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Utility Resolution Guide for ECCS Suction Strainer Blockage

Volume 4
Technical Support Documentation

Prepared By
BWR Owners' Group

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Utility Resolution Guidance

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Technical Support Documentation
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Technical Support Documentation Utility Resolution Guidance (URG) for ECCS Suction Strainer Blockage

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BWR Drywell Floor Flow Modeling Following Pipe Break Loss-of-Coolant Accident

Prepared

for

Boiling Water Reactor Owners' Group

Prepared by:

GE Nuclear Energy
San Jose, California, USA

September 1996

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TAG96-62

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Background

The BWR Owners' Group ECCS Suction Strainer Committee has investigated the flow field in the pool of water located on the drywell floor following a major pipe break loss-of-coolant accident. The overall objective is to estimate the degree of washdown of insulation and other debris to the wetwell via the numerous vent pipes (Mark I and Mark II)or the weir wall (Mark III) within the containment. The washdown process has been found to depend strongly on the water flow velocities on the floor. A one-dimensional analysis was performed by Bilanin of CDI (Appendix A) which estimated the average water velocity in the circumferential direction for various assumed splashdown area sizes emanating from the broken pipe (Mark I and II only). It was thought initially that the pool flow field is at least two-dimensional and that variations in the velocity in the vertical direction may be ignored. A preliminary scoping study was performed to evaluate the influence of the two-dimensional flow field using the potential flow method to provide indications as to what the two-dimensional flow field might look like. Results of this scoping study indicated a considerable variation in radial and circumferential flow velocities. It appeared from this scoping study that a more detailed three-dimensional analysis would be desirable using available fluid mechanics codes. A three-dimensional analysis was performed using the COMMIX code developed by the Argonne National Laboratory (Reference 1) for use in analyzing thermal hydraulic problems primarily in nuclear reactors. This analysis was initiated for the Mark II drywell configuration and was subsequently redirected to focus exclusively on the Mark III drywell configuration. This report documents the results of the three-dimensional study for Mark III, discusses the preliminary results for the Mark II configuration, and discusses the results relative to the previous one- and two-dimensional studies.

Model Configuration

The particular Mark III drywell configuration analyzed is that of the Perry Plant. The region analyzed is doughnut-shaped, confined on the inside by the reactor pedestal having an outside diameter of 9.63 m (31.58 ft) and on the outside by the weir wall having an inside diameter of 19.86 m (65.17 ft). The height of the weir wall is 4.72 m (15.5 ft).

Only one half of this doughnut shaped region extending from 0 to 180 degrees was analyzed because of symmetry considerations. Plane and elevation views of the analysis region showing the nodal structure used are given in Figures 1 and 2, respectively. In these figures the "I" index indicates the computational cell number in the radial direction (r-coordinate), the "J" index indicates the cell number in the circumferential direction (θ-coordinate) and the "K" index indicates the cell number in the vertical direction (z-direction). It is noted that in the analysis model I=1 to 7, J=1 to 16 and K=1 to 8. However, only the upper layer (i.e. K=8) has I=7 maximum whereas the remaining layers have I=6 maximum. It is also noted that inner cells with I=1 are all empty space.

It is assumed that the computational region is filled with water initially and that water enters vertically at the water surface level (K=8) into a 4.92 m^2 (53.0 ft^2) Splashdown Area located in the area containing cells I=3 to 5 and J=1 as indicated in the model plan view of Figure 1. The incoming flow rate to the Splashdown Area was 39,000/2=19,500 gpm (1.23 m^3/s) for the half section resulting a vertical flow velocity of 0.25 m/s (0.82 ft/s) . Water spills over the outside weir wall near the water surface elevation as denoted in the plane and elevation views of Figures 1 and 2 by the Drain Ring. The outside, vertical edge of the Drain Ring constituting cells I=7 and J=1 through 16 was assigned a surface permeability of 0.33 to account for the contraction that occurs as the water spills over the weir.

The water surface was assumed to be completely uniform and level (i.e. without waves) and had no surface depressions of any kind due to high flow velocities in some part of the flow field. Thus, the analysis region was treated as an enclosure with fixed boundaries having an inlet and an outlet as noted above since the COMMIX code has no free surface modeling capability. The density of the fluid was 973 kg/m^3 (60.7 lb/ft^3) and there were no heat sources or sinks in the system. The value of the constant user specified turbulent viscosity was set equal to a default value of 0.02 Ns/m^2 (0.01344 lbm/ft.s; 30 times the viscosity of water at 100^{0}F). A run was subsequently made with a turbulent viscosity value of 0.13 Ns/m^2 (0.08737 lbm/ft.s; 190 times the viscosity of water at 100^{0}F) to evaluate its influence on the overall flow field. The bottom and side walls of the region were assumed to be frictionless and hydraulic losses due to obstructions in the drywell were ignored in this Mark III analysis.

Analysis Results

The velocity field results obtained for the Mark III drywell configuration using the parameters and COMMIX analysis model described above are described first. Then the preliminary results obtained for the Mark II are summarized and discussed relative to the detailed results presented for Mark III. Finally, the COMMIX results are discussed relative to the Mark I and Mark II 1-D analysis results as documented in Appendix A.

Mark III Analysis Results

Velocity field plots are given in Figures 3 through 10 at several planes in the region analyzed. Figures 3 through 5 present the results for three different elevations, or constant K planes (z-planes). Figures 6 through 8 present the results for three different radial positions, or constant I planes (r-planes). Figures 9 and 10 present the results for four different angular positions, or constant J planes (θ-planes). The velocity vectors given in these figures represent the 2-D velocity vectors, or projection of the total velocity vector (consisting of three components in space), onto the plane shown in the a particular figure. Thus, the plane velocities shown may not be completely representative of the three-dimensional velocity vector. The lengths of the vectors represent the magnitude of the flow velocity relative to the length scale also given in the figures.

Several observations about the horizontal velocity field are made by considering Figures 3 through 5. From these figures it is seen that velocities are generally highest (up to about 25 cm/s (0.8 ft/s)) near the bottom of the pool in the J=1 to 9 region whereas in the J=10 to 16 region the velocities are quite low (about 0 to 5 cm/s (0.16 ft/s)). Also, velocities are quite high (up to about 15 cm/s (0.5 ft/s)) along the outer wall at both the middle elevation and the surface elevation as indicated in Figures 4 and 5. Velocities are generally much lower in other parts of the flow field at these elevations. From Figure 3 it is observed that flow velocities over the weir wall (at I=7) are quite uniform except near the Splashdown Area where the flow is about one half that of the average.

The vertical velocity field is given in Figures 6 through 8 at three radial positions. The strong influence of the incoming water jet entering with a considerable downward momentum at the center of the Splashdown Area is evident from Figure 6. The velocities are highest (about 25 cm/s (0.8 ft/s)) near the top of the pool. The high flow velocities persist as the jet proceeds downward to the bottom of the pool. Entrainment of water from the surrounding pool and mixing into the jet takes place as it moves downwards as seen in Figure 6. This causes the jet to slow down prior to being diverted horizontally near the bottom. Velocities are generally much lower in other parts of the entire flow field shown in this view.

The strong downward momentum of the incoming jet is also believed to be responsible for the relatively high velocities along the outside wall as seen from Figure 7. It is believed that the flow, still having considerable momentum as it flows along the floor, will resist to some extent to be turned sideways at the curved outside wall and is partially diverted upwards (seen in Figure 7) as well as sideways (seen in Figure 5). On the other hand, flow velocities are relatively low near the inner wall as indicated in Figure 8 and also in Figure 4 where some flow reversal is observed caused by jet entrainment.

Cross-sectional views of the flow field are given in Figures 9 and 10. The flow field at constant J=2 plane (right part of Figure 9) located near the Splashdown Area is dominated by the downward entrainment flow near the center region and by the upflow along the outside wall. At the more centrally located J=8 plane, the flow is toroidal in nature with strong flows along the floor and outside wall regions. The flow fields in the constant J planes shown in Figure 10 show similar trends as for the J=8 plane. However, it is noted that the flow velocities projected into the J=12 plane are generally very low (less than about 1 cm/s (0.03 ft/s)) except along the outside wall. This low velocity pattern was observed also in other planes with high J values, i.e. planes far removed from the Splashdown Area and also along the floor as seen in Figure 5.

The COMMIX model was rerun with a turbulent viscosity of 0.13 Ns/m^2 instead of the 0.02 Ns/m^2 value used for the calculation results presented above. Figures 11 and 12 give two plots corresponding to those given earlier in Figures 5 and 6, respectively. Comparison of the flow fields in these sets of figures (as well as other sets) shows no discernible differences. Hence, it is concluded that the turbulent viscosity and associated forces are not dominant in determining the flow field. This can be seen also from the Reynolds number for the jet calculated to be about 2,500 based on the highest turbulent viscosity value (0.13 Ns/m^2) and the hydraulic diameter of the Splashdown Area perimeter. The Reynolds number is defined here as the ratio of momentum to turbulent viscosity forces. Hence, it is clear that momentum forces in the incoming jet is the dominant force in determining the flow field.

Mark II Analysis Summary

The COMMIX analysis was initially focused on the Mark II configuration and the parameters used by Bilanin in his 1-D analysis (Appendix A). Some observations obtained from the preliminary Mark II analysis are given in the following:

1. Flow velocities are relatively high along the drywell floor in the Mark II configuration as was also observed in the Mark III configuration results above. The flow velocities along the floor are highest in the region up to J=8 where maximum velocities are up to about 20 cm/s (0.66 ft/s). There is a gradual reduction in the velocities for larger J values although not as much of a decrease as was observed in Figure 5 for Mark III.

2. Flow velocities at the pool surface are generally low (about 1 cm/s (0.03 ft/s)) except for the regions adjacent to the inner and outer walls where maximum velocities are as high as 5 cm/s (0.16 ft/s). It appears that the incoming flow is distributed along the floor (as in Mark III) and flows generally upwards to the drains distributed along the surface.

3. The momentum of the incoming jet dominates the flow field in the Mark II configuration as in the Mark III configuration although to a lesser degree because of the lower flow velocity into the Splashdown Area.

One-dimensional Analysis Summary

Velocities obtained by the Bilanin 1-D analysis in Appendix A are compared to the preliminary COMMIX analysis results summarized above. Results from the 1-D analysis indicated maximum velocities of about 6 cm/s (0.2 ft/s) and an essentially linear decrease for larger circumferential locations. The COMMIX analysis generally showed higher velocities of up to 20 cm/s (0.66 ft/s) in the vicinity of the Splashdown Area and somewhat lower velocities further away. The lower velocity in the 1-D analysis would be expected since they are average values for the entire circumferential cross-sectional flow area whereas in the COMMIX analysis, circumferential flow was concentrated along the floor. The flow velocities at large circumferential locations away from the Splashdown Area are predicted to be quite low in both analyses.

Discussion

Several modeling assumptions were made in the COMMIX analysis that could have some influence on the calculated flow field. However, this is not believed to be the case here because of the dominance of the momentum contained in the incoming jet. It was demonstrated that jet momentum forces dominate the flow field relative to turbulent viscosity forces. Accordingly, it is believed that neglecting wall friction and form losses also will have a negligible influence on the flow field. The size of the Splashdown Area could affect the flow field to some extent. The distribution of this momentum at the pool surface on the other hand may have some influence on the flow field.

The essential question to be answered by the flow calculations is how much of the shreds of fine fibers or debris will settle onto the drywell floor and be retained. Using the data given in Figure B-7 of Reference 2, it is apparent that little retention of fine fibers can be expected because of the relatively large flow velocities existing along the floor. Moreover, most of the pool is well mixed which along with high velocities at the floor elevation would preclude significant settling of fine fibers. The only region where some settling will occur is in the region furthest away from the Splashdown Area where the flow velocities are quite low. These observations apply generally to both the Mark II and Mark III drywell configurations.

Conclusions

The drywell floor water flow analysis results from the one-dimensional model of the Mark I and Mark II drywell configurations as documented in Appendix A showed that water velocities are relatively low. Therefore, the elevation of the pool water level will be nearly constant everywhere if surface wave actions are neglected. Velocity dead spots will likely exist in the pool particularly at circumferential locations 180 degrees away from the Splashdown Area. Break flow splashing and turbulence caused by the splashing will break up and transport most of the fine fibrous debris into the suppression pool through downcomer drains and vents. However, on the opposite side of the drywell from the break, low water velocities would likely result in settling of some debris.

For typical break flow rates, the water level above the downcomer for the Mark II plant configurations is predicted to be approximately 0.6 cm (0.02 ft.). This limits the water flow which is available to wash debris into the suppression pool such that only the smaller debris can be transported assuming that washdown of larger debris by surface wave action can be neglected. Predicted water flow velocities along the floor and in the pool generally are typically less than 6 cm/s (0.2 ft/s) based on the one-dimensional analysis.

The COMMIX code analysis results showed that flow velocities in the drywell pools are as high as 25 cm/s (0.8 ft/s) along the floors and mixing of the pool fluid is reasonably good in both the Mark II and Mark III configurations. This provide little opportunities for significant holdup and settling of fine fibrous debris on the drywell floor following a major pipe break since 50 percent of shreds of typical insulation fibers have settling velocities less than 0.1 cm/s (0.003 ft/s) (Reference 2).

In summary, results of the BWR drywell floor modeling analysis have not found velocities to be low enough to facilitate significant settling of typical fibrous debris in any of the containment designs studied. However, pool velocities may be low enough to allow heavier debris such as paint chip, iron oxide particles, cement dust and reflective metal insulation debris to settle onto the drywell floor.

References

1. P. L. Garner, R. N. Blomquist, and E. M. Gelbard,"COMMIX-1AR/P: A Three-Dimensional Transient Single-Phase Computer Program for Thermal Hydraulic Analysis of Single and Multicomponent Systems; Volume II: User's Manual", ANL-92/33, Vol. II, Argonne National Laboratory, Argonne, IL (September 1992).

2. G. Zigler, J. Brideau, D.V. Rao, C. Shaffer, F. Souto, and W. Thomas, "Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA Generated Debris", NUREG/CR-6224, Science and Engineering Associates, Inc., October 1995.

(2D) II Print II k8w3.plt II COMMIX RESULTS

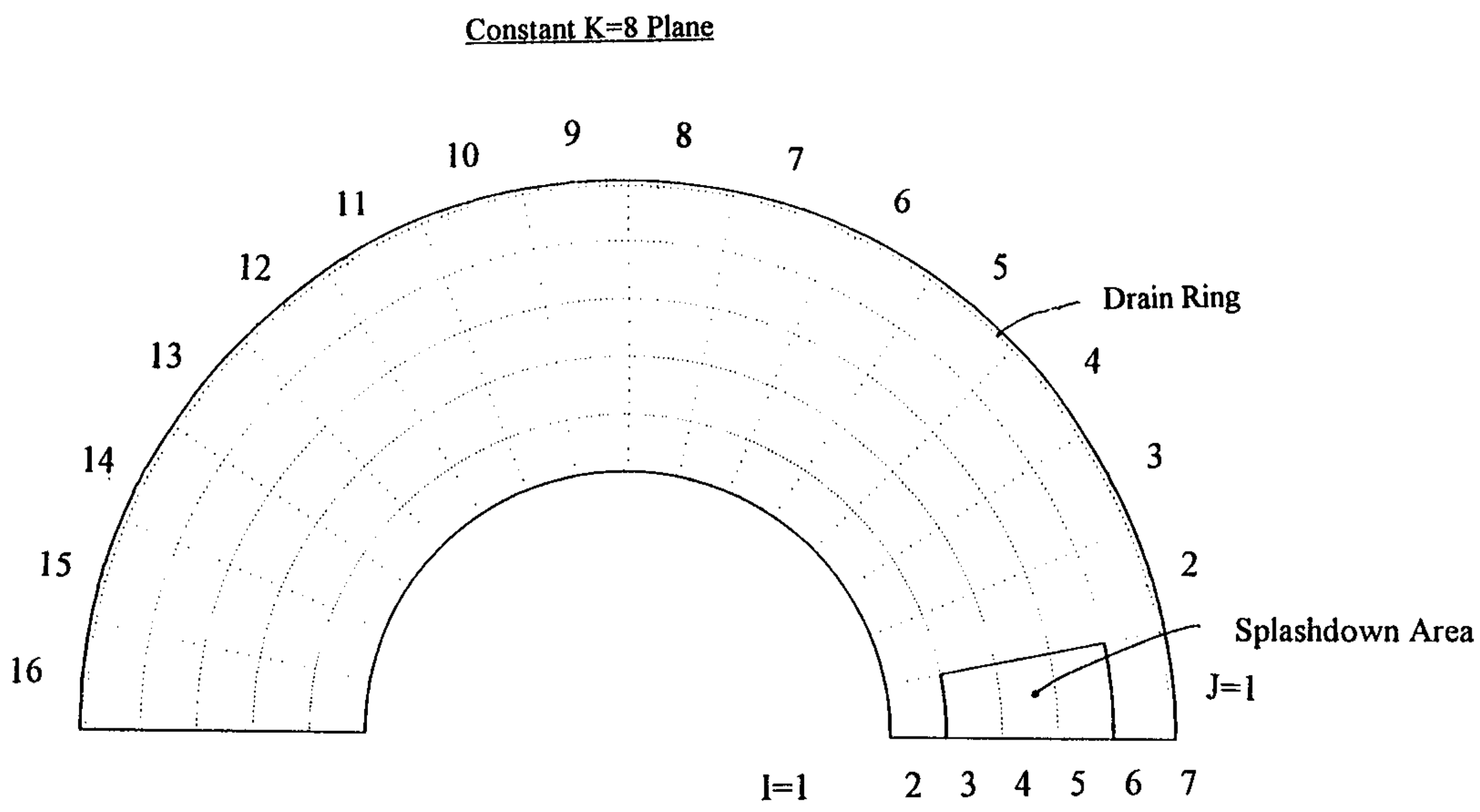

Figure 1 Plan View of Analysis Region at Water Surface (K=8)

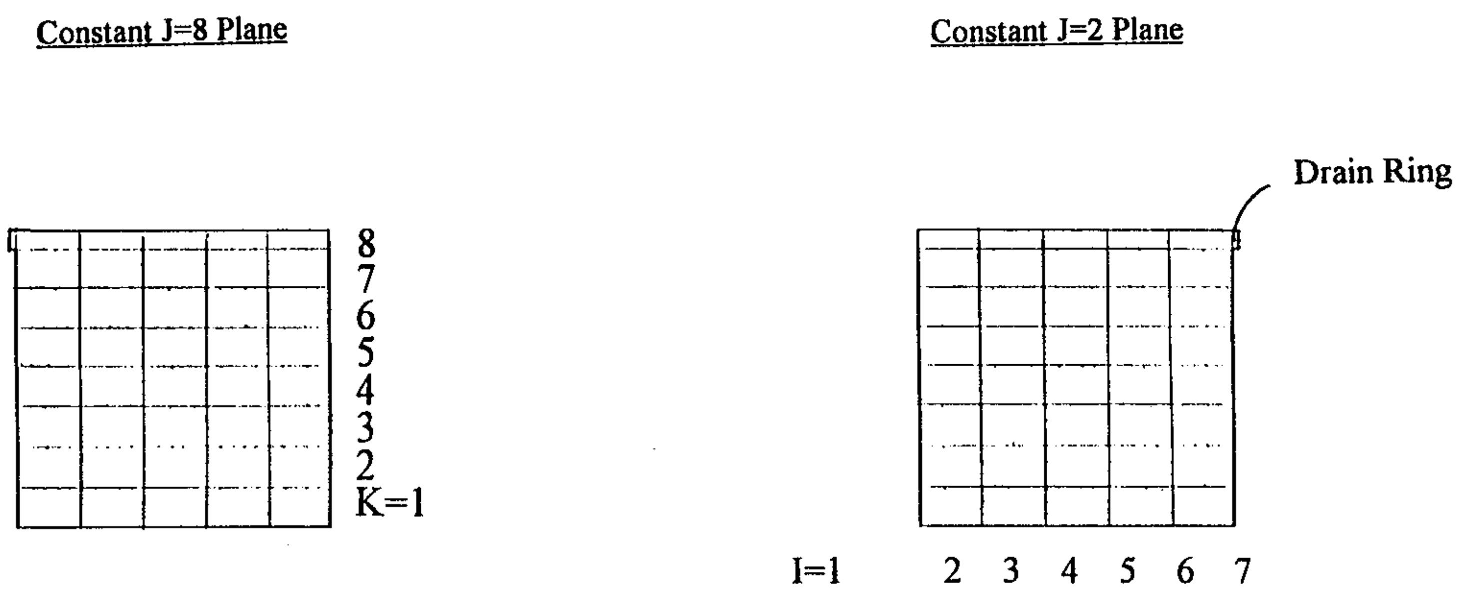

Figure 2 Elevation Views of Analysis Region for Two Angular Locations (J=2,8)

Constant K=8 Plane (at Water Surface)

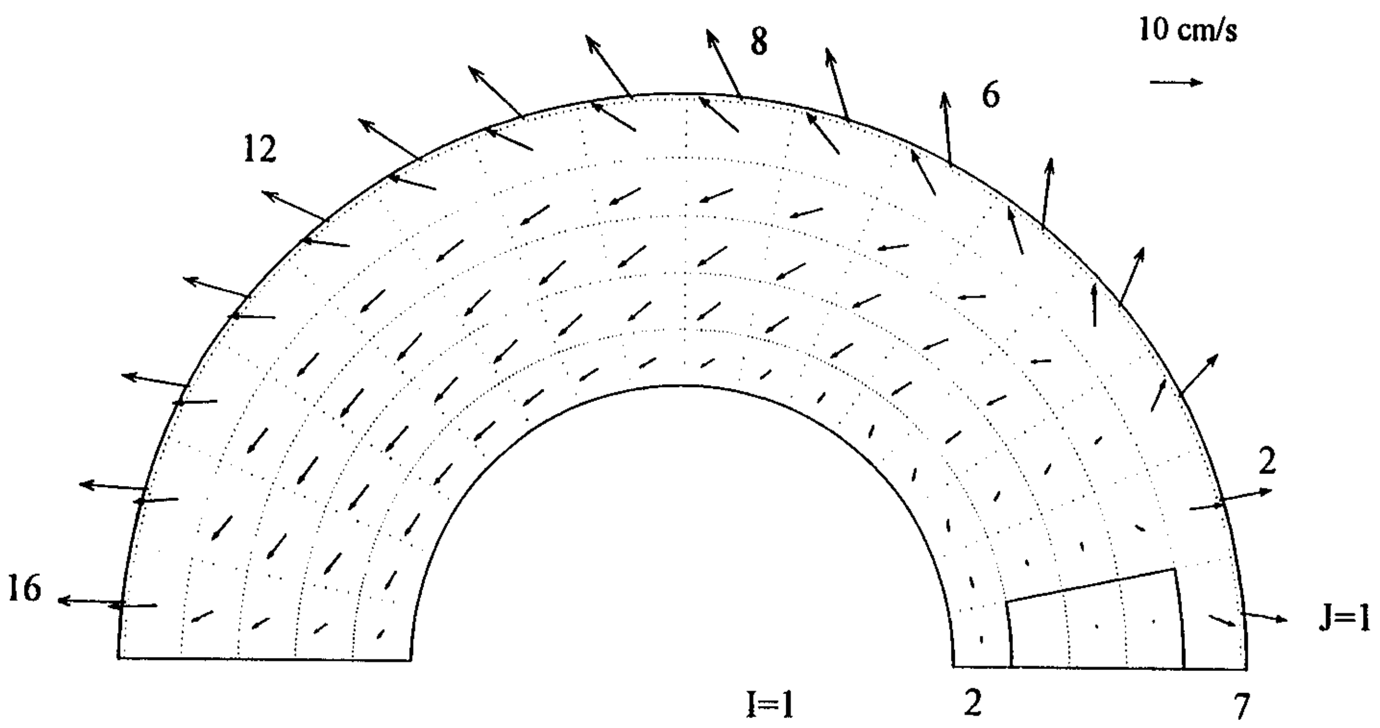

Figure 3 Calculated Velocity Field at Water Surface (K=8)

Constant K=4 Plane (Pool Middle Elevation)

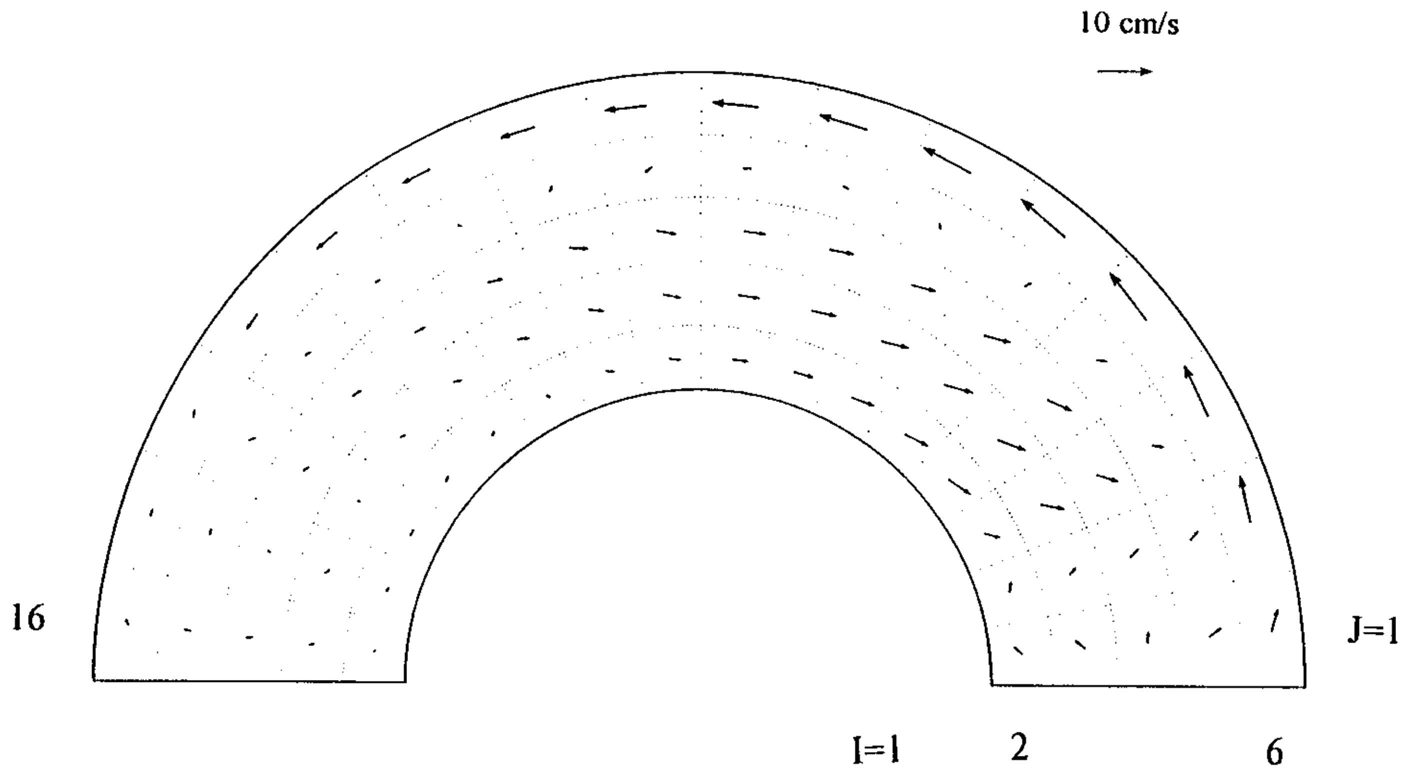

Figure 4 Calculated Velocity Field at Pool Middle Elevation (K=4)

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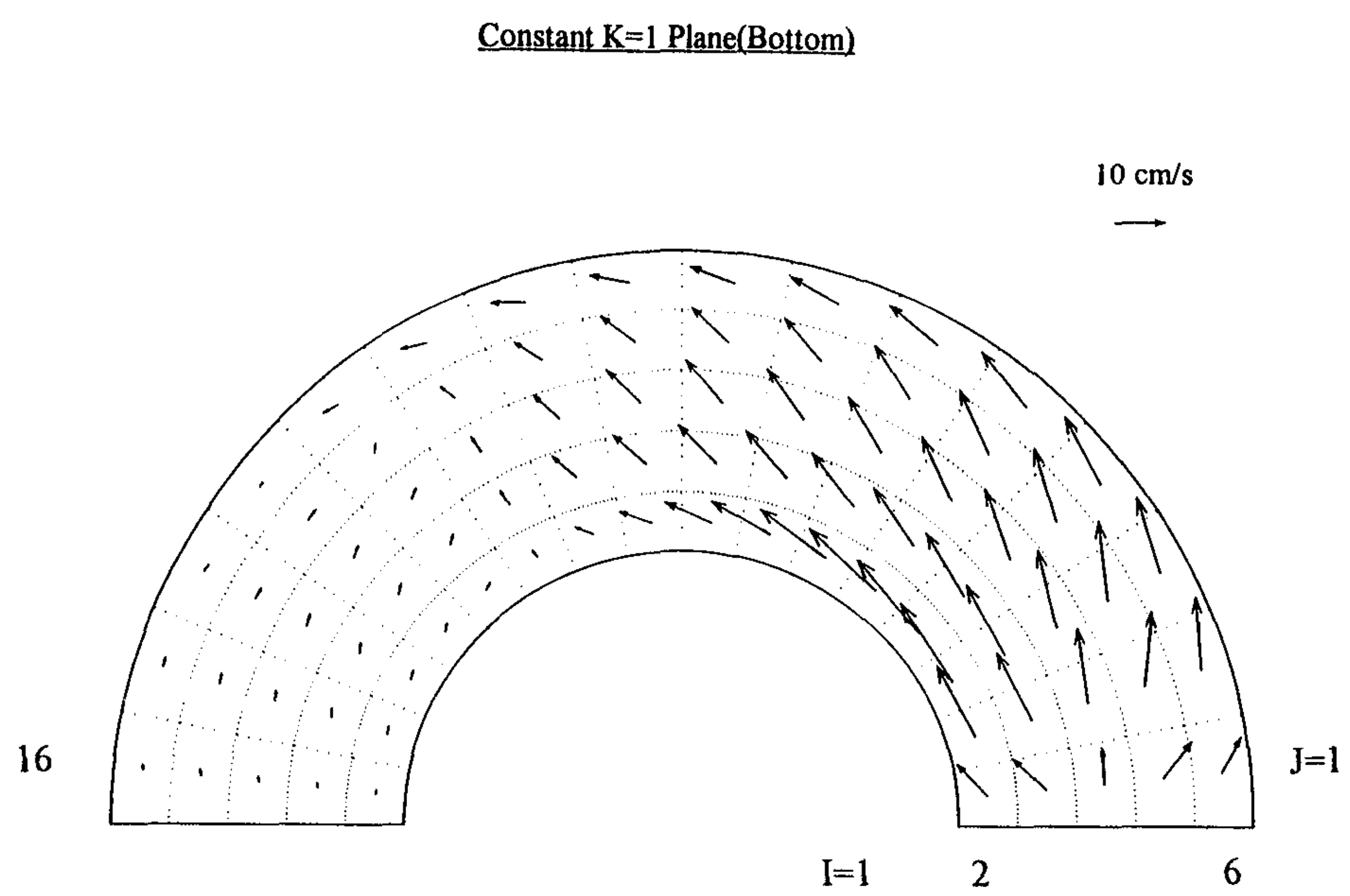


Figure 5 Calculated Velocity Field at Bottom (K=1)

Constant I=4 Plane (Center)

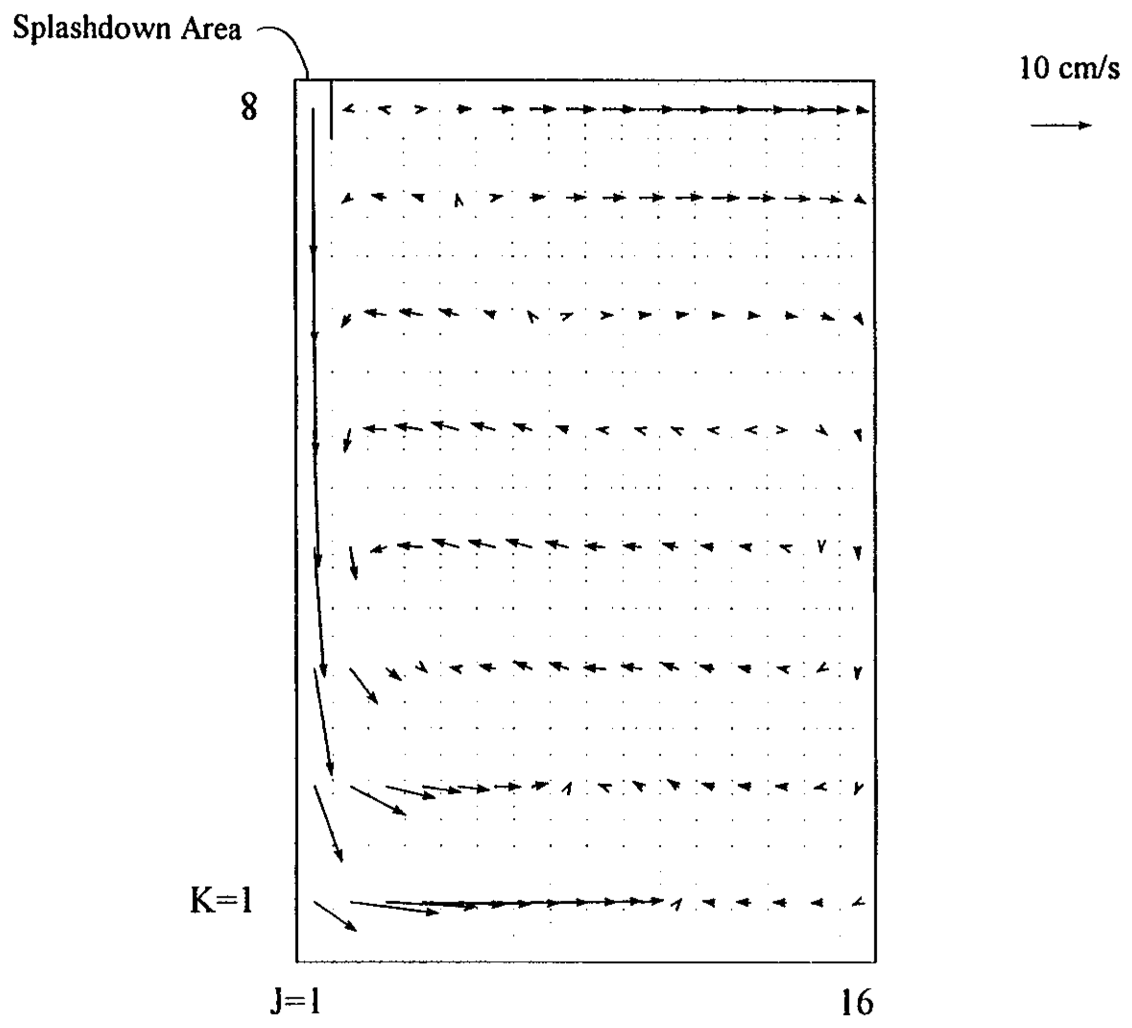


Figure 6 Calculated Velocity Field at Pool Center (I=4)

Constant I=6 Plane (Outside)

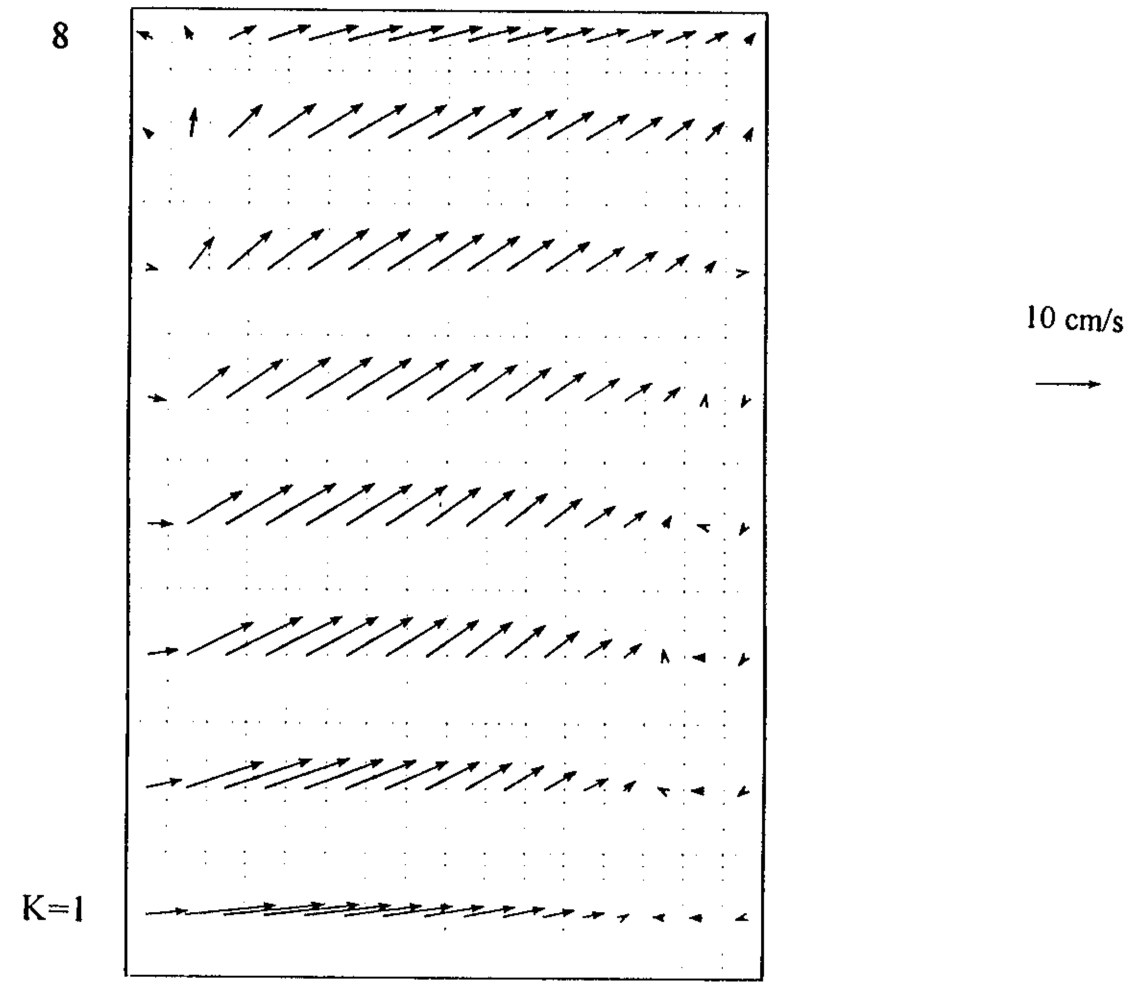


Figure 7 Calculated Velocity Field at Pool Outside Wall (I=6)

