Thus, the incremental best estimate risk from the outside-of-cavity DEG large LOCA with asymmetric loadings becomes:

$$\begin{aligned}\text{Risk} &= (0.2)(1\text{E-}3 \text{ man-rem/py}) \\ &= 2\text{E-}4 \text{ man-rem/py}\end{aligned}$$

Combining the two scenarios for DEG large LOCAs within and outside of the reactor cavity yields the following total risk for two loop plants:

$$\text{Risk} = 0.006 + 2\text{E-}4 = 0.006 \text{ man-rem/py}$$

Nominal estimate results for plants that use a two-loop configuration were adjusted to account for the added number of loops in some plants. A review of the GTAP A-2 owners group list indicates that these plants have an average of 3.1 loops. The nominal estimate becomes 0.009 man-rem/py.

Upper estimate risk calculations were made using procedures similar to those of the nominal estimates. The pipe rupture frequency of 1E-5 was allocated 80% to the primary loop and 20% to the reactor cavity by assuming the ratio of results from the SAI study. No corrections for the number of plant loops are necessary because this frequency is per plant year. The in-cavity failure rate of 2E-6 is 20 times higher than WASH-1400 for vessel rupture. The upper estimate cavity risk becomes:

Risk - (4E-11/py)(5.4E+6 man-rem) +
(8.2E-10/py)(4.8E+6 man-rem) +
(2.0 E-8/py)(5.4E+6 man-rem) +
(2.0E-6/py)(2300 man-rem)
= 0.12 man-rem/py

The upper estimate of primary loop breaks of 8E-6 is 12 times lower than WASH-1400 for large LOCAs. The upper estimate loop risk becomes:

Risk = 0.2 [(2E-11/py)(5.4E+6 man-rem) + (1.3E-11/py)(4.8E+6 man-rem) +
(3.9E-9/py)(5.4E+6 man-rem) + (8E-13/py)(2.7E+6 man-rem) +
(5.6E-10/py)(1E+6 man-rem) + (1.0E-10/py)(1.5E+5 man-rem) +
(2.4E-7/py)(2300 man-rem)
= 0.004 man-rem/py

Combining the two scenarios for upper estimate break frequencies yields the following total risk:

Risk = 0.12 + 4E-3 = 0.1 man-rem/py

Multiplying each of the risk calculations in these cases by the number of remaining plant years (16 plants x 23.6 yr = 377 py) results in the industry total public risk increase due to leak before break.

	Total Added Risk (man-rem)
Nominal Estimate	3.4
Upper Estimate	37
Lower Estimate	0

A nominal estimate for the total increase in core melt frequency for the proposed action was determined by summing the contributions for breaks inside the reactor cavity and out-of-cavity loop break systems interactions and then adjusting for the average number of loops.

Core melt increase = $3.1/2[9E\text{-}8 + 0.2(3E\text{-}6/250)] = 1E\text{-}7$/py

An upper estimate of the core-melt frequency increase was calculated by summing the contributions from reactor cavity pipe breaks (2E-6/py) and 20% of the out-of-cavity pipe break initiated core melt accidents.

Core melt increase = $2E\text{-}6 + 0.2(2E\text{-}7) = 2E\text{-}6$/py

Total core-melt frequency increase estimates are as follows:

	Increase in Core-Melt Frequency (Events/py)
Nominal Estimate	1E-7
Upper Estimate	2E-6
Lower Estimate	0

B. Occupational Exposure - Accidental

The increased occupational exposure from accidents can be estimated as the product of the change in total core-melt frequency and the occupational exposure likely to occur in the event of a major accident. The change in core melt frequency was estimated as 1E-7 events/yr. The occupational exposure in the event of a major accident has two components. The first is the "immediate" exposure to the personnel onsite during the span of the event and its short term control. The second is the longer term exposure associated with the cleanup and recovery from the accident.

The total avoided occupational exposure is calculated as follows:

$$D_{TOA} = NTD_{OA};\ D_{OA} = P(D_{IO}+D_{LTO})$$

where

D_{TOA} = Total avoided occupational dose

N = Number of affected facilities

T = Average remaining lifetime

D_{OA} = Avoided occupational dose per reactor-year

P = Change in core-melt frequency

D_{IO} = "Immediate" occupational dose

D_{LTO} = Long-term occupational dose.

