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Office of Nuclear Reactor Regulation
Mail Stop O10-A1
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT: Break Characteristics Model for Debris Generation Following a Design Basis Loss of Coolant Accident

PROJECT NUMBER: 689

Dear Ms. Black:

NEI met with NRC staff on April 29, 2003, to discuss activities related to Generic Safety Issue (GSI) 191, "Assessment of Debris Accumulation on PWR Sump Performance." During the meeting NEI presented an approach for defining the effective break area for use in determining debris generation following a design basis LOCA. NRC staff agreed to review the proposed approach and requested that NEI provide additional information, including how the approach meets applicable regulatory requirements. The enclosed document is provided in response to this request.

The document presents a model for use by Pressurized Water Reactor (PWR) plants that provides a more realistic representation of one aspect of debris generation modeling (break size) while maintaining an overall conservative representation of the debris generation potential of postulated breaks. The proposed model utilizes fracture mechanics as the basis for determining a break size that is realistically conservative. This break size is then used in determining the quantity of debris that would be generated from identified break locations.

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Please contact John Butler 202-739-8108, jcb@nei.org, or me if you have any questions on this transmittal.

Sincerely,

Anthony R. Pietrangelo

Enclosure

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Mr. Sunil D. Weerakkody, U. S. Nuclear Regulatory Commission
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Break Characteristics Model for Debris Generation Following a Design Basis Loss of Coolant Accident

October 10, 2003

Prepared by the Nuclear Energy Institute PWR Sump Performance Task Force

Table of Contents

1 Introduction

An important safety concern regarding long-term recirculation cooling following a Loss of Coolant Accident (LOCA) is the transport of debris materials to interceptors (i.e., trash racks, debris screens, suction strainers) inside containment and the potential for debris accumulation to result in adverse blockage effects. Debris resulting from a LOCA, together with pre-existing debris, could block the emergency core cooling system (ECCS) debris interceptors and result in degradation or loss of recirculation flow margin. Potential debris sources can be divided into three categories: (1) debris that is generated by the LOCA and is transported by blowdown forces (e.g., insulation, paint), (2) debris that is generated or transported by washdown, and (3) other debris that existed before a LOCA (dust, sand, etc.). Each debris source is separately evaluated to estimate the quantity and other characteristics necessary to assess the fraction that could be transported to the containment recirculation sump and its combined effect on recirculation flow.

An initial step in evaluating post-accident sump performance is the determination of the amount of debris generated from a postulated breach in the piping system. Current regulatory guidance calls for determination of the quantity and characteristics of debris generated by a postulated LOCA covering a range of break sizes, break locations, and other properties, in a manner that provides assurance that the most severe postulated LOCAs are calculated. Methods for determining debris generation typically utilize a bounding combination of deterministic and mechanistic methods to provide a conservative representation of the destructive behavior of a postulated break. These methods provide a conservative estimation of debris generation based upon models that are not representative of the expected behavior of pipe breaks.

This paper presents a model for use by Pressurized Water Reactor (PWR) plants that provides a more realistic representation of one aspect of debris generation modeling (break size) while maintaining an overall conservative representation of the debris generation potential of postulated breaks. The proposed model utilizes fracture mechanics as the basis for determining a break size that is realistically conservative. This break size is then used in determining the quantity of debris that would be generated from identified break locations.

The fracture mechanics techniques described in this paper are the same techniques that have been used successfully in the support of Leak-Before-Break (LBB) and the application of LBB to postulated leakage cracks in large reactor coolant piping in PWR's. These leakage cracks have leak rates well above the demonstrated PWR leak detection capabilities (typically 10 gpm), while at the same time have been shown to remain stable under all normal and off-normal plant operating conditions.

While the proposed treatment method and LBB applications utilize the same technical basis, the method proposed in support of debris generation differs substantially from an LBB application. LBB applications[1] typically use fracture mechanics to demonstrate that the probability of fluid system piping rupture is extremely low, and using this basis, local dynamic effects are excluded. The proposed fracture mechanic approach continues to include the local dynamic effects (e.g., debris generation) but uses fracture mechanics as a basis for determining the amount of debris that is generated by the postulated break via identification of an effective break area. Therefore, this method credits the demonstrated toughness of PWR piping, yet defines a conservative design input for sump performance evaluations.

The proposed model will be one of the options available for use by PWR licensees as a step in the overall analysis effort necessary to demonstrate compliance with regulatory requirements governing operation of the ECC and Containment Spray Systems (CSS).

[1] Applications utilizing the provisions of General Design Criterion 4 contained in Appendix A to 10 CFR Part 50, that allow the exclusion of local dynamic effects from the design basis for qualified piping systems.

2 Summary of Current Guidance and Need for Revised Approach

Upon initiation of a break in the reactor coolant system of a PWR, the forces released by the break have the potential to dislodge piping thermal insulation and other materials in the vicinity of the break. A portion of this material will be transported to the containment floor by the break flow and by containment sprays. Upon initiation of recirculation flow from the containment sump to the reactor coolant system, some of the debris in the lower containment elevations will be transported to and accumulate on the containment sump screens. The resultant increase in resistance to the flow by the debris accumulating on the containment sump screen has the potential to challenge the capability of the ECCS to provide long-term cooling to the reactor core.

In order to calculate plant response to a postulated event and the potential for significant blockage of the containment sump screens, it is necessary to take into account a wide range of phenomena and processes. These phenomena and processes are highly dependent upon plant design and operation as well as the specifics of the postulated LOCA event. The complexity and multivariate nature of the event progression, coupled with the absence of a comprehensive database addressing the full range of encountered phenomena inevitably leads to a calculation process that accounts for the resulting uncertainties in a conservative manner.

Typically the analyses investigating ECCS operation during the recirculation phase divide the process into three separate phases: (1) Debris Generation, (2) Debris Transport and (3) Debris Accumulation and Headloss. Each phase of the calculation process, while interdependent, involves its own set of phenomena and uncertainties. Known limitations in the knowledge base of these phenomena and associated calculation methods are typically accounted for in a bounding fashion during each phase of the process. The combined effect of these bounding calculations is a pessimistic prediction of ECCS recirculation performance that, while conservative, provides little insight into the realistically expected performance during a design basis event. An additional complicating factor is the realization that, unlike most design basis calculations, there is no unique set of conditions that can be repeatedly shown to represent the worst case for recirculation performance. This necessitates either a full scope set of calculations looking at an effectively boundless set of possible permutations and combinations of conditions, or a more limited set of calculations that combines conditions in a bounding, often unrepresentative, manner.

While it is not the intent of this paper to address the full set of calculations necessary to assess ECCS and CSS operation in the recirculation mode, it is informative to discuss current guidance and practices for the debris generation phase of the calculations.

<u>Break Size and Location</u>

Because the size and location of the break have a key influence on a number of key parameters that are specific to each plant's design and operation (e.g., debris generation quantities, debris transport capability, containment flood-up level and timing), it is not possible to predetermine the limiting break size or its location. The current practice is to analyze a full range of break sizes, ranging from the smallest break that has the potential to lead to ECCS recirculation operation to a full double-ended guillotine break of the largest reactor coolant system pipe. This full range of break sizes is postulated for a wide range of potential break locations to address factors such as variations in insulation materials on and around postulated break locations and proximity to the recirculation sump and its influence on debris transport.