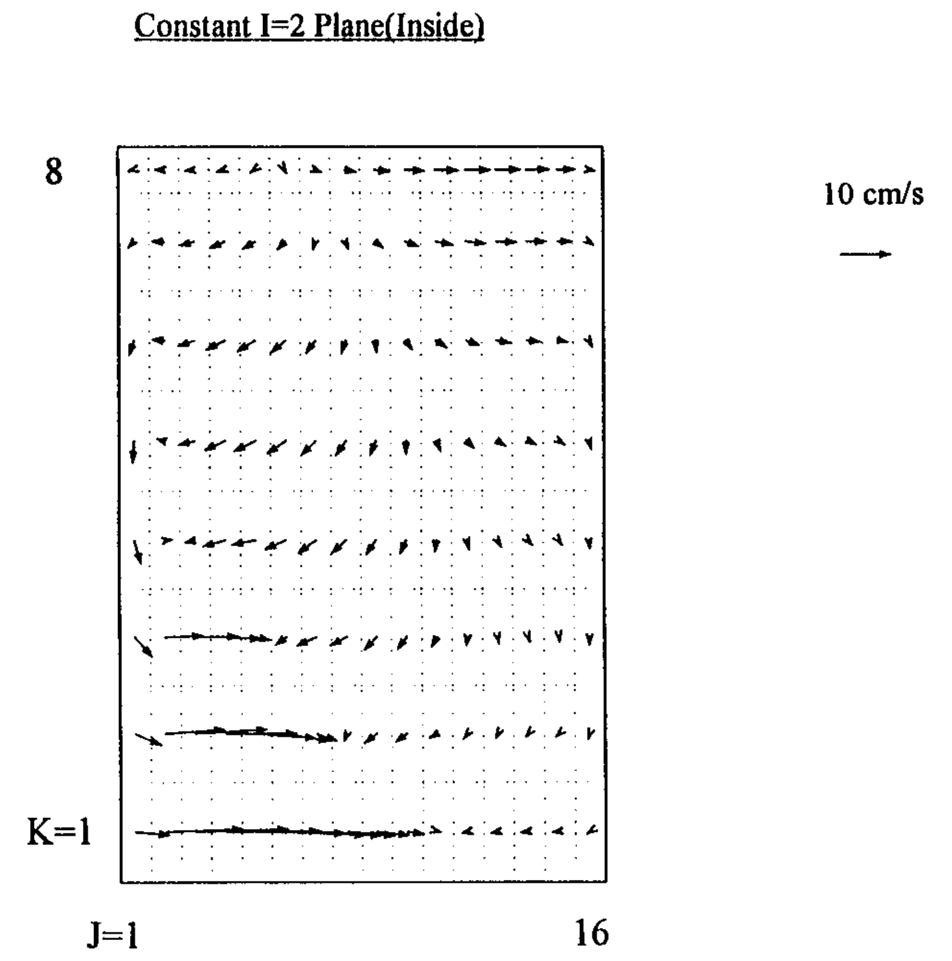


Figure 8 Calculated Velocity Field at Pool Inside Wall (I=2)

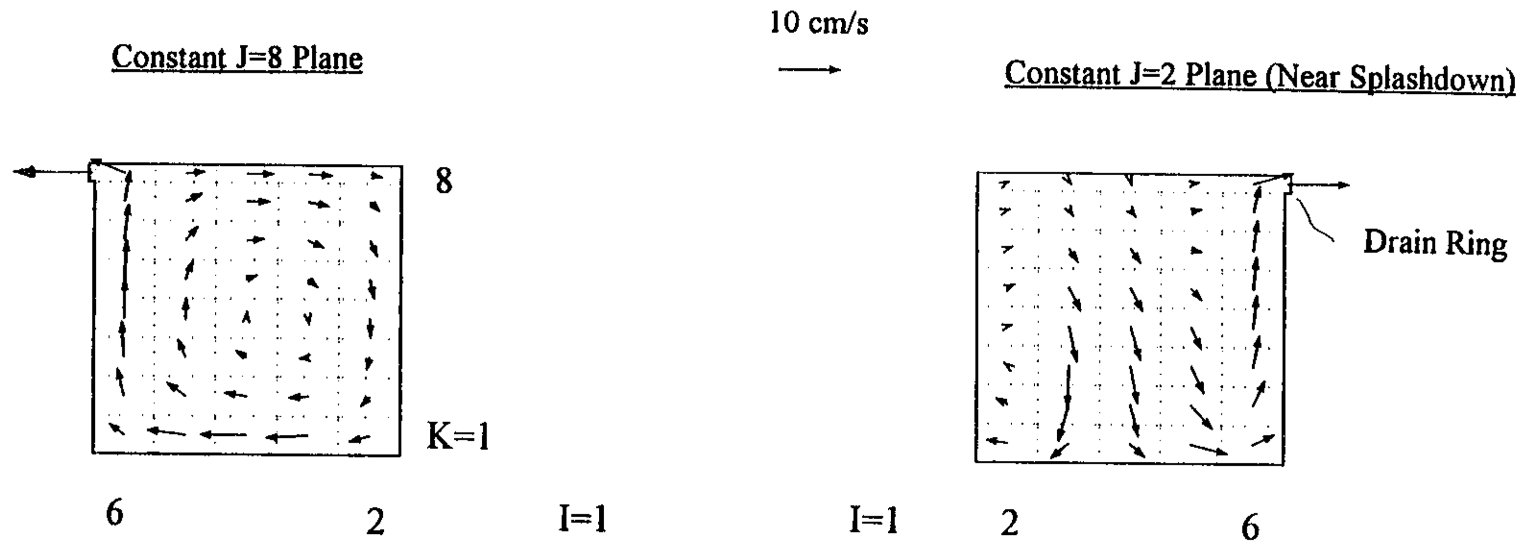


Figure 9 Calculated Velocity Field in Two Planes (J=8,2)

(2D) || Print || jw12.6.plt || COMMIX RESULTS

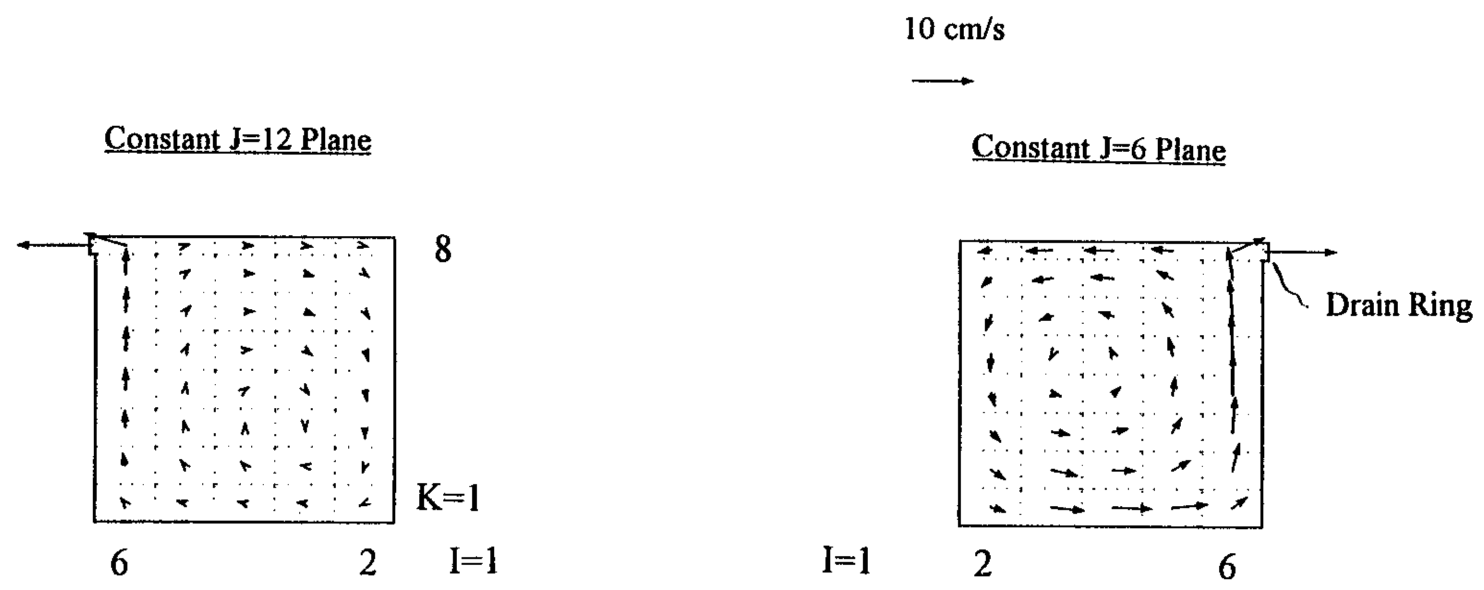


Figure 10 Calculated Velocity Field in Two Planes(J=12,6)

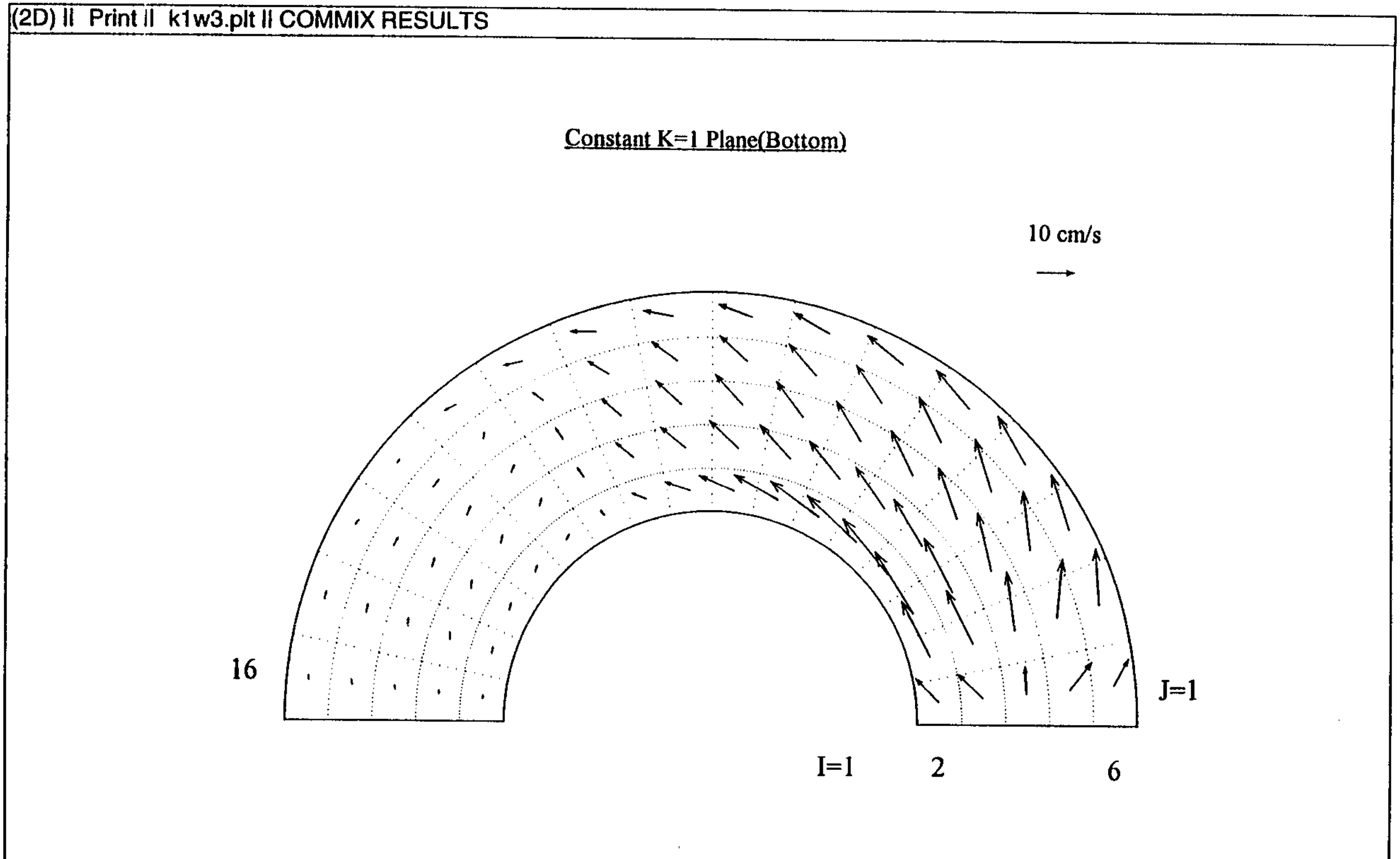


Figure 11 Calculated Velocity Field at Bottom with High μ_t (K=1)

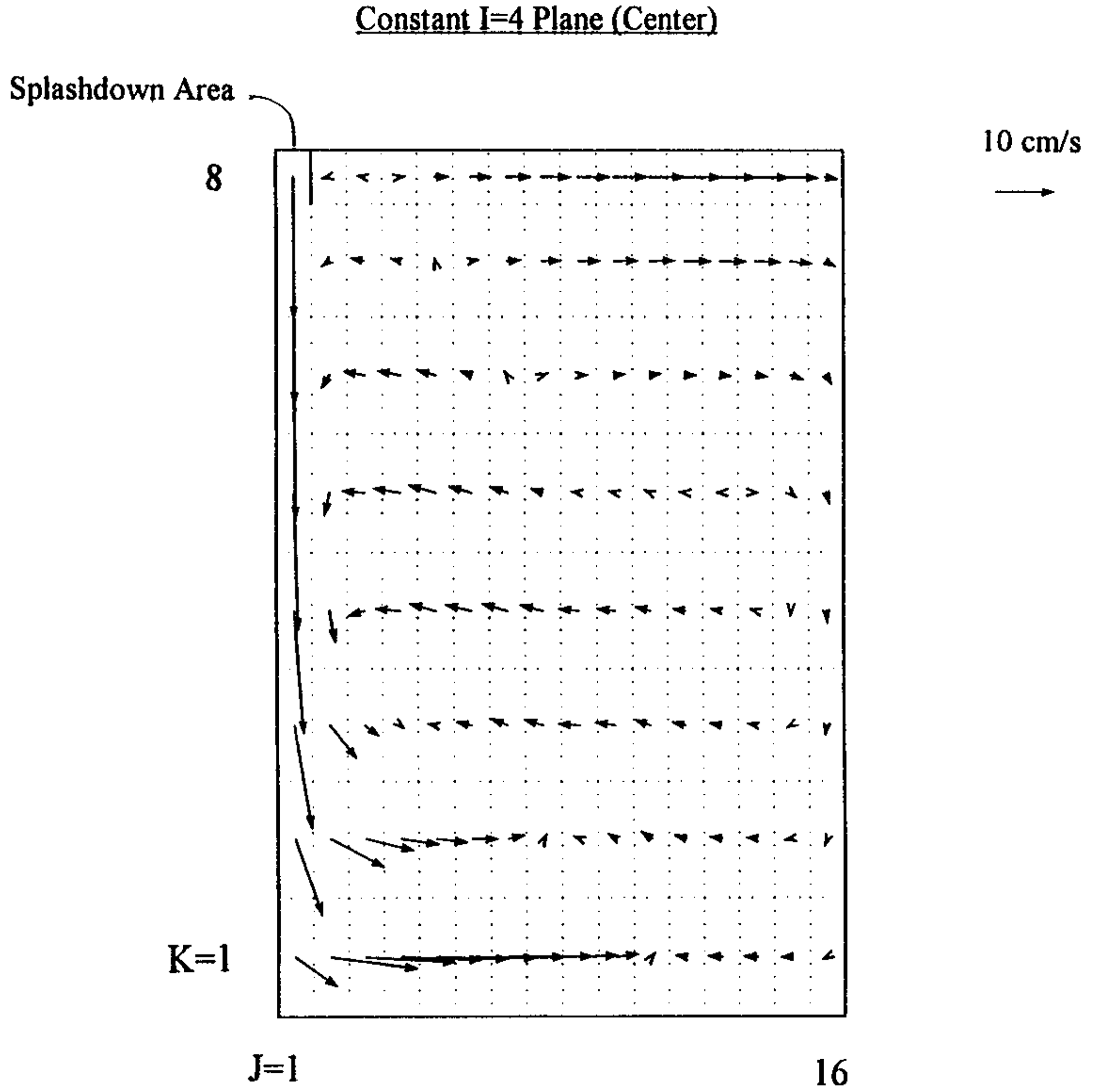


Figure 12 Calculated Velocity Field at Pool Center with High μ_t(I=4)

Appendix A

Drywell Floor Flow Modeling for Pipe Breaks

An analytic model is formulated to estimate the water flow velocities which should exist on the floor of a Mark II drywell during the time the ECCS system is supplying the RPV with make up flow. The model developed for Mark II containments is directly extendable to Mark I containments.

Nomenclature

ρ - water density
g - gravity
P - ave. hydrostatic pressure
δ - water height
h - downcomer height above floor
U - water velocity
D - downcomer diameter
r_2 - inner radius of drywell
r_2 - outer radius of drywell
Q - break flow
$\Delta\theta$ - 1/2 the angle over which break flow is assumed to impinge the drywell floor
C_D - downcomer drag coefficient
C_d - weir discharge coefficient
x - azimuthal distance around drywell
N - number of downcomers

Formulation (Refer to Figure 1)

The average pressure in the water column of height δ is

$$P = \frac{\rho g \delta}{2}$$

Assume that the break flow enters the pool floor uniformly over the area defined by $2\Delta\theta$. The vertical water velocity in this sector is $\frac{Q}{(r_2^2 - r_1^2)\Delta\theta}$
and is zero outside this sector. When the water elevation δ is above h, the downcomers behave as weirs. An estimate of the weir flow / downcomer of diameter D is ([1])

$$Q = C_d \frac{2}{3}\sqrt{2g}\pi D(\delta - h)^{3/2}$$

If there are N downcomers / reactor the total flow is NQ and the water velocity leaving the drywell floor through the downcomers but averaged over the floor area $\pi(r_2^2 - r_1^2)$ is

$$w_0 = C_d \frac{2}{3}\sqrt{2g}\frac{D}{r_2^2 - r_1^2}(\delta - h)^{3/2} N$$

[1] Daugherty and Franzini, Fluid Mechanics with Engineering Applications, McGraw Hill, 1965.

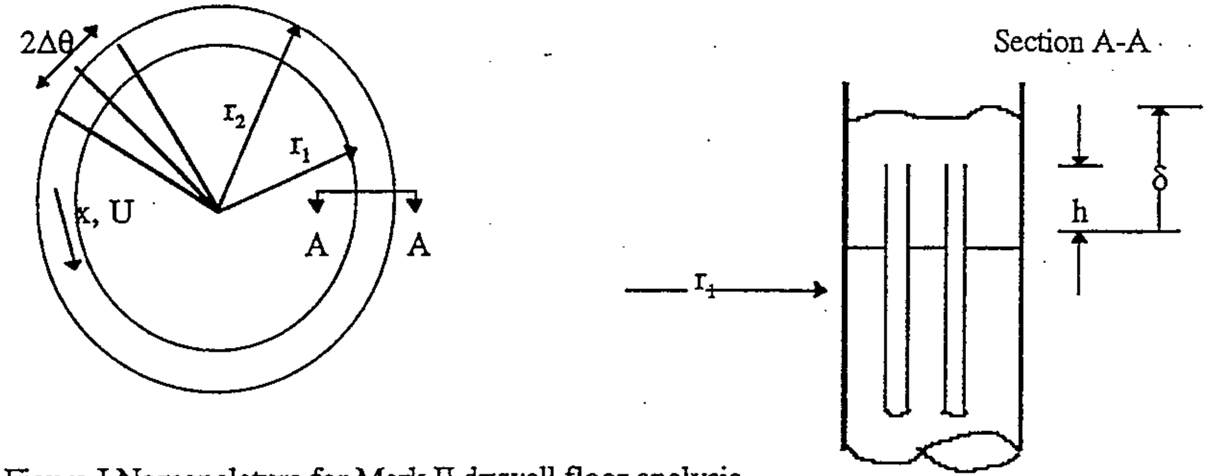


Figure I Nomenclature for Mark II drywell floor analysis

Conservation of mass in a differential volume $\Delta x \delta (r_2 - r_1)$ yields

$$\frac{dU\delta}{dx} = \frac{Q}{(r_2^2 - r_1^2)\Delta\theta} - C_d \frac{2}{3}\sqrt{2g}\frac{D(\delta - h)^{3/2}}{r_2^2 - r_1^2}N$$

where the first term on the RHS is zero when $x > \frac{(r_1 + r_2)}{2}\Delta\theta$

In this control volume conservation of momentum yields

$$\frac{d}{dx}(P + \rho U^2)\delta = -f_x$$

where f_x is the body force which results from the drag force of the downcomers on the fluid

$$f_x = \frac{C_D}{2}\rho\frac{U|U|DN}{\pi(r_2^2 - r_1^2)}$$

The formulation is complete with the specification of boundary conditions.

At x=0 and $\frac{r_1 + r_2}{2}\pi$, U=0. Therefore this becomes numerically a shooting problem to guess $\delta(0)$ and integrate to find U(x) and $\delta(x)$ where $U(\pi\frac{(r_1 + r_2)}{2}) = 0$.

A numerical example is programmed with the following inputs

h=1.5 ft
C_D=1.0
C_d=1.0
N=90
r_2=44 ft
r_1=14 ft
D=2 ft
Q=22ft^3/sec

$\Delta\theta = \pi/4$
g=32.2ft/sec^2

Results

On figure 2 is shown the nondimensionalized floor water velocity and water height δ versus distance around the pool for the assumption that the break flow is uniformily over $\pm\pi/4$, $\pm\pi/8$, and $\pm\pi/16$ radians. As can be seen the water velocity varies approximately linearly about the pool with the maximum velocity occuring at the outer edge of the region in which the break flow is assumed to fall to the dry well floor.

On figure 3 is shown the result of decreasing the break flow rate by a factor of 2 and 4. Note that the floor water velocities are directly proportional to break flow rate.

Mark I

Referring to Figure 4 shows a schematic of a Mark I drywell floor with eight main vents of diameter D. The main vents slant downward from the horizontal by α. Therefore the main vent viewed from inside the drywell is elliptical in cross section. Assume that the weir flow into this vent can be approximated by a v-notch with width D and height $\frac{D}{2}\cos\alpha$. The flow rate from this weir will be (1)

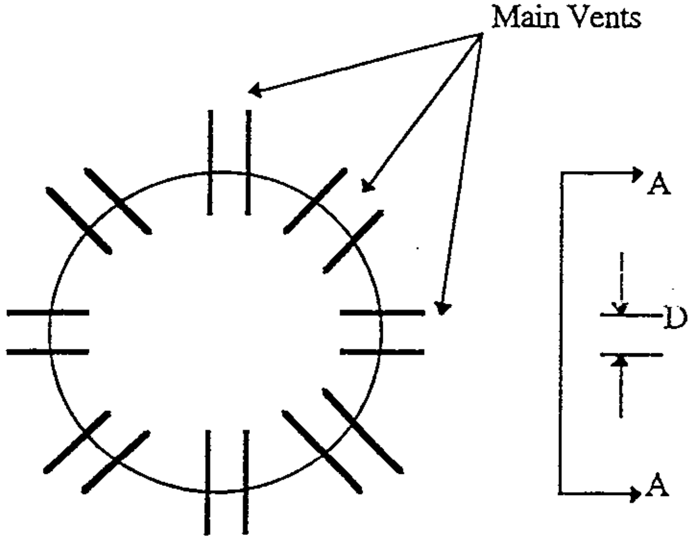
$$Q = C_d \frac{8}{15} \frac{\sqrt{2g}}{\cos\alpha} (\delta - h)^{5/2}$$

Since there are N main vents the total outflow will be QN and averaged over the pool floor area $\pi(r_2^2 - r_1^2)$ the outflow is $QN/\pi(r_2^2 - r_1^2)$. For Mark I analysis, the conservation of mass is rewritten

$$\frac{d}{dx} U\delta = \frac{Q}{(r_2^2 - r_1^2)\Delta\theta} - \frac{Cd}{\pi} \frac{8}{15} \frac{\sqrt{2g}}{\cos\alpha} \frac{(\delta - h)^{5/2} N}{(r_2^2 - r_1^2)}$$

where the first term on the RHS is zero when $x > \frac{(r_1 + r_2)}{2}\Delta\theta$

Figure 4



Section A-A

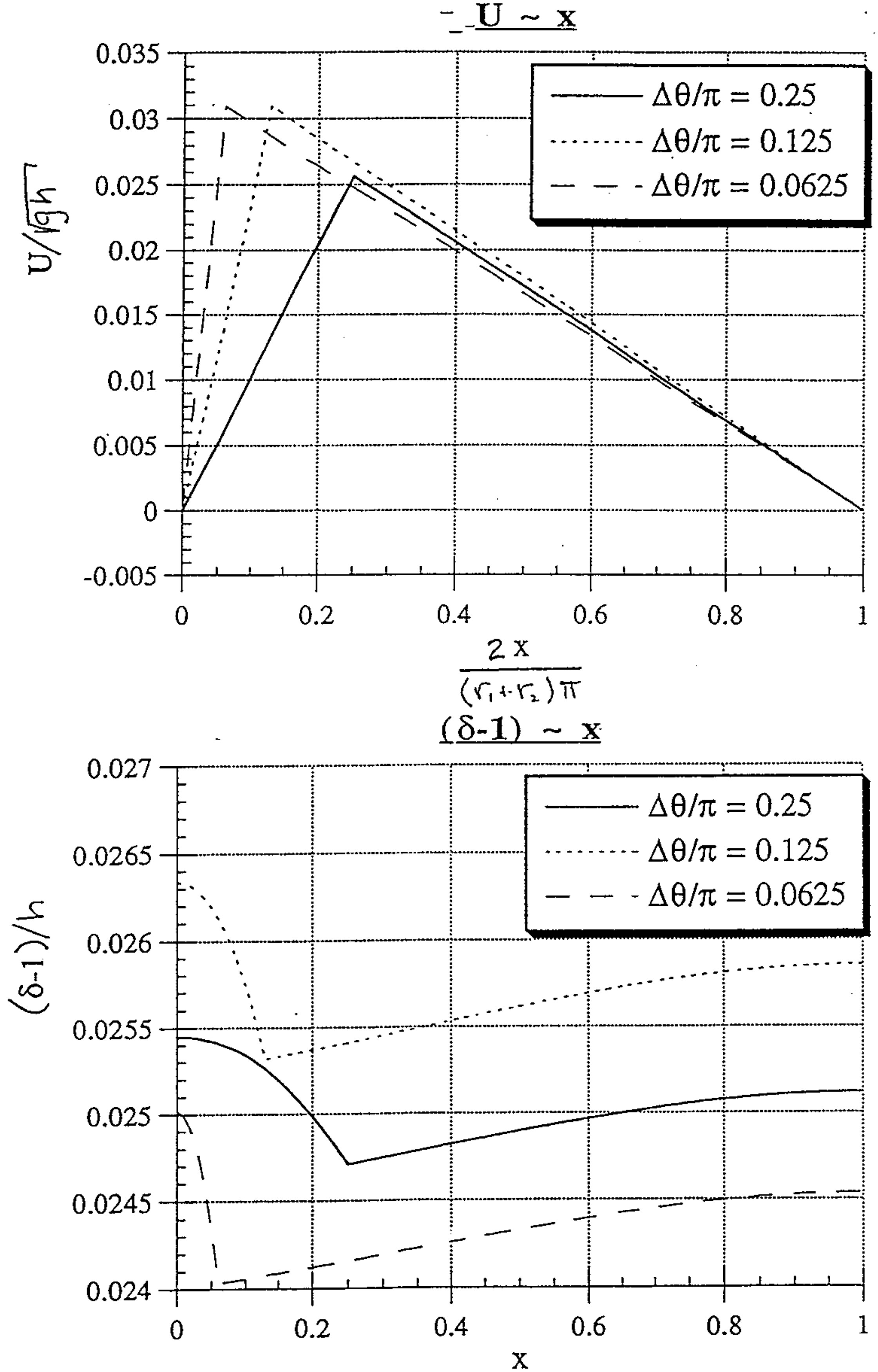


Figure 2. Drywell floor water velocity and water height as a function of break flow rate.

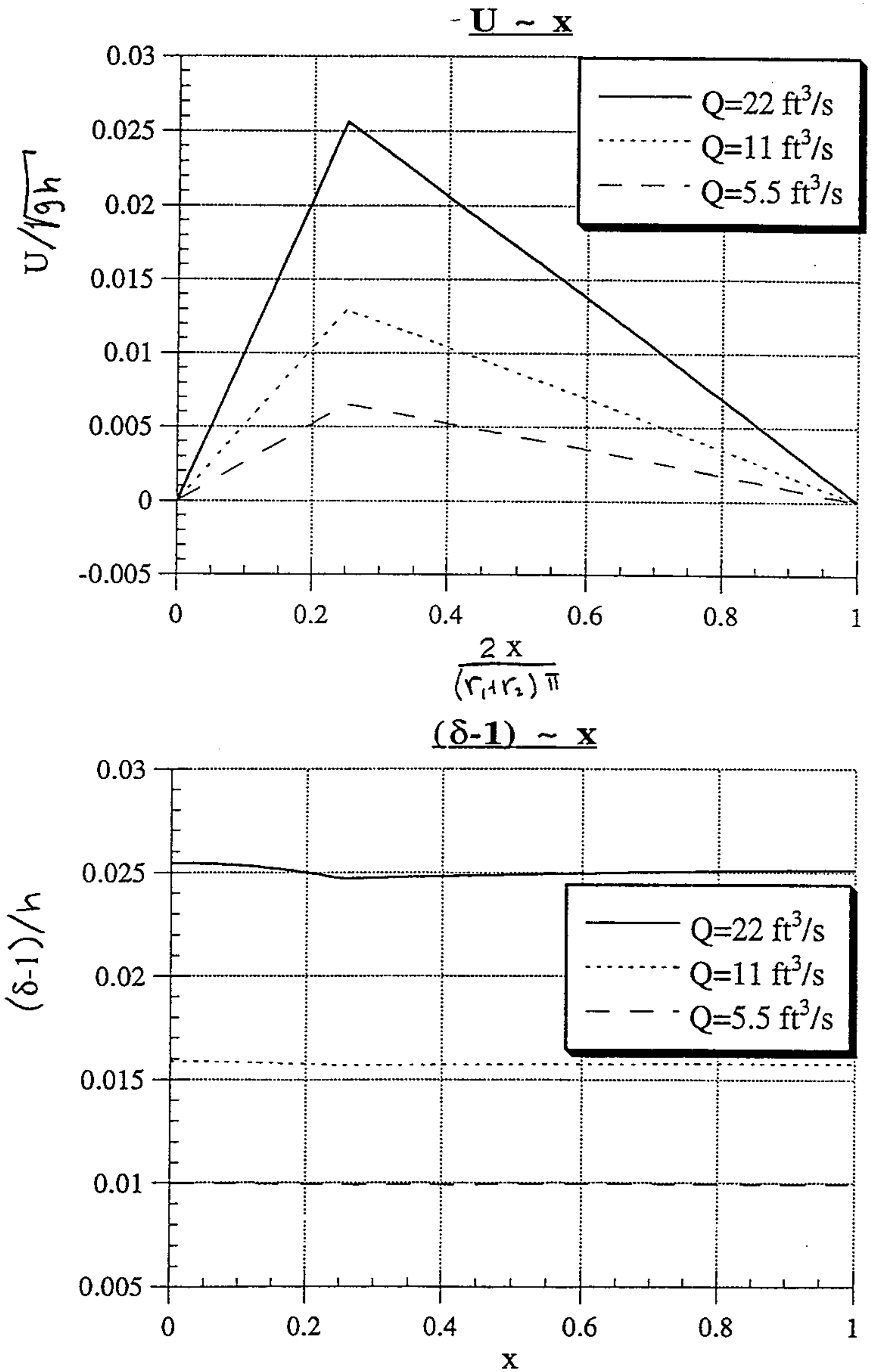


Figure 3. Drywell floor water velocity and water height as a function of break flow rate.

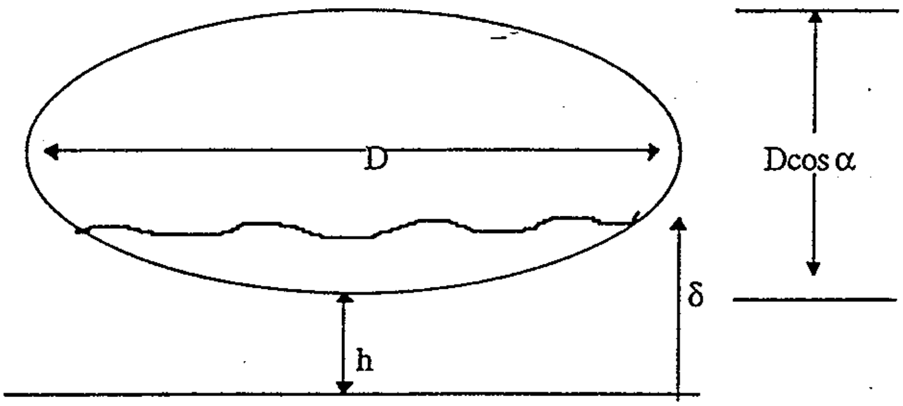


Figure 4 Nomenclature for Mark I drywell floor analysis

Predicting Transport

From the analysis described above, it is possible to estimate the transport of fibrous debris from the drywell floor to the supression pool. From transport tests at C.D.I.is known transport of fiber on floor versus gpm of flow per downcomer. From Alden tests undertaken for PPL transport over the weir as a function of U is known.

Approach

Fix Q and $\Delta\theta$
Predict U and gpm/downcomer
Specify mass of fiber on drywell floor
Determine fraction which transports by using transport data from Alden and C.D.I.

2

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OG94-661-161
September 13, 1994

Aleck W. Serkiz
US Nuclear Regulatory Commission
Two White Flint North - 10C9
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SUBJECT: **BWR OWNERS' GROUP ECCS SUCTION STRAINER COMMITTEE SUPPRESSION POOL SLUDGE PARTICLE SIZE DISTRIBUTION**

Attachment: Suppression pool sludge particle size distribution data from five BWR plants

Per your request we have evaluated suppression pool sludge particle size distribution data from samples obtained at five (5) BWRs. It is believed that these sludges consist of nearly 100% iron oxides. Plant specific data has been averaged to provide a suggested distribution for the NRC sponsored testing to be conducted at the Alden Research Labs (ARL). Please note that this data is considered preliminary in that additional samples will be available in the next few months. Due to variations in sampling techniques, these samples are not necessarily representative of the actual suppression pool sludges present.

For testing at ARL we would suggest the following particle size distribution based on data from the available suppression pool samples:

Particle Size (Microns)	*% of Suppression Pool Sludge*
2.5 (0-5)	81%
7.5 (5-10)	14%
42.5 (10-75)	5%

Recognizing the above limitations with respect to procurement of an appropriate this iron oxide material, we would suggest specifying an approximate linear particle size distribution such that 81 cumulative weight percent is less than 5 microns, 95 cumulative weight percent less than 10 microns, and 100 cumulative weight percent less than 75 microns.

If you have any questions rgarding this transmittal, please call the undersigned or the Committee Chairman, R. (Rocky) Sgarro at (610) 774-7914.