Results of the calculations are shown below. Uncertainties are conservatively propagated by use of extremes (e.g., upper bound D_{TO} + upper bound D_{LTO}).

	Increase in Core Melt Frequency (events/ reactor-yr)	Immediate[a] Occupational Dose (man-rem/ event)	Long Term[a] Occupational Dose (man-rem/ event)	Total Avoided Occupational Exposure) (man-rem)
Nominal Estimate	1E-7	1E3	2E4	0.8
Upper Estimate	2E-6	4E3	3E4	30
Lower Estimate	0	0	1E4	0

(a) Based on cleanup and decommissioning estimates, NUREG/CR-2601 (Murphy 1982).

C. Public Property

The effect of the proposed action upon the risk to offsite property is calculated by multiplying the change in accident frequency by a generic offsite property damage estimate. This estimate was derived from the mean value of results of CRAC2 calculations, assuming an SST1 release (major accident), for 154 reactors (Strip 1982). CRAC2 includes costs for evacuation, relocation of displaced persons, property decontamination, loss of use of contaminated property through interdiction and crop and milk losses. Litigation costs, impacts to areas receiving evacuees and institutional costs are not included. The damage estimate is converted to present value discounting at 10%. A 5% discount rate was also considered as a sensitivity case.

The following discounting formula is employed:

$$D = V \frac{e^{-It_i} - e^{-It_f}}{I}$$

where D = discounted value
V = damage estimate
t_i = years before reactor begins operation; 0 for operating plants
t_f = years remaining until end of life.
I = discount rate

For this proposed action, only operating reactors are affected, and the average number of years of remaining life is 23.5. Therefore, the 10% discount factor D/V = 9. The 5% discount factor equals 13.8. These values must be multiplied by the number of affected facilities (16) to yield the total effect of the action. Upper and lower bounds are values for Indian Point 2 and Palo Verde 3 calculated from Strip (1982). Results are as follows:

	Offsite Property Damage ($/event)	Discounted Offsite Property Damage [Lifetime Risk] ($/event)		Discounted Value of Additional Offsite Property Damage ($)	
		10%	5%	10%	5%
Nominal Estimate	1.7E+9	1.5+10	2.3E+10	2.4E+4	3.8E+4
Upper Estimate	9.2E+9	8.3E+10	1.3E+11	2.6E+6	4.1E+6
Lower Estimate	8.3E+8	7.5E+10	1.2E+10	0	0

D. Onsite Property

The effect of the proposed action on the risk to onsite property is estimated by multiplying the change in accident frequency by a generic onsite property cost. This generic onsite property cost was taken from Andrews et al. (1983). Costs included are for interdicting or decontaminating onsite property, replacement power and capital cost of damaged plant equipment. Onsite property damage costs were discounted using the following formula.

$$D = \left(\frac{V}{m}\right)\left[\frac{e^{-It_i}}{(I^2)}\right]\left[(1-e^{-Im})\left(1-e^{-I(t_f-t_i)}\right)\right]$$

where D = discounted value
V = damage estimate
m = years over which cleanup is spread = 10 years
t_i = years before reactor begins operation; 0 for operating plants
t_f = years remaining until end of life; 0 = 23.5 years
I = discount rate = 10% or 5%.

For this proposed action, the 10% discount factor equals 5.7 and the 5% discount factor equals 11. To obtain the total effect of the action, the per-reactor results are multiplied by the number of affected facilities (16). The uncertainty bounds given in the table reflect a 50% spread which was estimated to be indicative of the uncertainty level. The results are summarized below:

	Onsite Property Damage Estimate ($/event)	Discount Onsite Property Damage ($/event)		Discounted Value of Avoided Onsite Property Damage ($)	
		10%	5%	10%	5%
Nominal Estimate	1.65E+9	9.4E+9	1.8E+10	1.5E+4	2.9E+4
Upper Estimate	2.5E+9	1.4E+10	2.8E+10	4.6E+5	8.8E+5
Lower Estimate	8.2E+8	4.7E+9	9.0E+9	0	0

E. Occupational Exposure-Operational

Operational occupational exposure due to installation and maintenance of plant modifications is avoided by the proposed exemption to asymmetric blowdown loads during implementation and operation.