Current guidance calls for debris generation to be calculated for a number of postulated LOCAs of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated LOCAs are calculated. Proposed revision 3 to Regulatory Guide 1.82 (Reference 1) calls for the following postulated break sizes and locations to be considered.

1. Breaks in the hot leg, cold leg, intermediate leg, and, depending on the plant licensing basis, main steam and main feedwater lines with the largest amount of potential debris within the postulated zone of influence,

2. Large breaks with two or more different types of debris, including the breaks with the most variety of debris, within the expected zone of influence,

3. Breaks in areas with the most direct path to the sump,

4. Medium and large breaks with the largest potential particulate debris to insulation ratio by weight, and

5. Breaks that generate an amount of fibrous debris that, after its transport to the sump screen, creates a minimum uniform thin bed (1/8-inch layer of fiber) to filter particulate debris[2].

This process is applied in a deterministic fashion without consideration of the probability of a limiting size break occurring at the limiting location on the RCS. This can lead to a condition where the limiting break is controlled by a unique combination of break size, location and transport assumptions. This factor, in conjunction with other, more traditional, design basis assumptions (e.g., limiting single failure, maximum uncertainties on setpoints, timings, and flowrates) can easily lead to one or more extremely low probability events dominating calculation results.

[2] Screen blockage experiments have determined that a 1/8 inch layer of fiber combined with particulate debris can result in significant headloss. Fiber layers thicker than 1/8 inch result in lower headloss, while layers less than 1/8 inch are unstable and tend toward self destruction as headloss increases. The uniqueness of the set of conditions that result in a stable "thin-bed" of fiber and particulate is not addressed in current guidance.

Debris Generation

Given a postulated break size and location, the next step is to calculate (or estimate) the quantity and size distribution of debris that could be generated as a direct consequence of the break. The debris generation capability of a break is dependent on a number of factors including break size, break opening characteristics and break orientation, as well as characteristics of materials and structures surrounding the break. While a number of tests have been performed to investigate the mechanics of debris generation, these tests are limited in scope and both the tests and the resultant interpretation of test data have incorporated simplifying/bounding assumptions to address variability in key parameters.

One important simplifying assumption used in both debris generation tests and subsequent modeling of the tests is that the break opening time is instantaneous[3]. This is a carry-over assumption from thermal hydraulic analyses of reactor coolant system response performed in accordance with Appendix K to 10 CFR Part 50[4]. A consequence of this assumption for debris generation is the generation of an acoustic shock wave. This pressure wave is believed to be a major contributor to debris generation surrounding the break. Component insulation is destroyed initially by the blast effects of a shock wave that expands away from the break. This destruction is continued by the two-phase jet of fluid emanating from the break. Experiments show that the shock wave may cause substantial damage to even the most heavily reinforced insulating constructions (e.g., steel-jacketed RMI or fiber) if they are located sufficiently close to the break.

In order for a shock wave to occur, the break opening time (BOT) must be on the same order as the acoustic propagation time across the piping. If the BOT is long relative to the acoustic propagation time then a shock wave will not occur and debris generation will be predominated by jet impact and displacement forces. As part of its evaluation of the potential for shock waves following a double-ended guillotine break General Electric (Reference 2) estimated that a shock wave will not be generated for large bore piping break opening times greater than approximately 10 milliseconds. Realistic estimates of break opening times for a full double-ended rupture derived from mechanical response analyses show that the quickest opening time for large bore PWR piping is on the order of 100 milliseconds. However, the presumption of an instantaneous break opening time and resultant shock wave remains in regulatory guidance applicable to debris generation. Regulatory guidance contained in Revision 3 (proposed) to RG 1.82 states:

[3] Instantaneous break opening is simulated in debris generation tests through the use of fast opening rupture disks designed to open in a time span of approximately one millisecond.

[4] Appendix K to 10 CFR Part 50 covers required and acceptable features of ECCS evaluation models designed to address core response and ECCS cooling performance following a design basis LOCA event. These analyses are performed to demonstrate compliance with 10CFR50.46 performance criteria addressing peak cladding temperature, maximum cladding oxidation and core coolability (flow channel blockage resulting from fuel rod ballooning). Separate analyses, using other “non Appendix K” models are used to demonstrate compliance with 10CFR50.46 criteria not addressed by Appendix K.

> *The shock wave generated during postulated pipe break and the subsequent jet should be the basis for estimating the amount of debris generated and the size or size distribution of the debris generated within the zone of influence.*

The assumption of an instantaneous opening of the break is an unwarranted conservatism, and leads to a significant overestimation of the debris generation potential for a postulated break.

Determination of the amount of debris that is generated for a given break is also complicated by the complexity in modeling a three-dimensional jet of two-phase fluid expanding into a region composed of a multitude of materials in widely varying geometric configurations. A number of conservative simplifications of the problem have been proposed and used. One method for estimating the amount of debris generated by a postulated LOCA is to define a spherical zone of influence. The size of the zone of influence is dependent of the size of the break and on the materials considered within the zone. Once the zone of influence is defined, all materials within the zone are assumed to be damaged. The simplicity of these models inevitably results in an overestimation of the quantity of debris generated by a postulated break.

The quantity of debris that can be generated by a break based on assumptions and conservatisms cited above can be seen in results presented in NUREG/CR-6762 (Reference 3). The reported results of debris-generation simulations show debris volumes of 1700 ft^3 for a Large LOCA (> 6 in. diameter), compared to 40 ft^3 for a Medium LOCA (4 to 6 in. diameter) and 25 ft^3 for a Small LOCA (2 to 4 in. diameter).

In summary, current guidance calls for ECCS performance to be assessed in response to the most limiting set of conditions. One of the main controlling factors in calculations to assess ECCS recirculation performance is debris generation. The set of assumptions called for by current regulatory guidance result in ECCS performance being assessed in response to a spectrum of break sizes and locations. The probability of the limiting size break(s) occurring at specific locations is not accounted for in these calculations. The debris generation occurring at this limiting break size/location is then conservatively estimated based on models that are constructed from

1. unrealistic break characteristic assumptions (introducing phenomena that would not be expected to occur), and
2. a conservative expansion of limited test data, necessitated by the wide variety of materials and configurations involved and the large uncertainties associated with expansion of small scale experiments to PWR conditions.

The inevitable consequence of the current analysis methodology is an ECCS recirculation design that is based (focused) upon an extremely low probability event scenario.

In response to the large debris generation values resulting from the current approach, licensees may find it necessary to proactively reduce the debris generation potential in ways that may be detrimental to operation. Utilities may conclude that the only practical way to reduce the debris generation source term to a manageable size is to limit break size by installing (reinstalling) guard pipes, piping restraints, or other similar devices. The irony of such a change is that the justification for removal of such devices from plant designs origianlly was, in part, the low frequency of the same postulated breaks that would now be responsible for their return. The end result of such action is that the reactor coolant piping would be less accessible than was the case prior to these modifications. The modifications will result in less accessibility inside containment. This, in turn, will result in making the performance of some inspections no longer practical, cause other inspections to take longer, and cause plant personnel to receive increased doses for routine maintenance and inspection procedures.