Very truly yours,

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BWR Owners' Group Projects
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SJ Stark, GE
ECCS Suction Strainer Subcommittee

BWR OWNERS' GROUP

ECCS SUCTION STRAINER COMMITTEE

Suppression Pool Sludge Particle Size Distribution

		PERCENT WT. FRACTION					
Plant	***Cont. Type***	***0-1 Microns***	***0-5 Microns***	***0-10 Microns***	***10-75 Microns***	***> 75 Microns***	***Comments***
BWROG Test Material	-				*	27	Iron oxide has major effect on strainer head loss
VNC Calibration Standard	-	30	96	> 99	< 1	0	Iron oxide has <u>no</u> effect on strainer head loss
A.	III	65 75		100 100			Predominately iron and other metal oxides, some fiber impurities
B.	II		94	99.5	0.5	0	Mean particle size is 2.3 microns, essentially 100% iron oxide

* 41% less than 45 Microns; 32%, 45 to 75 Microns

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Plant	**Cont. Type**	**0-1 Microns**	**0-5 Microns**	**0-10 Microns**	**10-75 Microns**	**> 75 Microns**	**Comments**
C.	III	8	88	97	3	0	Sample contains organics and possible biological component; organic is oily in nature and appeared as droplets floating on surface of the sample; particulate spike at approximately 4 microns could be due to bacterial population
D.	I	14	85	97	3	0	
E.	III	18	75	86	14	0	Suppression Pool Vent
		14	65	82	18	0	Suppression Pool Floor
		15	65	80	20	0	Suppression Pool RHR "B"

3

Attachment 2

Suppression Pool Sludge Particle Size Distribution Data
Average Distribution Calculation

Plant	% of Suppression Pool Sludge Particle Size 2.5 um (0-5)	Particle Size 7.5 um (5-10)	Particle Size 42.5 um (10-75)	Particle Size Greater than 75 um
A	70	30	0	0
B	94	5.5	0.5	0
C	88	9	3	0
D	85	12	3	0
E	68	14	18	0
F	65	16	19	0
G	83	14	3	0
H	90	8	2	0
I	53	20	27	0
J	90	8	2	0
K	92	6	2	0
L	95	3	2	0
M	96	2	2	0
N	94	4	2	0
Average of 14 BWRs	83	11	6	0
Average provided to NRC via OG94-661-161 on 9/13/94	81	14	5	0

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BWR Owners' Group
Suppression Pool Sludge Generation Rate Data

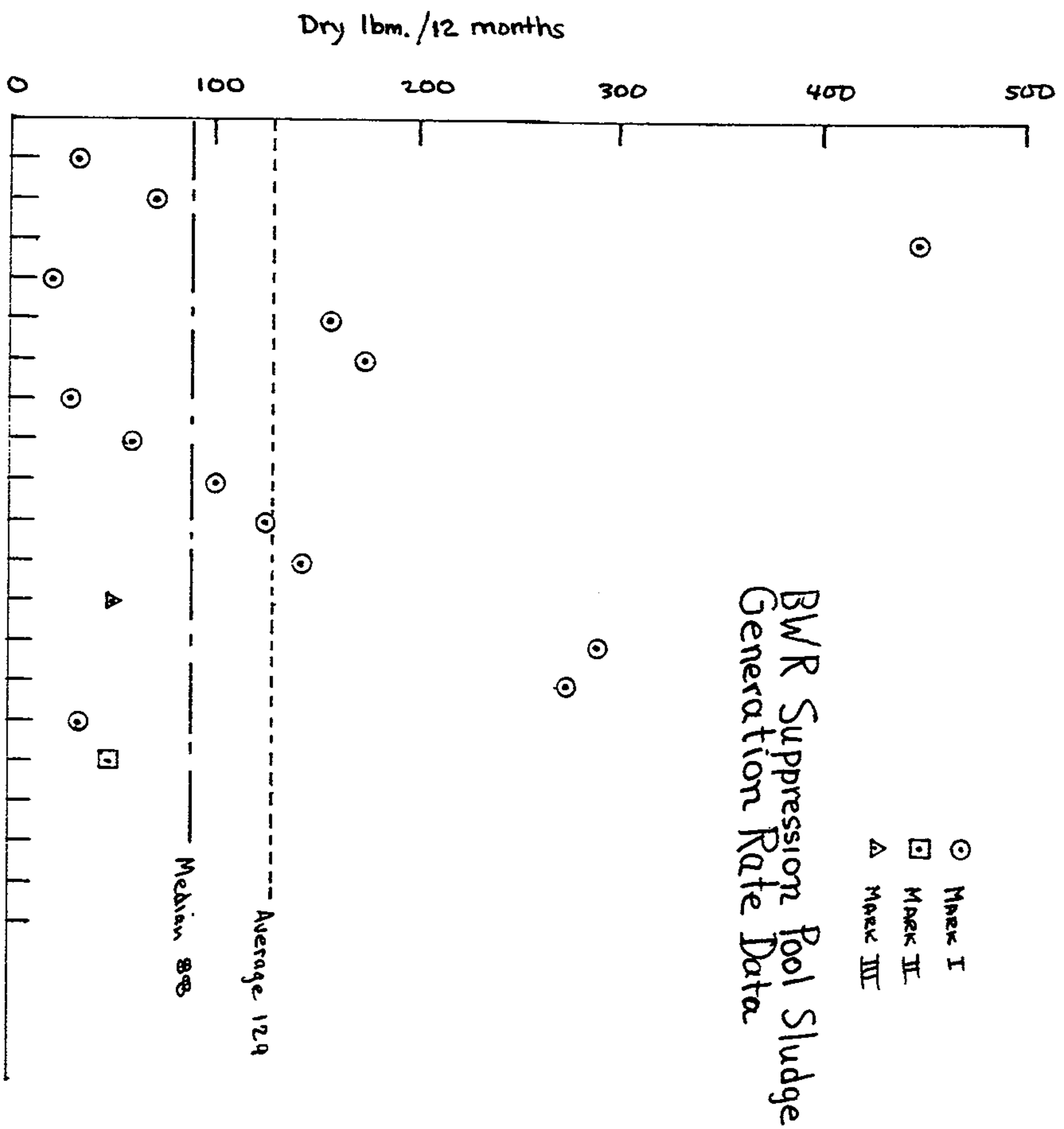
Plant	Cont. Type	Years	Wet lbm.	Dry lbm.(1)	Dry lbm. / year	Comments
Dresden 2	I	88-93	367	141	33	4.25 years, based on 10.2 lbm per filter
Dresden 3	I	88-94	1140	439	73	
Duane Arnold	I	85-88	3500	1350	449	"very conservative bounding calculation"
		90-95	240	92	21	4.5 years
Fitzpatrick	I	92-94	-	300	157	(2); 23 months
Millstone	I	92-94	907	349	175	2 years
Monticello	I	91-93	-	48	32	sludge pumped to resin liner, concentration estimated at 95% water 5% solids; 1.5 years
Oyster Creek	I	84-88	-	240	60	Volume estimate and density analysis
		88-92	-	408	102	
		93-94	-	200	126	19 months
Peach Bottom 2	I	91-94	1300	500	143	3.5 years
Perry	III	93-94	150	58	53	13 months (4)
Quad Cities 2	I	92-93	874	336	288	14 months (3)
		93-94	1188	457	274	20 months (3)
Vermont Yankee	I	83-95	1100	424	35	
WNP-2	II	86-94	1125	433	48	Includes initial construction debris, 9 years

(1) GE has analyzed the water content of three gravity filtered sludge samples from BWR suppression pools. The percent solid were 17.2%, 24.4%, and 38.5% (38.5% used to estimate the dry iron oxide generation rates).
(2) Sludge allowed to settle in separate radwaste tank, settled density was 65.5 lbm per cubic foot (specific gravity = 1.05)
(3) Subtracted dry filter weight from total wet weight of sludge and filter
(4) Miscellaneous outage debris may be included in wet sludge estimate

GE Nuclear Energy
Engineering Calculation Sheet

Sheet of

Subject	Originator T. Green	Date 6-8-95
Number	Verifier	Date



GE-NE-T23-00700-15-21
March 1996 (Rev. 1)

EVALUATION OF THE EFFECTS OF DEBRIS ON ECCS PERFORMANCE

WORK PERFORMED FOR THE
BWR OWNERS' GROUP
ECCS SUCTION STRAINER COMMITTEE

by

P. F. Kachel
Engineering & Licensing Consulting Services

Approved:

M. R. Wuestefeld, Manager
Engineering & Licensing Consulting Services

IMPORTANT NOTICE REGARDING CONTENT OF THIS REPORT

Please Read Carefully

1. Introduction

The Residual Heat Removal (RHR) and Core Spray (CS) systems have suction strainers installed in the suppression pool. These strainers are referred to here as Emergency Core Cooling System (ECCS) suction strainers. The purpose of the strainers is to filter out debris which could impact system operation. The ECCS suction strainers are installed to preclude the plugging of critical system orifices, and plant operating experience has shown the suction strainers to be effective in precluding plugging of downstream components. ECCS performance will be degraded if these strainers become clogged. Therefore, the BWROG has undertaken a program to redesign these strainers, and has completed testing of strainers with Electric Power Research Institute (EPRI) in Charlotte, NC. This report addresses the effect of debris which is assumed to pass through either the self-cleaning or the passive strainer.

During a postulated double ended guillotine pipe break Loss-of-Coolant Accident (LOCA) at any GE Boiling Water Reactor (BWR), a small fraction of available rust chips, paint chips, sand and fibrous insulation debris materials which are present in the suppression pool are postulated to be transported through the ECCS suction strainers and on to the Emergency Core Cooling System (ECCS) pumps. This evaluation assesses the impact of this debris on the downstream safety-related components such as pump cooling water hydrocyclone orifices, pump seals, RHR heat exchangers, core spray nozzles, containment spray nozzles, and lower tie plate fuel filters. This material can be operational debris which has been previously transported to the suppression pool, is transported to the pool during a LOCA, or is debris generated by the LOCA that is transported to the suppression pool. The quantity of operational debris can be minimized by good housekeeping procedures and by comprehensive foreign material exclusion (FME) programs which are being implemented effectively by BWR utilities.

The ECCS suction strainers are employed to preclude relatively large particles of foreign debris from entering emergency core cooling systems. Under turbulent LOCA conditions, suppression pools will be well mixed and suction strainers will collect or pass debris which is present in the pool. A fraction of the debris is expected to pass through these strainers. A larger fraction of fibrous debris is expected to pass through the self-cleaning strainers because of the scraping action that will break the fibrous debris into particles small enough to pass through the strainers. For passive strainers, when fibrous debris is present in a quantity large enough to coat the strainers, a complete precoat may be formed by the fibers, and virtually all other debris will then be effectively filtered onto the precoated strainer. For this condition a minimum quantity of debris would be expected to pass through. Without the fibrous "precoat", other debris such as iron oxide sludge of sizes smaller than the strainer holes would freely pass through the strainers. The purpose of this paper is to evaluate the impact of debris (which passes through the ECCS suction strainers) on the downstream components.

2. Particle Description

The fibrous debris (prototypically iron oxide sludge, rust, sand and epoxy particles) employed in the EPRI test program were either purchased or prepared for qualitative head loss testing purposes only. The epoxy particles are similar to what would be expected following degradation of safety-related epoxy coatings from steam impingement. A comparison of the size distribution of the rust and sand particles with actual plant debris has not been made.

The sample data obtained during the testing at EPRI is applicable for specific test conditions only. For example, the strainer head loss during the testing was higher than 150 inches of water, and most BWR ECCS pumps will not operate under these conditions due to inadequate Net Positive Suction Head (NPSH). In addition, the approach velocity could affect the filter efficiency of the strainer.

For plants which are considering installation of self-cleaning strainers, the debris materials are expected to be similar to the mockup test samples analyzed by GE (Reference 1). See Appendix 1 for details of the self-cleaning strainer test samples.

For plants with passive strainers, the test sample of Nukon fiber insulation was collected for particle characterization and was analyzed by GE (Reference 2) with the results as shown in Appendix 2. Note that all fibers collected were smaller than the strainer hole size. Appendix 3 provides a compilation of the various hole sizes for various plant strainers.

Other postulated lose particles could be the reflective metal insulation (RMI), but this debris is not expected to be small enough for a significant quantity to pass through the relatively fine-mesh ECCS pump suction strainers.

3. Safety and Operational Concerns

The safety and operational concerns associated with particles which pass through the ECCS suction strainers are:

1. The potential for failure of the ECCS pumps,
2. The potential for having inadequate heat transfer capability from the RHR heat exchangers,
3. The potential for plugging the core spray nozzles or the containment spray nozzles,
4. The potential for corrosion and chemical reaction with other reactor materials,
5. The potential for fuel bundle flow blockage and consequent fuel damage.

Each of these items will be addressed separately below.

4. Safety Evaluation

In order to generate a sufficient quantity of debris to obstruct flow through the ECCS suction strainers, a pipe break must be large enough to direct a steam–water jet to dislodge insulation material from surrounding pipes, structures, etc. In reality, a break of sufficient size to cause insulation material to be dislodged is very unlikely. It is much more probable that a small leak will develop prior to the occurrence of a break. Such a leak can be easily detected with existing plant instrumentation (technical specification limit of 5 gallons per minute unidentified leak), and the plant can be shut down for repairs before the leak develops into a break.

To degrade ECCS performance, this dislodged debris material would then have to travel to the suppression pool, mix with the suppression pool water, pass through the ECCS suction strainers, and be ingested into any or all of the motor driven ECCS pumps (typically five or six pumps, depending on the plant type).

The containment spray nozzles in BWR/2, 3, 4, and 5s are the most limiting components for which the ECCS suction strainer holes are sized (typically 0.125 inch holes) to prevent passage of foreign particulates of sufficient size to plug the drywell spray or suppression pool spray nozzles. For BWR/6s, the cyclone separator orifices for the ECCS pump seal flushing subsystem are the smallest orifices for which the strainer holes are sized (typically 0.09375 inch holes) to prevent plugging. See Appendix 3 for various strainer hole sizes.

For a LOCA, after considering the break (e.g., recirculation line) and a single failure, there will normally be at least one CS subsystem {low pressure core spray (LPCS) or high pressure core spray (HPCS) for BWR/5-6 plants} and one Low Pressure Coolant Injection (LPCI) subsystem available.

For BWR/3-4 plants, the LPCI subsystem injects into the jet pumps and into the vessel lower plenum, reflooding the core from below. In addition to reflooding the core, some LPCI flow will fill the bypass region surrounding the channels through the bypass holes and leakage through other openings such as the finger springs, located at the bottom of the fuel assemblies, used to steady the channel lower end. Any debris carried into the fuel bundle inlet region must stay suspended to contribute to blocking the lower tie plate. While this cannot be ruled out, it is unlikely that an entire fuel bundle lower tie plate will be blocked before the core is reflooded. In the long-term, the inlet flow rate to the core need only be enough to offset boiloff. This low inlet flow rate makes it even less likely that any significant amount of insulation material will remain suspended against the lower tie plate. Also, it is likely that due to the low velocities much of the debris will end up in the lower plenum. Certainly the heavier debris objects are more likely to end up there. Any buildup of debris in the bottom of the vessel will have no effect on the ability to cool the core and maintain it in a safe condition. It is considered incredible that the buildup would fill the lower plenum with debris.

For BWR/5-6 plants (plus Limerick and Hope Creek), LPCI injects directly into the bypass region (area between fuel channels and shroud) surrounding the core. For these plants any debris carried along with the LPCI flow will enter the bypass region. In order to reach the lower tie plate debris filter from the bypass region, the debris would have to pass through the small leakage paths between the core/lower plenum and bypass regions, or spill over the top of the fuel channels. Neither of these flow paths is expected to allow passage of enough debris to cause significant flow blockage at the lower tie plate debris filters.

The Core Spray System has nozzles sized to pass at least a 1/2" diameter sphere and will allow any debris (insulation materials or other) passing through the suction strainers to pass through the spray nozzles. No nozzle clogging is expected and only one CS system is required. CS cools the core from above and contributes to reflooding the core by filling the bypass region and the fuel channels.

The ECC Systems are only required to make up boil–off due to decay heat, and the remaining flow spills out the break.

For this debris analysis the particles evaluated are rust, paint chips, sand, and fibrous debris of random sizes and shapes. The rust chips are of low strength and will fracture into even smaller pieces upon interaction with other components. Similarly, the epoxy paint is also relatively brittle and will breakup as well. The sand will not melt or form a large enough agglomeration to significantly block flow (Reference 3). The glass fibers are "so fragile" (Reference 2) that they "have virtually no mechanical strength" (Reference 3). The rust, paint, and fiberglass debris that pass through the suppression pool strainers will be subjected to the ECCS flow rates and turbulence that will cause disintegration into particles of even smaller sizes than those described in Section 2, "Particle Description". This evaluation takes credit for the disintegration of rust, paint, and fiberglass particles.

4.1 The Potential for Failure of the ECCS Pumps

Seal Cooling Orifices

Orifices that control the flow to the ECCS pump seals are susceptible to plugging by particulates larger than the inlet seal cooling line hole diameter. Orifice holes in this application are 0.0625" and larger. Hard and round particulates smaller than 0.0625" would pass through the orifice. Loose strands of fiber less than 0.0625" in diameter may pass through the orifice, however large concentrations (blitz) of the fiber could plug the orifice. The consequence of a plugged orifice is high seal temperature and poor seal life (Reference 4).

Wear rings and bushings are specifically designed (hard materials) to resist wear due to hard particulates in the process fluid. If the concentration of hard particulates is unusually excessive, the impact could be a long-term deterioration in the pump performance, in the form of low pump head

(Reference 4). The requirement of 100 days of post LOCA operation is not considered long-term.

Seal Faces

New seal faces are lapped to very flat and smooth surfaces. The working gap between the faces is a fraction of a micron. This means that large particulates would pass over the seal faces, and would not enter the interface to destroy the smoothness of the face and cause leakage (Reference 4).

For the passive strainer with the holes sized at 0.125" (see Appendix 3. for listing of various strainer hole sizes) little fiber is expected to pass through after the initial filter bed is formed (Reference 5), and also little of the other debris (except for minimum sized iron oxide sludge) is expected to pass after the initial filter bed precoat is formed. For the self-cleaning strainer, more than 95% of the fiber and 100% of all the other debris is smaller than the pump seal cooling orifice diameter of 0.0625". Therefore, all materials would most likely pass through the orifice if 1% by volume of fiber (as estimated in Reference 6) does not cause a highly unlikely "blitz" which plugs the orifice. Since all particles are larger than a faction of a micron, they would not enter the pump seal face. Reference 7, which has reviewed several types of ECCS pumps for various BWR and Pressurized Water Reactor plants, notes that for shafts and bushings, debris in quantities of one percent or less of the pump fluid is likely to not constitute a major threat to the bushing integrity.

There were no problems with any of the pump seals or bushings from the debris passed during the blowdown event at Limerick, where the RHR pump strainer became unexpectedly clogged. Nine Mile Point 1 (NMP-1) does not have a ECCS suction strainer, but does have a 1/8" mesh strainer on the down-stream side of the pump. The suction to the pump has a very course grid with openings of 1" by 2". It is known that the grid has passed some chunks of wood that did not go through the ECCS pump but collected in the suction sump prior to the pump. Other condensate strainer material pieces have been observed down stream of the pump with no detrimental effect on the pumps. The ECCS strainers at Fitzpatrick have 1/4" square hole openings and have operated that way for many years without any pump problems.

The BWROG Interim Report published in December 1994 (Reference 8) states that iron oxide could contribute to accelerated seal wear due to abrasion, but this would be gradual and require months of wear to significantly degrade pump performance. Reference 7 acknowledges that pump vendors are not aware of any quantitative data which can be used to provide guidance in evaluating the likelihood of wear or seizure in the presence of particulates for these pumps. Reference 8 also states that "GE supplied ECCS pumps have been thoroughly tested to confirm design margin exists and to demonstrate that these pumps are rugged. Some of this testing involved extended operation under cavitating conditions. Initial testing involved running the pumps at desired capacities (4000 to 6000 gpm) and then reducing the suction pressure below the minimum required NPSH values until the impeller was cavitating. At the completion of this testing, the impeller was removed and inspected. There was no evidence of damage to the impeller from cavitation and only an indication of slight rubbing on the bottom impeller wear surfaces. No damage was evident on the case ring". Additional cavitation tests were performed without degradation of pump performance (Reference 8). As stated earlier a "blitz" or as stated in Reference 8, "clumps" of fibrous insulation have a potential to plug the cyclone separator inlet orifices on the ECCS pumps, which could cause the pump seals to heat up and wear faster, eventually leading to seal leakage that could potentially create a water management problem. Reference 8 concludes that the pumps are extremely rugged but do have clearances which could be adversely affected by fibrous insulation. The pump vendors have confirmed that these pumps would continue to run with significant insulation material in the process stream. Therefore, debris-induced damage is not a safety concern for any of the ECCS pumps.

4.2 The Potential for Inadequate Heat Transfer Capability from the RHR

Heat Exchangers

Significant impact on RHR heat exchanger performance can occur if a large quantity of debris is retained inside the heat exchangers causing blockage of the flow and/or fouling of the outer surfaces of the tubes (Reference 9). Flow from the suppression pool is channeled through the shell side of the RHR heat exchangers. The shell side flow velocity of a RHR heat exchanger varies from 2.5 to 5 ft/sec. At these velocities the flow will entrain the small particles without allowing them to settle in the heat exchanger. The most restrictive opening along the flow path is the spacing between adjacent tubes, which ranges from 0.25" to 0.5" in RHR heat exchangers. The tubes sizes are 0.75" or 1.0" diameter. The results of the size distribution analyses presented in Reference 1 are evaluated as follows:

1. The rust chips are the largest, but are very likely to break into smaller pieces. Considering the possibility that the largest chips get through the strainer holes and through the pumps without being broken up, (not considered credible), they may get stuck somewhere in the closely packed tubes of the tube bundle. They could then serve as nuclei to collect other debris. If this were to occur in a substantial quantity, fouling of the tube outer surface could take place and this would adversely impact heat exchanger performance. In addition, Cu-Ni tubes are used in some RHR heat exchangers. Iron oxide (Fe_2O_3 or Fe_3O_4) promotes oxidation and corrosion on the outside diameter of the Cu-Ni tubes and may contribute to fouling and/or thinning of the tubes.

2. Epoxy paint chips are small and light enough that they will be swept through the heat exchangers, and are of no concern.

3. The size of the sand grains are small enough that it is unlikely that they will be captured along the flow path, but may be heavy enough to settle in pockets of low velocity near the bottom of the heat exchanger. Since they will not settle on the outer surface of the tubes, they will not affect the heat exchanger performance.

4. Of the samples evaluated in Reference 1, only 0.1% of the fiber population had a length of 0.39" or greater. With this length it is unlikely they could attach to the outside diameter of even the smaller (0.75") diameter tubes. Moreover, it was reported in Reference 2 that the fibers were so fragile that any attempt to disperse the clumps caused extensive breakage of the longer fibers. These fibers also will be easily swept away and carried out of the heat exchanger without impacting heat exchanger performance.

In summary, a review of heat exchanger performance concludes that non-soluble insulation material will not deteriorate the performance of the as-is heat exchanger. The rust chips could present some potential impact to RHR heat exchanger performance. However, this concern is minimized by the fact that a large fraction of the bigger chips are so thin that they will flow through the heat exchangers while others will be broken into still smaller pieces by the rapid flow and therefore easily pass through the heat exchanger. The key factors in heat exchanger performance are the routine maintenance, inspection, and cleaning of the heat exchanger. Debris that pass through the ECCS suction strainers do not affect heat exchanger performance Therefore, there is no abnormal operational or safety concern with the identified debris on RHR heat exchanger performance, assuming they are properly maintained.

4.3 The Potential for Plugging of Core Spray Nozzles and Containment Spray Nozzles

During the review of plant drawings (References 9,10,&11) the minimum orifice diameter in the core spray headers was found to be 0.5". The containment spray nozzles were found to have orifices or openings sized from 0.125" to 1.5". It is highly unlikely that any of the identified debris in section 2, which would be expected to be much smaller by the time it reached the orifices, would be able to block the orifice. A very few longer particles would be expected to pass through the passive suction strainers (none were found in the debris that passed through the self cleaning strainer). There is no safety significance due to the small number of particles versus the large number of core spray and containment spray nozzles and orifices. Therefore, the expected debris will be of no safety concern for the core spray and containment spray operation.

4.4 The Potential for Corrosion and Chemical Reaction with other Reactor Materials

Iron oxide is found throughout the reactor system and will not chemically affect the reactor system components (Reference 3). Epoxy paint will not react chemically with any of the reactor materials. The sand (silica or alumina material) chemically is not detrimental to the reactor system, and mechanically will not melt or form a large enough agglomeration to significantly block flow paths such as fuel orifices (Reference 3). Fiberglass, at reactor temperatures, will not dissolve or melt and will remain as small fibers. These fibers are considered to have no mechanical strength, and will continue to fracture into smaller pieces that will not mechanically block flow paths (Reference 3). Therefore, there is no safety concern for corrosion or chemical reaction with other reactor materials due to the expected debris identified in section 2.

4.5 The Potential for Fuel Bundle Flow Blockage and Consequent Fuel Damage

A safety evaluation (Reference 12) by the GE Nuclear Energy Fuel Department has addressed the fiberglass debris as it might affect the new GE11 and GE13.

Reference 12 states that even though the fibrous insulation would not be expected to plug the debris filter, the consequences of plugging were considered from an ECCS cooling standpoint. As a result of these considerations, it was concluded that adequate core cooling would be provided during a LOCA. With normal core spray distribution, complete flow blockage of the fuel lower tie plate debris filter would allow adequate core cooling to be maintained. Consequently, it is very unlikely that excessive flow blockage of the lower tie plate debris filter would jeopardize adequate post-LOCA core cooling. Even for a core spray line break with failure of the diesel generator (D/G) powering the other core spray pump, the high pressure ECCS and some LPCI pump(s) will remain available for core cooling. The core spray line is located above the top of the core, so it will be possible to rapidly restore the vessel water level to above top of active fuel (TAF) once the vessel pressure is low enough to allow the LPCI pumps to inject. It is considered inconceivable for debris to plug all channels so that flooding could not occur from below. However, if the inlet to one or more fuel channels is totally blocked from below by debris, these bundles would receive radiation cooling to the channel walls as the bypass refills, then direct cooling from water spill-over from above once the water level is restored above the top of the fuel channels. Due to the expected core reflooding rate and the relatively small size of the core spray line break, it is GE Nuclear Energy's judgment that, on a best-estimate basis, the fuel in any blocked channels would remain well below the peak cladding temperature (PCT) limit of 2200°F.

The maximum particle sizes of the expected rust, iron oxide, epoxy paint, and sand are smaller than the fuel debris filter holes sizes and are likely to pass through without plugging. Therefore, there is no safety concern for fuel bundle flow blockage and consequent fuel damage due to all the debris identified in section 2.

5. Conclusion

This safety evaluation shows that adequate core cooling provided during a LOCA will not be compromised by the presence of rust, epoxy paint chips, sand, iron oxide sludge, and fibrous debris in the ECCS system or reactor core. It is concluded that there is no safety concern for the potential failure of the ECCS pumps, inadequate cooling capacity from the RHR heat exchangers, plugging of the core spray header nozzles, plugging of the containment spray nozzles, corrosion or chemical reaction with other reactor materials, or fuel bundle flow blockage.

6. References

1. Letter from Roger Caputi to Tom Green, Subject: "Particle Characteristics of Five Samples", dated October 11, 1995

2. Letter from Roger Caputi to Tom Green/John Embley, Subject: "Particle Characteristics of NUKON Fiber Insulation" dated September 5, 1995

3. Letter from B. D. Frew to P. F. Kachel, Subject: "Review of BWROG ECCS Samples", dated October 25, 1995

4. Letter 951026-1 from H. Tafarrodi to Pete Kachel, Subject: "Impact of Oxide Particles and Nukon Insulation Fiber on the Performance of the ECCS Pumps", dated October 31, 1995 and Revised November 1, 1995

5. Letter from T. A. Green to J. F. Klapproth, Subject: "NUKON FIBER PENETRATION OF ECCS PUMP SUCTION STRAINERS - - EFFECT ON LOWER CORE SUPPORT PLATE FUEL FILTERS", dated October 26, 1995

6. NUREG - 0897, Revision 1, dated October 1995, Item 3.2.2.4 "Particulate Ingestion"

7. Creare R & D Inc. TM - 962, "EMERGENCY COOLING PUMPS IN BOILING WATER REACTORS", by W. L. Swift, dated May 1, 1984

8. Interim Report of the BWROG ECCS Suction Strainer Committee, December, 1994

9. GE Drawing 131C8543 "Orifice" Core Spray Sprager

10. GE Drawing 137C7227 "Orifice" Core Spray Nozzles & HDW

11. GE Drawing 167B2284 "Orifice" Core Spray Nozzles & HDW

12. 10 CFR 50.59 Safety Evaluation of the GE11 and GE13 Fuel Bundle Debris Filter, prepared by J. L. Embley, dated September 7, 1995 (GE Class III Proprietary Information)

APPENDIX 1

SELF-CLEANING STRAINER TEST SAMPLES

Because of the nature of the first 3 samples, the overall examination involved a limited sieving and general qualitative examination of the sieved fractions.

Sample 1 (Rust):

A. The bulk sample had the appearance of semi-metallic gray-black/brown-black, irregular, oblong chips. The dimensions of the largest chips were typically in the range of 0.39" - 0.67" in length by 0.24" - 0.39" in width, with a thickness of the majority being 0.002" - 0.006" and a few being 0.008" - 0.012":

B. Sieve Results

Size Range	Weight %
ℓ > 0.039"	73
0.039" > ℓ > 0.006"	20.5
0.006" > ℓ > 0.003"	3.2
0.003" > ℓ > 0.002"	1.1
0.002" > ℓ	1.7

Sample 2 (Epoxy)

A. The bulk sample was composed of non-metallic white, irregular, angular flakes. The area of the largest chips were typically in the range of 0.20" - 0.28" in length by 0.08" - 0.16" in width, with a thickness of 0.008" - 0.009":

B. Sieve Results

Size Range	Weight %
$\ell > 0.039"$	42
$0.039" > \ell > 0.006"$	56
$0.006" > \ell > 0.003"$	1.0
$0.003" > \ell > 0.002"$	0.3
$0.002" > \ell$	0.1

Sample 3 (Sand)

Since the sieve test showed all particles were less than 0.039" but greater than 0.006", an examination of the sand was carried out under a stereomicroscope. Results from this exam showed the sample to be composed of uniform sand granules with a size range of 0.008" to 0.024".

Samples 4 and 5 (Fiberglass Fibers)

A. These two samples were virtually identical and are treated as one in the findings given below. (Sample 5 was somewhat dirtier and had a slightly higher fraction (~2X) of 0.197" fibers.

B. Observations of the 2 samples were carried out using both a stereomicroscope and a standard transmitted light microscope. In appearance and general handling characteristics these fibers matched the Nukon fiber insulation.

C. Quantitative results from the combined microscopic examinations are given below.

1. Diameter range: 0.00016" - 0.00036" (4 - 9 micron)

It should be noted that the values listed below are estimates, and the order of magnitude of the value rather than the value itself should be given the greater weight.

Nominal Length	Number %
0.39"	0.1
0.20"	1.0
0.08"	4.0
0.04"	10
<0.04"	85

APPENDIX 2

PASSIVE STRAINER TESTS SAMPLES

Nukon fiber insulation

Particle lengths varied from over 0.07" down to 0.005". Relatively few particles were found with this minimum length. A significant rise in number started in the 0.012" to 0.024" region and continued to show high populations in the 0.032" to 0.039") region. Beyond 0.039" the fiber population decreased markedly consistent with the results from the stereomicroscope observations. The diameter range was fairly narrow from 0.00016" to 0.00036" (4 - 9 micron)

APPENDIX 3

VARIOUS STRAINER HOLE SIZES

Plant	Strainer Hole Size
Fitzpatrick	1/4" square
Duane Arnold	1/8" circular
Hope Creek	1/8" between plates
LaSalle	0.60" x 0.60" wire screen
Quad Cites	1/8" circular
Dresden	1/8" circular
Oyster Creek	3/16" circular
Monticello	1/8" circular
Millstone	1/8" circular
Perry	3/32" circular
Grand Gulf	3/32" circular
Clinton	3/32" circular
River Bend	3/32" circular
NMP-2	3/32" circular
Hanford	3/32" circular
Limerick	12 x12 wire mesh (0.083" x 0.083") [0.060" square holes]

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 1, 1996

Mr. Rocky Sgarro
Pennsylvania Power and Light
Mail Code A6-1
2 North Ninth Street
Allentown, PA 18101

SUBJECT: BWR Drywell Debris Transport - Initial PIRT Report

Dear Mr. Sgarro:

Enclosed is a copy of the June 28, 1996, BWR Drywell Debris Transport Phenomena Identification and Ranking Tables (PIRT) report. This summary report is the result of PIRT panel meetings, which were held on May 15-17, 1996, in Albuquerque, New Mexico. We will utilize the phenomena identified and recommendations provided in planning our BWR drywell debris transport studies.

Although this report has not been reviewed in detail by the staff at this time, we are forwarding you this copy for the purpose of exchanging information in a timely manner.

If have questions related to this report, please call me. I am also taking the liberty of mailing copies to Tom Green and Alan Bilamin.

Sincerely,

Aleck W. Serkiz, Senior Task Manager
Generic Safety Issues Branch
Division of Engineering Technology
Office of Nuclear Regulatory Research

cc: w/PIRT Report
T. Green, GE/San Jose
A. Bilamin, Continuum Dynamics
PDR

cc: w/o PIRT Report
M. Marshall, Jr., RES/GSIB
C. Berlinger, NRR/SCSB
R. Elliott, NRR/SCSB
G. Wilson, INEL/LITC

Initial Issue
June 28, 1996

BWR Drywell Debris Transport Phenomena Identification and Ranking Tables (PIRT)

Gary E. Wilson (Idaho National Engineering Laboratory, Project Coordinator))
Brent Boyack (Los Alamos National Engineering Laboratory)
Mark Leonard (Innovative Technology Solutions Corp.)
Ken Williams (Flow Simulation Services, Inc.)
Lothar Wolf (University of Maryland)

Abstract

The NRC has recently issued a Regulatory Bulletin & accompanying Regulatory Guide (1.82, Rev. 2) which will require licensees of BWRs to develop a specific plan of action (including hardware backfits, if necessary) to preclude the possibility of early ECCS strainer blockage following a postulated LOCA. The postulated mechanism for strainer blockage is destruction of piping insulation in the immediate vicinity of the break and subsequent transport of fragmented insulation to the wetwell. In the absence of more definitive information, the Regulatory Guide recommends that licensees assume a dry well debris transport fraction of 1.0. Accordingly, the NRC has initiated research focused toward developing a technical basis to provide insights useful to regulatory oversight of licensee submittals associated with resolution of the postulated strainer blockage issue. Part of this research is directed towards development and application of analytical methods for more realistic definition of the debris transport through the drywell to the wetwell. To help focus this development into a cost effective effort, a panel, with broad based knowledge and experience, was formed to address the relative importance of the various phenomena that can be expected in plant response to postulated accidents that may produce strainer blockage. The resulting phenomena identification and ranking tables (PIRTs) reported herein will be used to help guide the analytical methodology development to a sufficient and efficient predictive capability. The phenomena occurring in BWR drywells was the specific focus of the panel, although the results reported herein may have additional utility.

Executive Summary

The NRC has recently issued a Regulatory Bulletin & accompanying Regulatory Guide (1.82, Rev. 2) which will require licensees of BWRs to develop a specific plan of action (including hardware backfits, if necessary) to preclude the possibility of early ECCS strainer blockage following a postulated LOCA. In the absence of more definitive information, the Regulatory Guide recommends that licensees assume a dry well debris transport fraction of 1.0. Accordingly, the NRC has initiated research focused toward developing a technical basis to provide insights useful to regulatory oversight of licensee submittals associated with resolution of the postulated strainer blockage issue. Part of this research is directed toward development of analytical methods for more realistic definition of the debris transport through the drywell to the wetwell. To help focus this development, a panel, with broad based knowledge and experience, was formed to apply the PIRT process to transport of break generated debris through BWR drywells. The first phase of the PIRT project executed in April-May 1996 and reported here, was focused toward timely development of initial PIRTs to guide the on-going analytical methodology development. The second phase, scheduled to be completed in approximately four months, is planned to use results from the analytical methodology development and application to confirm and refine the PIRTs. A third phase, after an additional three months, is tentatively planned, with the objective of further evaluation of the application of the analytical methodology to the strainer blockage issue.

The phenomena identification and ranking tables (PIRTs) developed by the panel will be used to help guide the analytical methodology development to a sufficient and efficient predictive capability. The highly important phenomena from those tables are summarized on the next page by drywell location (component), general phenomena type, and time in the transient (blowdown phase and post blowdown phase). Tables 1 and 2 in the body of the report list the full PIRTs developed by the panel.

Scenario and plant design selections are an important part of the PIRT process. Given the combinations of break locations and containment types, a large number of scenarios are possible. For the most part in the work accomplished to date, the panel focused on a single break type (high elevation main steam line break), a single containment type (Mark I), and no containment spray. However, obvious differences in phenomena importance in a recirculation line break low in the drywell, and in the other two containment designs, have been identified in the current PIRTs. The panel believes extensions to the broader range of potential scenarios should be recognized in the planned methodology development and application (see subsequent discussions). In addition, the currently planned exploratory CFD analysis of a recirculation line break low in the drywell is believed to be well taken in the context of expansion of the current PIRTs to the broader range of potential transients.