For this analysis, plants were broken into two groups; those requiring extensive modifications and the rest. A listing of each group and assumed modifications is given in the section on Industry Implementation Cost. Avoided implementation doses for the three plants requiring extensive modifications were based on a San Onofre estimate of 2600 man-rem/plant to install primary system pipe restraints at the RPV nozzles and modifying pump and steam generator supports for three loops. Some occupational doses will be incurred for the proposed action to upgrade leak detection systems. For these plants, it is estimated that 450 man-hours per plant inside containment at 45 mR/hr and 80 hours outside containment at 2.5 mR/hr would be required to install such modifications. No modifications to the core or core barrel were assumed. For this group, net avoided implementation doses were calculated as follows:

$$\text{Avoided installation dose} = 3[2600 - (0.0025\ (80) + 0.045\ (450))]$$
$$= 7700 \text{ man rem}$$

Implementation doses for the remaining thirteen plants were estimated as follows: 80% of total direct costs were assumed to be attributed to labor in radiation zones. These costs were converted to man-hours by dividing by the cost per man year (assumed to be $100K) and multiplying by 1800 man-hours/man-year. Man-rem estimates were calculated by assuming dose rates of 25 mR/hr inside containment and 2.5 mR/hr outside of containment. The lower value for containment work was assumed due to less extensive modifications and presumed better equipment access. Required activities are described further in Industry Implementation Costs.

Results of this analysis are as follows:

Activity	Direct Cost[a] ($/loop)	Number of Plants (Loops)	Man-Hours[b]	Dose Rate (R/hr)	Avoided Implementation Dose (man-Rem)
Install primary shield wall restraints and inspection port modifications	98000	13(40)[d,e]	56000	0.025	1400
Modify reactor coolant pump supports	20000	7(21)[d]	6000	0.025	150
Steam generator supports	120000	4(12)[d]	21000	0.025	520
Calibrate leak[c] detection system	N/A	11[f]	5000	0.025	(120)
Total					2000

(a) Stevenson 1980, except for shield wall and inspection port modifications. Costs for these activities are based on industry estimates for D.C. Cook.
(b) (Direct Cost)(Number of Loops)(1800 man-hr/man-yr)(0.8)/($1.0E+5/man-yr).
(c) Avoided doses are negative for these activities because they are required for the proposed action.
(d) Campbell 1979 and 1980.
(e) Ft. Calhoun was credited with 3 loops due to redundant cold legs.
(f) Two plants have verified adequate leak detection capability.

Occupational dose to maintain the modifications is also avoided. To estimate the amount, it was assumed that two additional man-weeks per plant-year would be spent inside containment if the modifications are made. This is due to inspection of the modifications and additional time required to gain access to primary system components. The total dose for the owners group is estimated below. Plants requiring extensive modifications have remaining lives totaling 56 plant-years. All other plant lives total 320 plant-years.

$$\text{Operational dose averted} = (80 \text{ man-hr/py})[(56 \text{ plant-years})(0.045 \text{ R/man-hr}) + (320 \text{ plant-years})(0.025 \text{ R/man-hr})]$$

$$= 840 \text{ man-rem}$$

Total avoided occupational doses due to implementation, operation and maintenance are shown below. Upper and lower estimates were developed using the following model (Andrews et al. 1983):

$$\text{Dose}_{\text{upper}} = 3 \text{ dose}_{\text{expected}}$$

$$\text{Dose}_{\text{lower}} = 1/3 \text{ dose}_{\text{expected}}$$

Activity	Dose Avoided (man-rem)
Implementation	9700
Operation, Maintenance	840
Total	1.1E+4
Upper Estimate	3.2E+4
Lower Estimate	3500

F. Industry Implementation Cost

Several levels of value to industry are seen as resulting from the proposed action. Potential design modifications that are avoided range from major component support upgrades to the addition of major new equipment, i.e. pipe restraints. Leak detection systems at some plants are already adequate. Modifications at other plants include an assessment and calibration of existing leak detection systems. The plants were divided into two groups based on assumed avoided plant modifications:

Plants Requiring Extensive Modifications:
- Haddam Neck
- Yankee Rowe
- San Onofre 1

Plants Requiring Some Modification:
- HB Robinson 2
- Zion 1,2
- Turkey Point 3,4
- RE Ginna
- Surry 1,2
- Point Beach 1,2
- DC Cook 1,2
- Ft. Calhoun.