Physical modification of the containment sump screen as a means to address GSI-191 concerns will likely be considered by many licensees. While an increase in sump screen area is an appropriate and perhaps necessary means to address GSI-191, large increases in sump screen area can have unintended consequences and every means should be taken to ensure that the size of the screen is appropriate for the issue. The large debris loadings resulting from non-mechanistic modeling of breaks could dominate the sizing requirements for containment sumps which, in turn, could lead to screen area requirements that lead to modifications that compromise other aspects of plant design and operation. Depending on the location of the containment sump, a large increase in screen area could result in an encroachment on reactor coolant piping. This may require additional plant modifications, such as the addition of piping restraints to preclude damage to the enlarged sump screen. Large increases in sump screen flow areas are also likely to greatly impede access inside containment. This would make maintenance and inspection activities more difficult, and potentially impractical.

3 Description of Proposed Model

As discussed in Section 2, current guidance calls for ECCS and CSS recirculation performance to be evaluated for a full range of break sizes across a full range of break locations. These calculations are performed to demonstrate that the ECCS can meet requirements for long-term cooling per 10 CFR 50.46(b)(5). In order to determine the quantity of debris that is generated as a direct consequence of the break it is necessary to specify the characteristics of the postulated break.

The following section summarizes the break characteristic models currently used to meet ECCS performance requirement specified in 10 CFR50.46. This is followed by a summary of the proposed approach for debris generation modeling.

3.1 Break Characteristic Models Currently Used to Meet 10 CFR 50.46

There are currently two general methods for assigning the break characteristics for use in meeting requirements of 10 CFR 50.46. These are:

1. Models for In-core Thermal Hydraulic Response and Mass/Energy Release (Appendix K Models)
2. Models for In-core Structural Response (LOCA Forces Models)

Appendix K to 10 CFR Part 50 covers required and acceptable features of ECCS evaluation models designed to address core response and ECCS cooling performance following a design basis LOCA event. These analyses are performed to demonstrate compliance with 10CFR50.46 performance criteria addressing peak cladding temperature (10CFR50.46(b)(1)), maximum cladding oxidation (10CFR50.46(b)(2) and core coolability[5] (10CFR50.46(b)(4)). Separate analyses are performed to demonstrate compliance with the core coolability criterion of 10CFR50.46 by calculating the impact of break forces on vessel internals (e.g., fuel assemblies).

3.1.1 Models for In-core Thermal Hydraulic Response and Mass/Energy Release

For the purpose of demonstrating compliance with 10 CFR50.46 (b)(1) and (b)(2) (peak cladding temperature and maximum cladding oxidation) and for determining mass and energy release for containment response calculations, the postulated break is assumed to be an instantaneous double-ended opening of a pipe up to and including the largest piping in the reactor coolant system. These calculations are deterministic in that they do not take into consideration the frequency of piping rupture of a given size. The calculations are also non-mechanistic since no

[5] Appendix K analyses address core coolability primarily from the impact of flow channel blockage resulting from clad ballooning.

known failure mechanism can lead to an instantaneous pipe rupture[6]. The assumption of an instantaneous break opening vs. a more defensible opening time (e.g., 100 milliseconds), has little impact on the associated in-core thermal hydraulic calculations.

The assumption of an instantaneous break opening time does have a significant impact on calculations performed for the purpose of meeting other 10 CFR50.46 requirements.

3.1.2 Models for In-core Structural Response

Calculations performed to demonstrate compliance with 10 CFR 50.46(b)(4) (coolable geometry) typically incorporate the likelihood of various break locations in accordance with the provisions of GDC-4, using “leak-before-break” analysis techniques. Such consideration allows the elimination from the design basis of the dynamic effects of pipe rupture in piping systems so qualified. For piping systems that have not been qualified for “LBB exclusion” the analyses determine the effective break area resulting from the postulated break and determine, based upon the break forces, existing structures and restraints, the piping displacement. The effective break area, taking into account limited displacement, results in an effective break area that is, in most cases, significantly less than the full pipe diameter. Further, in select applications, the calculations apply a realistic break opening time (BOT) based on the consideration of fracture mechanics and dynamic system structural analyses. Realistic BOT’s are typically calculated using finite element dynamic analysis methods based on the assumption that the crack is developed instantaneously. Although testing and analysis results indicate a finite crack propagation time, this is conservatively neglected in the BOT determination. While BOT has relatively little effect in the long term on the blowdown transient, a realistic time for the break to develop to its full break area can have a considerable effect in the initial stages of the blowdown.

3.2 Application of Current Models to Local Debris Generation

For the purposes of demonstrating compliance with 10CFR50.46(b)(5) (long-term cooling), either of the above two approaches for defining break characteristics could be considered, but each has noted limitations. The modeling characteristics used for in-core thermal hydraulic analyses (e.g., instantaneous break opening time) should not be considered appropriate for use in debris generation calculations because they are unrepresentative of break opening behavior and lead to an overly conservative estimation of debris quantities. The methods utilized for in-core structural analyses to demonstrate compliance with 10 CFR50.46(b)(4) are considered more appropriate for debris generation calculations, however, full application is constrained since experiments that have been conducted to determine debris generation have typically modeled instantaneous break openings using fast opening rupture disks. As such there is little

[6] For this discussion, instantaneous can be considered to be a break opening time less than ~10 milliseconds, i.e., the break opening interval necessary to generate a shock wave.

experimental data available to support the debris generation that would occur for realistic break opening times.

The exclusion from consideration of LBB qualified piping that is currently applied in structural analyses for 10CFR 50.46(b)(4) could be considered in debris generation calculations and was proposed by NEI in a letter to NRC dated October 4, 2002 [Reference 4]. The NRC has not provided a written response to the NEI request to utilize this approach; either accepting it as an appropriate method or by identifying the basis for its denial. Therefore, use of LBB piping exclusions remains a potential method for use in debris generation calculations.

The uncertainty associated with the schedule for NRC staff response on the proposed use of LBB exclusions led to the development of an alternative proposal that incorporates attributes of the two currently accepted pipe break characterization models.

3.3 Proposed Break Characteristic Model for Debris Generation

The proposed model utilizes fracture mechanics considerations to establish a maximum credible flaw size in qualified piping. The area associated with this flaw size is then increased by three orders of magnitude to determine the break size (area) to be used in debris generation calculations. The calculation of the debris quantities generated from these pipe break areas have as a basis a conservative estimate of the actual behavior of the piping material under normal and off-normal conditions.

The proposed model for determining the size of the pipe breach will utilize stable yet detectable leakage cracks already calculated for PWR primary coolant piping as a key input parameter. Compilations of stable leakage cracks that have been calculated for a number of PWR plants are presented in Table 3-1, along with the crack opening area for each crack. As seen in this table, the crack opening areas of the stable leakage cracks are quite small and would have little debris generating capability.