The validity (accuracy) of PIRTs is strongly dependent on the degree to which the available experimental and analytical data encompass the plant postulated accident response envelope. The panel believes the current detailed PIRTs in the body of the report, Tables 1 and 2, adequately reflect the information available during the initial PIRT development. However, the panel strongly supports the planned review of additional information expected to be available in approximately three months in the continuing analytical methodology development and application. This is considered to be particularly true if the scoping analyses suggested in a following paragraph are executed. The panel expects the planned PIRT refinements, based on the forthcoming information, will be significantly more useful in helping providing desired regulatory related insights.

With respect to the continued development and application of the analytical methodology, the panel's primary suggestion relates to performing "scoping" analyses early in the continuing methodology development. The details of the suggested effort are described in the first bullet of Section 3.2 in the body of the report. It may be noted the panel believes the suggested scoping

BWR drywell debris transport highly important phenomena

Component	Phenomenon type	Phenomenon	Highest of the highly ranked phenomena
Blowdown phase			
Drywell open areas	Thermal hydraulic related	Pressure driven flows (bulk flows)	✓
		Localized flow field	✓
		Flashing of break liquid effluent①	✓
	Debris transport & depletion related	Advection/slip	✓
		Debris fragmentation	✓
		Gravitational settling	✓
Drywell structures	Thermal hydraulic related	Porosity	✓
		Recirculation (streaming) deluge	✓
	Debris transport & depletion related	Recirculation deluge (streaming) related transport	✓
		Entrapment/impaction	✓
		Adhesion	✓
		Runoff/re-entrainment	
Drywell floor	Thermal hydraulic related	Pool formation①	
		Pool overflow (timing issue this phase)①	
		Surface wetting (before pool formation)	
		Pool flow dynamics①	
	Debris transport & depletion related	Pool transport (to/through vent)①	
		Adhesion	
		Settling①	
		Impaction	
Post-blown down phase			
Drywell structures	Thermal hydraulic related	Condensation②	✓
		Film draining under gravity②	✓
	Debris transport & depletion related	Film related transport②	✓
		Runoff/re-entrainment②	✓
Drywell floor	Thermal hydraulic related	Pool overflow	✓
		Pool formation	✓
		Pool flow dynamics②	✓
	Debris transport & depletion related	Pool transport (to/toward vent)②	✓
		Settling	✓
		Debris fragmentation, including baffle effect①	✓
Vent entrance	Thermal hydraulic related	Localized liquid flow field②	✓
	Debris transport & depletion related	Advected mass②	✓

Notes: ① Applies only to the recirculation line break.
② Applies only in the case of drywell overflow to vent.

analyses will have a secondary benefit. That is, the results can be expected to aid in the planned PIRT confirmation and refinement discussed above.

The panel was requested to provide advice regarding methods for analyzing BWR debris transport. The panel was able to develop perspectives in this regard. The panel believes the defensibility of the technical adequacy of the final analytical methodology will depend strongly on an adequate validation of that methodology. This leads to the following panel perspectives (in generally decreasing order of importance):

① The inherited MELCOR aerosol models should be shown to be adequate (validated) to model debris transport.

② The ability of MELCOR to adequately model mass, momentum and energy transport between adjacent volumes should be demonstrated. A suggested technique is to show consistency between MELCOR and independent flow field evaluations such as those generated with a CFD model.

③ The proposed use of MELCOR as part of the analytical methodology will require the addition of several models for PIRT identified highly important phenomena (see Section 3.2). It is important such models be shown to be adequate for their intended purpose. The scoping studies already noted above should be useful in helping define "adequate".

④ The quality of the validations noted in the above three items will increase in proportion to the amount of applicable experimental data available for that use. Acquisition of additional data through "bench top" experiments can be of considerable worth. This in turn implies a well conceived scaling rationale structure to enable full use of any new data. This is equally true of existing experimental data. Prototypical characterization (amount, size, constituents) of the debris generated by a LOCA is of special concern. It is not clear to the panel that sufficient data is available for a well executed validation of the debris generation model(s). Debris generation must be a key feature in the analysis methodology. Limited means to demonstrate the prototypicality of the model(s) will diminish the quality of the validation.

⑤ The uncertainty quantification approach proposed to be used in conjunction with the methodology development is considered to be a worthy effort. The panel believes the one–at-a-time sensitivity studies should be performed with two objectives: 1) help in further confirmation and refinement of the PIRTs and, 2) help define adequacy in the context of model requirements. However, the panel cautions that satisfaction of objective 2) is <u>more defensible</u> if the uncertainty is developed in a <u>quantified statistic</u>. That is, the uncertainty is cast in terms of a probability distribution, rather than statements such as "it is highly unlikely that • • • •". The statement, "it is highly unlikely that • • • •", implies use of a bounding approach. If a bounding approach method is selected to determine uncertainty, then the panel notes the bounding values <u>must</u> be well <u>justified</u> and <u>documented</u>.

Finally, the panel believes application of the PIRT process was more successful than initially might have been expected in resolving the highly complex problem of interest into a tractable issue. The extension of the standard methodology to identify high level integral system processes, and relate the phenomena to these processes, early in the effort was of particular benefit. This addition to the PIRT process was instrumental in helping the panel to address, for the first time, an application that required consideration of the cross-coupling of the distinctly different debris related phenomena, from the more normal thermal hydraulic processes treated in previous PIRT studies. The panel suggests these lessons learned should be considered in future applications of the PIRT process.

BWR Drywell Debris Transport
Phenomena Identification and Ranking Tables (PIRT)

1. Introduction

1.1 Background - The NRC has recently issued a Regulatory Bulletin & accompanying Regulatory Guide (1.82, Rev. 2)[1] which will require licensees of BWRs to develop a specific plan of action (including hardware backfits, if necessary) to preclude the possibility of early ECCS strainer blockage following a postulated LOCA. The postulated mechanism for strainer blockage is destruction of piping insulation in the immediate vicinity of the break and subsequent transport of fragmented insulation to the wetwell. In the absence of experimental data and analytical results, demonstrating significant retention of debris in the drywell, the Regulatory Guide recommends that licensees assume 100% of debris, generated as a consequence of the LOCA, is transported from the drywell to the suppression pool. The current recommendation to use a dry well debris transport fraction of 1.0 can pose significant design impacts for some licensees.

A review of incidents that have occurred to date indicate two general categories of ECCS strainer blockage mechanisms. One (an incident in the Barsebäck plant in Sweden involving the spurious opening of a safety valve) involves debris generation in the drywell due to blast effects of high-velocity coolant discharge from the primary coolant system onto piping insulation. Similar effects are expected if a pipe running through the drywell should rupture. Transport of fibrous debris to, and collected on, ECCS strainers reduces NPSH and degrades pump performance. The second category are US incidents in which degraded RHR pump performance was observed as a consequence of pre-existing debris and sludge in the suppression pool collecting on ECCS strainers. This category has already been addressed through a separate NRC bulletin which requires periodic cleaning of BWR suppression pools.

Characterization of the debris and amount generated as a consequence of a LOCA in a BWR drywell is being addressed through an experimental program supported by the BWR Owner's Group (BWROG). Information from the NRC research[2], and to a limited extent from the BWROG work, constitutes the baseline for the PIRT project described herein.

1.2 Objectives

1.2.1 USNRC BWR Debris Transport Research - The primary objective of the NRC research program is to identify analytical methods and experimental evidence, and thereby develop a technical framework for evaluation of licensee submittals related to mitigation of strainer blockages.

1.2.2. PIRT Project - The primary objectives[3] of the project for the PIRT panel are to:

1) Use the Phenomena Identification and Ranking Table (PIRT) process to identify phenomena and to rank their importance as related to transport of LOCA generated debris within US BWR drywells,
2) Use the PIRT tables to advise the NRC staff in the analysis of BWR drywell debris transport, from the perspectives of phenomena modeling and identification of present computer codes best suited for such analyses,
3) Advise the NRC staff regarding potential methods to characterize the estimated uncertainties in code predictions and the application of calculations to predict actual plant behavior,
4) Advise the NRC staff regarding the panel's views about the success expectancy for the analysis approach presented to the panel.

These objectives are planned to be achieved through a two-phase PIRT application. The first phase, reported here, was focused toward timely development of initial PIRTs to guide the on-going analytical methodology development. The second phase, scheduled to be completed in approximately four months, is planned to use results from the analytical methodology development and application to confirm and refine the PIRTs. A third phase, after an additional three months, is tentatively planned, with the objective of further evaluation of the application of the analytical methodology to the BWR strainer blockage issue.

1.3 Report Structure - The primary topic of interest, the PIRTs, are provided in Tables 1 - 2 in Section 2. The highly ranked phenomena, extracted from these tables, are also summarized in the Executive Summary. The base conditions for which the PIRTs were developed are also provided in Section 2, in subsections preceding the tables. Phenomena descriptions and ranking rationales (as referenced in Tables 1 - 2) are provided, respectively in Appendices A and B. Details of the PIRT panel insights regarding the PIRTs, the debris transport related research, and other aspects of the strainer blockage issue are given in Section 3. These results are also summarized in general order of importance in the Executive Summary. Documents more directly related to the PIRT development are referenced throughout the report and identified in Section 4. Other sources of information, that completed the general information base available to the panel prior to the first meeting, or developed in association with panel meetings, are summarized in Appendix C.

2. BWR Drywell Debris Transport PIRTs

2.1 PIRT Process Overview - The information obtained through the application of the PIRT process[4,5,6] identifies the requirements which will be imposed on research supporting analytical tools used to simulate accident scenarios. In addition, those requirements are prioritized with respect to their contributions to the reactor phenomenological response to the accident scenario. Because it is not cost effective, nor required, to assess and examine all the parameters and models in a best estimate code in a uniform fashion, the methodology focuses on those processes and phenomena which dominate the transient behavior, although all plausible effects are considered. This screening of plausible phenomena, to determine those which dominate the plant response, insures a sufficient and efficient analysis. PIRTs are not computer code-specific, that is, PIRTs are applicable to the scenario and plant design regardless of which code may be chosen to perform the subsequent safety analysis. This also adds to the efficiency and generality of the process.

A typical application of the PIRT process is conceptually illustrated in Figure 1 and described as follows. The PIRT process focuses on phenomena/processes that are important to the particular scenario, or class of transients, in the specified NPP (i.e., those that drive events). Plausible physical phenomena and processes, and their associated system components are identified. From a modeling perspective, phenomena/processes important to a plant response to an accident scenario can be grouped in two separate categories: 1) higher level system interactions (integral) between components/ subsystems , and 2) those local (within) to a component/subsystem. The identification of plausible phenomena is focused toward component organization, but experience has indicated it can be most helpful to relate the phenomena to higher level integral system processes. Often time can be saved when it can be demonstrated a higher level integral system process is of low importance during a specific time phase. A subsequent and equally important step is the partitioning of the plant into components/subsystems. This latter step is a significant aid in organizing and ranking phenomena/processes.

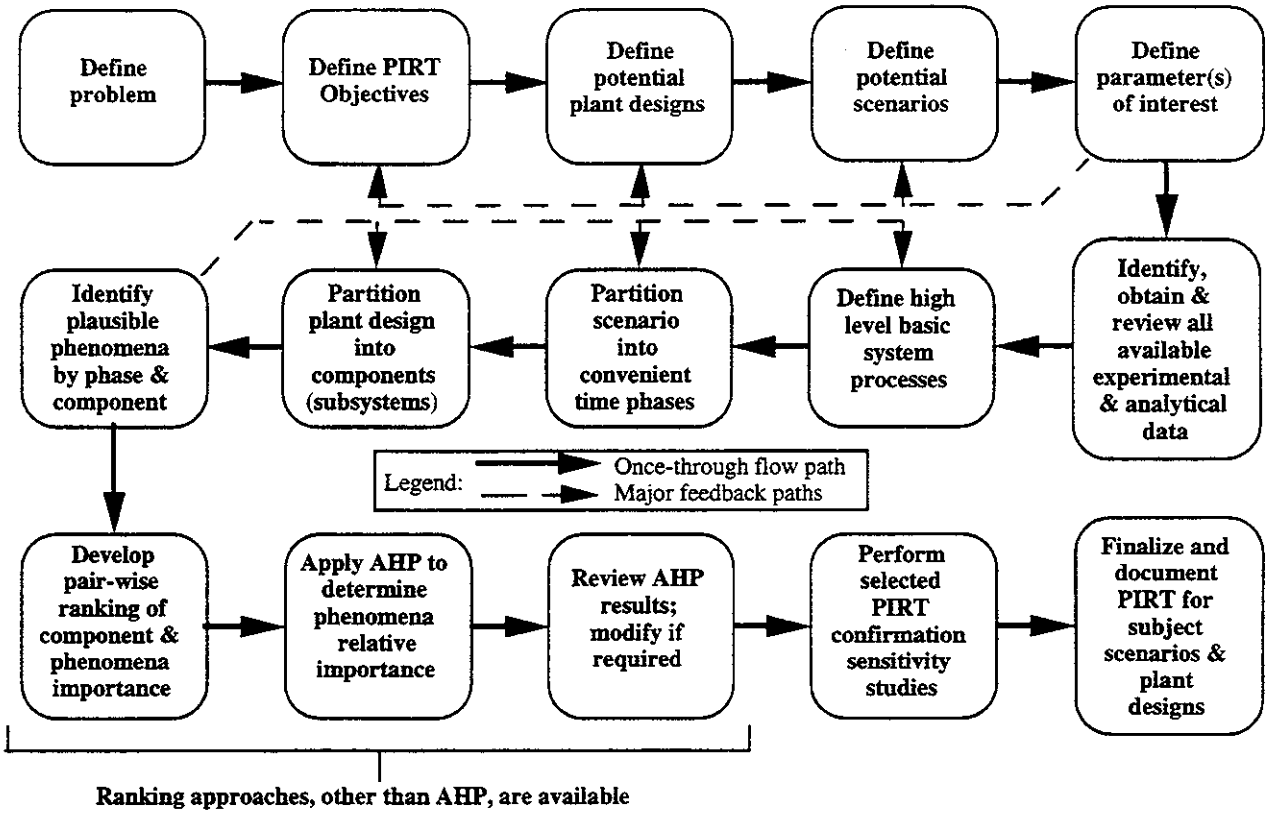

Figure 1. Illustration of typical PIRT process.

The phenomena/processes are then ranked with respect to their influence on the primary evaluation criteria to establish PIRTs. Primary evaluation criteria (or criterion) are normally based on regulatory safety requirements such as those related to restrictions in fuel rods (peak clad temperature, hydrogen generation, etc.) and/or containment operation (peak pressure, ECCS performance, etc.). The rank of a phenomenon or process is a measure of its relative influence on the primary criteria (criterion). The identification and ranking are justified and documented.

The relative importance of phenomena are time dependent as an accident progresses. Thus, it is convenient to partition accident scenarios into time phases in which the dominant phenomena/ processes remain essentially constant; each phase is separately investigated. The processes and phenomena associated with each component are examined as are the inter-relations between the components. Cause and effect are differentiated.

The processes and phenomena and their respective importance (rank) are judged by examination of experimental data, code simulations related to the plant and scenario, and the collective expertise and experience of the evaluation team. Independent techniques to accomplish the ranking include expert opinion[1], subjective decision making methods (such as the Analytical Hierarchy Process [AHP]), and selected calculations. The final product of application of the PIRT process is a set of tables (PIRTs) documenting the ranks (relative importance) of phenomena and processes, by transient phase and system component. Supplemental products include descriptions of the ranking scales, phenomena and processes definitions, evaluation criteria, and the technical rationales for each rank. In the context of the PIRT process application to drywell debris transport, the primary elements of interest are described in the following Sections 2.2 through 2.8. The PIRTs resulting from this specific application are documented in Section 2.9.

2.2 Primary Parameter of Interest - This is the criterion that was defined and used to judge the relative importance of the phenomena/processes important to drywell debris transport. For the present PIRT endeavor, it was obvious that this parameter must be the *fraction of debris mass generated within the "break region" that is transported to the wetwell vent entrance.*

2.3 Plant Design(s) Considered - For US BWRs there are three different containment types: Mark I, Mark II and Mark III. There are a total of 37 BWR plants of which 23 have a Mark I design. It was determined that the best approach for the initial PIRT exercise was to focus first on a Mark I design because of its unique features, and then highlight differences expected to impact the other two containment designs. Because containment spray activation is not automatic in the Mark I and II designs, and because the Mark III design does not contain a spray system, it was concluded spray effects would not be specifically included in the PIRTs. In summary, given the panel's time constraint, the initial approach was to develop a "generic" PIRT that is common to all three designs, but possibly containing "exception" statements that are design specific.

2.4 Accident Scenario(s) Considered - Considerable effort has been given to specify volumetric debris generation depending on various scenarios (LLOCA, MLOCA, steam and recirculation line break, etc.) in NUREG/CR-6224[2]. This study further included the failure probability of numerous weld locations, various elevations and numerous systems piping. These results concluded that this "spectrum of breaks" can lead to a large variation in volume of debris generated; namely varying from 2 to over 112 ft^3 of debris. Thus, the specific accident scenario considered may have some effect on the relative importance of some phenomena. To accommodate

[1] As described in Section 2.8, the ranking in this effort focused on expert opinion based on experimental and analytical data, and the panel members' broad experience in the field of interest.

this variable within the time constraints available to complete the PIRT, the panel agreed to consider the following scenario as the primary basis for ranking phenomena:

- Large LOCA: The intent was to base the PIRT on a bounding accident scenario (in terms of debris generation). Primary consideration was given to a steam line break, but major differences in recirculation line breaks were also recognized to account for break effluent fluid conditions.
- Full-power operation at the time of break initiation.

Again the initial approach was to develop a "generic" PIRT with respect to the different scenario conditions, but possibly containing "exception" statements that are scenario dependent.

2.5 Partitioning of Drywell into Components - The panel was fortunate that the prior work[2] provided a consistent framework for partitioning the drywell into the four components pictorially illustrated in Figure 2 and described below:

- Open area: The free flow area, excluding the potential pool in the bottom of the drywell.
- Structures: All solid boundaries and barriers to the flow stream, including drywell walls, pipes, cabinets, walls, grates, etc.
- Floor: That area where a potential, essentially liquid, pool may form in the lower drywell elevations.
- Vent entrance: The inlet area of the vent where significant interactions with the open area and/or floor components may take place.

Boundary conditions:

Based on discussions related to the opinion that break flow was adequately characterized by the proposed methodology, and that the development of debris generation models was already well focused, it was determined these sources were best characterized as boundary conditions to the PIRT work. Therefore, there was no need to define components for these regions. That is, the PIRT process did not give consideration to primary coolant break flow, determination of a representative debris "size distribution" or other characteristics of the debris source. Debris characterization was an assumed input boundary condition. However, the influence of break flow and debris characterization on the PIRT components (i.e., interactions) was considered. It should be noted that the panel did not necessarily agree that debris generation and characterization is a

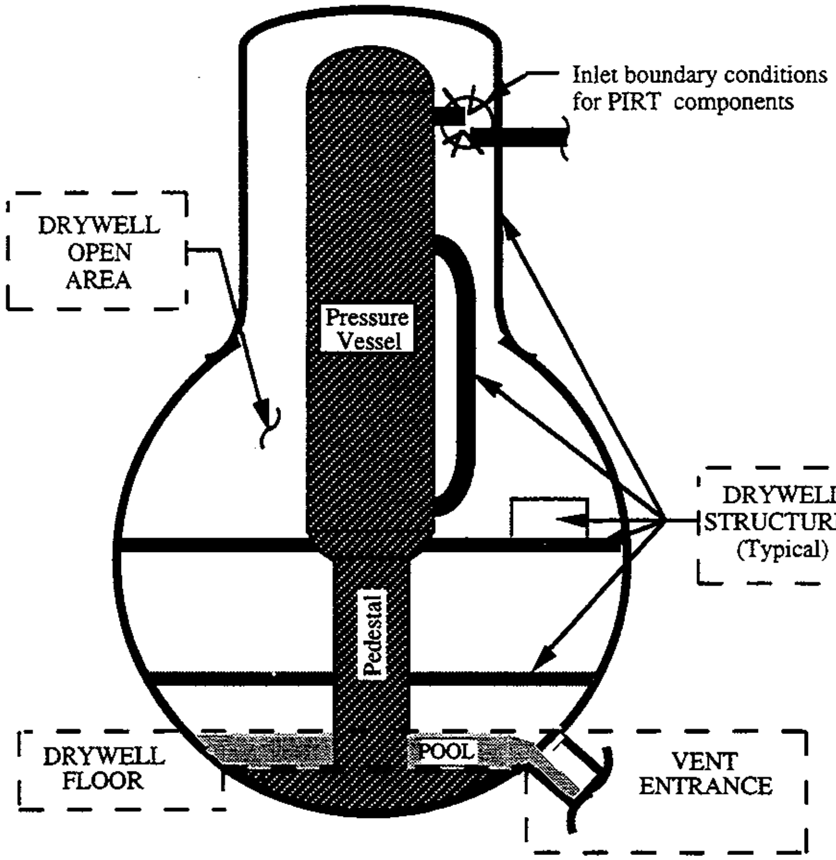

Figure 2. Component partitioning of drywell.

closed issue from the perspective of the overall research. However, in the absence of a definitive characterization of the debris source for particular accident scenarios, it was agreed that the PIRT development could better proceed by considering these elements as boundary conditions. Similarly, existing information related to suppression pool behavior, formed vent exit boundary conditions for the PIRT development.

2.6 Partitioning of Scenario into Time Phases - Again the prior work[2] provided a clear resolution to this objective:

- Blowdown: From break initiation through that point where the initial, dynamic, high energy nature of the break flow has decayed to essentially constant conditions (≈ zero to 100 s for LLOCA). The specific time at which this phase terminates increases with decreasing break size.

- Post-blowdown: End of blowdown to that point in time when debris transport has become essentially insignificant (≈ 30 min for LLOCA). The boundary between blowdown and this phase included the specification that debris washdown by containment sprays, if present, was contained in the post-blowdown phase.

2.7 System Level Processes - As a first step in the PIRT process, phenomena believed to have some significance to the plant behavior were identified by the previously defined component and time phase partitioning. Early in the process it was determined that major system level interactions were important to identification of the plausible phenomena, and were even more important in the subsequent ranking effort. Therefore, the following five high level system processes were adopted to aid in the effort:

① Gas/vapor transport - Flow of noncondensibles and steam through free stream paths and around structures.
② Suspended water transport - Flow of liquid through free stream paths and around structures
③ Water depletion/accumulation/surface transport - Capture, storage, and flow of liquid on the surface of drywell internal structures.
④ Debris transport - Flow of debris through free stream paths and around structures, including transport via gas/vapor media, liquid films, pool surfaces and within pools.
⑤ Debris depletion - Capture and storage of debris by structures and liquid pools, including growth or fragmentation of the debris.

Features of these processes are pictorially illustrated in Figures A1-A9 in Appendix A.

It may be noted that these processes were used in their broadest sense solely as an aid in organizing the phenomena into tractable groups for further consideration in the ranking of relative importance. In this sense, relating a particular phenomenon to a system level process helps to define the context in which the importance of the phenomenon was judged.

2.8 Description of Phenomena Ranking Scale - It was agreed that the use of the labor intensive AHP ranking methodology was not within the scheduler constraints of the PIRT effort. Accordingly, it was decided that the low, medium and high rank scheme should be adopted, where, from prior PIRT applications the following general interpretations serve as guidelines:

Code development and assessment:

- Low = Phenomena has small effect on the primary parameter of interest. Phenomena should be represented in the code, but almost any model will be sufficient,
- Medium = Phenomena has moderate influence on the primary parameter of interest. Phenomena should be well modeled; accuracy maybe somewhat compromised,
- High = Phenomena has dominant impact on the primary parameter of interest. Phenomena should be explicitly and accurately modeled.

And for code uncertainty quantification:

- Low = Combined uncertainty of phenomena maybe determined in a bounding fashion, or may be eliminated when justified,
- Medium = Phenomena should be evaluated to determine if uncertainty should be treated individually as are high ranks, or in a combined manner as are low ranks,
- High = Phenomena uncertainty should be individually determined and then combined statistically with other uncertainty sources (root mean square, Monte Carlo sampling, etc.).

During the actual ranking the panel found it helpful to differentiate between the lowest of the low, and highest of the high ranks. Therefore, a numerical ranking scheme of 1 to 5 was adopted with the following meaning:

1 = Lowest importance
2 = Low importance
3 = Moderate importance
4 = High importance
5 = Highest importance

Because these numerical ranks better reflect the panel's ranking conclusions they have been maintained in Tables 1 and 2.

2.9 PIRTs

2.9.1 Blowdown - The PIRT for this phase is provided in Table 1. The structure of the table is:

- Column 1 - Component in which phenomenon occurs. The components are described in Section 2.5 and Figure 2.
- Column 2 - General phenomenon type.

- Column 3 - Higher level system process with which the phenomenon is associated. These processes are described in Section 2.7.
- Column 4 - Phenomena being ranked.
- Column 5 - Cross reference number for phenomenon description given in Table A1 in Appendix A. Additional pictorial descriptions are provided in Figures A1-A6 as cross referenced in Table A1.
- Column 6 - Phenomenon relative importance rank. The ranking scheme is described at the end of Section 2.8.
- Column 7 - Cross reference number for ranking rationale given in Table B1 in Appendix B.

2.9.2 Post-Blowdown - The PIRT for this phase is provided in Table 2. The structure of this table is similar to Table 1, except the phenomena descriptions are provided in Table A2 and Figures A7-A9 in Appendix A, and the ranking rationales are given in Table B2 in Appendix B.

2.9.3 Ranking Summary - The highly important phenomena from Tables 1 and 2 have also been summarized in the table in the Executive Summary at the beginning of this report.

Table 1. BWR debris transport blowdown phase PIRT (1 of 2)

Component	Phenomenon type	System level process	Phenomenon	Description ①	Rank ②	Ranking rational③
Drywell open areas	Thermal hydraulic related	Gas/vapor transport	Pressure driven flows (bulk flows)	1	5	1
			Mixing (noncondensibles)	2	1	2
			Localized flow field	3	5	3
			Turbulence	4	2	4
		Suspended water transport (incl. gravitational settling)	Flashing of break liquid effluent	5	1 (5)④	5
			Droplet interactions	6	1	6
			Condensation (droplet formation)	7	1	7
		Water surface transport depletion/accumulation/ (implied surface orientation)	Condensation (structural)	8	1	8
			Film Dynamics	9	1	8
	Debris related	Debris transport	Advection/slip	10	5	9
			Agglomeration	11	3	10
			Debris/Flow field coupling	12	1	11
			Debris fragmentation	13	5	12
		Debris depletion	Gravitational settling	14	5	13
			Condensation on particles	15	3	14
			Stephan flow (diffuseophoresis)	16	1	15
			Thermophoresis	17	1	15
Drywell structures	Thermal hydraulic related	Gas/vapor transport	Heat transfer	18	3	16
		Water surface transport depletion/accumulation/ (implied surface orientation)	Porosity	19	5	3
			Film shear	20	1	17
			Surface wetting (condensation, impact)	21	3	18
			Film draining under gravity	22	3	19
			Recirculation (streaming) deluge	23	5	20
	Debris related	Debris transport	Resuspension	24	3	21
			Agglomeration	25	2	22
			Recirculation deluge (streaming) related transport	26	5	20
			Film related transport	27	3	19
			Runoff/reentrainment	28	4	23
		Debris depletion	Entrapment/impaction	29	5	24
			Adhesion	30	5	24

Table 1. BWR debris transport blowdown phase PIRT (2 of 2)

Component	Phenomenon type	System level process	Phenomenon	Description	Rank	Ranking rational
Drywell floor	Thermal hydraulic related	Water surface transport depletion/accumulation/ (implied surface orientation)	Pool formation	31	1 (4)	25
			Pool overflow (timing issue this phase)	32	1 (4)	25
			Heat transfer to structure	33	2	26
			Surface wetting (before pool formation)	34	4	27
			Pool flow dynamics	35	1 (4)	25
	Debris related	Debris transport	Resuspension	36	2	28
			Pool transport (to/through vent)	37	1 (4)	25
		Debris depletion	Agglomeration in pool	38	1 (3)	25
			Adhesion	39	4	27
			Settling	40	1 (4)	25
			Impaction	41	4	27
Vent entrance⑤	Thermal hydraulic related	Gas/vapor transport	Pressure driven flow (bulk) (vapor/gas)⑥	42	2	29
			Localized vapor flow field	43	2	29
		Water surface transport depletion/accumulation/ (implied surface orientation)	Localized liquid flow field	44	2	29
	Debris related	Debris transport	Advected mass	45	2	29
		Debris depletion	De-entrainment	46	2	29
			Re-entrainment	47	2	29

Notes

①: See Appendix A for phenomena descriptions.

②: 1 = Low importance to debris transport through the drywell to the wetwell; 3 = Moderate importance; 5 = High importance

③: See Appendix B for ranking rationales.

④: First entry (without brackets) applies to main steam line break; Value in () applies to the recirculation line break; Typical of all entries so indicated.

⑤: For this transient and this phase, the panel concluded that this component was of importance rank 2.

⑥: Includes effect of baffles

Table 2. BWR debris transport post-blowdown phase PIRT (1 of 2)

Component	Phenomenon type	System level process	Phenomenon	Description ①	Rank ②	Ranking rational③
Drywell open areas	Thermal hydraulic related	Gas/vapor transport	Pressure reduction due to condensation	48	≤ 2	30
			Mixed convection flow	49		
			Natural circulation	50		
			Thermal stratification	51		
			Steam-air distribution (species separation)	52		
			Mixing	53		
			Plume	54		
			Diffusion	55		
			Spray-induced turbulence	56		
		Suspended water transport (incl. gravitational settling)	Liquid transport	57		
			Flow regime	58		
			Spray source	59		
	Debris related	Debris transport	Debris advection/slip	60		
			Stephan flow (diffuseophoresis)	61		
			Thermophoresis	62		
			Debris T-H interactions (both directions)	63		
		Debris depletion	Gravitational settling	64		
			Removal by airborne liquid	65		
			Condensation on particles (growth, change in characteristic)	66		
			Agglomeration (growth, change in characteristics)	67		
Drywell structures	Thermal hydraulic related	Gas/vapor transport	Condensation	68	2 (5)④	31
		Water surface transport/depletion/accumulation/ (implied surface orientation)	Film draining under gravity	69	2 (5)④	31
			Film shear	70	2	32
			Recirculation deluge	71	2	33
	Debris related	Debris transport	Entrapment/Impaction	72	I⑤	N/A
			Resuspension into flow stream	73	2	34
			Film related transport	74	2 (5)④	31
			Recirculation deluge related transport	75	2	33
		Debris depletion	Agglomeration	76	2	22
			Mechanical entrapment	77	3	35
			Runoff/reentrainment	78	2 (5)④	31

Table 2. BWR debris transport post-blowdown phase PIRT (2 of 2)

Component	Phenomenon type	System level process	Phenomenon		Rank	Ranking rational
Drywell floor	Thermal hydraulic related	Gas/vapor transport	Heat transfer to structure	79	3	36
		Water depletion/accumulation/surface/transport (implied surface orientation)	Pool overflow	80	5	37
			Asymmetric effects	81	1	38
			Pool formation	82	5	39
			Pool flow dynamics	83	2 (5)④	40
	Debris related	Debris transport	Pool transport (to/toward vent)	84	2 (5)④	40
		Debris depletion	Settling	85	5	41
			Impaction	86	I	N/A
			Resuspension into drywell open area	87	I	N/A
			Agglomeration	88	1 (3)⑥	22
			Debris fragmentation	89	1 (5)⑥	42
Vent entrance	Thermal hydraulic related	Gas/vapor transport	Press-driven flow (bulk)	90	1	43
			Localized vapor flow field	91	I	N/A
		Suspended water transport (incl. gravitational settling)	Advected liquid mass	92	1	44
		Water depletion/ accumulation/surface transport (implied surface orientation)	Localized liquid flow field	93	I (5)④	39
	Debris related	Debris transport	Advected mass	94	I (5)④	45
		Debris depletion	De-entrainment	95	I	N/A
			Entrainment	96	I	N/A

Notes

①: See Appendix A for phenomena descriptions.

②: 1 = Low importance to debris transport through the drywell to the wetwell; 3 = Moderate importance; 5 = High importance

③: See Appendix B for ranking rationales.

④: First entry (without brackets) assumes there is no overflow of the drywell pool. Ranking in () if drywell overflows.

⑤: Indicated phenomena has insignificant effect on debris transport through the drywell to the wetwell (typical of all entries so indicated and listed solely to show phenomena was considered, but rejected as having any significance).

⑥: First entry (without brackets) applies to main steam line break; Value in () applies to the recirculation line break; Typical of all entries so indicated.

3. Panel Insights Regarding BWR Debris Transport

3.1 PIRT Related Insights

- The information provided in Table 3 was developed by the NRC to aid in the panel deliberations.

Table 3. Goals and associated success criteria for NRC/NRR's development and application of an analysis framework for evaluating debris transport.

Goal	Approach	Product	Success Criteria
Identify & rank important phenomena	PIRT	PIRT report	a) Consensus ranking of most important phenomena b) Whether identified phenomena can be incorporated into calculational tools identified
Develop calculational methodology	Evaluate applicable calculational tools & transport models (i.e., MELCOR/CFD), test against available information or related use. Use PIRT panel experts & other analysts if needed	Calculational methods & models to estimate fraction of debris transported to wetwell	Calculational methodology which accounts for important phenomena & judged by experts to have a basis for acceptance & application
Apply calculational methodology	Perform MARK I, II & III reference plant calculations	Containment specific calculations which estimate fraction of debris which might be transported to the wetwell	Provide insights into important plant modeling requirements & calculational method(s) influences
Estimate calculational sensitivities & uncertainties	Perform sensitivity & uncertainty analyses for reference plant conditions	Parametric trends & uncertainty estimates for debris transport fractions	a) Confirm ranking of controlling phenomena & use of selected codes/models b) Ability to perform evaluations without need to commit extensive resources
Respond to NRR USER needs to understand key phenomena & their significance in review of licensee submittals	Use insights gained from above to judge licensee estimates of debris transport in the drywell	Technical findings report with condensed guidelines	Clarity & ease of applying insights from methodology & sample plant calculations

- An important part of the PIRT process is scenario selection. Given the combinations of break locations and containment types, a large number of scenarios are possible. For the most part, the panel focused on a single break type (high elevation main steam line break), a single containment type (Mark I), and no containment spray. The panel believes extensions to the broader range of potential scenarios should be recognized in the planned methodology development.

- Given the limited experimental and analytical information available, the panel welcomed and used the results from the partial scoping cases prepared by panel members. These were a FLOW3D calculation[7] prepared by K. Williams and a MELCOR calculation[8] prepared by M. Leonard. It was considered entirely appropriate to use this information to enrich the panels understanding of potential phenomena importance. However, it is recognized the resulting PIRTs may be somewhat biased toward the MSLB transient. In addition, the desired simplicity in the scoping analyses may have reduced the desired generic nature of the PIRTs. That is, there is some risk the panel produced a PIRT more reflective of the high-elevation main steam line break (MSLB) and the Mark I containment, than of other transients. Thus, the panel strongly believes the basic sensitivity studies proposed in Section 3.2 should also be used to help confirm (or refute) the phenomena ranks given in Tables 1 and 2.

A primary outcome of the PIRT process is the apparent general condition that the blowdown phase tends to dominate the amount of debris transported to the wetwell as reflected in Table 1. In a similar manner, the drywell structures tend to be somewhat the more important component with respect to most of the phenomena related to debris transport. These two findings should be given due consideration in the analytical resolution of the debris transport issue in the drywell.

3.2 Methodology Related Insights

- The PIRT panel believes that the overall BWR LOCA-debris drywell transport problem is inherently very complex. Consistent with the PIRT methodology, the panel has successfully "decomposed" the problem into a finite number of processes (5) and components (4). We believe that during our two meetings the panel made significant progress in the identification of a limited number of high-importance, or controlling phenomena. Analytical methods for characterizing drywell transport are only now evolving. The panel is not aware of any existing analytical tool (computer code) available to handle these phenomena in a fully integrated manner. Even state-of-the-art separate effects computational fluid dynamics (CFD) codes are lacking in some key aspects, e.g., inertial impaction of debris. Therefore, some significant analytical model development is necessary and appropriate. The resulting methodology will need to be exercised for sensitivity and parametric calculations to develop a basis for uncertainty quantification. Additionally, it appears likely that these analytical modeling efforts would benefit greatly from some new experimental data at a "separate effects" level.

The panel concluded that conducting some well focused "scoping analyses" were very helpful in its consideration of this difficult problem. Therefore, we advise that the following limited-scope, "gedanken models" be undertaken. These results would help increase the understanding of the relative importance of phenomena, i.e., ranking. The results should form the basis for a scrutinizable, overall understanding of the debris-transport problem and its underlying and controlling phenomena.