For plants requiring extensive modifications, data developed for modification to primary system component supports and vessel nozzle restraints by San Onofre were used (Baskin 1980). Total reported costs were divided by three to obtain a per-loop cost. Costs for contingencies were ignored. Results are as follows:

	Per-Loop Costs ($K)
Direct Costs (materials, field costs)	901
A/E Support	333
NSSS Supplier Support	716
Utility Support	166
Escalation (1979-1982)	740
Total	2856

In addition, Baskin reports that 40 days of replacement power would be purchased. At $300K/day (Andrews et al. 1983), the total replacement power costs are $12M per plant.

It is conservatively assumed that all three plants will require upgrading to their leak detection systems. This may include calibration of current flow measurement systems and revisions to technical specifications. Costs for these upgrades are based on labor estimates of 0.25 man-yr. At $100K per man-yr, total costs are $25K/plant.

Total implementation costs for the three plants were calculated as follows:

Implementation costs = (Total Number of Loops)(Avoided Cost per Loop) + (Number of Affected Plants)[(Replacement Power Avoided Cost) - (Leak Detection Costs)]

= (11)($2.86E+6) + 3[$1.2E+7 - $2.5E+4]

= $6.7E+7

Implementation costs for the remaining plants are derived from UCRL-15340 (Stevenson 1980) and industry estimates including San Onofre. Results are indicated below:

Modification	Cost
Primary Shield Wall Restraint and Inspection Port Modification (Hot and Cold Leg)	$230K/loop
Reactor Coolant Pump Supports	$ 52K/loop
Steam Generator Supports	$310K/loop
Reactor Vessel Supports	$ 19K/loop
Reactor Coolant Component Walls	$230K/plant

The shield wall restraints and inspection port modifications are to control ruptures in the reactor cavity. These costs were escalated in 1982 dollars based on estimates for DC Cook units and are assumed to include all overheads, material and labor. All other costs listed are based on work by Stevenson. The original work did not appear to include engineering, NSSS supplier and utility support costs. An additional 134% was assumed for these costs based on the San Onofre data. All costs were also increased by an additional 19% for escalations between 1980 and 1982.

All modifications would not be required at all plants. Based on Owners Group analyses (Campbell 1979), it was assumed that the following number of modifications would be performed.

Modification	Number of Plants	(Loops)	Owners Group Avoided Cost
Primary Shield Wall Restraint and Inspection Port Modification	13	(40)	\$9200K
Reactor Coolant Pump Supports	7	(21)	\$1100K
Steam Generator Supports	4	(12)	\$3700K
Reactor Vessel Supports	0		0
Reactor Coolant Compartment Walls	0		0
Total			\$14000K

Shield wall restraints and inspection port modifications were assumed to be required at all plants. Pump and steam generator support work was assumed to be needed at plants identified by the owners group. Reactor vessel supports were assumed not to be needed by any plants. Stevenson discusses them as mainly a seismic restraint. Reactor coolant compartment wall anchors are only required for the safe shutdown earthquake (SSE) and LOCA load combinations. Thus they were not used in this analysis.

Needs for replacement power to modify remaining plants were not identified in the available data. It was assumed for plants requiring pump and steam generator support modifications that some replacement power would be needed (four plants). For this analysis, it was assumed that one half of the incremental outage time of San Onofre would be needed or 20 days. Total outage days would be 80. Costs for replacement power at \$300K/day total \$24M.

Costs for modifying leak detection systems are assumed the same for plants requiring some modification as for plants with extensive modifications. It was assumed that only 11 of the 13 plants need upgrading. Costs for this work total \$2.8E+5.

Net avoided costs for plants with some modifications were calculated as follows:

Net Avoided Implementation Costs = Primary System Modifications + Replacement Power - Leakage Detection Systems.