For the purposes of conservatively calculating debris generation for a postulated through-wall flaw, the breach area associated with the stable leakage crack is increased by a factor of 1000. Use of a pipe breach area that is three orders of magnitude larger than the calculated area of the associated stable leakage crack results in maximum pipe breach areas for use in evaluating debris generation as follows:

- For B&W / Framatome plants 83 in^2
- For Combustion Engineering plants 40 in^2
- For Westinghouse plants 40 in^2

Using a circular hole for the break geometry, the equivalent hole diameters for the break areas identified above are calculated as:

- For B&W / Framatome plants 10.28 inch diameter
- For Combustion Engineering plants 7.10 inch diameter
- For Westinghouse plants 7.10 inch diameter

The geometry of the circular hole is assumed to be in the pipe centered at the midpoint of the through-wall crack or flaw. The proposed model could be applied to piping segments for which fracture mechanics analysis results are available for determining stable leakage crack areas. Piping segments for which the calculation of stable leakage cracks do not exist will assume the full cross sectional area of the inside diameter of the pipe for the purposes of debris generation.

It is important to note that the proposed break model is used only for the determination of dynamic effects impacting local debris generation. All other phenomena affecting long-term cooling, such as break flow, global effects within containment, debris transport, and screen blockage, will utilize a full range of break sizes and locations (up to full double-ended guillotine rupture of largest pipe).

3.4 Comparison of Current and Proposed Break Characteristic Models

Table 3-2 provides a comparison of key attributes of current break modeling used to demonstrate compliance to 10 CFR 50.46 and break modeling proposed for use in calculating debris generation potential for postulated breaks.

Table 3-1: Stable Leakage Crack Sizes for PWR Primary Loop Piping

Westinghouse Designed Plants

Pipe OD (in)	Pipe Wall Thickness (in)	Stable Crack Length [Note 1] (in)	Crack Opening Area (in^2)
32.12 – 37.75	2.21 – 3.27	2.5 – 8.55	0.030 – 0.040

CE Designed Plants

Case	Pipe Wall Thickness (in)	Stable Crack Length [Note 1] (in)	Crack Opening Area (in^2)
Circumferential Crack in Pump Discharge	3.0	7.0	0.040
Circumferential Crack in Hot Leg	3.75	7.0	0.040
Axial Slot in Pump Suction Elbow	3.0	4.0	0.040
Circumferential Crack in Pump Suction Elbow	3.0	11.0	0.040
Circumferential Crack in Pump Discharge	2.5	7.0	0.040

B&W Designed Plants

Applicable Plants	Piping Segment	Stable Crack Length [Note 1] (in)	Crack Opening Area (in^2)
Plants A, B, C, D, E, and F	Cold Leg, Straight	9.2	0.075
	Cold Leg, Elbow	9.0	0.075
	Hot Leg, Straight	8.0	0.068
	Hot Leg, Elbow	10.8	0.083
Plant G	Cold Leg, Straight	9.39	0.065
	Cold Leg, Elbow	9.41	0.074
	Hot Leg, Straight	11.39	0.074
	Hot Leg, Elbow	12.63	0.083

Notes: 1) Stable crack length is based on a leak rate of 10 gpm.

Table 3-2
Comparison of Break Characteristic Models for Debris Generation

Break Characteristic Models	**(A) In-core T/H, M&E Model**	**(B) In-core Structural Response Model**	**(C) LBB Application Proposed in 10/04/02 NEI Letter**	**(D) Fracture Mechanics Approach**
Current Application	Used to support analyses that demonstrate compliance to 10 CFR 50.46(b)(1) and 10.46(b)(2)	Used to support analyses that demonstrate compliance to 10 CFR 50.46(b)(4)	Proposed for use in analyses that demonstrate compliance to 10 CFR 50.46(b)(5)	Proposed for use in analyses that demonstrate compliance to 10 CFR 50.46(b)(5)
Break Opening Time	Instantaneous	Varies by vendor. Ranges from instantaneous to value determined by fracture mechanics and structural analysis.	Instantaneous	Instantaneous
Break Locations	All high-energy RCS piping (no exclusions)	All non-LBB qualified piping (LBB piping excluded per GDC-4)	All non-LBB qualified piping (debris generation from LBB piping excluded per GDC-4)	All high-energy RCS piping (no exclusions)
Effective Break Size	Full double-ended guillotine	Effective break area determined based upon calculated piping displacement for non LBB piping	Full double-ended guillotine	Break size for debris generation determined using fracture mechanics principles
Similar Applications	In-core T/H analyses performed to demonstrate compliance with 10 CFR 50.46(b)(1) and (2)	Structural calculations performed to demonstrate compliance with 10 CFR 50.46(b)(4)	Similar to In-core Structural Response model but conservatively retains modeling of instantaneous DEG opening for non LBB piping.	Similar to In-core T/H analysis model with exception of fracture mechanics based break size modeling in lieu of double-ended break

4 Technical Bases for Proposed Break Characteristic Model for Debris Generation

Significant testing and analyses have been performed to characterize the behavior and response of flaws that may be present in reactor coolant piping. These efforts have provided a comprehensive and realistic basis for defining stable through-wall cracks in large PWR reactor coolant piping. The fracture mechanics analytical techniques, applied reactor coolant system loadings, actual material properties, and installed leak detection capabilities are discussed below. Combined in a comprehensive plant-specific analysis, these techniques demonstrate that a conservatively postulated through-wall crack would be large enough to be detected by plant leak detection systems, yet remain stable in the full power operating environment, including faulted loading conditions (References 5, 6, and 7).

The following discussion is applicable to and includes both stainless steel and carbon steel piping with stainless steel clad.

4.1 Piping System Loading Conditions

The loads resulting from both normal operating conditions and faulted plant conditions are applied in the evaluation of both the stability and leakage of through-wall cracks or flaws. These conditions conservatively bound other loading conditions on the piping systems of interest. The components for normal loads are pressure, dead weight and thermal expansion.

Normal condition loads are used in the leak rate calculations. For a given length crack or flaw, the application of normal operating condition loads determines the flow area and leakage rate.

For the faulted condition loading, loads associated with the safe shutdown earthquake (SSE) are considered in addition to the normal loads. This load combination is used in the demonstration of crack stability.

4.2 Material Characterization

Material properties for the fracture mechanics evaluations are taken from the certified material test reports (CMTRs). Properties are determined both at room temperature and/or at operating temperature. Forged and cast stainless steels both typically have high fracture toughness values. However, cast stainless steels are subject to thermal aging during service. This thermal aging causes an elevation in the yield strength of the material and a degradation of the fracture toughness. Detailed fracture toughness testing has been performed for cast stainless steel, the results of which are used to establish the end-of-service life (40 or 60 years, as determined by

the plant) fracture toughness values for specific materials. Detailed fracture toughness testing has also been performed for the low alloy ferritic steel pipe materials and associated weldments.

4.3 PWR Primary Loop Piping Leak Rate Determination

The determination of leakage crack size is based on the leak detection capability of the plant leak detection systems.