Scoping Analysis

Scoping analyses should be undertaken for the overall drywell debris-transport problem from "beginning to end." That is, to consider debris transport phenomena from LOCA initiation through the post-blowdown time period. Furthermore, the analysis should focus on the above five system-level processes. At a minimum it should consider at least two fundamental aspects: vapor- and liquid-transported debris. Finally, the analysis would be most useful if completed on a component-level basis. To help define our vision of the level of effort for this endeavor, it should take approximately two to three weeks of engineering effort.

As an example of this scoping analysis process we note the panel's performance of similar efforts on the initial vapor-phase transport of debris. This first used a fine-mesh CFD simulation complemented with a scoping type MELCOR study. These results were then supplemented with an "integral" consideration wherein the vapor volumetric "source" within the drywell was considered in terms of "turnovers" resulting from the isentropic blowdown of the vessel's liquid inventory. These integral and differential approaches were used to cross-check one another. The analysis approaches selected should emphasize usage of fundamental conservation laws and physical principles. If desired, additional examples of this "gedanken" process for the drywell problem could be generated.

Specifically, we recommend the scoping studies consider at least the following issues:

① Early time vapor transport to characterize the maximum size of debris particle that can be advected with the flow, and

② Particle inertial impaction upon structural elements versus gravitational settling, and

③ Liquid accumulation onto the drywell floor to characterize the potential for spill-over into the vents.

Preliminary information indicated that a MSLB event would not flood up to the vents. Consideration of these three debris transport processes would likely give a lower bound to the fraction of debris transported into the vents.

This "analytical decomposition" of the overall problem should produce an excellent framework for prioritization of the necessary methodology development and application activities. Finally, these scoping analyses should help focus the efforts necessary to better quantify uncertainty of the overall analytical methodology.

In summary, the suggested analysis should be done quickly to help focus a cost-effective model development and application, and to produce results that can easily be understood by an independent audience that is engineering-educated, but not familiar with the details of the drywell debris transport problem.

- The panel believes the value of CFD tools for understanding the "pressure driven flows (bulk flows)" has been demonstrated. In addition, the proposed utilization of the CFD results to support noding decisions for other analysis codes, such as MELCOR, is considered appropriate. However, the operative words are "understanding" and "support", much in the same sense as the scoping studies are suggested above. That is, the CFD analyses should provide guidance for other analysis tools.

- The panel has expressed concerns regarding the application of MELCOR models in which a containment would be divided into an arbitrary number of 1-dimensional (lumped parameter) nodes (volumes) to evaluate debris transport and deposition. Under conditions in which local, or even large-scale variations and asymmetries in the drywell flow field are anticipated, the ability to justify the selection of cell-to-cell flow areas and associated flow resistances in a lumped parameter model is a crucial element of demonstrating the credibility of resulting calculations. It is recommended that the NRC (or its contractors), model mass, momentum and energy transport between adjacent volumes in a manner that can be demonstrated to be consistent with independent flow field evaluations such as those generated with a CFD model.

- The PIRTs identify several highly-ranked (level 5) phenomena/processes that are not currently modeled in MELCOR. These include the following.

Blowdown phase:

- Drywell Open areas: (1) localized flow field, and (2) debris fragmentation
- Drywell structures: (1) entrapment/impaction, and (2) adhesion

Post-blowdown phase:

- Drywell floor: debris settling in a deep pool of water

There appears to be a reasonable assurance that models can be provided for these processes. Care should be taken to ensure that these models are appropriately characterized where possible, using simple problems and standard criteria based on dimensionless groups, to display the characteristics of the models. The scoping studies discussed above should be used to the maximum extent possible to guide the model development.

- Interpretation of scaled experimental results, code model improvements and code validation (assessment/benchmarking) are all strongly related to developing and using a sufficient scaling rationale(s). The panel believes the scaling issue must be well addressed and documented in the methodology development.

- The panel recognizes that the limited experimental data that is available places constraints on the degree of model validation that can be accomplished. However, the panel believes model/code validation must be addressed and documented to the maximum extent possible in the methodology development. This is particularly true in the case where existing MELCOR aerosol models form the basis for debris transport characterization.

- The panel recognizes the high value of additional "benchtop experiments" to supplement existing data. Such new data can be particularly helpful in the development and validation of analytical models.

3.3 Other Insights

- The panel was encouraged by Eric Haskin's description of the sensitivity analysis approach[9] proposed for the drywell debris transport study, particularly as a basis for evaluating uncertainty in the predictive tools. The one-at-a-time sensitivities approach is considered a reasonable second step beyond the simple problems for new models proposed in Section 3.2. However, the panel's experience suggests that use of uncertainty statements of the nature of "it is considered highly unlikely that • • • •", rather than more probabilistic, and/or bounding type statements, will receive significant comment. The panel believes it will be necessary to construct a defense for the "highly unlikely" type arguments and suggests that defense may be founded in the research objective to provide insights useful to evaluation of licensee submittals. It remains the licensee's obligation to prove the safety of the proposed strainer designs.

3.4 Summary of Insights - The insights described in this section have been summarized, in their general order of importance, in the Executive Summary at the beginning of this report.

4. References

[1] USNRC, *Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident*, Regulatory Guide 1.82, Revision 2 (May 1996).

[2] G. Zigler et al., *Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA Generated Debris*, NUREG/CR-6224, Science And Engineering Associates, Inc. (October 1995).

[3] *SUMMARY MINUTES: First BWR Debris Transport PIRT Meeting*, Transmitted by letter GEW-12-96 (May 20, 1996).

[4] TPG (Technical Program Group), *Quantifying Reactor Safety Margins: Application of CSAU to a LBLOCA*, NUREG/CR-5249, EG&G Idaho, Inc. (1989)

[5] TPG (Technical Program Group), *Quantifying Reactor Safety Margins: Application of CSAU to a LBLOCA*, Nuclear Engineering and Design 119 (1990):
B. E. Boyack et al., Part 1: An overview of the CSAU Evaluation Methodology,
G. E. Wilson et al., Part 2: Characterization of Important Contributors to Uncertainty,
W. Wulff et al., Part 3: Assessment and Ranging of Parameters,
G. S. Lellouche et al., PART 4: Uncertainty Evaluation of LBLOCA Analysis Based on TRAC-PF1/MOD1,
N. Zuber et al., Part 5: Evaluation of Scale-Up Capabilities of Best Estimate Codes,
I. Catton et al., Part 6: A Physically Based Method of Estimating PWR LBLOCA PCT.

[6] R. A. Shaw, T. K. Larson and R. K. Dimenna, *Development of a Phenomena Identification and Ranking Table (PIRT) for Thermal-Hydraulic Phenomena During a PWR LBLOCA*, NUREG/CR-5074, EG&G Idaho, Inc. (1988).

[7] Ken A. Williams, *CFD Simulation of BWR Drywell Response to MSLB Event*, Flow Simulation Services, Inc. (May 15, 1996).

[8] Mark Leonard, *Insights from Some Quick MELCOR Calculations*, Innovative Technology Solutions Corp. (May 1996).

[9] Eric Haskin and Francisco Souto, *Sensitivity Analysis Approach Proposed for Debris Drywell Transport Study*, University of New Mexico and Science And Engineering Associates, Inc. (May 1996).

Appendix A

Phenomena descriptions for BWR debris transport PIRTs

Table A1. Phenomena descriptions for BWR debris transport blowdown phase PIRT (1 of 3)
(Reference number relates to entry in Table 1 in the report main body)

Reference Number	Phenomena	Phenomena Description	See Figure
1	Pressure driven flows (bulk flows)	Net (macro-scopic) flow characteristics of the drywell atmosphere.	A1
2	Mixing (noncondensibles)	Mixing (or stratification) of noncondensible gases in the drywell atmosphere (N2 or air) with the two-phase break effluent.	A1
3	Localized flow field	Flow direction and/or velocities that differ from the bulk (net) atmosphere flow characteristics due to localized geometries	A1
4	Turbulence	Local fluid vortexes or flow eddies created by flow around obstacles	A1
5	Flashing of break liquid effluent	Phase transformation (liquid ⇨ vapor) due to expansion across choked break plane.	A1
6	Droplet interactions	Mechanical interactions between suspended water droplets due to diffusion, settling, or any other process causing relative motion.	A1
7	Condensation (droplet formation)	Phase transformation (vapor ⇨ liquid) as steam cools during its motion through the drywell atmosphere creating nucleation-size water droplets	A1
8	Condensation (structural)	Heat and mass transfer from steam in the drywell atmosphere to surfaces of drywell structures associated with steam condensing on cooler structures.	A1
9	Film dynamics	The interaction between gas flow in the drywell atmosphere and liquid (condensate) films on structure surfaces; including interfacial shear, surface instability and droplet re-entrainment.	A1
10	Advection/slip	Transport of airborne debris within the carrier gas medium.	A2
11	Agglomeration	Mechanical interaction among suspended debris particles by which two or more small particles combine to form a larger conglomerate particle.	A2
12	Debris/Flow field coupling	The influence of local variations in fluid flow field on debris transport and deposition.	A1/A2
13	Debris fragmentation	Break up of relatively large pieces of debris into smaller particles that can be re-entrained into the flow stream due to fluid shear created (for example) by locally-high flow velocities at constricted flow areas.	A2
14	Gravitational settling	Downward relocation (sedimentation) of debris in the drywell atmosphere onto structure surfaces under the force of gravity.	A2
15	Condensation on particles	Heat and mass transfer from steam in the drywell atmosphere to surfaces of suspended debris particles with steam condensing onto particle surface.	A2

Table A1. Phenomena descriptions for BWR debris transport blowdown phase PIRT (2 of 3)

Reference Number	Phenomena	Phenomena Description	See Figure
16	Stephan flow (diffusiophoresis)	Transport of debris particles toward deposition surfaces due to concentration gradients of atmosphere contents (dominated by steam concentration gradients created by condensation on drywell structures).	A2
17	Thermophoresis	Transport of debris particles toward deposition surfaces due to temperature gradients within the atmosphere and between the atmosphere and bounding structures.	A2
18	Heat transfer	Cooling of drywell atmosphere due to heat transfer to structures	A3
19	Porosity	Variations in fluid flow area and flow resistance due to distributed structures within the drywell.	A3
20	Film shear	The interfacial interaction between gas flow in the drywell atmosphere and liquid (condensate) films on structure surfaces	A3
21	Surface wetting (condensation, impact)	Formation of a liquid film on structure surfaces due to condensation of steam from the atmosphere or impaction of water droplets onto structure surfaces.	A3
22	Film draining under gravity	Downward, free-surface flow of liquid (water) films on structure surfaces by gravity	A3
23	Recirculation (streaming) deluge	Large flow rate of liquid effluent from a low-elevation (e.g., recirculation line) break in the reactor coolant system onto drywell structures	A3
24	Resuspension	Re-entrainment of debris previously deposited on structure surfaces into the atmosphere flow stream due to local fluid/structure shear forces.	A4
25	Agglomeration	Mechanical interaction among debris particles on structure surfaces (i.e., within a liquid film) by which two or more small particles combine to form a larger conglomerate particle.	A4
26	Recirculation deluge (streaming) related transport	Relocation of debris from drywell structures due to interactions with the deluge of liquid from recirculation pipe breaks.	A4
27	Film related transport	Relocation of debris along structure surfaces due to flow of liquid films under the force of gravity.	A4
28	Runoff/re-entrainment	Re-suspension of debris on structure surfaces into the flow stream as liquid films drain off of structures.	A4
29	Entrapment/impaction	Capture of debris particles on structure surfaces due to either transport into confined spaces due to localized flow fields or inertial impaction.	A4

Table A1. Phenomena descriptions for BWR debris transport blowdown phase PIRT (3 of 3)

Reference Number	Phenomena	Phenomena Description	See Figure
30	Adhesion	Permanent retention of debris particles on a structure surface due to mechanical interactions with a rough surface or other forces	A4
31	Pool formation	Creation of a pool of water on the drywell floor sufficiently deep to allow overflow into wetwell transfer piping due to the accumulation of water from all sources higher in the drywell (e.g., film drainage, droplet settling)	A5
32	Pool overflow (timing issue this phase)	Transport of water from drywell floor into wetwell vent pipes.	A5
33	Heat transfer to structure	Heat transfer between water on drywell floor and bounding structures	A5
34	Surface wetting (before pool formation)	Wetting of drywell floor due to steam condensation or settling of suspended water droplets.	A5
35	Pool flow dynamics	2- or 3-dimensional flow patterns and velocities within the pool of water on the drywell floor; includes free-surface (vertical) velocity profile and turbulent mixing (circulation) flows.	A5
36	Resuspension	Re-entrainment of debris into the atmospheric flow stream from the drywell floor due to high shear forces at the surface of the floor	A6
37	Pool transport (to/through vent)	Relocation of debris in the pool of water on the drywell floor toward wetwell vent pipe entrances	A6
38	Agglomeration in pool	Mechanical interaction among debris particles in the pool of water on the floor by which two or more small particles combine to form a larger conglomerate particle.	A6
39	Adhesion	Permanent retention of debris particles on the drywell floor due to mechanical interactions with a rough surface or other forces	A6
40	Settling	Downward relocation (sedimentation) of debris within the pool of water on the drywell floor under the force of gravity.	A6
41	Impaction	Capture of debris on the surface of the drywell floor (or water pool) due to inertial deposition.	A6
42	Pressure driven flow (bulk) (vapor/gas)	Vent entrance component phenomenon similar to Ref. No. 1, this table	
43	Localized vapor flow field	Vent entrance component phenomenon similar to Ref. No. 3, this table	
44	Localized liquid flow field	Vent entrance component phenomenon similar to Ref. No. 3, this table	
45	Advected mass	Vent entrance component phenomenon similar to Ref. No. 10, this table	
46	De-entrainment	Capture of debris from the bulk flow stream	A5
47	Re-entrainment	Resuspension of debris in the bulk flow stream	A6

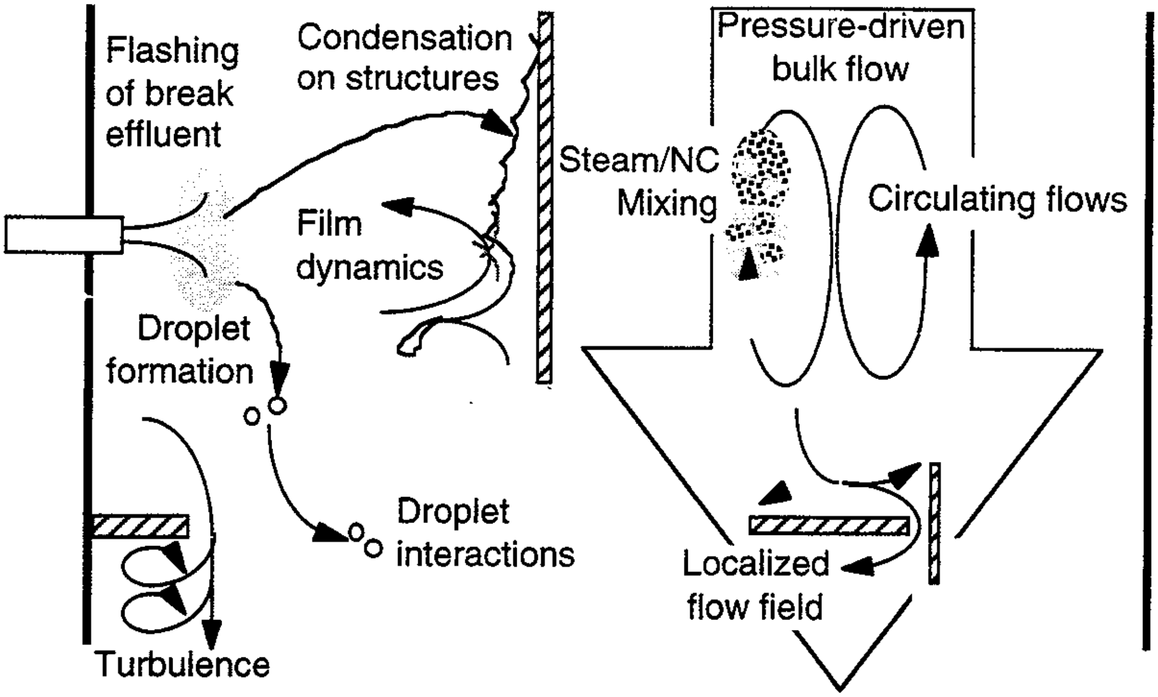

Figure A1. Thermal-hydraulic Pocesses in Drywell Open Areas
-- Blowdown Phase --

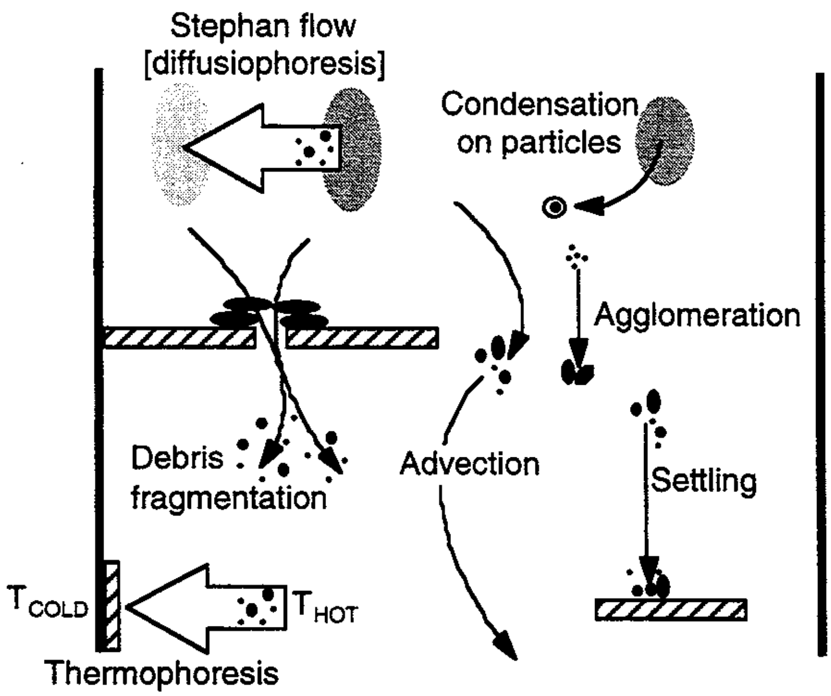

Figure A2. Transport / Deposition Processes for Debris in Drywell Open Areas
-- Blowdown Phase --

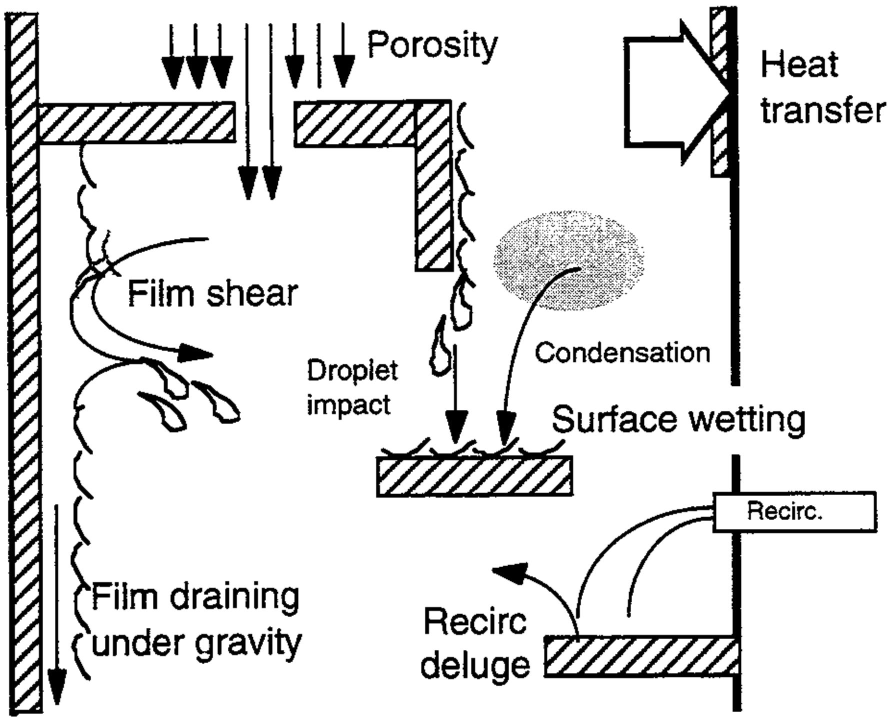

Figure A3. Thermal-hydraulic Processes on Drywell Structures
-- Blowdown Phase --

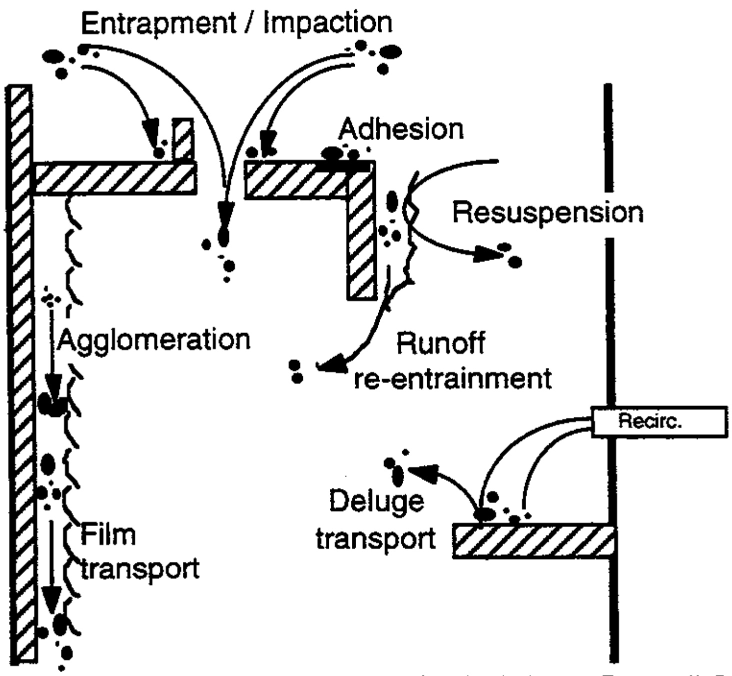

Figure A4. Transport / Deposition Processes for Debris on Drywell Structures
-- Blowdown Phase --

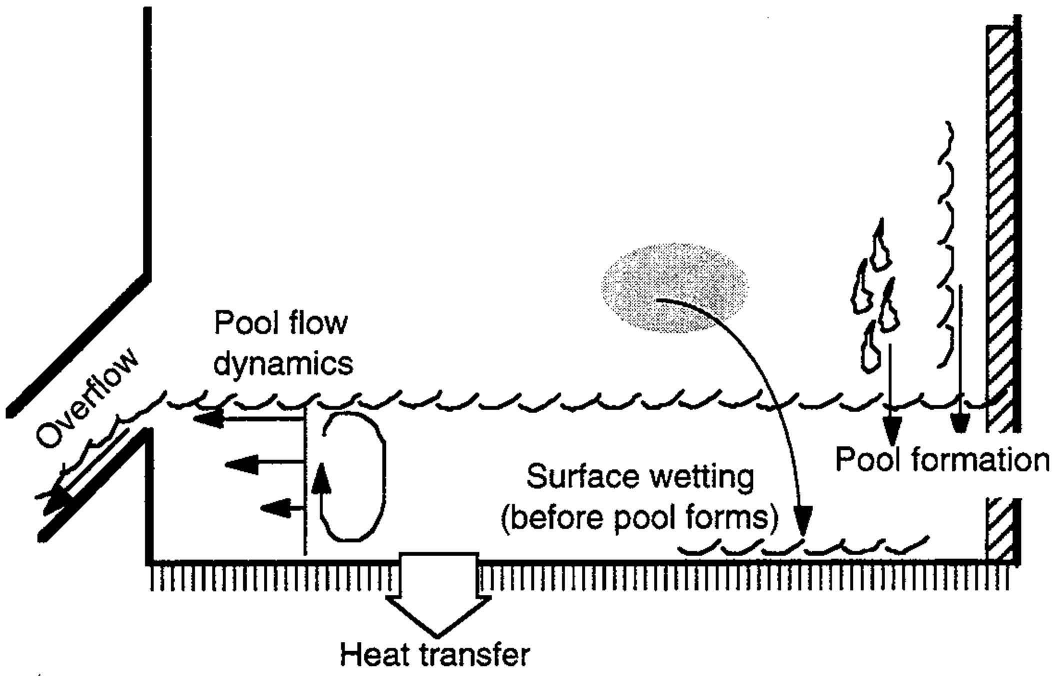

Figure A5. Thermal-hydraulic Processes on the Drywell floor
-- Blowdown Phase --

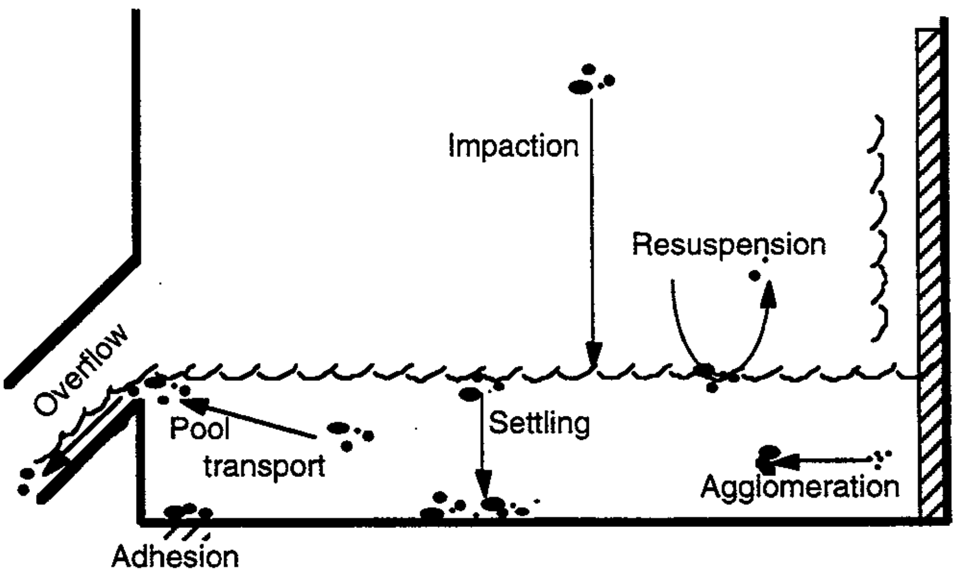

Figure A6. Transport / Deposition Processes for Debris on the Drywell floor
-- Blowdown Phase --

Table A2. Phenomena descriptions for BWR debris transport post-blowdown phase PIRT (1 of 3)
(Reference number relates to entry in Table 2 in the report main body)

Reference Number	Phenomena	Phenomena Description	See Figure
48	Pressure reduction due to condensation	Drywell pressure reduction due to reduction in vapor volume fraction	A7
49	Mixed convection flow	Mixed forced and gravity induced flow	A1
50	Natural circulation	Localized flow driven by buoyancy forces	A1
51	Thermal stratification	Formation of vertical temperature gradiant	
52	Steam-air distribution (species separation)	Flow patterns promoting species separation	
53	Mixing	Similar to Ref. No. 2, Table A1	
54	Plume	Centralized local flow pattern	
55	Diffusion	Migration due to localized phenomena	
56	Spray-induced turbulence	Local fluid vortexes or flow eddies created by spray interactions	A1
57	Liquid transport	Movement of liquid by forced and gravity induced forces	
58	Flow regime	Flow field with standerized attributes	
59	Spray source	Effects of spray source configuration and attributes	
60	Debris advection/slip	Similar to Ref. No. 10, Table A1	
61	Stephan flow (diffuseophoresis)	Similar to Ref. No. 16, Table A1	
62	Thermophoresis	Similar to Ref. No. 17, Table A1	
63	Debris T-H interactions (both directions)	Similar to Ref. No. 12, Table A1	
64	Gravitational settling	Similar to Ref. No. 14, Table A1	
65	Removal by airborne liquid	Capture by airborne liquid	
66	Condensation on particles (growth, change in characteristic)	Similar to Ref. No. 15, Table A1	
67	Agglomeration (growth, change in characteristics)	Similar to Ref. No. 11, Table A1	
68	Condensation	Phase transformation (vapor ⇨ liquid) as steam cools during its motion through the drywell atmosphere (on structures and suspended particles)	A7
69	Film draining under gravity	Downward, free-surface flow of liquid (water) films on structure surfaces by gravity	A7
70	Film shear	The interfacial interaction between gas flow in the drywell atmosphere and liquid (condensate) films on structure surfaces	A7
71	Recirculation deluge	Large flow rate of liquid effluent from a low-elevation (i.e., recirculation line) break in the reactor coolant system onto drywell structures.	A7

Table A2. Phenomena descriptions for BWR debris transport post-blowdown phase PIRT (2 of 3)

Reference Number	Phenomena	Phenomena Description	See Figure
72	Entrapment/Impaction	Capture of debris particles on structure surfaces due to either transport into confined spaces due to localized flow fields or inertial impaction.	N/A
73	Resuspension into flow stream	Re-entrainment of debris previously deposited on structure surfaces into the atmosphere flow stream due to local fluid/structure shear forces	A7
74	Film related transport	Relocation of debris along structure surfaces due to flow of liquid films under the force of gravity.	A7
75	Recirculation deluge related transport	Relocation of debris from drywell structures due to interactions with the deluge of liquid from recirculation pipe breaks.	A7
76	Agglomeration	Mechanical interaction among debris particles on structure surfaces (i.e., within a liquid film) by which two or more small particles combine to form a larger conglomerate particle.	A7
77	Mechanical entrapment	Capture of debris in local structural 'pooling points' -- i.e., locations that allow the accumulation and storage of draining condensate and associated transported debris.	A7
78	Runoff/reentrainment	Re-suspension of debris on structure surfaces into the atmosphere flow stream as liquid films drain off of structures	A7
79	Heat transfer to structure	Heat transfer between water on the drywell floor and bounding structures	A8
80	Pool overflow	Transport of water from drywell floor into wetwell vent pipes	A8
81	Asymmetric effects	Preferential transport of water towards a sub-set of the total number of wetwell vent pipes due (for example) to azimuthmal asymetries in water source locations (e.g., recirculation line break location)	A8
82	Pool formation	Creation of a pool of water on the drywell floor sufficiently deep to allow overflow into wetwell transfer piping due to accumulation of water from all sources higher in the drywell (e.g., film drainage, droplet settling)	A8
83	Pool flow dynamics	2- or 3-dimensional flow patterns and velocities within the pool of water on the drywell floor; includes free-surface (vertical) velocity profile and turbulent mixing (circulation) flows	A8
84	Pool transport (to/toward vent)	Relocation of debris in the pool of water on the drywell floor toward wetwell vent pipe entrances.	A9
85	Settling	Downward relocation (sedimentation) of debris within the pool of water on the drywell floor under the force of gravity	A9

Table A2. Phenomena descriptions for BWR debris transport post-blowdown phase PIRT (3 of 3)

Reference Number	Phenomena	Phenomena Description	See Figure
86	Impaction	Capture of debris on the surface of the drywell floor pool due to inertial impaction	N/A
87	Resuspension into drywell open area	Re-entrainment of debris into the atmospheric flow stream from the drywell floor due to high shear forces at the surface of the floor	N/A
88	Agglomeration	Mechanical interaction among debris particles on the drywell floor by which two or more small particles combine to form a larger conglomerate particle.	A9
89	Debris fragmentation	Break up or relatively large pieces of debris on the drywell floor (pool surface) into smaller particles due to inertial impact of liquid break effluent from a recirculation line break.	A9
90	Press-driven flow (bulk)	Similar to Ref. No's. 1 & 42, Table A1	
91	Localized vapor flow field	Similar to Ref. No's. 3 & 43, Table A1	
92	Advected liquid mass	Similar to Ref. No's. 10 & 45, Table A1	
93	Localized liquid flow field	Similar to Ref. No's. 3 & 44, Table A1	
94	Advected mass	Similar to Ref. No's. 10 & 45, Table A1	
95	De-entrainment	Similar to Ref. No. 46, Table A1	
96	Entrainment	Similar to Ref. No. 47, Table A1	

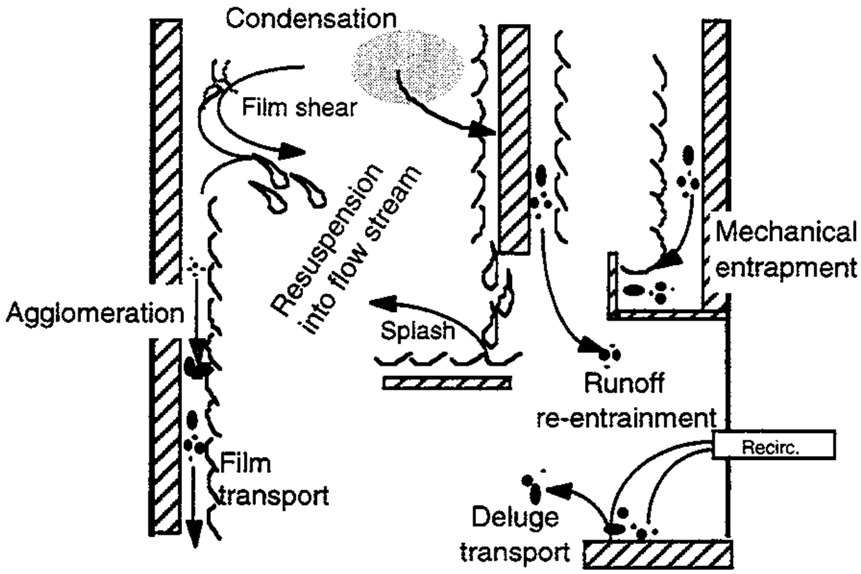

Figure A7. Thermal-hydraulic & Debris Transport / Deposition Processes on Drywell Structures
-- Post-blowdown Phase --

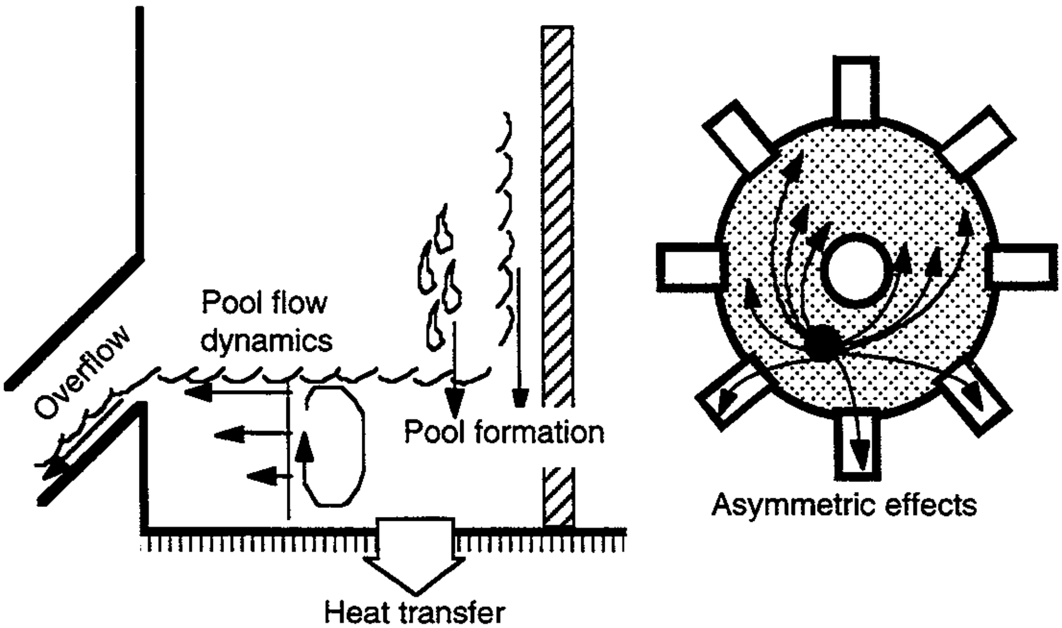

Figure A8. Thermal-hydraulic Processes on the Drywell floor
-- Post-blowdown Phase --

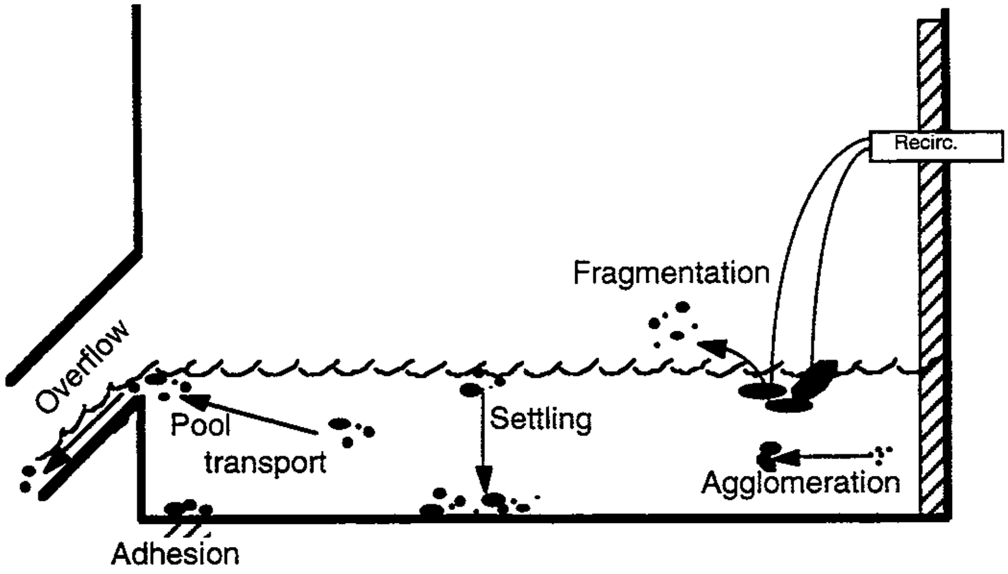

Figure A9. Transport / Deposition Processes for Debris on the Drywell floor
-- Post-blowdown Phase --

Appendix B

Ranking rationales for BWR debris transport PIRTs

Table B1. Ranking rationales for BWR debris transport blowdown phase PIRT (1 of 3)
(Reference number relates to entry in Table 1 in the report main body)

Reference Number	Phenomena	Ranking Rationale
1	Pressure driven flows (bulk flows)	The dominant transport medium for all debris that reaches the wet well. CFD calculations (Williams for 0.6 s of MSLB) show large fraction of drywell free volume is in unidirectional flow.
2	Mixing (noncondensibles)	Only significant influence is effect on steam condensation which during 60 s blowdown is relatively small. Jet thermal mixing is not controlling bulk flow.
3	Localized flow field	Most significant debris depletion mechanism (impaction).
4	Turbulence	An effect of channeling, thus of lower order importance.
5	Flashing of break liquid effluent	Low for MSLB; high for recirculation line break. Relates to amount of liquid exiting break.
6	Droplet interactions	Assumes w/o spray. Does not significantly affect debris transport because does not influence the bulk flow field.
7	Condensation (droplet formation)	No significant condensation in open area (excluding break boundary region). This is freestream condensation associated with the rapid depressurization. For the MSLB, it was assumed that the droplet formation occurred within the source cell of approximately 10 m^3.
8	Condensation (structural)	Of low importance to T/H behavior in open area because the volume of steam condensing is small compared to an open area volume.
9	Advection/slip	Dominant debris transport mechanism (correlates with pressure driven flows).
10	Agglomeration	(a) For individual (small) fibers, need to agglomerate large numbers of fibers to have a significant effect on growth. (b) for larger agglomerates (clumps), further agglomeration does not have a significant effect on depletion and sweepout of small fibers is a moderate effect at last. That is, there are two major contributions. The first is that several strands could agglomerate (10s of fibers) but not change the behavior of the agglomerated debris. The second is that big chunks could sweep out individual strands but only do so in a moderate way because the big pieces fall out so quickly.
11	Debris/Flow field coupling	Debris density is too low to result in significant influence.
12	Debris fragmentation	Significant source of small size particles with attendant increased transport through the drywell.
13	Gravitational settling	Dominant debris depletion mechanism for debris characterized by a high percentage of large size particles.