= \$1.4E+7 + \$2.4E+7 - \$2.8E+5

= \$3.8E+7

To generate upper and lower estimates for costs, it was assumed that estimates are within 50% of the nominal estimate. Results for industry implementation costs are summarized below:

Plants with Extensive Modifications	\$6.7E+7
Plants with Some Modifications	\$3.8E+7
Total	\$1.1E+8
Upper Estimate	\$1.6E+8
Lower Estimate	\$5.3E+7

G. Industry Operation and Maintenance Costs

Industry avoided operation and maintenance costs were developed based on the assumption that additional restraints will result in additional inspections and restrict access to steam generators, reactor coolant pumps and reactor nozzles. Based on the values used for occupational dose estimates, this labor is assumed to total 80 man-hours/plant-year. At \$100K/man-year and 44 man-wk/man-yr, the annual cost is \$4540/plant. The present value of this quantity for 16 plants over 23.5 years with upper and lower estimates are as follows:

	Discount Rate 10%	5%
Present Value of Operation and Maintenance Costs	= \$6.5E+5	1.0E+6
Upper Estimate	= \$9.8E+5	1.5E+6
Lower Estimate	= \$3.3E+5	5.0E+5

H. NRC Implementation Support Costs

NRC Avoided Implementation costs are estimated to be 0.5 man-year of labor to review plant modifications. This is partially offset by an estimate of 0.25 man-year to review leak detection system upgrades and revisions to plant technical specifications. Net NRC cost savings are as follows:

Avoided NRC Implementation Support Costs:

16 plants (0.25 man-yr/plant @ $100,000/man yr) = $4.0E+5

Upper Estimate = $6.0E+5

Lower Estimate = $2.0E+5

No additional NRC costs during operations are expected.

7.0 CONCLUSIONS

The summary results for the value-impact assessment are shown below. The nominal estimates for cost and dose indicate that the proposed action should be recommended. The uncertainty bounds do not show negative benefits for either dose or cost. The upper estimate is very positive. The following observations can also be made:

- This action did not address costs of core and core support modifications. Adding these costs would increase the negative impact of the exemption.
- The schedule for avoided plant modifications assumed backfitting to add only an increment of downtime to normal outages. If not, the additional avoided costs for replacement power would increase the negative impact obtained.
- The dose avoided for this action is primarily occupational dose during equipment installation. This dose is being weighed against statistical estimates of public and occupational dose for rare events.
- Cost results are not sensitive to discount rates used in this analysis.

Summary of Value-Impact Assessment

Value (man-rem)			Impact ($)					
Nominal Est.	Upper Est.	Lower Est.	Nominal Est.		Upper Est.		Lower Est.	
			10%	5%	10%	5%	10%	5%
1.1E+4	3.2E+4	3500	-1.1E+8	-1.1E+8	-1.6E+8	-1.6E+8	-5.3E+7	-5.3E+7

REFERENCES

Aldrich, D. C., et al. 1982. Technical Guidance for Siting Criteria Development. NUREG/CR-2239, Sandia National Laboratories, Albuquerque, New Mexico.

Andrews, W. B., et al. 1983. Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development. NUREG/CR-2800 (PNL-4297), Pacific Northwest Laboratory, Richland, Washington.

Baskin, K. P. 1980. Letter to Mr. D. L. Zuemann of the US NRC dated February 13, 1980. Docket No. 50-206. Southern California Edison, Rosemead, California.

Bush, S. H. 1977. "Reliability of Piping in Light Water Reactors." IAEA-SM-218/11. International Symposium on Application of Reliability Technology to Nuclear Power Plants. International Atomic Energy Agency, Vienna, Austria.

Campbell, T. E. et al. 1980. Westinghouse Owners Group Asymmetric LOCA Loads Evaluation Phase C. WCAP 9748, Westinghouse Electric Corp., Pittsburgh, Pennsylvania.

Campbell, T. E. et al. 1979. Westinghouse Owners Group Asymmetric LOCA Loads Evaluation - Phase B. WCAP 9628, Westinghouse Electric Corp., Pittsburgh, Pennsylvania.

Campbell, T. E. and J. N. Chirigos, et al. 1982. Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Through-Wall Crack. WCAP 9558, Rev. 2, Class 2, Westinghouse Electric Corp., Pittsburgh, Pennsylvania.