LEAK DETECTION

Early detection of leakage in components of the reactor coolant pressure boundary (RCPB) system is necessary to identify deteriorating or failed components and minimize the release of fission products. Regulatory Guide (R.G.) 1.45 (Reference 8) describes acceptable methods to select leakage detection systems for the RCPB.

R.G. 1.45 specifies that at least three different detection methods should be employed. Plant sump level monitoring and airborne particulate radioactivity monitoring are specifically recommended. A third method can be either monitoring of condensate flow rate from air coolers or monitoring of airborne gaseous activity.

R.G. 1.45 also recommends that flow rates from identified and unidentified sources should be monitored separately, the former to an accuracy of 10 gpm and the latter to an accuracy of 1 gpm. (Note that plants with coolant activity levels sufficiently low as to suggest radiation monitoring will not detect leakage with an accuracy of 1 gpm have implemented alternate leakage monitoring methods.) Indicators and alarms for leak detection should be provided in the main control room. The sensitivity and response time for each leakage detection system used should be such that each is capable of detecting 1 gpm or less in one hour.

All US PWR's meet or exceed the leak detection guidance of the preceding paragraph. Specific leak detection capabilities of a plant are identified in its technical specifications.

LEAK RATE CALCULATIONS

The first step for calculating the leak rates is to determine the crack opening area when the pipe containing a postulated through-wall flaw is subjected to normal operating loads. Using the crack opening area, leak rate calculations are performed for the two-phase choked flow condition. From the actual pipe stress analysis, deadweight, normal 100% power thermal expansion and normal operating pressure loads are used in the calculation of the crack opening area and hence the leak rate. All loads are combined by the algebraic summation method.

It is noted that a through-wall circumferential flaw is postulated in the piping that would yield a leak rate of 10 gpm. A flaw that results in a 10 gpm flow rate is used to assure a factor of 10 in margin between the calculated leak rate compared to the leak detection capability of the plant.

4.4 Fracture Mechanics Evaluation

The stability of a calculated leakage crack or flaw is demonstrated based on material properties and faulted applied load conditions. Based on extensive analyses, significant margins on crack stability have been demonstrated for the calculated leakage cracks.

4.4.1 Local Failure Mechanism

The local mechanism of failure is primarily dominated by the crack tip behavior in terms of crack-tip blunting, initiation, extension and finally crack instability. Local stability will be assumed if the crack does not initiate at all. It has been accepted (Reference 9) that the initiation toughness measured in terms of J_{Ic} from a J-integral resistance curve is a key material parameter defining crack initiation. If, for a given load, the applied J-integral value is shown to be less than the J_{Ic} of the material, then the crack will not initiate (Reference 9).

If the initiation criterion is not met, then stability is said to exist when the applied tearing modulus value is less than the material tearing modulus value, and the applied J-integral value is less than the J_{max} value of the material.

4.4.2 Global Failure Mechanism

Determination of the conditions which lead to failure in stainless steel is done with plastic fracture methodology because of the large amount of deformation accompanying fracture. One accepted method for predicting the failure of ductile material is the plastic instability method, based on traditional plastic limit load concepts, but accounting for strain hardening and taking into account the presence of a flaw. The flawed pipe is predicted to fail when the remaining net section reaches a stress level at which a plastic hinge is formed. The stress level at which this occurs is termed as the flow stress. The flow stress is generally taken as the average of the yield and ultimate tensile strength of the material at the temperature of interest. This methodology has been shown to be applicable to ductile piping through a large number of experiments (Reference 9).

5 Compliance with Applicable Regulations

5.1 Regulatory Requirements

Title 10, Section 50.46 of the Code of Federal Regulations (10 CFR 50.46) requires that licensees design their ECCS systems to meet five criteria, one of which is to provide the capability for long-term cooling. Following successful system initiation, the ECCS shall be able to provide cooling for a sufficient duration that the core temperature is maintained at an acceptably low value. In addition, the ECCS shall be able to continue decay heat removal for the extended period of time required by the long-lived radioactivity remaining in the core. The requirements of 10 CFR 50.46 are in addition to the general ECCS cooling performance design requirements found elsewhere in 10 CFR Part 50, in particular the system safety function requirements in General Design Criterion (GDC) 35 of Appendix A to 10 CFR Part 50.

The Containment Spray System is required to meet, in part, GDC 38 and GDC 40 of Appendix A to 10 CFR Part 50. These criteria specify requirements regarding heat removal from the reactor containment following any loss-of-coolant accident and to control fission products that may be released into the reactor containment.

5.2 Current Regulatory Guidance

The regulations are not specific as to the manner in which ECCS "capability for long-term cooling" is to be demonstrated. The regulations are also not specific as to whether or how debris generation, as a direct result of a design basis LOCA, is to be determined. Methods that are acceptable to the NRC for determining whether designs maintain a "capability for long-term cooling" and that meet regulatory requirements are currently specified in regulatory guidance. The applicable regulatory guide for this purpose is Regulatory Guide 1.82, *Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident*, Revision 3 (proposed). This regulatory guide has undergone significant revision since its initial release in 1974, reflecting new insights and results of ongoing research. The revisions also reflect significant changes in the regulatory treatment of debris generation. As is discussed in the following section, the regulatory treatment has progressed from a fully non-mechanistic treatment (Rev. 0) which only accounts for the effect of debris generation on containment sump performance, to a mechanistic treatment that allows for consideration of the probability of pipe rupture (Rev. 1), to a mechanistic treatment with no allowance for consideration of the probability of pipe rupture (Rev. 2 & proposed Rev. 3).

The following paragraphs summarize the evolution of regulatory guidance addressing debris generation following a LOCA event.

<u>Regulatory Guide 1.82, Revision 0</u>

The containment recirculation portions of the ECCS and CSS for U.S. PWRs were originally designed and licensed in conformance with Regulatory Guide 1.82 Revision 0[7] or predecessor guidance. In accordance with guidance contained in Revision 0 to RG 1.82, the 'capability for long-term cooling" was demonstrated in a non-mechanistic fashion, by assuming 50% of the containment sump screen area was unavailable for flow due to blockage.

Debris Generation Guidance 1974-1985

- o Applicable guidance contained in Regulatory Guide 1.82, Revision 0
- o Non-mechanistic treatment
- o Assume accident debris results in 50% blockage of containment sump screen(s)

<u>Regulatory Guide 1.82, Revision 1</u>

Regulatory Guide 1.82 was revised in November 1985 as part of the resolution to Unresolved Safety Issue (USI) A-43, "Containment Emergency Sump Performance." The staff concluded at that time that no new requirements would be imposed on licensees; however, the staff did recommend that Revision 1 to RG 1.82 be used as guidance for the conduct of 10 CFR 50.59 reviews dealing with change out and/or modification of thermal insulation installed on primary coolant system piping and components. As part of this revision, guidance was added that called for "*evaluation or confirmation of …debris effects (e.g. debris transport, interceptor blockage, and head loss) …to ensure that long-term recirculation cooling can be accomplished*."