Table B1. Ranking rationales for BWR debris transport blowdown phase PIRT (2 of 3)

Reference Number	Phenomena	Ranking Rationale
14	Condensation on particles	Mechanism to increase the effective size of debris particles. Presumes wetted particles particle exit break boundary. Presumes a wet source, i.e., that the debris was wetted in the 10 m^3 source volume.
15	Stephan flow (diffuseophoresis) & Thermophoresis	Insignificant to debris transport in this phase.
16	Heat transfer	Moderate-order effect on debris transport because of influence on containment depressurization. Diminishes the peak pressure (driving potential) for the flow to the wetwell but the effect is moderate).
17	Film shear	Velocity not high enough to produce significant shear.
18	Surface wetting (condensation, impact)	Dominant effect to establish films that ultimately transport debris to drywell floor; also mechanism to promote debris capture on structures.
19	Film draining under gravity	Mechanism to transport debris captured in structural films.
20	Recirculation (streaming) deluge	One of the dominant mechanisms (with vapor/gas transport) of debris during the recirculation line break.
21	Resuspension	Lower order effect in context of likelihood, i.e., the debris impacted and adhered, the gas/vapor flows decrease with time, and thus there is only a moderate likelihood that debris will be resuspended.
22	Agglomeration	Lower order effect (i.e., debris growth vs direct removal).
23	Runoff/reentrainment	Important contributor to the airborne debris mass.
24	Entrapment/impaction & Adhesion	Dominant mechanism for removing high-speed airborne debris where departure from fluid streamlines is important.
25	Pool formation, Pool overflow (timing issue this phase), Pool flow dynamics, Pool transport (to/through vent), Agglomeration in pool & Settling	Importance is related to effect on next time phase (MSLB), e.g., the pool entrapment and transport effects during the post-blowdown phase.
26	Heat transfer to structure	Importance to surface wetting.
27	Surface wetting (before pool formation), Adhesion & Impaction	Moderate effect related to debris depletion from flow stream to vent. Moderate effect on retention of debris at the drywell floor.
28	Resuspension	Importance related to likelihood in context of low free stream velocity.

Table B1. Ranking rationales for BWR debris transport blowdown phase PIRT (3 of 3)

Reference Number	Phenomena	Ranking Rationale
29	Pressure driven flow (bulk) (vapor/gas), Localized vapor & liquid flow fields, Advected mass, De-entrainment & Re-entrainment	a) Mark I & II: Component is of overall less importance to debris transport than other components because essentially all debris arriving at component is transported to the drywell. This conclusion applies only to the Mark I and II containments. b) This conclusion may not apply to the Mark III

Table B2. Ranking rationales for BWR debris transport post-blowdown phase PIRT (1 of 1)
(Reference number relates to entry in Table 2 in the report main body)

Reference Number	Phenomena	Ranking Rationale
30	All referenced phenomena listed under the drywell open area component in Table 2 in the report main body	Assigned rank for this component is 2 because airborne debris during this phase is negligible. See drywell turnover figure showing about 7 turnovers during the blowdown phase. An important related factor of this component assignment is that no process/phenomena will have a rank higher than the component rank.
31	Condensation, Film draining under gravity, Film related transport & Runoff/ re-entrainment	For a Mark I there is not sufficient primary coolant inventory condensed to fill the drywell to the spillover point for the vents.
32	Film shear	Moderate contributor to resuspension.
33	Recirculation deluge & related debris transport	Relates to reduced likelihood that a significant amount of debris remains in this phase for washdown.
34	Re-suspension into flow stream	Only important in fast draining films which already transport debris.
35	Mechanical entrapment	Moderate as related to expected area/volume available to accumulate debris. Unknown characteristic of the system. The amount of entrapment depends upon what fraction of the drywell structures are oriented such that they would trap debris.
36	Heat transfer to structure	Relates to pool stratification and turbulence and attendant effect on debris transported to wetwell.
37	Pool overflow	Dominant mechanism for debris transport to wetwell.
38	Asymmetric effects	Low potential to significantly affect amount of debris transported to wetwell.
39	Pool formation & Localized liquid flow field	Important contributor to debris transport to wetwell.
40	Pool flow dynamics & Pool transport (to/toward vent)	Relates to fraction of debris available for overflow transport.
41	Settling	Major contributor to pool entrapment of debris.
42	Debris fragmentation	Recirculation spill over churns pool.
43	Press-driven flow (bulk)	Very little suspended debris in gas/vapor field; low vapor velocity.
44	Advected liquid mass	Only contribution is by runoff from structures.
45	Advected mass	Relates to potential of flow conditions to carry over debris to wetwell

Appendix C

Information base used in the application of the PIRT process
to debris transport in a BWR drywell

Documents developed external to the BWR debris transport PIRT project

1 W. W. Durgin and J. Noreika, *The Susceptibility of Fibrous Insulation Pillows to Debris Formation Under Exposure to Energetic Jet Flows*, NUREG/CR-3170, Alden Research Laboratory (March 1983).
2 Kevin W. Brinckman, *Results of Hydraulic Tests on ECCS Strainer Blockage and Material Transport in a BWR Suppression Pool*, EC-059-1006, Revision 0 (May 1994).
3 SEA, *A Methodology for Estimating BWR Drywell Transport Fractions During Blowdown and Washdown*, SEA N0. 93-554-06-A:12, Science And Engineering Associates, Inc. (July 1995).
5 Aleck W. Serkiz et al., *An Overview of the BWR ECCS Strainer Blockage Issues*, NUREG/CP-0149, Volume 3, Brookhaven National Laboratory (March 1996) pp175-199.
6 George E. Hecker et al., *Experiments of ECCS Strainer Blockage and Debris Settling in Suppression Pools*, NUREG/CP-0149, Volume 3, Brookhaven National Laboratory (March 1996) pp201-225.
7 Gilbert L. Zigler and D. V. Rao, *The Strainer Blockage Assessment Methodology Used in the BLOCKAGE Code*, NUREG/CP-0149, Volume 3, Brookhaven National Laboratory (March 1996) pp227-235.
8 D. V. Rao et al., *Proposed Methodology for Modeling LOCA Debris Transport in BWR Drywells*, Science And Engineering Associates, Inc. (February 5, 1996).
9 G. E. Wilson, *Statistically Based Uncertainty Analysis for Ranking of Component Importance in the Thermal Hydraulic Safety Analysis of the Advanced Neutron Source Reactor*, EGG-NE-10078, EG&G Idaho, Inc. (1992).
10 T. Saaty, *Decision-Making For Leaders*, Belmont, CA, Lifetime Learning Publications, Wadsworth Inc. (1982).

Documents associated with the first PIRT meeting

11 Gilbert L. Zigler, *NUREG/CR-6224 Overview*, Science and Engineering Associates (April 1996).
12 Clint Shaffer, *Overview of Proposed Analytical Methods for Addressing Debris Transport Problem*, Science and Engineering Associates (April 16, 1996).
13 George Hecker, *Why Use CFD For Drywell Transport?*, Alden Research Laboratory (April 1996).
14 D. V. Rao, *Phenomenological Considerations in Drywell Debris Transport*, Science and Engineering Associates (April 16-18, 1996).
15 Gary E. Wilson, *PIRT Process Considerations*, Idaho National Engineering Laboratory (April 1996).

Documents associated with the second PIRT meeting

16 D. V. Rao et al., *Drywell Debris Transport Methodology: Responses to PIRT Panel Request for Information*, SEA No. 96-3104-06-A:1, Science And Engineering Associates, Inc. (May 1996).
17 D. V. Rao, *Accident Progression Scenarios for BWR*, Science And Engineering Associates, Inc. (May 1996).
18 D. V. Rap et al., *SEA/ARL Proposed Methodology for Important Phenomena Identified by PIRT Panel*, Science And Engineering Associates, Inc. (May 1996).
19 Mark Leonard, *Basic Information on Non-Spherical Particle Transport Properties*, Innovative Technology Solutions Corp. (May 1996).
20 John E. Brockmann, *Aerosol Physics*, Sandia National Laboratories (May 1996).

21 Lothar Wolf and Mark Leonard, *Collection of Schematics Describing Important Physical Phenomena for Debris Transport in BWR Containment During and After LOCA*, University of Maryland and Innovative Technology Solutions Corp. (May 1996).
22 Lothar Wolf, *Description of Coupled Thermohydraulics and Aerosol Phenomena in LWR Containment*, University of Maryland (May 1996).
23 Lothar Wolf, *Suggestions for Dimensionless Presentations of Major Aerosol Transport Processes in LWR Containments*, University of Maryland (May 1996).
24 Lothar Wolf, *Overview of Experimental and Analytical Results of Containment LOCA and Aerosol Behaviors*, University of Maryland (May 1996).
26 Lothar Wolf, *Summary of Unpublished German Experiments on Insulation Damages and Floating Behavior*, University of Maryland (May 1996).
27 K. Mun and L. Wolf, *GOTHIC Computation and Comparisons with Data of Marviken (BWR) Test 17*, University of Maryland (May 1996).
28 K Mun and L. Wolf, *GOTHIC Computation of BWR Mark I LOCA with Spray Operation*, University of Maryland (May 1996).

7

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

July 25, 1996

Mr. Rocky Sgarro
Pennsylvania Power and Light
2 North Ninth Street
Mail Code A6-1
Allentown, PA 18101

SUBJECT: COMMENTS ON DRAFT UTILITY RESOLUTION GUIDANCE SECTIONS 3.1.4, 3.2.1.1, 3.2.2.2, AND 3.2.3.4

Dear Mr. Sgarro:

On April 1, 1996, the Boiling-Water Reactor Owners Group (BWROG) submitted four draft sections for their Utility Resolution Guidance (URG) document to the staff. The four sections submitted were: Section 3.1.4 entitled "Backflush," Section 3.2.1.1 entitled "Postulated Break Locations," Section 3.2.2.2 entitled "Suppression Pool Transport and Settling," and Section 3.2.3.4 entitled "ECCS Pump NPSH Calculations." All sections were marked draft and dated March 31, 1996. It was requested that the staff review the sections and provide the BWROG with any preliminary comments the staff might have. The staff does not normally review draft documents; however, due to the tight schedule that licensees are on for implementation of the requested actions in NRC Bulletin 96-03, the staff is providing comments on the draft sections to the BWROG so that these comments may be addressed by the BWROG prior to submitting the completed URG to the staff for review. It is hoped that this, in turn, will facilitate a faster review on the final document.

The staff has completed a preliminary review of these sections and our comments are attached. It should be noted that since we have not seen all of the sections in the document, that our comments should not be construed as an endorsement of these sections or a partial endorsement of the URG document. This is because many of the sections are interrelated, and we cannot perform a complete review without having the whole document with which to work. However, we have attempted to raise any staff concerns early so that the BWROG may work on resolving them prior to the submittal of the completed URG document. No response is required to this letter, however, the staff will carefully review the final URG and supporting documents to ensure that our concerns stated herein have been adequately addressed.

If you have any questions regarding this letter, please contact either Rob Elliott at (301) 415-1397 or Michael Marshall at (301) 415-5895.

Sincerely,

Carl Berlinger, Chief
Containment Systems and Severe Accident Branch
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation

Enclosure: As stated

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Enclosure

STAFF COMMENTS ON THE BWROG'S DRAFT URG SECTIONS 3.1.4, 3.2.1.1, 3.2.2.2 AND 3.2.3.4

Comments on URG Section 3.1.4, "Backflush"

1) Under "Backflush Requiring Shutdown of ECCS Pumps," Page 3, Line 15: It is not clear that backflushing requires ECCS pump shutdown to be viable. This is especially apparent when you discuss the possibility of providing backflush without pump shutdown on the very next page. Backflushing may not even require securing flow to the reactor vessel. For plants with individual pump suctions, it may be possible to backflush one strainer while injection flow continues through another pump. There appear to be many more ways to accomplish backflushing than has been addressed by the BWROG in this section.

2) Section 2.2.3 (f) of Regulatory Guide 1.82, Revision 2 (RG 1.82) states: "The performance characteristics of a passive or an active strainer for the debris types and amounts postulated should be supported by appropriate test data."

   Under "Design Considerations," page 7, Item 4: The BWROG recommends that test data be used in the event a backflush system is used as a primary success path. The staff notes that this practice would be appropriate whether using backflush as a primary success path or as a defense-in-depth measure. Several of the design considerations in this section would probably be good engineering practice regardless whether the backflush is for defense-in-depth or as a primary success path.

3) Under "Design Considerations," page 8, item 9: This item should be updated to be consistent with NRC Bulletin 96-03. The bulletin states that instrumentation and alarms which are relied upon for initiating backflush should be Type A, Reg Guide 1.97, Post-Accident Monitoring Equipment. This would require that these instruments be added to the technical specifications.

4) Under "Design Considerations," page 8, item 12: This item should be updated to be consistent with NRC Bulletin 96-03. The bulletin states that new LCO's are not needed; however, if a backflush system or strainer component is inoperable, the inoperability of that component or subsystem should be evaluated for its impact on the operability of the ECCS and the appropriate action statement entered.

5) Section 2.3 of RG 1.82 states: "If relying on operator actions to prevent the accumulation of debris on suction strainers or to mitigate the consequences of the accumulation of debris on the suction strainers, safety-related instrumentation that provides operators with an indication and audible warning of impending loss of NPSH for ECCS pumps should be available in the control room."

   Under "Design Considerations:" the URG states that in the event a backflush system is used as a primary success path, safety-grade equipment be used unless supporting technical analysis can justify an exemption from 10 CFR 50.46 requirements. The URG also states that instrumentation and alarms that indicate a need to backflush a strainer be identified. However, unlike RG 1.82, the URG does not specify that operator instrumentation that provides indication and audible warning of impending loss of NPSH for ECCS pumps be available in the control room.

6) General Comment - Backflush is meant to be used with appropriate measures to delay or prevent as much as possible the onset of strainer blockage (e.g., larger strainers, suppression pool cleaning, etc.). Just adding backflush by itself would probably not be considered adequate by the staff. The URG is lacking on guidance in this area.

Section 3.2.1.1, "Postulated Break Locations"

1) Under "Introduction," Page 1, Lines 9 through 12 and Lines 21 through 27: The focus by the BWROG position on the postulated pipe break locations that need to be analyzed by a licensee appears to be on break locations "which have the greatest potential for debris generation" based on pipe stress analysis methodology. 10CFR50.46 states in part, that ECCS cooling performance "must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated." There is no language in the rule that implies only the most probable break locations need to be evaluated. This section as currently written, however, clearly states that the BWROG believes that only the most probable breaks need to be analyzed. While this may be a good starting point from which to start an analysis, it does not by itself appear to be sufficient to meet the intent of the rule. The Standard Review Plan (SRP), Section 3.6.2 and Branch Technical Position (BTP) MEB 3-1 were developed for looking at dynamic and environmental effects of a LOCA (e.g., pipe whip), and are not considered sufficient for meeting the requirements of 10CFR50.46 for the ECCS suction strainer clogging issue.

SRP Sections 6.3 and 15.6.5 deal with evaluation of the ECCS, and more specifically, the adequacy of ECCS performance. The review procedures of 15.6.5 specifically requires reviewers to ensure that "A variety of break locations and the complete spectrum of break sizes were analyzed." This is clearly the intent of 10CFR50.46. Regulatory Guide 1.82, Revision 2 (RG 1.82) Regulatory Position 2.3.1.5 states the following: "As a minimum, the following postulated break locations should be considered. (a) Breaks on the main steam, feedwater, and recirculation lines with the largest amount of potential debris within the expected zone of influence, (b) Large breaks with two or more different types of debris within the expected zone of influence, (c) Breaks in areas with the most direct path between the drywell and wetwell, and (d) Medium and large breaks with the largest potential particulate debris to insulation ratio by weight." The staff believes that the RG provides a more complete scope of breaks needed to meet the intent of 10CFR50.46. The staff notes, however, that there was not enough detail provided for the pipe stress analysis or weld location approaches, described in the URG, to determine if there would be significant overlap between that approach and the one in the RG.

2) Under "Introduction," Page 1, Line 12: The focus of the BWROG position in this section appears to be on the "fibrous debris source term." The staff reminds the BWROG that the focus should be on all debris sources which may clog suction strainers. Other Debris may still cause clogging problems (e.g., paints/coatings, plastic, rope, etc.) Plants with small NPSH margins and high strainer approach velocities may still have problems with reflective metallic insulation (RMI). This comment applies in general to all URG sections.

3) General Comment: URG 3.2.1.1 cites Reg. Guide 1.46 several times. Reg. Guide 1.46 was withdrawn by the NRC in March 1985 because the July 1981 revision of SRP 3.6.2 provided more current information concerning the matters covered in the guide. The URG should not cite regulatory documents that have been withdrawn.

4) General Comments: First, RG 1.82, states the following in Regulatory Position 2.3.1.3: "Identify all sources of fibrous materials in the containment such as fire protection materials, thermal insulation, or filters that are present during operation." The URG focuses too much on fibrous and not enough on all debris sources. NRC Research has indicates that RMI can, under certain circumstances, represent a potential clogging mechanism (see Reference 1 transmitted to R. Sgarro by letter dated July 9, 1996), as can containment coatings, and other assorted debris. Keep in mind that some plants have very small NPSH margins and high strainer approach velocities. The focus of all guidance should be to minimize the potential for clogging of ECCS strainers by <u>all</u> potential debris sources. Therefore, debris source term estimation should account for other debris sources. Second, the BWROG seems to struggling with different methodologies for different plant licensing situations. A more logical approach would be to develop a simplified generic BWROG recommended methodology, and let the individual plants modify it as necessary for application to their plant.

<u>Section 3.2.2.2, "Suppression Pool Transport and Settling"</u>

1) Under "Introduction," Page 1, Lines 8 through 10: the URG directs licensees to account for debris normally present in the suppression pool as well as debris transported from the drywell. However, the BWROG usage of the words "normally present" may lead to exclusion from the analysis of known foreign materials in the pool. When foreign materials are dropped in the suppression pool, they should be immediately retrieved. If retrieval is not possible, the impact of that material in combination with other material in the pool and that may reach the pool during an accident should be evaluated and appropriate action taken.

2) Under "BWROG Guidance," Page 5, Line 22: The URG should be more specific as to what debris would be suspended by the high energy phase of a LOCA.

Specifically, the phrase "slightly greater than 1.0" could be interpreted differently by every plant. How much greater than 1.0 does a specific gravity have to be for a certain debris type to be suspended and what is the basis for the value selected? The URG should be more specific to ensure consistent utility response to NRC Bulletin 96-03.

3) Under "Basis," Page 7, Line 27: the URG incorrectly states that the testing documented in the two NUREG documents was performed using a test setup similar to a Mark II containment type. In fact, these NUREG studies were based on a Mark I containment type.

4) Under "Basis," Page 7, Line 28 through Page 8, Line 3: There is no evidence to support this statement. On what basis would a licensee be able to demonstrate that debris would not be suspended from the pool floor?

5) General Comment: Section 3.2.2.2 of the URG does not acknowledge the potentially large uncertainties inherent in an analysis of accident conditions so that credit may be taken for suppression pool settling. For example, there are major uncertainties related to the variability of amounts and size distributions of generated debris, and in the transport of debris to and within the suppression pool. There are also variability uncertainties related to the amount of pool agitation caused by washdown from the drywell, suppression pool cooling, etc. Consideration should also be given to whether or not suppression pool cooling causes resuspension of sludge or debris on the floor of the suppression pool.

6) General Comment: URG 3.2.2.2 does not provide calculational methodologies for plant-specific evaluations. Instead, the guidance in URG 3.2.2.2 regarding plant-specific evaluations is very general. In addition, there is no guidance on how to deal with the calculational uncertainties described above.

Section 3.2.3.4, "ECCS Pump NPSH Calculations"

1) General: the draft document does not provide a method or methods for calculating head loss across the strainers (i.e., no correlations are provided). We assume that the BWROG is still in the process of developing recommended correlations for strainer head loss.

2) General: this section also focuses too much on just fibrous debris and corrosion products. Consistent with NRC Bulletin 96-03, licensees need to focus on all potential types of debris which could cause clogging of the ECCS suction strainers (e.g., paint chips, concrete dust, etc. in addition to fibrous material and corrosion products). Regulatory Position 2.2.1(f) states that the potential for foreign materials (e.g., tape, wire, paper, plastic, etc.) in the suppression pool and their potential to impact the suction strainer head loss should be considered in the evaluations. URG 3.2.3.4 provides no guidance to licensees regarding consideration of these types of foreign materials in the suppression pool.

3) Page 2, Lines 4 through 9: the URG states that "bathtub curve" head loss trends for ECCS strainers due to the possible ranges of fibrous debris quantities with given amounts of particulate debris on the strainers applies only to conventional truncated cone strainers, and does not apply to alternate strainer geometries. The only alternate geometries referred to in URG 3.2.3.4 are the star and stacked disc designs, and it is not clear that these have been subjected to sufficient testing to demonstrate that the "bathtub" head loss characteristic does not occur (with the possible exception of the 60 point star strainer). The URG assertion regarding the applicability of the "bathtub curve" to different strainer designs needs to be verified with test data, and the theoretical basis for the phenomena explained. Our review of the test data from the flow tests performed by the BWROG on one of the stacked disk strainer designs indicates that the testing was limited, and was not of sufficient depth or breadth to conclusively demonstrate that such strainer designs are not susceptible to the "thin film" effects observed with conventional strainer designs.

Note that the bathtub curve effects and thin film effects were not observed in the head loss tests performed at Alden Research Laboratory in support of the resolution of ECCS strainer blockage in BWRs. Tests conducted with small quantities of fibrous materials together with particulate debris resulted in the formation of non-uniform beds on the strainer surface.

The claim by the BWROG in URG 3.2.3.4 that the newer strainer designs are not susceptible to the thin film head loss effects may be due to the formation of non-uniform debris beds on these strainers for tests conducted with relatively small quantities of fibrous materials. However, the test data we have reviewed did not provide any details on the characteristics of the debris beds formed on the strainers. In addition, while apparently not observed to date, there is no phenomenological basis for concluding that a thin, uniform debris bed could not be formed on the newer strainer configurations. The fact that such beds have not been observed to date may be coincidental. Sufficient test data under such conditions is needed to assure that the thin film debris beds will not be formed on the newer strainer designs. The BWROG is requested to show how their data is sufficient to draw this conclusion.

4) Under Guidance, Page 6, Lines 6 and 7: the guidance advises licensees to use the "expected" amount of sludge/corrosion products in performing their plant-specific evaluations, and to use the "maximum expected quantity" of fibrous debris. The more prudent recommendation would be to use the "maximum expected" amount of sludge/corrosion products to better assure adequate strainer sizing rather than to use just the expected amount. The "expected" quantities are not defined, but we assume they represent the best estimates of such quantities. Also, maximum expected quantities of other debris types should be accounted for (e.g., paint chips, concrete dust, tie wraps, etc.).

5) Under Guidance, Page 6, Lines 8 through 11 and Lines 26-28: has the BWROG considered the need for margin in the NPSH calculation due to the uncertainties involved with the rest of the analysis?

6) Under Guidance, Page 7, Lines 7 through 9: In order to assume a clean strainer at the start of an accident, the licensee must have a basis on which to make that assumption (e.g., strainer/suppression pool inspections and cleanings every refueling outage).

7) Under Guidance, Page 7, Lines 11 through 16: If the ECCS cooling performance model required by 10 CFR 50.46 is updated by the licensee, then a report to the staff must be made consistent with the requirements of 10 CFR 50.46.

8) General: Regulatory Position 2.1.1(b) of RG 1.82, Rev. 2, states than an assessment should be made of the ECCS pump's susceptibility to degradation from debris ingestion. URG 3.2.3.4 makes no mention of performing this type of assessment. Is this covered in another part of the URG?

References

1) Science and Engineering Associates, Inc. Technical Letter Report entitled, "Experimental Investigation of Head Loss and Sedimentation Characteristics of Reflective Metallic Insulation Debris," Dated May 1996.

8

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 20, 1996

Mr. Rocky Sgarro
Pennsylvania Power and Light
2 North Ninth Street
Mail Code A6-1
Allentown, PA 18101

SUBJECT: COMMENTS ON DRAFT UTILITY RESOLUTION GUIDANCE SECTIONS 3.1, 3.2.2, 3.2.4, AND 3.4

Dear Mr. Sgarro:

On April 1, 1996, the Boiling-Water Reactor Owners Group (BWROG) submitted four draft sections for their Utility Resolution Guidance (URG) document to the staff. The four sections submitted were: Section 3.1.4 entitled "Backflush," Section 3.2.1.1 entitled "Postulated Break Locations," Section 3.2.2.2 entitled "Suppression Pool Transport and Settling," and Section 3.2.3.4 entitled "ECCS Pump NPSH Calculations." All sections were marked "Draft" and dated March 31, 1996. The staff subsequently provided comments to you by letter dated July 25, 1996. In a meeting with the BWROG on June 5, 1996, the staff was given four additional draft sections for review. The four sections submitted were: Section 3.1 entitled "Evaluation of Resolution Options," Section 3.2.2 entitled "Other Drywell Debris Sources," Section 3.2.4 entitled "Suppression Pool Debris Sources," and Section 3.4 entitled "Self-Cleaning Strainer." These sections were also marked "Draft" and dated May 28, 1996. It was requested that the staff review these additional draft sections and provide the BWROG with any preliminary comments the staff might have.

The staff does not normally review draft documents; however, due to the tight schedule that licensees are on for implementation of the requested actions in NRC Bulletin 96-03, the staff is providing comments on the draft sections to the BWROG so that these comments may be addressed by the BWROG prior to submitting the completed URG to the staff for review. It is hoped that this, in turn, will facilitate a faster review on the final document.

The staff has completed a preliminary review of the four sections received on June 6, 1996, and our comments are attached. It should be noted that since we have not seen all of the sections in the document, our comments should not be construed as an endorsement of these sections or a partial endorsement of the URG document. This is because many of the sections are interrelated, and we cannot perform a complete review without having the whole document with which to work. However, we have attempted to raise any staff concerns early so that the BWROG may work on resolving them prior to the submittal of the completed URG document. No response is required to this letter, however, the staff will carefully review the final URG and supporting documents to ensure that our concerns stated herein have been adequately addressed.

If you have any questions regarding this letter, please contact either Rob Elliott at (301) 415-1397 or Michael Marshall at (301) 415-5895.

Sincerely,

Original signed by:

Carl H. Berlinger, Chief
Containment Systems and Severe Accident Branch
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation

Enclosure: Staff Comments

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Enclosure

COMMENTS ON BWROG OF URG SECTIONS: 3.1.4, "BACKFLUSH;" 3.2.1.1, "POSTULATED BREAK LOCATIONS;" 3.2.2.2 "SUPPRESSION POOL TRANSPORT AND SETTLING;" and 3.2.3.4, "ECCS PUMP NPSH CALCULATIONS"

URG Section 3.1, Evaluation of Resolution Options

1. The conformance, or non-conformance, with RG 1.1 (see pages 5 and 8 of Section 3.1) is of concern to the staff. The staff discourages the use of containment overpressure in determining NPSH margin. This is because containment pressure will not likely be available over the long term. In addition, it is possible to postulate excessive containment leakage (i.e., failure of a containment penetration). It is not recommended that licensees seek NPSH credit through a licensing basis change to take credit for containment overpressure.

2. Page 6, Lines 12 through 15: The staff cannot evaluate the validity of the statement without the supporting section 3.1.1.1 and an explanation of how this conclusion was reached.

3. Page 6, Lines 25 through 29: Given the thin fiber bed effect when combined with sludge on standard trucated cone and cylindrical strainer design, it seems unlikely that a licensee will be able to justify the adequacy of their existing strainers; however, comment 2 above also applies to this part of the URG.

4. Page 7, Line 7: Licensees should also quantify nonfibrous debris which can potentially plug strainers.

5. Page 13, Lines 22-24: Please explain how this method would be used to meet 10CFR50.46. If this is done to reduce the maximum amount of fiber debris from a given break, it may be potentially okay. If its purpose is to replace fibrous insulation at the "most likely breaks" only, the staff may take exception as noted in our previous comments on the URG transmitted to R. Sgarro on July 25, 1996. The staff's concern is to make sure that licensee's evaluation is of sufficient scope to ensure compliance with 10CFR50.46.

URG Section 3.2.2, Other Drywell Debris Sources

1. Page 2, Lines 20 through 24: Has the BWROG testing identified any material which would not transport?

2. Page 3, Lines 1 through 12: How does a licensee determine the debris source term from their FME controls? For instance, how is FME effectiveness measured?

3. Page 4, Lines 11 through 15: How does a licensee establish their baseline value of fibrous, non-insulation debris in the drywell?

4. Page 7. Line 12: The staff has not been provided with any data which substantiates use of any transport factor less than 1.0.

5. Page 8. Item 2 and Page 13. Item 2: What is the correlation of NPSH margin to verification of baseline values (i.e.. how much margin is needed for a one cycle verification. how much for two cycle. etc.). This should be established in order to ensure a consistent industry response.

6. Page 8. Item 3. Page 13. Items 1 and 3. and Page 17. Item 1: How does a licensee determine what is "not transportable?" Criteria should be provided. Also. is there a need to periodically perform the review to ensure that conditions haven't changed?

7. Page 9. Lines 20 through 22: Can utilities realistically control their dirt/dust source term through FME/housekeeping? If so. how is the effectiveness of their FME program measured? It seems that a more appropriate way would be to use a bounding assumption combined with FME.

8. Pages 11 and 12: No discussion is provided on verifying the adequacy of existing coatings (e.g.. has the licensee verified that qualified coatings have not degraded over time due to irradiation. misapplication. etc.). and ensuring that the licensee's coating log accurately reflects the coatings in the plant (both qualified and unqualified).

9. Page 13. Item 3: How does the licensee establish their quantity of fixed particulate debris assumed to be available for transport?

10. Page 14. Item 4: How is the effectiveness of the programmatic controls for particulates measured?

11. Page 15. Line 4: Why is latent debris only a problem after containment pressure is reduced?

12. Page 15. Lines 7 through 9 and Page 16. Lines 11 through 14: How about gravity as a transport mechanism to the drywell floor? How about break flow as a transport mechanism to the suppression pool?

13. Page 15. Lines 9 through 12 and Page 16. Lines 14 through 16: What is the basis for this statement? The staff has not seen any information that expects latent debris to be a small quantity.

14. Page 15. Lines 24 through 27: Has consideration been given to replacing unqualified or indeterminate coatings with qualified? Or performing in-situ testing to qualify coatings if possible?

15. Page 2. Lines 1 through 7: Please substantiate the claim that unqualified paints (latent debris) will not be transported to the wetwell in the same time frame as qualified paints.

16. Page 14. Lines 19 through 25: The distinction between latent debris and fixed debris is unclear. Please clarify (especially the difference in generation and transport).

Section 3.2.4. Suppression Pool Debris Sources

1. Page 3. Line 20: Optimizing FME to a zero source term is not impractical. it's impossible. Sludge is generated during normal operation when the suppression pool is inaccessible.

2. Page 8. Lines 2 through 5: This is good guidance. except that no methodology is provided on how to determine the amounts of fibrous debris which may be washed down.

3. Page 12. Line 9: If any plants in the survey had higher than 150 lbm per year sludge generation. then this could be a non-conservative assumption for a licensee if their subsequent plant specific evaluation determines a higher generation rate than the recommended value.

4. Page 13. Lines 4 through 7: Again. has consideration been given to minimizing the source term for coatings such as removal of unqualified coatings or qualification through in-situ testing?

5. Pages 13 through 15: Because of the non-uniformity of sludge on the pool floor. how can this procedure be accurate? Why not perform successive cleanings to determine sludge generation rate?

6. Page 18. Line 7: What is definition of "substantially greater than 1.0?"

7. Page 18. Lines 22 through 23: Should this evaluation should also include irradiation and aging effects?

8. Page 6. Lines 20 through 24: Does the BWROG intend to provide guidance on how to conservatively account for head loss caused by different types of fibrous debris?

9. Page 10. Lines 15 through 17: Since plants are being advised to collect sludge samples to establish sludge generation rates. the size distribution associated with that sludge should be compared to particulate size distribution being recommended by the BWROG.

URG Section 3.4. Self Cleaning Strainers

1. Page 4. Lines 10 through 13: Justification for not needing instrumentation and alarms would be needed. The staff believes that instrumentation and alarms are need to ensure operability.

2. Page 4. Lines 18 through 21: This surveillance frequency has not been justified to the staff nor does this document provide such justification.

3. Page 5. Lines 16 through 19: This methodology is still under evaluation by the staff.

4. Page 6. Item 11: GE Rept. GE-NE-T23-0700-15-21. Rev. 1. March 1996. is identified. Although this evaluation is introduced into the self-cleaning strainer design section. the implications of effects of debris which could pass through the strainer has much broader implications and we should get a copy for review.

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OMB No.: 3150-0011
NRCB 93-02

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

May 11, 1993

NRC BULLETIN NO. 93-02: DEBRIS PLUGGING OF EMERGENCY CORE COOLING SUCTION STRAINERS

Addressees

For Action:

All holders of operating licenses for nuclear power reactors.

For Information:

All holders of construction permits for nuclear power reactors.

Purpose

This bulletin notifies licensees of a previously unrecognized contributor to the potential loss of net positive suction head (NPSH) for the Emergency Core Cooling Systems (ECCS) for Light Water Reactors during the recirculation phase of a loss-of-coolant accident (LOCA). All operating reactor licensees are requested to take the recommended actions, and are required to provide the U.S. Nuclear Regulatory Commission (NRC) with a written response describing the actions taken associated with this bulletin.

Background

On December 3, 1985, the NRC issued Generic Letter 85-22, "Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage," which recommended that Regulatory Guide 1.82, Revision 1, "Water Sources for Long-Term Recirculation Cooling Following a Loss of Coolant Accident," be used as guidance for 10 CFR 50.59 reviews dealing with modification of thermal insulation. Regulatory Guide 1.82, Rev. 1 discussed, among other things, the blockage of strainers due to fibrous debris. On September 30, 1992, the NRC issued Information Notice (IN) 92-71, "Partial Plugging of Suppression Pool Strainers at a Foreign BWR," which provided information on the plugging of two ECCS suction strainers by mineral wool insulating material that had been dislodged by steam from an open safety valve. On April 26, 1993, the NRC issued Information Notice 93-34, "Potential for Loss of Emergency Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment," which described two incidents of debris identified in suppression pools associated with damaged and clogged strainers, and the identification of a significant source of material in a PWR with the potential to restrict the flow through the sump debris screen.