Campbell, T. E. and J. N. Chirigos, et al. 1981. Tensile Toughness Properties of Primary Piping Weld Metal for Use Mechanistic Fracture Evaluation. WCAP 9787, Class 2, Westinghouse Electric Corp., Pittsburgh, Pennsylvania.

Harris, D. O. and Fullwood, R. R. 1976. An Analysis of the Relative Probability of Pipe Rupture at Various Locations in the Primary Cooling Loop of a Pressurized Water Reactor Including the Effects of Periodic Inspection. SAI-001-PA, Science Applications Inc., Palo Alto, California.

Heaberlin, S. W., et al. 1983. A Handbook for Value-Impact Assessment. PNL-4646 (Draft), Pacific Northwest Laboratory, Richland, Washington.

Hosford, S. B., et al. 1981. Asymmetric Blowdown Loads on PWR Primary Systems. NUREG-0609, U.S. Nuclear Regulatory Commission, Washington, D.C.

Lu, S. 1981. Probability of Pipe Fracture in the Primary Coolant Loop of a PWR Plant, NUREG/CR-2189. U.S. Nuclear Regulatory Commission, Washington, D.C.

Murphy, E. S., and G. M. Holter. 1982. Technology, Safety and Costs of Decommissioning Reference Light Water Reactors Following Postulated Accidents. NUREG/CR-2601, Pacific Northwest Laboratory, Richland, Washington.

Stevenson, J. D. 1980. Cost and Safety Margin Assessment of the Effects of Design for Combination of Large LOCA and SSE Loads. UCRL-15340, Lawrence Livermore Laboratory, Livermore, California.

Strip, D. R. 1982. Estimates of the Financial Consequences of Nuclear Power Reactor Accidents. NUREG/CR-2723, Sandia National Laboratories, Albuquerque, New Mexico.

PDR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JAN 24 1984

MEMORANDUM FOR: Frank J. Miraglia, Assistant Director for Safety Assessment, DL

THRU: John A. Zwolinski, Section Leader
Operating Reactors Assessment Branch, DL

Gary M. Holahan, Chief
Operating Reactors Assessment Branch, DL

FROM: Rudy O. Karsch, Lead Project Manager for Equipment Qualification, MPA B-60
Operating Reactors Assessment Branch, DL

SUBJECT: EQUIPMENT QUALIFICATION PROGRAM STATUS REPORT, WEEK ENDING JANUARY 20, 1984

Equipment Qualification Review Meetings for Yankee Rowe, Robinson and Kewaunee were completed on January 17, 18, and 20, respectively. In addition to the resolution of the specific equipment problems identified by the Franklin Research Center TER and compliance to 10 CFR 50.49 (the EQ Rule), the licensees surveillance inspection program and preventative maintenance program for qualified equipment was reviewed. Yankee Rowe will provide additional discussion on their JCO for PORV leak detection, e.g., a backup is provided by tail-pipe temperature sensing. Robinson did not provide required operating times for any equipment reviewed by the TER. They will provide this as part of their submittal. Kewaunee mistakenly asserted that numerous items of equipment outside of containment are not subjected to a harsh environment and thus not subject to the EQ Rule. In fact, it was determined that they are subject to a harsh environment, radiation only. Their submittal will reflect this change and documentation kept on file to support the qualified status of equipment in this category. The meetings with Yankee Rowe, Robinson and Kewaunee provided closure of all open items and no extension requests are anticipated at this time. A list of attendees is enclosed.

Numerous licensees have requested clarification of the overlap between the EQ Rule's completion deadlines and the schedule requirements of RG 1.97 as they relate to post-accident monitoring equipment (10 CFR 50.49, Section b.3 and footnote 4). The specific question is: Equipment is installed in the plant which peforms a RG 1.97 function, but does not meet all the functional requirements and is not proposed to satisfy RG 1.97. It will be replaced in the future with qualified equipment to satisfy RG 1.97, but not until after the qualification deadline for 10 CFR 50.49. Is it subject to the EQ Rule? This matter was discussed with ELD (Bill Shields) and the following interpretation is consistent with the regulations: If the equipment in question does not fall into the category of safety-related electric equipment defined by 10 CFR 50.49 sections b.1 and b.2, and is not proposed to satisfy RG 1.97, it does not require qualification. However, at the time it is identified by any documents or schedule, as a proposed fix under RG 1.97, it becomes subject to the EQ Rule and, therefore, subject to the EQ Rule's qualification deadline.