For the purpose of defining break or rupture locations, Revision 1 to RG 1.82 refers the user to Standard Review Plan (SRP) Section 3.6.2[8], which provides guidance for selecting the number, orientation, and location of postulated ruptures within a containment. SRP 3.6.2 provides instruction and guidance to NRC staff reviewers regarding break and crack location criteria and methods of analysis for evaluating the dynamic effects associated with postulated breaks and cracks in high- and moderate-energy fluid system piping. SRP 3.6.2 is the primary review guidance for ensuring that a design meets the requirements of General Design Criterion (GDC) 4. GDC 4 requires that structures, systems, and components important to safety shall be designed to accommodate the effects of postulated accidents, including appropriate protection against the dynamic and environmental effects of postulated pipe ruptures.

Compliance with GDC 4 requires that nuclear power plant structures, systems, and components important to safety be designed to accommodate the effects of, and be compatible with, environmental conditions associated with normal operation, maintenance, testing, and

[7] Regulatory Guide 1.82, *Sumps for Emergency Core Cooling and Containment Spray Systems*, Revision 0, June 1974
[8] Standard Review Plan, Section 3.6.2, "*Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping*"

postulated accidents, including loss-of-coolant accidents. These structures, systems, and components are to be protected against pipe-whip and discharging fluids. GDC-4 allows such dynamic effects to be excluded from the design basis if the probability of pipe rupture is shown to be extremely low.

For determination of debris generation from identified break locations, RG 1.82 Revision 1 identifies a multiple region insulation debris model developed in NUREG-0897 as an acceptable model.

Debris Generation Guidance 1985-1996

- o Applicable guidance contained in Regulatory Guide 1.82, Revision 1
- o Available for use, however, PWR licensees not required to adopt and revise design basis
- o Break locations determined per guidance contained in SRP 3.6.2 and Branch Technical Position EMEB 3-1
- o SRP 3.6.2 provides guidance for exclusion of dynamic effects of break locations (in accordance with GDC-4) based on low probability of piping rupture under design basis conditions
- o Debris generation from identified break locations determined using experimentally developed multi-region insulation destruction model

<u>Regulatory Guide 1.82, Revision 2</u>

Regulatory Guide 1.82 was revised a second time in May 1996 to alter the debris blockage evaluation guidance for boiling water reactors. While the Introduction section notes that only the section concerning BWRs were changed from Revision 1, a noted change to sections applicable to PWRs is the deletion of any reference to SRP section 3.6.2 for use in determining break locations.

Debris Generation Guidance 1996-2003

- o Applicable guidance contained in Regulatory Guide 1.82, Revision 2
- o Available for use, however, PWR licensees not required to adopt and revise design basis
- o Removed allowance for consideration of extremely low probability of rupture per SRP 3.6.2, BTP EMEB 3-1 and GDC-4.
- o No specific guidance on break locations or break sizes for PWRs. BWR guidance revised to include consideration of debris generation from a range of break sizes, locations and other properties to provide assurance that most severe postulated LOCAs are calculated.

Debris generation from identified break locations determined using experimentally developed insulation destruction models

Regulatory Guide 1.82, Revision 3 (Draft)

In a proposed revision 3 to RG 1.82, guidance for debris generation is revised primarily to provide more detailed guide for PWRs. Consistent with Revision 2, the guidance calls for determination of debris generation for a range of break sizes, break locations, and other properties to provide assurance that the most severe postulated LOCAs are calculated.

Debris Generation Guidance 2004(?)

- o Proposed revision 3 to Regulatory Guide 1.82
- o Consistent with Revision 2, no allowance for consideration of extremely low probability of rupture per SRP 3.6.2, BTP EMEB 3-1 and GDC-4.
- o PWR guidance revised to include consideration of debris generation from a range of break sizes, locations and other properties to provide assurance that most severe postulated LOCAs are calculated.
- o Debris generation from identified break locations determined using experimentally developed insulation destruction models

5.3 Precedence for Consideration of Fracture Mechanics in Meeting 10CFR50.46 Criteria

5.3.1 GDC-4 – Leak Before Break

In October 1987, General Design Criterion (GDC) 4 in Appendix A to 10 C.F.R. Part 50 was revised to allow the use of fracture mechanics to exclude dynamic effects from the design basis of qualified piping (i.e., piping for which the probability of rupture can be demonstrated to be extremely low). Specifically:

> *"**Criterion 4 - Environmental and dynamic effects design bases**. Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. <u>However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping</u>."*
> [Emphasis added]

The broad-scope rule introduced an acknowledged inconsistency in the design basis by excluding the dynamic effects of postulated pipe ruptures while retaining non-mechanistic pipe rupture for containments, ECCS, and environmental qualification (EQ) of safety-related electrical and mechanical equipment.

The NRC staff subsequently clarified its intended treatment of the containment, ECCS, and EQ in the context of LBB applications in a request for public comment on this issue that was published on April 6, 1988 (53 FR 11311). In its clarification the staff stated that the effects resulting from postulated pipe breaks can be generally divided into local dynamic effects and global effects. Local dynamic effects of a pipe break are uniquely associated with that of a particular pipe break. These specific effects are not caused by any other source or even by a postulated pipe break at a different location. Examples of local dynamic effects are pipe whip, jet impingement, missiles, local pressurization, pipe break reaction forces, and decompression waves in the intact portions of that piping or communicating piping. Global effects of a pipe break need not be associated with a particular pipe break. Similar effects can be caused by failures from such sources as pump seals, leaking valve packings, flanged connections, bellows, manways, rupture disks, and ruptures of other piping. Examples of global effects are gross pressurizations, temperatures humidity, flooding, loss of fluid inventory, radiation, and chemical condition.

The application of LBB technology eliminates the local dynamic effects of postulated pipe breaks from the design basis. However, global effects may still be caused by something other than the postulated pipe break. Since the global effects from the postulated pipe break provide a reasonably conservative design envelope, the NRC staff continue to require the consideration of global effects for various aspects of the plant design , such as EQ, ECCS, and: the containment.

5.3.2 Industry Proposal to Apply GDC-4 Exclusion to Local Debris Generation

In a letter dated October 4, 2002 (Reference 4), NEI provided its view on the application of the LBB considerations of GDC-4 to local debris generation from a postulated break. NEI presented the position that debris generation, as a result of break jet expansion and impingement forces, is a dynamic effect uniquely associated with pipe rupture and, as such, is appropriately encompassed within the scope of the revised GDC-4.

In its letter, NEI made the following points:

- Debris generation within the zone of influence of a break is a local dynamic effect covered by GDC-4

  The dynamic effects addressed by GDC-4 are delineated in the Federal Register notice that modified GDC-4 (52 FR 41288): *"Dynamic effects of pipe rupture covered*

by this rule are missile generation, pipe whipping, pipe break reaction forces, jet impingement forces, decompression waves within the ruptures pipe and dynamic or nonstatic pressurization in cavities, subcompartments and compartments." The initial blast wave exiting a DEGB and the ensuing break jet expansion and impingement forces are the dominant contributors to debris generation following a LOCA. Other contributors are pipe whip and pipe impact.