9305110015

Description of Circumstances

Information Notice 93-34 and Supplement 1 to IN 93-34 issued on May 6, 1993, described clogging of ECCS pump suction strainers that occurred at the Perry Nuclear Plant, a BWR-6. The event involving deposition of filter fibers on residual heat removal strainers occurred in March 1993, two months after the strainers at Perry had been replaced and the suppression pool had been thoroughly cleaned. Subsequent to issuance of IN 93-34, the licensee chemically analyzed the debris on the strainer. The debris consisted of glass fibers from temporary drywell cooling filters that had been inadvertently dropped into the suppression pool, and corrosion products that had been filtered from the pool by the glass fibers adhering to the surface of the strainer. A small amount of the fibrous filter material was also found in the suppression pool near the weir wall.

The licensee used the fibrous material as a filter in the drywell in three cooling air return ducts that each have a surface area of about 5.6 square meters [60 square feet]. In addition, there were six similar air filters in containment. The purpose of the filters is to maintain cleanliness in containment and the drywell during reactor outages. It has been the licensee's practice to replace the filter material at the end of each outage and to leave the material in the drywell and containment during operation of the plant at power. As a result of the March 1993 event, the licensee will remove the filters from the drywell and containment prior to startup. They will also remove fibrous insulation from ducting in the pool-swell area of the containment. In addition, Perry has thoroughly cleaned the drywell, containment, and suppression pool to reduce dirt, dust, and foreign material which could contribute to the clogging of the strainers. They have also initiated a program to periodically monitor RHR strainer differential pressure. They have replaced their strainers, significantly increasing the flow area. In addition, a procedure has been developed to use existing equipment to backflush the strainers in the event of clogging.

Discussion

Section 50.46 of Title 10 of the Code of Federal Regulations (10 CFR 50.46) requires that each operating nuclear power plant have an ECCS that provides, among other characteristics, long-term cooling capability. For the LOCA sequence, this long-term cooling capability is provided through recirculation of the coolant from the Boiling Water Reactor (BWR) suppression pool or the Pressurized Water Reactor (PWR) sump back to the reactor vessel. Appendix A to 10 CFR Part 50, General Design Criterion (GDC) 35, "Emergency Core Cooling," GDC 36, "Inspection of Emergency Core Cooling System," and GDC 37, "Testing of Emergency Core Cooling System," require appropriate design, inspectability and testability of the Emergency Core Cooling Systems. This includes the ability of the ECCS to provide long-term core cooling.

NUREG-0897, Rev. 1, "Containment Emergency Sump Performance," which was written in conjunction with resolution of Unresolved Safety Issue (USI) A-43, addressed transport of fibrous thermal insulation from the containment to the strainers during a LOCA. Resolution of USI A-43 was based in part on strainer head loss tests with fibrous thermal insulation obstructing flow. The consequences on head loss of the filtering action of the fibrous material on the strainer was beyond the scope of USI A-43. The staff has in place a program to systematically evaluate the larger implications of the Perry experience and the aforementioned foreign reactor event. This will include consideration of strainer area, containment housekeeping, pool cleanliness, and measures to cope with clogged strainers.

The Perry event showed that filtering of corrosion products, dust, and other debris from the drywell, as occurred at Perry, may cause an unexpectedly rapid loss of net positive suction head for the ECCS pumps when they are needed to perform their intended function. Fibrous air filters and other temporary material appear to be likely sources of such fibrous material. This bulletin deals with the presence of such material in reactor containments.

Requested Actions

All holders of operating licenses for nuclear power reactors, immediately upon receiving this bulletin, are requested to take the following actions:

Identify fibrous air filters or other temporary sources of fibrous material, not designed to withstand a LOCA, which are installed or stored in your primary containment. Take any immediate compensatory measures which may be required to assure the functional capability of the ECCS. Take prompt action to remove any such material. Because of the low probability of a LOCA event, the staff considers removal of this material at the next shutdown, or within 120 days, whichever comes first, to be sufficiently prompt. If the facility is currently in a shutdown, you are requested to remove such material prior to restart.

Reporting Requirements

All action addressees are required to submit the following written reports:

1. Within 30 days of the date of this bulletin, a written response stating whether the actions requested above have been or will be performed. If the use of such material is identified, this written response shall also include the locations and quantity of use, any immediate compensatory measures taken, and the current schedule for removal of the material.

2. Within 30 days of completion of the requested actions, a report confirming completion.

3. If an addressee proposes not to take the actions requested in this bulletin, provide to the NRC staff, within 30 days of the date of this bulletin, your proposed alternative course of action and a justification for any deviations from the requested actions.

Address the required written reports to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555, under oath or affirmation under the provisions of Section 182a, Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f). In addition, submit a copy to the appropriate regional administrator.

Backfit Discussion

The operability of the ECCS suction for recirculation is required to meet a condition of a plant operating license and the requirements of Section 50.46 of 10 CFR. The actions requested by this bulletin represent a new staff position and are considered necessary to ensure that licensees are in compliance with existing NRC rules and regulations where these conditions are applicable. Therefore, this bulletin is being issued as a compliance backfit under the terms of 50.109(a)(4), and is being issued as an immediately effective action [10 CFR 50.109(a)(6)].

Paperwork Reduction Act Statement

This bulletin contains information collection requirements that are subject to the Paperwork Reduction Act of 1980 (44 U.S.C. 3501 et seq.). These requirements were approved by the Office of Management and Budget, approval number 3150-0011.

The public reporting burden for this collection of information is estimated to average 60 hours per response, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the collection of information. Send comments regarding this burden estimate or any other aspect of this collection of information, including suggestions for further reducing reporting burden, to the Information and Records Management Branch (MNBB-7714), U.S. Nuclear Regulatory Commission, Washington, D.C. 20555; and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-3019, (3150-0011), Office of Management and Budget, Washington, D.C. 20503.

Compliance with the following request for information is purely voluntary. The information would assist NRC in evaluating the cost of complying with this bulletin:

(1) the licensee staff time and costs to perform requested inspections, corrective actions, and associated testing;

(2) the licensee staff time and costs to prepare the requested reports and documentation;

(3) the additional short-term costs incurred as a result of the inspection findings such as the costs of the corrective actions or the costs of down time;

(4) an estimate of the additional long-term costs which will be incurred in the future as a result of implementing commitments such as the estimated costs of conducting future inspections or increased maintenance.

If you should have any questions about this matter, please contact one of the technical contacts listed below or the appropriate NRR project manager.

James G. Partlow
Associate Director for Projects
Office of Nuclear Reactor Regulation

Technical contacts: Roger Woodruff, NRR
(301) 504-2917

John B. Hickman, NRR
(301) 504-3017

Attachment:
List of Recently Issued NRC Bulletins

LIST OF RECENTLY ISSUED NRC BULLETINS

Bulletin No.	Subject	Date of Issuance	Issued to
93-01	Release of Patients After Brachytherapy Treatment with Remote Afterloading Devices	04/20/93	Brachytherapy Licensees Authorized to Use After-loading Brachytherapy Unit(s) Capable of Delivering Dose Rates Greater than 500 RADS (centigray) per Hour at 1 Centimeter
90-01, Supp. 1	Loss of Fill-Oil in Transmitters Manu-factured by Rosemount	12/22/92	All holders of OLs or CPs for nuclear power reactors.
92-03	Release of Patients after Brachytherapy	12/08/92	<u>For Action</u> - Brachytherapy Licensees Authorized to use the Omnitron Model 2000 High Dose Rate (HDR) Afterloading Brachytherapy Unit <u>For Information</u> - None
92-01, Supp. 1	Failure of Thermo-Lag 330 Fire Barrier System to Perform its Specified Fire Endurance Function	08/28/92	<u>For Action</u> - All holders of operating licenses for nuclear power reactors. <u>For Information</u> - All holders of construction permits for nuclear power reactors.
92-02	Safety Concerns Relating to "End of Life" of Aging Theratronics Tele-therapy Units	08/24/92	<u>For Action</u> - All Teletherapy Licensees <u>For Information</u> - None

OL = Operating License
CP = Construction Permit

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

February 18, 1994

NRC BULLETIN 93-02 SUPPLEMENT 1: DEBRIS PLUGGING OF EMERGENCY CORE COOLING SUCTION STRAINERS

Addressees

For Action:

All holders of operating licenses or construction permits for boiling-water reactors.

For Information:

All holders of operating licenses or construction permits for pressurized-water reactors.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this bulletin supplement:

(1) to inform Action and Information addressees about the vulnerability of emergency core cooling system (ECCS) suction strainers in boiling-water reactors (BWRs) and containment sumps in pressurized-water reactors (PWRs) to clogging during the recirculation phase of a loss-of-coolant accident (LOCA).

(2) to request that Action addressees take the appropriate actions to ensure reliability of the ECCS in view of the information discussed in this bulletin supplement regarding the vulnerability of the ECCS strainers to clogging.

(3) to require that Action addressees report to the NRC whether and to what extent the requested actions will be taken and to notify the NRC when actions associated with this bulletin supplement are complete.

Background

The NRC staff concerns related to the potential loss of post-LOCA recirculation capability due to insulation debris were discussed in Generic Letter 85-22 (December 3, 1985), "Potential for Loss of Post-LOCA Recirculation Capability due to Insulation Debris Blockage" which documented the NRC's resolution of Unresolved Safety Issue (USI) A-43, "Containment Emergency Sump Performance." Although the staff concluded at that time that no new requirements would be imposed on licensees and construction permit holders, the staff did recommend that Regulatory Guide 1.82, Revision 1, "Water Sources for Long-Term

Recirculation Cooling Following a Loss-of-Coolant Accident," be used as guidance for the conduct of 10 CFR 50.59 reviews dealing with the modification of thermal insulation installed on primary coolant system piping and components. NUREG-0897, Revision 1, "Containment Emergency Sump Performance" (October 1985), contained technical findings related to USI A-43, and was the principal reference for developing the revised regulatory guide.

Since 1985, the NRC staff has issued several information notices on this subject because of new developments.

On May 19, 1988, the NRC issued Information Notice 88-28, "Potential for Loss of Post-LOCA Recirculation Capability due to Insulation Debris Blockage," which discussed an item reported under 10 CFR Part 21 concerning the deterioration of drywell insulation and the potential for the aluminum foil coating of the insulation to block ECCS strainers during a LOCA.

On January 30, 1990, the NRC issued Information Notice 90-07, "New Information Regarding Insulation Material Performance and Debris Blockage of PWR Containment Sumps," which discussed experiments exposing NUKON insulation to high temperatures and alkaline conditions. The results of these tests indicated that head loss across the insulation material increases significantly after about 24 hours due to a reduction in flow area caused by compaction of the fiberglass material at elevated pH conditions.

On September 30, 1992, the NRC issued Information Notice 92-71, "Partial Plugging of Suppression Pool Strainers at a Foreign BWR," which reported on the plugging of two ECCS strainers at the Barsebäck Unit 2 BWR in Sweden on July 28, 1992. The strainers were plugged by mineral wool insulation that had been dislodged by steam from a pilot-operated relief valve that opened while the reactor was at 3100 kPa [435 psig]. Two of the five strainers on the suction side of the containment spray pumps were in service and became partially plugged with mineral wool. Following an indication of high differential pressure across both suction strainers 70 minutes into the event, the operators shut down the reactor and backflushed the strainers.

Information Notices 93-34 and 93-34 Supplement 1, "Potential for Loss of Emergency Cooling Function due to a Combination of Operational and Post-LOCA Debris in Containment," were issued on April 26, 1993, and May 6, 1993, respectively. They described several instances of clogging of ECCS pump strainers including two that occurred at the Perry Nuclear Plant, a domestic BWR 6. The first Perry event entailed clogging of residual heat removal strainers by operational debris. The second Perry event involved the deposition of filter fibers on residual heat removal strainers. The debris consisted of glass fibers that had been inadvertently dropped into the suppression pool from temporary drywell cooling filters, and corrosion products that had been filtered from the pool by the glass fibers adhering to the surface of the strainer. On May 11, 1993, in response to this event, the staff issued NRC Bulletin 93-02, "Debris Plugging of Emergency Core Cooling Suction Strainers," which requested that both PWR and BWR licensees (1) identify fibrous air filters and other temporary sources of fibrous material in containment not designed to withstand a LOCA and (2) take prompt action to remove the material and ensure the functional capability of the ECCS. All licensees have responded to the bulletin and the staff has reviewed their responses. Discussions are continuing with several licensees. Resolution of the bulletin is anticipated for all licensees in the near future.

In response to these two events, the Perry Nuclear Plant licensee took several actions. These included an increase in the suction strainer area from 1.9 to 3.9 m^2 [20 to 42 ft^2], provisions for a suction strainer backflush capability, and improved measures to maintain a high level of cleanliness in the suppression pool.

Description of Circumstances

After learning of the Barsebäck event, the staff performed approximate calculations for all domestic BWRs, based partly on information obtained from resident inspectors on the design of each BWR and partly on approximations based on general BWR features (such as the close proximity of steam lines leaving the drywell). These calculations showed the potential for loss of net positive suction head (NPSH) of ECCS pumps in some large-break LOCA scenarios in U.S. BWRs.

In the meantime, the regulatory authorities of Sweden and other northern and central European countries have viewed the Barsebäck incident as a precursor event related to potential loss of ECCS cooling due to LOCA-generated debris. They initiated a safety reanalysis effort, coupled with experiments directed at estimating the following: (1) amount of insulation destroyed by the steam jet, (2) resulting composition of debris, (3) amount of debris transported to the suppression pool, (4) extent of insulation debris buildup on strainers, and (5) resultant pressure drop across the blocked strainer under the postulated conditions. The staff compared the recently obtained results of this work with information in NUREG/CR-2982, Revision 1, "Buoyancy, Transport, and Head Loss of Fibrous Reactor Insulation" (July 1983), which was developed as part of the resolution of USI A-43. This comparison showed that the method of fragmenting insulation materials used in U.S. experiments done in support of USI A-43 may not be representative of the scenario following a large LOCA, and that the extent of debris generation due to the jet resulting from a postulated pipe break as reported in NUREG-0897 (1985) was underestimated. Also, the second event at the Perry Nuclear Plant described in IN 93-34 demonstrated that small particles, in combination with debris fibers, significantly increased the pressure drop across the strainers.

Upon completion of the approximate calculations, the staff contracted for a plant-specific study using a BWR 4 as a model to more accurately quantify the effect of LOCA-generated debris on available NPSH. A draft report, "Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA Generated Debris," was published on January 20, 1994. Initial results from this study indicated that the available NPSH margin for the ECCS pumps may be inadequate following dislodging of insulation caused by a LOCA and transport of insulation debris to the suction strainers. This study presently does not consider (1) the effect of corrosion products and other particulates (which were demonstrated in the second Perry event and in experiments to significantly increase the pressure drop across the strainers), (2) the dense packing of debris on the surface of the strainers (which was observed in the Barsebäck incident and in experiments performed in Europe), and (3) the loss of insulation from any pipe but the pipe postulated to break (that is, loss of insulation due to impact of the steam jet on adjacent piping). While the modeling of the transport of debris to the strainers may be conservative, and no credit was taken for accident containment pressure in calculating the available NPSH margin, the staff would expect the analysis results to yield a net reduction in the available NPSH had all these effects been included. Work is now being done to include these effects in the BWR 4 model.

Members of the NRC staff and representatives of domestic BWR licensees attended an OECD/NEA workshop on the Barsebäck incident held in Stockholm, Sweden, on January 26 and 27, 1994. Representatives from other countries at this conference discussed actions taken or planned to prevent or mitigate the consequences of BWR strainer blockage. These actions including changes in insulation materials in containment from mineral wool to fiberglass or a reflective metallic insulation, increasing the (total) area of BWR strainers from 5 m^2 to more than 100 m^2, providing instrumentation to monitor the differential pressure across the strainers during an accident, providing or improving the capability to backflush the strainers, and providing guidance in the emergency operating procedures concerning the correct response to this problem.

The NRC staff has met on two occasions with the Boiling Water Reactor Owners' Group (BWROG) on this issue. These meetings have been productive and both sides have expressed their intent to continue a discussion of the resolution of this issue using the meeting format. Further meetings are anticipated soon and the BWROG has informed the staff of interim actions it is planning to address this issue. In addition, the BWROG has presented two analyses to the NRC staff which form the basis for continued operation while this issue is being resolved. These are discussed further below.

Discussion

10 CFR 50.46 requires that each BWR and PWR must have an ECCS that is designed so that the calculated cooling performance following a postulated LOCA conforms to the acceptance criteria set forth in the regulation. These calculations are done assuming a single failure. Experience from recent operating events, as discussed above, demonstrates that excessive buildup of debris from thermal insulation, corrosion products, and other particulates on ECCS pump strainers has the potential to cause a common-mode failure of the ECCS. The staff presently considers these concerns plant specific because there is such variability of insulations installed, strainer or debris screen sizes and NPSH margins available.

Based on the operating events at Barsebäck and Perry and information from on-going domestic and foreign programs, the staff considers it important to address the issue of strainer blockage. In particular, the results of the recent plant-specific BWR 4 analysis done for the NRC staff and information from the recently completed OECD/NEA conference in Stockholm, discussed above, indicate that immediate interim actions are warranted and prudent until the NRC staff and the BWROG complete studies of the recently identified phenomena and until appropriate actions, based on these studies, can be implemented. The recently identified phenomena include the increased pressure drop across the ECCS strainers due to corrosion products and other particulate matter, and the effect of compaction of debris on the strainers to a greater extent than measured in the work leading to resolution of USI A-43.

Because of the larger surface area of the screens surrounding PWR sumps, the staff considers it acceptable to wait until further studies are performed before determining the need for further action at PWRs.

The NRC considers the interim actions given below to be adequate based on the low probability of the initiating event.

Actions Requested

The NRC requests that pending final resolution of this issue, Action addressees take the following interim actions to enhance the capability to prevent or mitigate loss of the ECCS following a LOCA due to strainer clogging.

- Provide training and briefings to apprise operators and other appropriate emergency response personnel of the information contained herein and in the referenced information notices regarding the potential for suppression pool strainer clogging.

- Assure that the emergency operating procedures make the operator aware of possible indications of ECCS strainer clogging and provide guidance on mitigation.

- Institute procedures and other measures to provide compensatory actions to prevent, delay, or mitigate a loss of available NPSH margin under LOCA conditions. Such measures should be consistent with providing the design basis emergency system functions for core and containment cooling. Actions to assure sufficient core and containment cooling may include:

    - Reduction of flow (consistent with delivering the required ECCS flow) through the strainers to reduce head loss and extend the time for debris deposition

    - Operator realignment of existing systems to allow backflushing of clogged strainers

    - Operator realignment of existing systems to allow injection to the core from water sources other than the suppression pool

    - Intermittent operation of the containment sprays, when possible, to reduce the transport of debris to the strainers

    - Other plant-specific measures which assure availability of sufficient core and containment cooling to meet the design basis of the plant

Action addressees should complete these requested interim actions within 90 days of the date of this bulletin supplement.

Action addressees are encouraged to work with the BWROG to obtain a final resolution of this issue.

Reporting Requirements

All Action addressees are required to submit the following written reports:

(1) Within 60 days of the date of this bulletin supplement, a report indicating whether or not the addressee intends to comply with the actions requested above, description of planned actions, and the schedule for completing them. If an addressee chooses not to take the requested actions, the report shall contain a description of a proposed alternative course of action, the schedule for completing this alternative course of action, and a justification for any deviations from the requested actions.

(2) Within 30 days of completion of the requested actions, a report confirming completion.

Address the required written reports to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555, under oath or affirmation under the provisions of Section 182a, Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f). In addition, submit a copy to the appropriate regional administrator.

Backfit Discussion

Adequate flow from the ECCS is required to meet a condition of a plant operating license and the requirements of 10 CFR 50.46. The actions requested by this bulletin supplement represent a new staff position and are necessary to ensure that licensees are in compliance with existing NRC rules and regulations where these conditions are applicable. Therefore, this bulletin supplement is being issued as a compliance backfit under the terms of 50.109(a)(4).

Paperwork Reduction Act Statement

The information collections contained in this request were approved by the Office of Management and Budget, clearance number 3150-0011, which expires June 30, 1994. The public reporting burden for this collection of information is estimated to average 200 hours per response, including the time for reviewing instructions, searching existing data sources, gathering and maintaining the data needed, and completing and reviewing the collection of information. Send comments regarding this burden estimate or any other aspect of these collections of information, including suggestions for reducing this burden to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-3019, (3150-0011), Office of Management and Budget, Washington D.C. 20503, and to the U.S. Nuclear Regulatory Commission, Information and Records Management Branch, (MNBB-7714), Washington, D.C. 20555.

Compliance with the following request for information is voluntary. The information would assist the NRC in evaluating the cost of complying with this bulletin supplement:

(1) the licensee staff time and costs to perform requested procedure reviews and implementation of changes;
(2) the licensee staff time and costs to prepare the requested reports and documentation;

(3) the additional short-term costs incurred to address the changes, such as the costs of the corrective actions or the costs of down time; and

(4) an estimate of the additional long-term costs that will be incurred as a result of implementation commitments.

If you have any questions about this matter, please contact one of the technical contacts listed below or the appropriate NRR project manager.

Jack W Roe for
Steven A. Varga
Acting Associate Director for Projects
Office of Nuclear Reactor Regulation

Technical contacts: Rob Elliott, NRR
(301) 504-1397

John B. Hickman, NRR
(301) 504-3017

Attachment:
List of Recently Issued NRC Bulletins

10

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

April 26, 1993

NRC INFORMATION NOTICE 93-34: POTENTIAL FOR LOSS OF EMERGENCY COOLING FUNCTION DUE TO A COMBINATION OF OPERATIONAL AND POST-LOCA DEBRIS IN CONTAINMENT

Addressees

All holders of operating licenses or construction permits for nuclear power reactors.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to alert addressees to potential problems regarding operational and post-accident debris that could block emergency core cooling pump strainers in a boiling water reactor (BWR) or containment emergency sump screens in a pressurized water reactor (PWR). It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances

The following paragraphs discuss instances that involve either the actual clogging of emergency core cooling (ECC) or residual heat removal (RHR) pump suction strainers or the potential for clogging of suction strainers in both BWRs and PWRs.

Perry (BWR-6)

On May 22, 1992, during refueling outage 3 at the Cleveland Electric Illuminating Company, Perry Nuclear Plant, the licensee performed an inspection of the suppression pool floor and all suction strainers in the suppression pool using an underwater video camera mounted on a robotic submarine. The licensee found debris on the suppression pool floor and on RHR "A" and "B" suction strainers. The debris consisted of general maintenance-type material and a coating of fine dirt that covered most of the surface of the strainers and the pool floor. As a corrective action, the licensee vacuumed the suppression pool and cleaned the strainers during a mid-cycle outage in January 1993. After cleaning the strainers, it became evident that the RHR "A" and "B" strainers were deformed. The strainers are conical shaped devices made of 18 gauge stainless steel perforated plate with 0.18 cm

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[0.07 inch] diameter holes at 0.287 cm [0.113 inch] centers with internal cruciform shaped stiffener plates for support. The deformation consisted of a buckling of the stainless steel plate between the stiffeners and 3 small cracks were observed on one of the strainers. The licensee replaced the deformed strainers in February 1993, prior to startup from the mid-cycle outage. When the licensee reviewed the historical data on RHR "A" and "B" pump suction pressure and strainer differential pressure, it found no significant trend in pump suction pressure.

After the licensee cleaned the suppression pool and replaced the strainers, an event occurred at the plant in March 1993, during which several safety relief valves (SRVs) were manually lifted and RHR was then used for suppression pool cooling. The licensee inspected the strainers to assess their condition after use and found that the RHR "B" strainer was again coated with debris. A test was run on the RHR "B" pump with the strainer in the as-found condition to determine pump operability and was terminated after 10 hours at the direction of the system engineer when pump suction pressure dropped from an initial reading of 44.1 kPa gauge [6.4 psig] (after pump start) to 0 kPa gauge [0.0 psig]. A second test that used improved suction pressure instrumentation was run on the same loop with similar results (pump suction pressure dropped to 0 kPa gauge [0.0 psig] after 18 hours). The licensee continued to run that pump for an additional 8 hours during the second test, and observed no further decrease in pump suction pressure. Also, in both tests, no change in system flow rates or pump motor amperage was observed.

<u>Grand Gulf</u> (BWR-6)

Entergy Operations' Grand Gulf Nuclear Station experienced similar problems with RHR suction strainers. On March 18, 1988 and again on July 2, 1989, the RHR "A" pump before-start suction pressure fell below the inservice inspection (ISI) acceptance criteria of 17.2 kPa gauge [2.5 psig]. The licensee determined that the low suction pressure was caused by a clogged strainer that takes suction from the suppression pool. The licensee developed more stringent suppression pool cleanliness requirements and more restrictive pump suction pressure limits to ensure that the strainers are cleaned when pump after-start pressures reach the new limits. After an initial cleaning including hydrolazing the walls and floor, the licensee also established a requirement for vacuum cleaning the suppression pool at the end of every refueling outage. Since the July 1989 problem occurred, Grand Gulf has not observed any additional instances of before-start or after-start suction pressures falling below the minimum requirement of 17.2 kPa gauge [2.5 psig].

<u>North Anna</u> (Westinghouse, PWR)

Virginia Power Company's North Anna, Unit 1 personnel removed the mirror insulation from the steam generators (SGs) as part of their SG replacement program and discovered that most of the unqualified silicon aluminum paint covering the SGs had come loose from the SG exterior surface and was only being supported by the insulation jacketing. The pieces of paint ranged in size from sheets 0.61 m [2 feet] wide to dust particles. The same paint had also been used on the pressurizer and was also loose. The quantity of this coating is significant, approximately 1,087 square meters [11,700 square feet] in containment. Although the loose paint is held in place by the insulation during normal operation, it could be exposed during a design basis accident, if there was a pipe or component breach in the vicinity of the SGs or pressurizer causing the insulation jacketing to be removed. Paint fragments could potentially reach the containment sump and reduce the net positive suction head (NPSH) of ECC system pumps that take suction from the sump.

Discussion

It is important that emergency coolant is provided to maintain the reactor at safe temperature levels during all postulated design basis accident conditions. This function is performed by the ECC systems. In the long-term cooling, suction for these systems is either the containment sump (for PWRs) or the suppression pool (for BWRs). In addition, the RHR system provides suppression pool cooling for BWRs during normal operation and transients.

As a result, it is important that adequate NPSH be provided to the pumps throughout the long-term cooling phase. One threat to NPSH is the clogging of the strainers on the suction inlet. Excessive deposits of foreign material on the surfaces of strainers in the suppression pools of BWRs or debris screens in the containments of PWRs can lead to challenges to pump operability. These deposits can reduce the hydraulic head at the suction nozzles of emergency core cooling pumps to less than the net positive suction head required to ensure that the pumps operate without cavitation. If cavitation occurs, the pumps may fail to deliver adequate flow to maintain the integrity of the fuel cladding and the containment pressure boundary.

These recent events, as well as the stuck open relief valve event referenced in NRC Information Notice 92-71, "Partial Plugging of Suppression Pool Strainers at a Foreign BWR," have revealed that debris can be created during the LOCA event as well as during normal operation. Therefore, the complete evaluation of the effects of debris on the performance of safety related systems would consider the combination of both sources.

Related Generic Communications

(1) NRC INFORMATION NOTICE 92-71: "Partial Plugging of Suppression Pool Strainers at a Foreign BWR"

(2) NRC INFORMATION NOTICE 88-28: "Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage"

This information notice requires no specific action or written response. If you have any questions about the information in this notice, please contact one of the technical contacts listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

Brian K. Grimes, Director
Division of Operating Reactor Support
Office of Nuclear Reactor Regulation

Technical contacts: B. Wetzel, NRR
(301) 504-1355

J. Kudrick, NRR
(301) 504-2871

Attachment:
List of Recently Issued NRC Information Notices

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

May 6, 1993

NRC INFORMATION NOTICE 93-34, SUPPLEMENT 1: POTENTIAL FOR LOSS OF EMERGENCY COOLING FUNCTION DUE TO A COMBINATION OF OPERATIONAL AND POST-LOCA DEBRIS IN CONTAINMENT

Addressees

All holders of operating licenses or construction permits for nuclear power reactors.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) issued Information Notice (IN) 93-34 because of the possible problems that could occur with operational and post-accident debris blocking emergency core cooling pump strainers in a boiling-water reactor (BWR) or containment sump screens in a pressurized-water reactor (PWR). The IN was based, in part, on an event that occurred at the Perry Nuclear Plant. The NRC is issuing this supplement to IN 93-34 to alert addressees to additional information relating to that event. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances

IN 93-34 described clogging of emergency core cooling (ECC) pump suction strainers at the Perry Nuclear Plant, a BWR-6. The latest strainer clogging event occurred in March 1993, 2 months after the licensee had replaced the strainers and thoroughly cleaned the suppression pool. After the IN was issued, the licensee chemically analyzed the debris on the strainer. The debris consisted of fibers from air filter material that had been inadvertently introduced into the suppression pool and corrosion products that had been filtered from the pool by the fibers adhering to the surface of the strainer. A small amount of the fibrous filter material also was found in the suppression pool near the weir wall.

The licensee uses the fibrous material in the drywell in three air filters that each have a surface area of about 5.57 square meters [60 square feet]. In addition, there are six similar air filters in containment. The purpose of the filters is to provide filtered air in containment and the drywell during reactor outages. It has been the licensee's practice to replace the filter material at the end of each outage and to leave the material in the drywell and containment during operation of the plant at power. As a result of the

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March 1993 event, the licensee intends to remove the filter material from the drywell before startup to eliminate this source of fibrous material from the drywell.

Discussion

NUREG-0897, Rev. 1, "Containment Emergency Sump Performance," which was written in conjunction with resolution of Unresolved Safety Issue (USI) A-43, addresses transport of fibrous thermal insulation from the containment to the strainers during a loss-of-coolant accident (LOCA). Resolution of USI A-43, in part, was based on strainer head loss tests with fibrous thermal insulation obstructing flow. USI A-43 did not address the consequences on head loss of the filtering action of the fibrous material on the strainer. The Perry event showed that filtering corrosion products, dust, and other debris from the drywell during a LOCA may cause an unexpectedly rapid loss of net positive suction head for the ECC pumps when they are needed to perform their intended function.

Related Generic Communications

- NRC Information Notice 92-71: "Partial Plugging of Suppression Pool Strainers at a Foreign BWR"
- NRC Information Notice 88-28: "Potential for Loss of Post LOCA Recirculation Capability Due to Insulation Debris Blockage"

This information notice requires no specific action or written response. If you have any questions about the information in this notice, please contact the technical contact listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

Brian K. Grimes, Director
Division of Operating Reactor Support
Office of Nuclear Reactor Regulation

Technical contact: Roger W. Woodruff, NRR
(301) 504-2917

Attachment:
List of Recently Issued NRC Information Notices

11

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555

August 12, 1994

NRC INFORMATION NOTICE 94-57: DEBRIS IN CONTAINMENT AND THE RESIDUAL HEAT REMOVAL SYSTEM

Addressees

All holders of operating licenses or construction permits for nuclear power reactors.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to alert addressees to the problem associated with debris recently discovered in the containment and the residual heat removal (RHR) system at some BWR sites. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances

The following are instances in which debris were found in BWR suppression pools.

LaSalle County Station, Unit 1

On April 26 and May 11, 1994, while in a refueling outage, the licensee made two dives into the Mark II design suppression pool to clean the emergency core cooling system (ECCS) suction strainers of a small amount of debris which caused less than 1 percent clogging. The diver found that the strainers had experienced no apparent damage or deformation of the strainer faces. However, while in the pool on both occasions, the divers found and removed an assortment of operational debris. On the first dive, the diver removed a hardhat, a pair of anti-contamination coveralls, a 15.2 meter (m) [50 ft] length of Tygon tubing, 3 nuts, and a 4.6 m [15 ft] length of black duct tape. On the second dive, the diver removed four lengths of 1.9 cm [3/4 in] hose ranging in length about 8 m [25 ft] to about 46 m [150 ft], three lengths of Tygon tubing ranging in length from 6 m [20 ft] to 15 m [50 ft], a short length of 5 cm by 10 cm [2 in by 4 in] wood, and a flashlight.

The diver also noted that a sediment had formed on the suppression pool floor ranging in thickness from 0.3 cm to 5 cm [1/8 in to 2 in]. The suppression pool floor is a level floor with raised ridges in a waffle pattern. The 5 cm [2 in] accumulations of sediment were found in the raised corners of the waffle pattern. The licensee took a sample of the sediment and had it analyzed. The analysis results showed that the filterable solid consisted of over 99 percent iron oxide, with trace amounts of nickel, copper, and chrome. The filtrate contained 120 parts per billion (ppb) dissolved nickel. No organic constituents were found. Radiologically, the sample consisted of 75 percent cobalt-60 with small contributions from manganese-54, cobalt-58, and iron-59. The licensee concluded that the sample contained normal system corrosion products with no contaminants such as hydraulic fluid or solvents. Before startup, the licensee cleaned all ECCS strainers of debris and removed all objects from the pool.

River Bend Nuclear Station

On June 13, 1994, while the plant was in a refueling outage, the licensee conducted an inspection of the ECCS suction strainers and the Mark III design suppression pool. During the inspection, 16 objects were located in the suppression pool. One of these objects, a plastic bag, was removed from the residual heat removal system "A" suction strainer. The other objects that were removed from the pool included a hammer, grinding wheel, slugging wrench, socket, hose clamp, bolt, nut, step-off-pad, two ink pens, antenna, scaffold knuckle, short length of rope, and used tape. Most of these items were not listed in the station suppression pool lost item log. These findings prompted the licensee to take the following corrective actions: (1) remove all items from the suppression pool, (2) inspect all accessible areas for additional debris, (3) verify the strainers for all ECCS pumps to be clean, and (4) increase surveillance of the suppression pool work area to minimize additional objects dropped into the pool. The licensee is reviewing its policies and practices regarding loose objects in and around the grating areas in the containment to determine their adequacy.

The licensee also found sediment in the suppression pool. During the previous refueling outage, that ended in September 1992, the licensee drained and cleaned the pool. However, the licensee was unable to completely clean the pool. After draining the pool, there was still about 0.3 m [1 ft] of water inside the weir wall that was "mucky." During the current refueling outage, the licensee used a portable cleanup system to clean the water in the pool. By the end of the outage, water clarity in the pool significantly improved. However, a layer of sediment still remains on the pool floor. The licensee is planning to install a permanent pool cleanup system two outages from now; however, the planned system will not be able to remove the sediment inside the weir wall.

The following is an instance in which debris was found in the RHR system.

Quad Cities Unit 1

On July 14, 1994, during a post-maintenance test run of the "A" loop of the RHR system, test data indicated that the RHR torus cooling/test return valve, valve 1001-36A, was plugged. When the 36A valve was opened for inspection, the remains of a plastic bag were found shredded and caught within the anti-cavitation trim which was installed during the recent outage. Some of the material appeared to have travelled the entire way through the anti-cavitation trim. The majority of the material was found lodged on the suction side of the valve trim. A few small pieces of plastic were found in the Mark I design suppression pool and removed.

Subsequent to the July 14 event, the licensee observed reduced flow from the "C" RHR pump and initiated further investigation. On July 23, 1994, licensee maintenance personnel removed a drain plug on the volute of the "C" RHR pump and used a boroscope to inspect the pump internals. A 10 cm [4-inch] diameter wire brush wheel and a piece of metal were found wrapped around a vane of the pump. The licensee had opened the RHR system during the outage to work on the RHR 7D valve and removed a butterfly valve on the common suction line (valve RHR 6B). The licensee retrieved the wire wheel brush, the metal and two washers from the pump.

Discussion

The events described above illustrate the potential for adverse effects on emergency core cooling system performance due to debris. The debris resulted from inadequate control of foreign material inside the containment or resulted from inadequate inspection after maintenance activities were performed on a safety system (the RHR system).

Previous NRC generic communications have noted that ECCS strainer clogging represents a potential threat to the reliable functioning of the ECCS pumps throughout a design basis accident. These previous NRC communications have dealt with the potential to clog ECCS strainers with debris generated during plant work activities, debris from a loss-of-coolant accident (LOCA), or a combination of these. The types of debris described herein are further examples of debris licensees have found in their suppression pools or in the RHR system. Although the licensees in the LaSalle and River Bend cases stated that the debris found in their suppression pools would have been insufficient to clog strainers, these debris in combination with the debris generated during a LOCA could accelerate a loss of net positive suction head for the ECCS pumps or cause other types of damage to the strainers. For example, corrosion sediment in the pool could clog the strainers if debris such as the anti-contamination clothing were already drawn to the strainer surface where the clothing could filter the finer particles of sediment out of the water. In addition, damage to strainers might occur if some of the heavier tools were to strike the strainers during the early stages of a LOCA blowdown.

Previous NRC generic communications also have addressed plant events where debris was found in safety systems, namely the auxiliary feedwater system and the safety injection system, and resulted in reduced flow during testing of the systems. The Quad Cities event discussed above involved debris in the RHR system. The debris in this case could potentially cause a transient, result in failure to mitigate a transient or accident, or result in damage to equipment.