 XA 7pp.

ENCLOSURE 1

EQUIPMENT QUALFICATION MEETING

ROBINSON

JANUARY 18, 1984

Attendees	Affiliation
G. Requa	NRC/DL/ORB#1
Frank Gilman	CP&L - HBR
George Honma	CP&L - HBR
Tillie Hudson	CP&L
Rudy Karsch	NRC/DL/ORAB
Paul Shemanski	NRC/NRR/DE/EQB
Max W. Yost	INEL
Fred Roy	Patel Engineer
Roland K. Ho	EPM
H. M. Fontecilla	NUTECH
R. D. Condello	NUTECH
Tommy Le	NRC/IE/DQASIP
Peter Yandow	CP&L Harris
Jerry Waldorf	CP&L Brunswick/Corp.
Patrick Carier	CP&L Brunswick/Corp.

EQUIPMENT QUALFICATION MEETING

MAINE YANKEE

JANUARY 17, 1984

Attendees	Affiliation
Peter Erickson	NRC/DL/ORB5
A. E. Finkel	NRC/R-I
C. J. Anderson	NRC/R-I
D. A. Hansen	YAEC
R. R. McCoy	YAEC
W. G. Jones	YAEC
S. P. Fournier	YAEC
Andrew D. Hodgdon	YAEC
R. M. Mitchell	YAEC- Rowe
D. E. Yasi	YAEC
J. A. Kay	YAEC-Lic. Engineer
L.L. Richardson	YAEC
S. F. Urbanowski	YAEC
Rudy Karsch	NRC/DL/ORAB
Paul Shemanski	NRC/DE/EQB
Max W. Yost	INEL
Tommy Le	NRC/IE/DQASIP
Bob LaGrange	NRC/NRR/DE/EQB

EQUIPMENT QUALFICATION MEETING

KEWAUNEE

JANUARY 20, 1984

Attendees	Affiliation
Don Neighbors	NRC/DL/ORB#1
Rudy Karsch	NRC/DL/ORAB
Paul Shemanski	NRC/NRR/DE/EQB
Max Yost	INEL
Charles Schrock	WPSC
John G. Thorgersen	WPSC
Eric Schmieman	Self
Sherry Bernhoft	WPSC
Larry Price	Gasser/Nimohawk
Tommy Le	NRC/IE/QASIP
George Flowers	VEPCO/I&C
Paul Conner	VEPCO/I&C
Dick Smith	VEPCO/OPS-Licensing

1/19/84

Enclosure 2

Equipment Environmental Qualification Review Meetings

With the Licensees

Schedule Status

Plant	Meeting Scheduled	Expected Submittal	Project Manager
Cook 1 and 2	Completed	10/13/83	D. Wigginton
Three Mile Island 1	Completed	1/31/84	J. Van Vliet
Point Beach 1 and 2	Completed	Received	T. Colburn
Prairie Island 1 and 2	Completed	1/3/84	D. DiIanni
Peach Bottom 2 and 3	Completed	1/5/84	G. Gears
Oyster Creek	Completed	1/9/84	J. Lombardo
Monticello	Completed	1/28/84	H. Nicolaras
Davis-Besse 1	Completed	1/13/84	A. DeAgazio
Crystal River 3	Completed	1/31/84	R. Hernan
San Onofre 1	Completed	1/31/84	W. Paulson
Palisades	Completed	2/13/84	T. Wambach
Farley 1 and 2	Completed	2/28/84	E. Reeves
Yankee Rowe	Completed	3/1/84 ~~2/17/84~~	P. Erickson
Robinson	Completed	3/2/84 ~~2/18/84~~	G. Requa
Kewaunee	Completed	3/1/84 ~~2/20/84~~	D. NEIGHBORS ~~M. Grotenhuis~~
Zion 1 and 2	1/25/84	2/27/84	J. Norris
Quad Cities 1 and 2	1/26/84	2/28/84	R. Bevan
Dresden 2 and 3	1/26/84	2/29/84	R. Gilbert
Oconee 1, 2, 3	1/31/84	2/29/84	J. Suermann
Brunswick 1 and 2	2/2/84	3/5/84	S. Mackay
Duane Arnold	2/7/84	3/7/84	M. Thadani
St. Lucie 1	2/8/84	3/8/84	D. Sells
Turkey Pt. 3 and 4	2/9/84	3/9/84	D. McDonald
Hatch 1 and 2	2/14/84	3/14/84	G. Rivenbark