- Debris generation does not fall within the scope of functional and performance requirements for containment, ECCS and EQ that were retained in the GDC-4 revision

The rule change acknowledged inconsistencies in the design basis by excluding the dynamic effects of postulated pipe ruptures while retaining non-mechanistic pipe rupture for containments, ECCS, and environmental qualification (EQ) of safety-related electrical and mechanical equipment. As stated in 53 FR 11311, , "*…local dynamic effects uniquely associated with pipe rupture may be deleted from the design basis of containment systems, structures and boundaries, <u>from the design basis of ECCS hardware</u> (such as pumps, valves, accumulators, and instrumentation), and from the design bases of safety related electrical and mechanical equipment when leak-before-break is accepted.*" (Emphasis added)

- For PWR licensees, LBB considerations for debris generation would be applied as part of a revision to design bases that specifically incorporates mechanistic processes addressing debris generation, debris transport and debris blockage.

The ECCS recirculation designs for most PWR plants in the U.S. are based on guidance provided in Revision 0 of Regulatory Guide 1.82, *Sumps for Emergency Core Cooling and Containment Spray Systems*. This guidance accounts for screen blockage in a non-mechanistic fashion by assuming that one-half of the vertical screen area of the sump is unavailable for recirculation flow. Since the impact of LOCA-generated debris on sump blockage is not addressed directly through this approach, consideration of leak-before-break for LOCA-generated debris would have no affect on ECCS designs that utilize this guidance in their design bases. Subsequent revisions to Regulatory Guide 1.82 (Revision 1 – November 1985, Revision 2 – May 1996, Revision 3 – draft) have incorporated a more mechanistic process that provides a more phenomenologically accurate, but conservative, estimate of the debris blockage that PWR sumps could experience following a LOCA.

In a letter to NRC dated April 30, 2003 (Reference 10), NEI recommended that the proposed revision 3 to Regulatory Guide 1.82 incorporate language that acknowledges treatment of debris generation under the LBB provisions of GDC-4. Specifically, NEI recommended that the following paragraph be included in the proposed revision to Regulatory Guide 1.82:

"Consistent with the requirements of 10 CFR 50.46, debris generation should be calculated for a number of postulated LOCAs of different sizes, locations, and other properties sufficient to

provide assurance that the most severe postulated LOCAs are addressed. In accordance with GDC-4, dynamic effects associated with postulated pipe ruptures (including local debris generation) may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping."

5.3.3 Status of NRC Response to NEI Proposal

NRC has not issued a written response to the NEI positions on use of GDC-4 to exclude local debris generation as a local dynamic effect for qualified piping. NRC staff have stated during public meetings that they believe that the requested exclusion of local debris generation is not in accordance with the requirements of 10 CFR 50.46. Specifically, 10 CFR50.46(c)(1) which defines LOCAs as "*hypothetical accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, from breaks in pipes in the reactor coolant pressure boundary* <u>*up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system*</u> [emphasis added]." The preliminary staff position appears to preclude the use of GDC-4 in analyses performed to meet the "long-term cooling" requirements of 10 CFR 50.46 criteria. Although, it is noted that NRC has reviewed and approved break-size exclusions allowed by GDC-4 in analyses performed to meet the "coolable geometry" criterion of 10 CFR 50.46.

5.3.4 Use of Fracture Mechanics to Meet 10CFR50.46 "Coolable Geometry" Criterion

Subsection (b) of 10 CFR50.46 specifies 5 criteria that must be met by the ECCS. They are:

1) Peak cladding temperature. The calculated maximum fuel element cladding temperature shall not exceed 2200º F.

2) Maximum cladding oxidation. The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

3) Maximum hydrogen generation. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

4) Coolable geometry. Calculated changes in core geometry shall be such that the core remains amenable to cooling.

5) Long-term cooling. After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

The first three criteria (peak cladding temperature, maximum cladding oxidation and maximum hydrogen generation) are met through the use of approved ECCS models that meet requirements of 10CFR50.46(a). These models meet either the requirements of Appendix K to 10 CFR50.46 or make use of NRC approved best estimate models. These “core response” models model the full range of break sizes (up to and including full double-ended guillotine break of the largest pipe in the reactor coolant system) in accordance with 10CFR50.46(c)(1). The assumptions on break opening time range from “instantaneous” (a requirement for all Appendix K models) to 1 millisecond (for some NRC approved Best Estimate LOCA models). Fracture mechanics considerations are not taken into account in either the Appendix K models or Best-estimate models.

The fourth criterion (Coolable Geometry) is demonstrated through the performance of dynamic analyses of the assembled reactor vessel, internals, and fuel and is performed for a range of postulated LOCAs in accordance with applicable regulatory guidance. The results of these analyses provide assurance that the forces resulting from the postulated LOCAs will not result in fuel assembly deformation to an extent that would lead to a loss of “coolable geometry.”

For most, if not all PWRs, the range of LOCAs that are considered is limited through application of Leak-Before-Break considerations, supported by fracture mechanics. Using NRC approved guidance, forces resulting from breaks in LBB qualified piping are not included in the set of analyses performed to demonstrate compliance to 10 CFR 50.46(b)(4).

The proposed use of fracture mechanics to demonstrate compliance with the fifth criterion (long-term cooling) is significantly more conservative than the NRC approved methods used to demonstrate “Coolable Geometry.” In the modeling that will be used to demonstrate long-term cooling following a postulated LOCA, a full range of break sizes (up to full double-ended guillotine rupture of largest pipe) will continue to be addressed for all relevant phenomena with the exception of the dynamic effects which impact local debris generation. All other phenomena affecting long term cooling (e.g., break flow, global effects within containment, debris transport, screen blockage) will model a full range of break sizes and locations.

6 Retained Safety Margins

The determination of debris generation that results as a direct consequence of the local dynamic effects of a postulated pipe break is a single step in the larger effort necessary to assess the recirculation performance of the ECCS and CSS following a design basis LOCA.

In order to calculate plant response to a postulated pipe break event and the potential for significant blockage of the containment sump screens, it is necessary to take into account a wide range of phenomena and processes. Figure 6-1 illustrates some of the phenomena and processes that must be considered. These phenomena and processes are highly dependent upon plant design and operation as well as the specifics of the postulated LOCA event. The complexity and multivariate nature of the event progression, coupled with the absence of a comprehensive database addressing the full range of encountered phenomena inevitably leads to a calculation process that accounts for the resulting uncertainties in a conservative manner.

As noted before, typically the analyses investigating ECCS operation during the recirculation phase divide the process into three separate phases: (1) Debris Generation, (2) Debris Transport and (3) Debris Accumulation and Headloss. Each phase of the calculation process, while interdependent, involves its own set of phenomena and uncertainties. Known limitations in the knowledge base of these phenomena and associated calculation methods are typically accounted for in a bounding fashion during each phase of the process. Thus, it is important to note that the more realistic treatment of the debris generation phase using the break characteristics model described in this paper, neither eliminates nor alters the conservative treatment of other phenomena and processes. As such, the overall results from the analyses will retain a significant degree of conservatism.

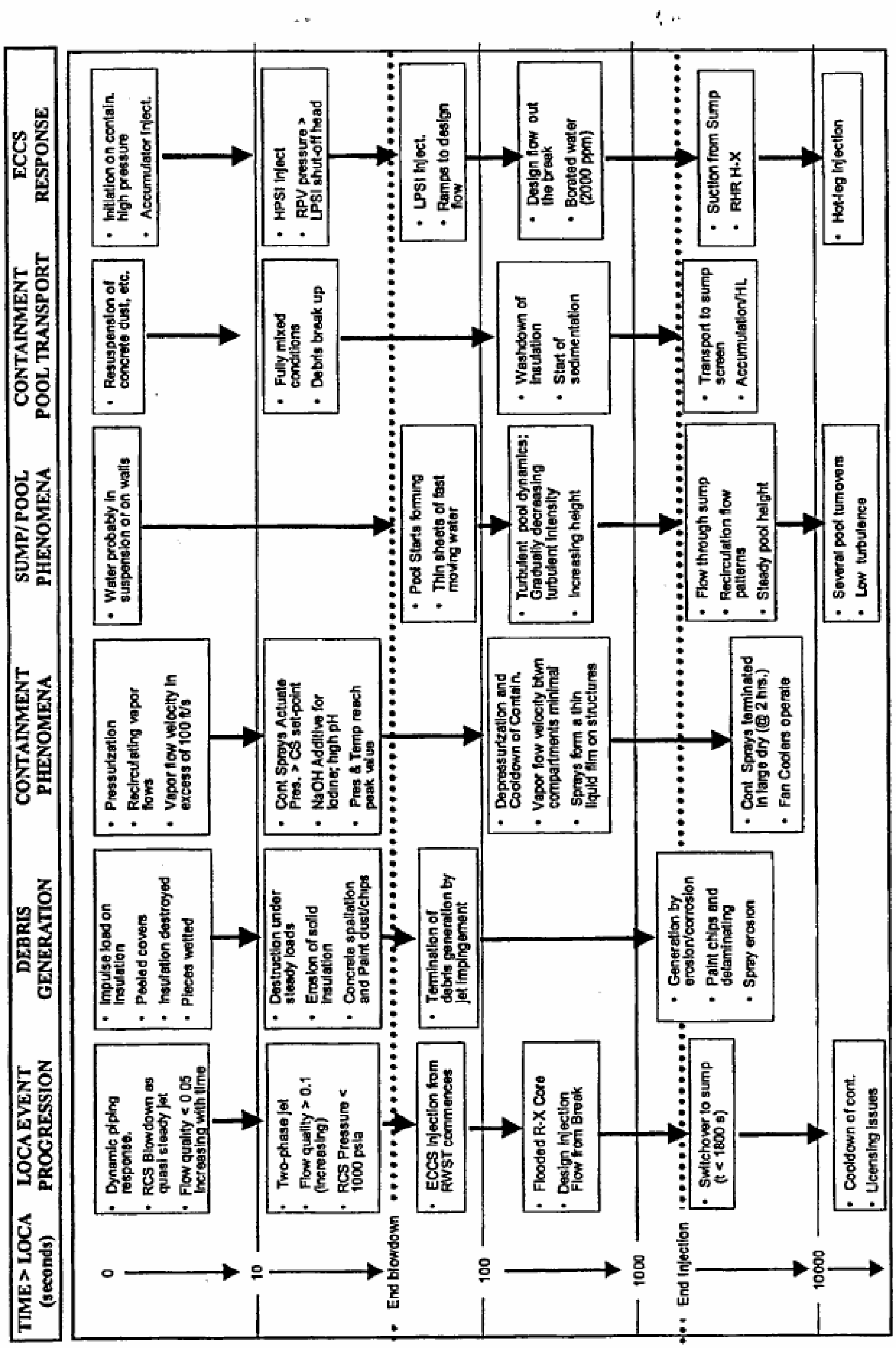

Figure 6-1
PWR LLOCA Accident Progression in a Large Dry Containment (Ref. 1)

7 Summary

This paper outlines a method of using fracture mechanics analysis techniques to define pipe break areas for the evaluation of consequential debris generation for post-accident containment sump performance evaluation. The proposed break characterization model is based on stable leakage crack sizes that generate detectable leaks and have already been calculated for PWR primary coolant piping and, in some cases, surge line piping. The debris generated from the proposed break characteristic model areas are meaningful with respect to sump performance and are based on the actual behavior of the piping material under normal and off-normal conditions.

For added margin, the proposed break characteristic model incorporates a factor of 1000 applied to the flow area of a stable through-wall flaw that produces a 10 gpm leakage rate. The geometry of the breach will be taken to be a circular hole in the pipe of interest.

Fracture mechanics analysis techniques have been used successfully, in conjunction with plant leak detection systems, to determine the size of stable cracks for PWR primary loop piping. The leakage flow of these stable cracks has been evaluated to be 10 gpm, or a factor of 10 above the leak detection capability of PWR plants.

It is therefore concluded that the proposed break characteristic model based on calculated stable leakage cracks using proven fracture mechanics techniques provides an acceptable, conservative, yet realistic approach for the evaluation of containment sump performance.

8 References

1) Regulatory Guide 1.82, Revision 3, *Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident*, February 2003 (proposed)

2) DRF A74-00003, *Evaluation of Existence of Blast Waves Following Licensing Basis Double-Ended Guillotine Pipe Breaks*, Moody, F.J., Green, T.A., August 1996 (Provided in Technical Support Documentation of NEDO-32686-A, Utility Resolution Guidance for ECCS Suction Strainer Blockage)

3) NUREG/CR-6762, Volume 1, *GSI-191 Technical Assessment: Parametric Evaluations for Pressurized Water Reactor Recirculation Sump Performance*, August 2002

4) Letter, Alex Marion, NEI to Gary Holahan, NRC, *Application of Leak-Before-Break Technology to Pipe Break Debris Generation and Request for Public Comment Opportunity*, October 4, 2002

5) WCAP-15131, Revision 1, *Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the D. C. Cook Units 1 and 2 Nuclear Power Plants*, [Proprietary] (This topical report is representative of a large number of plant specific analyses performed for Westinghouse designed plants.)

6) CEN-367-A, Revision 000, *Leak-Before-Break Evaluation of Primary Coolant Loop Piping in Combustion Engineering Designed Nuclear Steam Supply Systems*

7) BAW-1847, Revision 1, *The B&W Owners Group Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping of B&W Designed NSS*, September 1985 [Proprietary] (This topical report is representative of the evaluations performed for the B&W designed plants.)

8) USNRC Regulatory Guide 1.45, *Reactor Coolant Pressure Boundary Leakage Detection Systems*, May 1973

9) NUREG-1061, Volume 3, Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, *Evaluation of Potential for Pipe Breaks*, November 1984

10) Letter, Anthony Pietrangelo, NEI to NRC Rules and Directives Branch, *Comments on Draft Regulatory Guide DG-1107, Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident, (68 Fed. Reg. 13338)*, April 30, 2003