Contacts: P. Shemanski, EQB, 492-8289; R. Karsch, ORAB, 492-8563

1/19/84

Equipment Environmental Qualification Review Meetings
(Continued)

Plant	Meeting Scheduled	Expected Submittal	Project Manager
Arkansas 1	2/15/84	3/15/84	G. Vissing
LaCrosse	2/16/84	3/16/84	R. Dudley
Arkansas 2	2/29/84	3/29/84	R. Lee
Rancho Seco	3/7/84	4/7/84	S. Miner
Big Rock Point	3/14/84	4/16/84	R. Emch
Nine Mile Point 1	3/15/84	4/16/84	R. Hermann
Calvert Cliffs 1 and 2	3/16/84	4/17/84	D. Jaffe
Surry 1 and 2	3/19/84	4/19/84	D. Neighbors
North Anna 1 and 2	3/20/84	4/20/84	L. Engle
Fort Calhoun	3/23/84	4/23/84	E. Tourigny
Trojan	3/27/84	4/27/84	C. Trammell
Cooper	3/29/84	4/29/84	B. Siegel
Fitzpatrick	3/30/84	4/30/84	H. Abelson
Beaver Valley 1	4/5/84	5/7/84	P. Tam
Haddam Neck	4/10/84	5/10/84	J. Lyons
Millstone 1	4/11/84	5/11/84	J. Shea
Millstone 2	4/11/84	5/11/84	K. Heitner
Ginna	4/17/84	5/17/84	G. Dick
Vermont Yankee	4/18/84	5/18/84	V. Rooney
Pilgrim	5/22/84	6/22/84	P. Leech

JAN 24 1984

CRGR met this week to discussed RG 1.89, "Environmental Qualification of Electrical Equipment Important to Safety for Nuclear Power Plants." The discussion was very lengthy and RG 1.89 will probably not be issued in the immediate future. DL concurred in the RG contingent on several additions and changes being made which became the basis for NRR's concurrence. Some of the requested changes were not incorporated, and an explanation was not provided. This may further delay issuance of RG 1.89.

McGuire 1 is essentially in compliance with the EQ Rule. When McGuire 2 completed its EQ review, a parallel effort for McGuire 1 resolved all open items for both plants. An SER will be required for McGuire 1 to document this action. A status report and schedule of upcoming meetings is enclosed. The expected submittal dates for the completed plants represent negotiated dates. TVA plants are not yet scheduled. It is expected that they will choose dates within the next two weeks. Maine Yankee is still arguing about their post-accident environment. This should be put to bed shortly, which leaves Indian Point 2/3 and Salem 1/2 as the lone unexplained hold-outs. Project managers for the above plants should encourage their licensees to commit to a meeting date to expedite completion of the EQ effort. Please notify Rudy Karsch (x28563) in ORAB concerning these matters.

/S/

Rudy O. Karsch, Lead Project Manager
for Equipment Qualification, MPA B-60
Operating Reactors Assessment Branch
Division of Licensing

Enclosures:
1. Meeting Attendees
2. Schedule of Upcoming Meetings

cc: D. Eisenhut/R. Purple
G. Lainas
T. Novak
V. Noonan
DL BC's
DL PM's
B. Shields (ELD)
J. Joyce (ICSB)

DISTRIBUTION
Central File
~~ORAB Rdg~~
GHolahan
JZwolinski
RKarsch

ORAB:DL	ORAB:DL	C:ORAB:DL
RKarsch:dm	JZwolinski	GHolahan
01/23/84	01/23/84	01/23/84