Related Generic Communications

- NRC Information Notice 93-34 and Supplement 1: "Potential for Loss of Emergency Core Cooling Function due to a Combination of Operational and Post-LOCA Debris in Containment"
- NRC Bulletin 93-02 and Supplement 1: "Debris Plugging of Emergency Core Cooling Suction Strainers"
- NRC Information Notice 92-85: "Potential Failures of Emergency Core Cooling Systems caused by Foreign Material Blockage"
- NRC Information Notice 92-71: "Partial Plugging of Suppression Pool Strainers at a Foreign BWR"
- NRC Information Notice 88-87: "Pump Wear and Foreign Objects in Plant Piping Systems"
- NRC Information Notice 88-28: "Potential for Loss of Post LOCA Recirculation Capability Due to Insulation Debris Blockage"

This information notice requires no specific action or written response. If you have any questions regarding the information in this notice, please contact one of the technical contacts listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

Brian K. Grimes, Director
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Office of Nuclear Reactor Regulation

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Attachment:
List of Recently Issued NRC Information Notices

12

PERFORMANCE OF CONTAINMENT COATINGS DURING A LOSS OF COOLANT ACCIDENT

REPORT
PREPARED BY

BECHTEL POWER CORPORATION

FOR THE

BWR OWNERS' GROUP

ON BEHALF OF

GENERAL ELECTRIC COMPANY

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DISCLAIMER OF RESPONSIBILITY

Important Notice Regarding Contents of this Report

PLEASE READ CAREFULLY

The information contained in this document is furnished for the purpose of providing members of the BWR OWNERS' GROUP (the "Group") with an engineering evaluation regarding expected performance of typical BWR containment coatings that are subject to design basis accident conditions which include steam line and recirculation piping line breaks. Each member of the Group is responsible for determining the adequacy or usefulness of the information in this document for its particular purposes.

The only undertakings of General Electric Company ("GE") respecting information in this document are contained in the applicable contract (standing purchase order) between GE and the individual utility member of the Group in effect at the time this document is issued. The only undertakings of Bechtel Power Corporation (Bechtel) respecting information in this document are contained in the applicable contract between Bechtel and GE. Nothing contained in this document shall be construed as changing any of the foregoing contracts.

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PERFORMANCE OF CONTAINMENT COATINGS DURING A LOSS OF COOLANT ACCIDENT

CONTENTS PAGE

1.0 Background Information

1.1 Coatings/Systems Typically Found In BWR Containments

Three primary coating systems have historically been used inside the primary containments of BWR nuclear plants. These consist of the following:

- Untopcoated inorganic zinc
- Inorganic zinc with epoxy topcoat for enhanced decontaminability
- Epoxy primer/topcoat for steel and concrete and epoxy surfacer systems for concrete

In addition to those primary systems, alkyd, vinyl, silicone, silicone alkyd and silicone acrylic systems are sometimes applied on minor pieces of equipment and components. Alkyds have historically been the most commonly used of these other coatings.

- **Inorganic Zinc (IOZ)**

  Inorganic zinc coatings consist of either alkyl or metal silicates, (sometimes referred to as water glass) binders plus zinc dust pigment. Alkyl silicates such as ethyl silicate are generally partially hydrolyzed during the paint manufacturing process. After application, these coatings complete hydrolyzation through reaction with atmospheric moisture resulting in a stable silicate binder. The cured film generally contains approximately 75-85% metallic zinc. The cured coating therefore consists of only inorganic components. Surface preparation for these materials is critical as they are very intolerant to organic surface contamination. The cure proceeds slowly , especially in low humidity, because of the necessity to react with atmospheric moisture. Zinc oxide formation should be avoided . Because of the cure mechanism, shrinkage does not occur. Temperature resistance can be as high as 750°F which is the melting point of zinc. Radiation resistance is also high, being tested above 9E10 rads without failure.

  One coat of inorganic zinc is applied to carbon steel surfaces at a dry film thickness of 2 to 5 mils (0.002 to 0.005") for corrosion protection. Inorganic zinc coatings are left untopcoated unless decontamination is a significant consideration or where color or light reflectance are important. In the drywell one coat of inorganic zinc has been used; whereas, two coats are often applied in the immersion zones of the wetwell and torus areas. The total dry film thickness for immersion service has usually been 4 to 8 mils.

- **IOZ/Epoxy**

  Inorganic zinc coatings with epoxy topcoats are used where decontamination is a factor or light reflectance or color is required. Inorganic zinc with epoxy topcoats are not normally recommended for immersion service. Inorganic zinc materials are more difficult to decontaminate than the epoxy materials. Even though inorganic zinc with epoxy topcoats have been successfully used for many years, there are some inherent facts which are important to be aware of with this system. The inorganic zinc coating has a weak internal or cohesive strength. Epoxies, on the other hand, have high internal strength. Epoxy materials also shrink as they cure. This can add stress at the inorganic zinc/epoxy interface. The addition of the epoxy topcoat also reduces the temperature resistance of the coating system to the temperature resistance of the epoxy. The IOZ thickness remains in the 2 to 5 mil range and the epoxy thickness varies from 3 to 8 mils when used as a topcoat for zinc.

- **Epoxy**

Epoxy materials are characterized by an oxirane group in the structure of the organic resin molecule. Epoxy resins make up the binder portion of the paint film. The epoxy molecule must be reacted with a curing agent to effect cure. The curing agents can be polyamides, polyamines, or amine adducts. Phenolic modifications are also sometimes used. The characteristics of the cured film vary with the curing agent. In general, polyamides have good water resistance; whereas, phenolics have better high temperature resistance. Pigments are added to modify the density and performance characteristics of the cured film. Fillers may be added as extenders or to enhance temperature resistance, radiation resistance of other properties of the coating. Epoxies resins have good radiation resistance with a damage threshold of greater than 2E8 rads . Some epoxies have shown resistance to 7E9 rads in both air and water.

Epoxy systems used for metal surfaces usually consist of a primer and finish coat applied at 3 to 6 mils per coat. In some applications a 3 coat system is used at 4 to 8 mils per coat. Some systems are self priming and some use a separate primer.

Epoxy systems are also applied to concrete surfaces. Where decontamination is not a primary consideration, thin film clear sealers are used. Where decontamination of concrete surfaces is a consideration, surfacers and fill coats are used to smooth out the surface prior to application of the epoxy topcoat. Thickness on walls ranges from 3 to 15 mils depending on the system and the number of coats. On floors where wear is a consideration, the thickness can be 15 to 40 mils for self leveling epoxies or as high as 100 to 200 mils for aggregate filled surfacers.

- **Alkyd**

Alkyd resins consist of the reaction product of a polybasic acid, a polyhydric alcohol, and a monobasic fatty acid or oil. Alkyd resins vary in properties mainly due to the type and amount of oil modification. Alkyd resins cure by the oxygen conversion of the oil portion of the molecule. The acid number or content is determined by the polybasic acid used in the polymerization of the alkyd resin. Temperature resistance ranges up to approximately 180° F. Silicone modifications can increase the upper temperature resistance properties. Radiation damage threshold is greater than 1E8 rads.

The nominal thickness of alkyd, silicone alkyd, vinyl and acrylic systems vary from 1 to 5 mils. High heat silicone is usually applied in two or three coats to achieve a total nominal dry film thickness of 3 mils.

1.2 Aging Process

- **Oxidation**

The organic binders utilized in polymeric coatings gradually disintegrate with age. This disintegration by oxidation is normally a surface phenomenon. The primary cause is solar or nuclear radiation. Moisture, oxygen and heat also play a part in the binder degradation. Not all coatings will deteriorate at the same rate. Epoxies will show severe surface chalking while inorganic zinc coatings will oxidize but are almost immune to chalking. Alkyds and acrylics will show moderate chalking. Chalking results from the destruction of the chemical bonds at the surface of the coating film due to radiation combined with oxygen + moisture + heat found in the dry-well and wet-well of a BWR. Ultra-violet light from the sun and fluorescent lights are also significant contributors to chalking. During construction the coatings are often exposed to sunlight for long periods of time. Embrittlement is usually caused by heat and radiation. The chemical cross linking of the resins is accelerated by the heat and radiation resulting in what may be referred to as "overcuring of the film".

- **Heat**

The affects of heat on coatings can have severe adverse affects on the physical properties of the coating film. The degree of effect of the heat on the coatings depends on many variables. The temperature resistance of various resins and the coatings formulated from those resins vary. For example:

Coating Material	Temperature Resistance
Silicone	1200°F
Inorganic Zinc	700°F
Epoxy	200°F - 400°F
Alkyd	150°F - 200°F
Vinyl	150°F - 180°F

The effects of heat on the coating system is also dependent on the temperature and the duration of the exposure. Short term excursions at high temperatures may not harm the coating while long term exposure will have a serious affect on the coating. Dry heat and wet heat (immersion and condensation) will also affect the service life of the coating in different ways. Wet heat is usually much more aggressive in causing performance problems with protective coatings while coatings exposed to dry heat are usually less adversely affected. Cycling of the coatings through a range of temperature conditions and wet and dry conditions can cause damage to a coating system. Heat aging of the coating systems can accelerate the cross-linking of the polymers causing embrittlement in some coatings and softening of others. Often coatings that have been exposed to higher temperatures will go through a color change. For example a white coating may turn tan or brown when exposed to high temperature. This type of color change may be cosmetic in nature and not have any affect on the functional properties of the coating or may be indicative of the deterioration of the resin matrix. In general, coatings which have become embrittled through heat aging are more susceptible to impact damage, disbondment and blistering. Epoxies, for example, that have been heat aged can be cross linked to the point where there is excessive film shrinkage resulting in high stresses in the cured film. The epoxy coatings generally have high bond strengths and can usually withstand these stresses. However, such stresses may lead to cracking and disbondment if the adhesion is weak or marginal. Since the tensile strength of epoxy coatings are generally much higher than that of inorganic zinc, highly stressed epoxy coatings applied over inorganic zinc primers may cause the inorganic zinc primer to split within itself resulting in disbondment of the epoxy topcoats.

- **Radiation**

Radiation resistance varies with the polymer selected and with the binder, pigments, and other variables in the coating formulation. Organic or polymeric materials are degraded by the breaking of chemical bonds due to high level energy from radiation. Initially the radiation energy promotes the cure of polymers but with time the cross linking goes to termination and the absorbed energy then breaks the chemical bonds within the coating film. Various resins have different damage thresholds. The damage threshold is considered to be the point at which the coating looses 25% of its physical properties or the point at which the coating will no longer perform as intended. Generally speaking, the addition of inert fillers and pigments increases radiation resistance of polymers. For example, epoxies which are high molecular weight polymers, are more susceptible to radiation degradation than inorganic zinc materials which have inorganic binders. Pigments and fillers added to polymeric binders interfere with the transmittance of radiation into the paint film. Listed below are some relative radiation resistance of coating types most likely to be found in the containment of a BWR. The accumulated dose is given in rads of γ radiation:

COATING	EXPOSURE	RESULT
• Inorganic Zinc	2E10	No Damage
• Epoxy	>2E8	General resin threshold damage
	>1E9	Epoxy coatings formulated for use in containment
	>7E9	Some tested epoxy coatings
• Alkyd	5 to 7E8	Damage Threshold

Heat, which often builds up in the samples during irradiation, can be as degrading as the radiation itself. As stated earlier, the detrimental affects of oxygen are compounded when combined with radiation. Irradiation in confined spaces without air circulation has a more detrimental effect on a coating than irradiation in areas where there is good air circulation. This is because heat is not dissipated in those confined spaces and ozone (from the ionizing radiation) is more prevalent. This has been confirmed where test results were compared after sample panels were irradiated in closed cans and others were irradiated in the open.

- **Moisture**

The permeability or moisture resistance of coatings varies. A coating will absorb moisture up to an equilibrium state where the release of moisture occurs at the same rate as absorption. Moisture can react with some alkyd resins and pigments. This can lead to the reduction of adhesion of the coating to the substrate. Moisture is less reactive with epoxy resins but can cause degradation if soluble salts are present or water reactive solvents are still present in the film. All coatings are permeable to some degree. As the moisture enters the film, the film swells and loses mechanical strength. Contamination on the substrate can also promote the penetration of water through the paint film and onto the underlying substrate. When water accumulates on the substrate, shear and tensile stresses are set up in the paint film . If these forces exceed the molecular bond strength of the coating to the substrate, the coating will fail by blistering, disbonding or delamination. Inorganic zinc coatings being a very porous film allow the easy ingress and outflow of water. In fact, water can actually increase the cure and resultant bond strength of the inorganic binder system.

- <u>Immersion</u>

In immersion, the effects of moisture on coating performance is amplified. The water, especially deionized water, readily penetrates the paint film. Carbon dioxide is absorbed into the water which causes the water to become acidic (pH of 5 to 5.5 can occur). Tight molecular structures reduce permeability in a coating film. Some coatings such as alkyds and acrylics are not suitable for water immersion. Inorganic zinc is good for immersion in neutral or slightly alkaline water, but the silicate binder will slowly dissolve in hot water.

1.3 Factors Affecting Performance and Failure Mode

Protective coatings are used primarily to provide corrosion protection for ferrous substrates. Coatings can also provide other benefits such as aesthetics, color, lighting enhancement and ease of cleaning or decontamination. The quality of the applied coating system is based on the type of the coating system as well as the quality of the coating application and inspection program. The best coating systems will not perform their intended function if not properly applied. The most important factors which affect the coating performance are surface preparation, dry film thickness, application/curing of the film and operating exposure.

- <u>Surface Preparation</u>

One of the most critical steps in the application of a quality coating system is the often tedious and time consuming process of surface preparation. The quality of the surface preparation is most often controlled through implementation of standards from the Steel Structures Painting Council (SSPC). There are numerous SSPC surface preparation standards which range from minimal preparation by Hand Tool Cleaning (SSPC SP-2) to the most demanding surface preparation by Abrasive Blast to White Metal (SSPC SP-5). The quality of the applied coating system and its ability to properly adhere to the substrate is directly proportional to the degree of surface preparation. The type of surface preparation to be used is generally dependent on the coating material and the service conditions to which the applied coating is to be exposed. The better the surface preparation the better the adhesion of the coating to the substrate. There are three adhesion components which affect the integrity of the bonding of the coating system to the substrate:

1. **Chemical Bond** - A chemical bond results form a chemical reaction between the coating and the substrate. The chemical bond is attributed to the resin used in the coating and its ability to form an adhesive bond to the substrate.

2. **Polar Bond** - The polar bond is established by the attraction of the resin molecule to the substrate, similar to a magnetic attraction. Polar resins are negatively and positively charged portions of the coating film that are attracted to the oppositely charged substrate molecules. Cleanliness of the prepared substrate is critical to the formation of polar bonds since distance between the coating and the substrate is critical to bond strength.

3. **Mechanical Bonds** - The adhesion properties that result from this bond are attributed to the roughness or profile of the substrate. The shape and height of the profile is important in providing the mechanical grip required for good adhesion of the coating. Also, a roughened substrate will provide a larger surface area for the coating to bond to. The required surface profile is dependent on the coating to be applied and is specified in addition to the degree of cleanliness required by various SSPC Surface Preparation Standards.

Defects which result from improper surface preparation include:

1. **Loss of Adhesion or Disbondment** - Loss of adhesion is the separation of the coating system from the substrate. This failure mode can result from insufficient surface preparation which does not remove surface contamination such as rust, mill scale, dirt, moisture, oil or grease. The absence of or an insufficient profile can also cause loss of adhesion. Thermal expansion and contraction of the substrate can cause premature failure of the coating if the surface preparation and/or surface cleanliness is inadequate or unsatisfactory. The main goal of surface preparation is to prevent a coating failure due to a loss of adhesion.

2. **Loss of Cohesion or Delamination** - Loss of cohesion is the separation within the coating system (e.g. separation of the topcoat from the primer). Delamination is the loss of adhesion between coats and is sometimes referred to as cohesive failure. This failure mode can result from the use of non-compatible coating materials, application of the topcoat to a contaminated primer and failure to observe minimum and maximum recoat times.

3. **Blistering** - Blistering of a protective coating is a localized lifting of the coating from the substrate. Blistering can also occur between coats. Blistering occurs most often due to solvent or moisture entrapment in the coating film. However, it can also occur when there is poor adhesion of the coating to the substrate. If the coating has poor adhesion, moisture can penetrate the coating film and occupy the space between the coating and the substrate or between coats. Pressure can build up in this area creating the blister. Blister formation is usually progressive in nature either expanding in population or size or both.

4. **Craters** - Craters or fish eyes can be caused by improper surface preparation if there is oil or grease left on the substrate. Coating is applied over these contaminants will not wet out the substrate and form a continuous film. The areas where the coating has pulled back from the substrate appear as craters.

- **Dry Film Thickness (DFT)**

The dry film thickness of a coating represents the amount of the coating that has been deposited on the substrate after cross-linking and/or solvent evaporation has been completed. The thickness of the coating is specified by the manufacturer and varies depending on the generic coating family, intended service, volume solids and thixotropic properties. The coating manufacturer will usually provide a DFT range which provides a minimum and maximum limit for optimum performance characteristics. The application of the coating at excessive thickness can cause as many problems as applying the coating at insufficient thickness. Problems caused by excessive DFT include:

1. Retardation of the coating cure due to latent solvent release.
2. The formation of blisters due to solvent and water entrapment.
3. Incomplete cure of the film due to premature outer surface curing of the thick film.
4. High stresses within the coating film resulting from shrinkage of the thick film.
5. Mud cracking of the IOZ due to excessive thickness.

Problems caused by low DFT include:

1. Poor flow characteristics and poor film formation.
2. A reduction in barrier protection of the coating film.
3. A reduction in the service life of the coating.

- **Application / Curing Conditions**

The environmental conditions at the time of application can have an affect on the coating film formation and the cure of coating film. The air temperature and the temperature of the substrate can affect the curing rates of certain coatings. For example, epoxies cannot chemically cross link below certain minimum temperatures. Some solvents will not evaporate below a minimum temperature and some coatings will not flow out properly if the temperature is too cold. On the other hand, high temperatures can prevent proper surface wet out of the substrate by "flashing off" the solvents too rapidly. Application of certain coatings at high temperatures can also cause pin holing and blistering. Temperature is especially critical when working with concrete coatings. It is much easier to apply coatings to concrete when the temperature is decreasing. This will reduce moisture evolving from the concrete surface and increase coating penetration into the concrete. Relative humidity is also an important condition during the application of protective coatings. IOZ requires moisture for film development. If zinc dust oxidizes prior to the silicate film formation, the bond to the zinc is diminished and the coating film will be soft and exhibit a reduction in cohesive strength. High humidity can be detrimental to some coatings by retarding evaporation of solvents and adversely affecting cure.

- **Operating Exposure**

The service life of a coating system is affected by the environment to which it is exposed. Some of the common operating conditions that can impact the service life or performance of a coating system are:

1. **High Humidity** - Coatings are regularly exposed to high humidity in many applications. This type of exposure is common in process areas both indoors or outdoors. High humidity can cause condensation of water, and sometimes chemical laden water, on the film. Moisture causes deterioration of the film and accelerates the aging process of most organic coatings. At very high humidities a coating can absorb moisture vapor which can cause blistering if surface contaminants were not all removed during surface preparation. See section 1.2 for further discussion of the effects of moisture on coating films. The wetwell/torus coatings are certainly exposed to constant moisture. During outages and various other times the drywell coatings may also be exposed to high humidities.

2. **Chemical Contamination** - The proper selection and application of protective coatings in areas where corrosive chemicals are used will protect equipment, facilities and the environment from the damaging effects of the chemicals. However, chemicals can weaken or destroy the molecular bonds or otherwise attack the resins, pigments, fillers, plasticizers, etc. in the coating film.

3. **Radiological Contamination** - Exposure to ionizing radiation can accelerate the aging process of protective coatings. See Section 1.2 for further discussion of the effects of radiation on coatings.

4. **Temperature Fluctuations** - Thermal expansion and contraction can have adverse effects on the coating system. The coating system must be properly applied and have the cohesive and adhesive strength to withstand movement of the substrate. This is necessary when the containment shell or the wetwell/torus expands or contracts, especially during a LOCA. Temperature changes are common during reactor start ups or shut downs. Some BWR containments have seen wide fluctuations in temperatures during plant operation. Some of those temperatures have been very high and of long duration.

5. **Physical Abuse** - The coating system must be durable enough to resist mechanical damage from outside sources such as workers, traffic and every day wear and tear.

1.4 What Constitutes Qualified and Non-Qualified Coatings?

Coatings are qualified for use inside containment through a series of tests. Those tests are shown in Table I which cross references the ANSI standards with the ASTM replacement standards. The main tests pertinent to a coatings use inside the containment are:

- Design Bases Accident (DBA)
- Radiation Tolerance

Those tests show whether or not the coating(s) will be able to survive a LOCA without disbonding and detaching from the surface. Sometimes the test panel preparation includes artificial aging to assure that the coatings can survive a LOCA after years of normal plant operation.

A coating or a coating system is qualified by testing steel panels or concrete blocks that are prepared and coated in the manner in which the coating is intended to be used in a plant situation. The panel/block preparation and coating application are documented. The coating material is documented to trace the material to manufacturer batch numbers and coating formulation data. The test shows that a specific coating system, made up of coating formula X as coat one, and coating formula Y as coat two, is qualified for use inside containment when properly applied within a certain thickness range over a surface prepared in a specific manner.

You will note from the above description that the surface preparation, coating materials and coating application parameters are tied together. A significant change in any of the parameters mentioned will make the applied coating system non-qualified or, at best, indeterminate. Some examples of qualified coating materials which cannot be considered qualified as applied to a surface inside containment are:

- EXAMPLE 1 - A coating material was qualified using test panels where the coating was applied to steel panels abrasive blasted to a SSPC SP10 near white blast with an anchor profile of 1.5 - 2.5 mils. In the field, the same coating was applied to containment structural steel that was prepared by power wire brushing. The power wire brushing produced a clean but burnished surface. Though the coating may provide adequate service under normal ambient conditions, it is likely to disbond and detach from the surface during a LOCA event since the surface was polished smooth by the power wire brush. The power wire brushing was significantly inferior as a surface preparation for coatings work when compared to the surface preparation used to qualify the coating system for use inside containment. This is a case where a qualified material was applied over a non-qualified surface preparation and this makes the applied coating system non-qualified.

- EXAMPLE 2 - A qualified coating material has been applied inside containment but the quality control inspectors did not carry out all the necessary inspections or document the surface preparation and coating application properly. Since proper surface preparation and coating application are essential to the performance of the coating, the applied coating system must be considered non-qualified. The coating system, as described in this example, would be better described as indeterminate than as non-qualified. However, there may be tests which can be performed to make this system a qualified coating system as applied. Those tests include adhesion tests, microscopic film examination, coating cure tests and destructive dry film thickness tests. If the test results are positive and there is no reason to believe the surface preparation was improper or application was not done correctly, an engineering evaluation may result in the applied coating system becoming qualified.

This distinction between a truly unqualified coating and an indeterminate coating is important when comparing Examples 1 and 2. The coating system in Example 1 is unqualified and will almost assuredly fail if subjected to the dynamic forces of a LOCA. On the other hand, it is unlikely that the coating in Example 2 will fail if subjected to the dynamic forces of a LOCA even if the tests are not carried out to qualify the applied system.

As can be seen from the two examples above, there are various reasons why a coating system may be called unqualified or indeterminant. Improper and missing QA/QC documentation are probably the most common reasons for coating systems to be placed on the unqualified coatings list. The failure mode and the type of debris generated by the failed coating will often depend on what makes the coating unqualified. The probable mode of failure of an applied unqualified or indeterminate coating system can often be assessed through an engineering evaluation. Some of the possible reasons for making an applied coating system unqualified are:

1. Use of unqualified materials
2. Inadequate surface preparation
3. Improper material mixing
4. Insufficient application temperature to achieve proper cure
5. Excessive dry film thickness
6. Recoat window exceeded
7. Surface contamination between coats
8. Lack of QA/QC documentation
9. Film thickness slightly outside of qualification test range
10. Application parameters outside of specification
11. Outstanding NCR on applied coating

Those are only some of the possible reasons for considering a coating unqualified. The first 7 items listed above would make the coating application suspect in that they could cause the coating to fail during a LOCA event. However items 8 through 11 may not cause the coating to fail by delamination, disbondment or otherwise peeling or chipping off the substrate and creating debris during a LOCA event.

The failure of a coating system to pass the flame spread, smoke generation, abrasion resistance or the decontaminability tests would make that coating unqualified for use inside containment. However if it passes the radiation tolerance and DBA tests it would not be expected to fail during a LOCA event and would not generate debris. It is important to understand the difference between a coating system being qualified and it being a safety related problem. The coating will be a safety related problem during a LOCA event if it were to form debris that could plug screens or equipment, could plug or damage the reactor core, or could result in chemical reactions that would mitigate the safe shutdown of the reactor. There are many coating systems and coating materials that would stay intact on the substrate during a LOCA event and would not be a safety related problem even though they could not pass all the qualification tests.

As can be seen from the above discussion, the reason for a coating being put on the unqualified coatings list may be critical to the expected performance of that coating during a LOCA event. An engineering evaluation may be performed to determine the impact unqualified and indeterminant coatings will have on the ECCS during a LOCA. An engineering evaluation of the indeterminate or unqualified coatings in a plant must consider plant specific requirements and becomes a plant specific process. However, such a process may be needed if the total amount of unqualified or indeterminate coating is determined to be a problem for the strainers or ECCS in general. The bottom line is still to determine the probability of a coating system failing during a LOCA event, the time into the event a failure is likely to occur, the failure mode, and how all this affects the ECCS efficiency.

2.0 Loss of Coolant Accident (LOCA)/Design Bases Accident (DBA) Test

2.1 General

Coatings are qualified for use inside containment by being put through a series of tests. Those tests are aimed at making sure that the coating (when properly applied) will perform satisfactorily and as expected during normal plant operations and during a LOCA event. The main safety concern is that the coating may fail and cause debris that will adversely affect the safe shutdown of the reactor. The coating's ability to handle the effects of the accumulated radiation doses, the temperatures, the pressures and the dynamic changes during a LOCA event is critical.

In order to test the ability of a coating to remain intact during a LOCA event, the coating is applied to steel panels or concrete blocks, irradiated and then subjected to a simulated LOCA event in an autoclave. This test exposes the coated panels to high temperatures, steam, water immersion, pressure swings and cold spray solutions. The test parameters and duration are selected to duplicate as closely as possible the conditions the coating will be subjected to inside containment during a LOCA. For original qualification, most coatings were tested using a set of temperature, pressure and the time parameters which were expected to umbrella plant specific curves. The umbrella test parameters were designed to stress the coating at least as severely as specific plant LOCA conditions would. The drywell conditions during a LOCA are so much more severe (from a dynamic forces standpoint) than are the wetwell/torus conditions that the test was limited to the drywell conditions. As long as the coating was suitable for long term immersion in high purity water and could be qualified for use in the drywell, the coating would be qualified for use in the wetwell/torus. A typical drywell test curve reaches 340°F and 70 psi, whereas the wetwell/torus conditions would not exceed 212°F and would typically remain below 35 psi.

2.2 Test Curves

Copies of the ANSI and ASTM umbrella test curves used for DBA testing of coatings for BWR drywells are attached. The attached curves require the coated panels to be exposed to temperatures as high as 340°F with abrupt changes of as much as 90°F causing thermal shock. The pressure changes are rapid and subject the coating to sudden decompression. The combination of pressure and temperature changes can be devastating to a coating system. Some containment coatings have been successfully tested to curves requiring exposure to conditions as high as 385°F. The 385°F exposure is much more destructive to coatings than a 340°F exposure.

The umbrella curves were selected to represent the conditions to which the vast majority of the containment coatings would be subjected during a LOCA. These curves do not simulate the higher temperatures and jet impingement to which a coating will be exposed in the immediate area of a pipe break. However, a very small quantity of coating would be subjected to the high pressure water or steam impingement from a pipe break. It is assumed that any coating that is within a few pipe diameters of a line break will be removed by the initial shock wave or the hot jet impingement. This assumption was partially validated in one test at Oak Ridge National Laboratory in which a series of IOZ coated test panels were exposed to direct impingement by super heated steam. In that test the steam temperature was higher than 400°F and the pressure in the autoclave reached 70 psi within 10 seconds. The IOZ was eroded from the panels by the direct steam impingement in that test. The IOZ debris was in the form of zinc powder ranging in particle size from 4 to 20μ.

2.3 Irradiation

Some of the coated panels are irradiated in a gamma radiation field prior to DBA testing. The dose rate is usually between 1E6 and 2E7 rads per hour. The standard accumulated dose used for a BWR is 1E9 rads. Some coatings have been DBA tested after irradiation to accumulated doses as high as 7E9 rads. The effects of irradiation on various coatings was previously discussed.

2.4 Pressure

By itself, the pressure exerted on the coatings inside containment during a LOCA event is not really detrimental to the coatings. The pressure does force moisture into the coatings and that moisture can be damaging. The release of that pressure can result in severe damage to the coatings. The length of time at pressure, the amount of the pressure drop, and the rate of the pressure drop all affect the way in which the coating will react to the pressure fluctuations during the DBA test and during a LOCA event. The longer a high pressure is maintained, the more difficult it is for the coating to resist delamination, blistering, or disbondment during a subsequent depressurization.

The greater the degree and speed of depressurization, the greater the chance of coating failure in the form of blistering, disbondment or delamination. The standard test curves require exposure to more rapid and greater pressure changes than actual plant curves would dictate. Therefore, by using the standard umbrella test curves instead of less severe plant specific test curves for qualifying coating systems for use inside containment, there is a built-in safety factor. This conservatism means that there is a high probability that coatings originally qualified to the standard test curves will probably survive a LOCA event intact even if they have been placed on the unqualified coatings list because they were applied under borderline conditions.

The synergistic effect of temperature and pressure can be significant with some coatings such as alkyds and vinyls which are thermoplastic and soften when heated. The moist heat in the containment during a LOCA make epoxies more flexible and, in some cases, that added flexibility allows them to blister during pressure drops without cracking or dislodging from the surface.

2.5 Temperature

The umbrella DBA curves envelope the exposure conditions at most, if not all, plants. The 6 hours at 340°F followed by a cold spray solution induced thermal shock used for DBA testing is much more severe than the more gradual temperature changes shown in most plant specific curves. The other thermal shocks built into the ANSI/ASTM umbrella curves are also more severe than the more gradual changes indicated in plant specific curves. The high temperatures used in the DBA tests affect various coatings differently. The saturated wet heat would be particularly detrimental to alkyds and other thermoplastic coating systems that may be applied as a manufacturer's standard on off the shelf equipment. Those types of coatings do not have good resistance to high temperatures or to moisture under pressure due to softening, permeation and loss of physical properties.

2.6 LOCA/DBA Dynamics

The points during a DBA test at which dynamic pressure or temperature changes occur or when cold spray hits the coated surface represent the times into the LOCA simulation when the coating is most stressed and is most likely to fail. The organic coatings become more permeable at elevated temperatures and tend to have water or water vapor penetrate into the coating during the periods when pressure is high. The epoxy coatings are not softened by the heat but have increased flexibility due to the combination of heat and moisture. Alkyds are softened significantly and the molecular bonds within the film are attacked by the heat and moisture. When the depressurization takes place and the cold spray hits the coating, the coating is subjected to a severe thermal shock, erosion and an explosive depressurization within the coating film. The greater the permeability of the coating film, the faster the film can release the internal pressure, and the less impact the depressurization has on the coating. Generally, the greater the bond strength of the coating to the substrate, the less impact the thermal shock stresses have on the coating adhesion. The other dynamic condition which will impact a coating inside containment is the movement of the substrate such as the expansion and contraction of the drywell shell plate or the wetwell/torus shell. The dynamic forces can be expected to have a significant effect on an alkyd coating or a mixed system using an alkyd based primer.

The effect of direct spray or steam impingement on the coating will vary with the distance between the coating and the steam or spray source and with the time into the LOCA event. Resistance of the film to the erosive or blast effects of steam or spray will be less if the coating is soft or if weak blisters have formed in the film. The condition of the coating film changes with the time into the event.

The key factor is that the coating will not peel from the surface and form debris unless acted on by outside forces. The dynamic forces exerted during temperature and pressure changes and containment sprays supply the stresses needed for failed coating to detach and create debris. Once the dynamic portion of the LOCA event is finished, the coating may fail by blistering or even cracking, but it is <u>highly unlikely</u> that debris will be produced. The DBA testing is often cut off after 96 hours (or even sooner) since the large temperature swings and pressure fluctuations are finished at that time. If a coating has not failed by disbondment, delamination or disintegration at the end of 4 days, that coating can be expected to remain on the surface and not form any significant debris during the remainder of the cool down period.

2.7 LOCA/DBA General Effects on Coatings

The containment coatings that have been prequalified by radiation tolerance and DBA testing are expected to be unaffected by a LOCA event except for a small amount of coating affected by direct high pressure, high temperature water or steam impingement following a line break. Coatings adversely affected by the LOCA conditions, other than a main line break, may fail by blistering, cracking, disbondment, delamination, peeling or chalking or by some combination of those failure modes.

- **Blistering** may occur during the increasing and constant pressure portions of the LOCA event. The blistering may be caused by moisture reacting with soluble salts on the substrate or within the film, water reacting with the film or substrate causing gases to be released, or water deteriorating the primer without significantly adversely affecting the tensile properties of the finish coat(s). However, blistering is far more likely to occur during the pressure drops if cold containmet spray is used to cool the drywell and the wetwell/torus. The film becomes subject to explosive expansion from the internal pressure buildup and the external pressure release. Even the more rigid thermoset epoxies can blister because the film is flexible enough to deform without cracking due to the plasticizing effect of the hot moisture permeation over time.

- **Cracking** is not a defect that would normally be expected to occur during a LOCA event unless the coating was already deteriorated prior to the LOCA event. Cracking could be caused by coating shrinkage or embrittlement or the coating's inability to expand or contract with the substrate movement. However, except for IOZ, the heat and moisture have a plasticizing or softening effect on the coating during a LOCA event which gives the coatings the resilience to prevent cracking unless crack initiation had occurred prior to the event. Even blisters that form during the LOCA event do not normally develop cracks until the coating cools down and dries out. The cooling down to ambient temperature and drying out will cause shrinkage and stresses in the applied coating which can result in cracks on or around blisters. However, by the time that occurs, the LOCA event is over and coating debris is not a safety concern.

- Disbondment or delamination of a coating could occur during a LOCA event due to a number of reasons. A coating may disbond if it was applied over an inadequately prepared surface or if the coating was misapplied resulting in a lack of surface wet-out or a generally weak bond. In those cases the coating would probably be on the unqualified coating list. A coating may delaminate between coats if the surface is contaminated between coats, if the drying times between coats are too long (or too short in some cases), or if there is an incompatibility between the materials used for the various coats. When a thermoset coating is applied over a thermoplastic primer (i.e. an epoxy topcoat applied over an alkyd primer), it is quite probable that disbondment will occur and that type of system should be on the unqualified coatings list. If a coating has been physically damaged or exposed to damaging heat or chemical environments, the coating bond to the surface may have been weakened. This could cause at least partial disbondment of the damaged coating during a LOCA event. It should be noted that delamination or disbondment may occur in small isolated areas or in large areas. The delaminated or disbonded coating may or may not dislodge from the surface and become debris. Dislodgment in sheets is unlikely other than during the major dynamic phases of a LOCA event. In the case of a system such as the epoxy applied over the alkyd, the expansion and contraction stresses exerted by the epoxy during even minor pressure or temperature fluctuations could be enough to cause severe disbondment or delamination of the softened and moisture affected alkyd primer and lead to dislodgment of the sheets.

- Peeling is usually associated with a combination of delamination or disbondment and coating film breaks. The coating must crack, be cut or otherwise break in order for it to peel. Peeling is usually associated with dislodging of the disbonded coating. Peeling during a LOCA would probably only occur during times when the coating is being impinged by the containment spray or by streams of air or steam that would exert a force behind the coating. The coating would have to be cut, cracked or split in order to allow the force of a weak stream to act behind the coating surface. A very high pressure or high volume stream could break a weak film and cause it to peel but this condition would only be expected to exist in the case of a line break.

- Chalking is usually the result of oxidation, chemical attack on the surface of the coating or the surface of the resin being degraded by UV light. UV light is not a major concern inside containment once a plant goes into operation. Oxidation is a concern only in those cases where the drywell and wetwell/torus are not under nitrogen blankets. Oxidation and UV degradation could have occurred during plant construction but the coating at startup should have been in good condition. Since the BWR containment spray is relatively high purity water, chemical attack is not expected to be a problem. Ionizing radiation could cause surface chalking of coatings but this would be significant only on coatings with very low radiation tolerance. The coating systems that would likely be found inside containment would be minimally affected. In the unlikely case where chalking is evident, the chalked surface would erode in the form of particles when subjected to steam or spray impingement or fluid/air flows. The LOCA/DBA event in a BWR would not be expected to cause the coating to chalk.

2.8 Coating Failure/Disbondment vs. Time Into LOCA/DBA

The distinction between coating deterioration and failure, as it applies to safety issues in the event of a LOCA, is crucial. As long as the coating remains on the surface and does not contribute debris to the containment, it is not a safety concern with regard to plugging strainers or reducing the efficiency of the ECCS. Hydrogen generation from zinc or other chemical side effects are not the issue here. Again we point out that all qualified containment coatings can be expected to go through a LOCA event without failure except for the areas in the immediate vicinity of a line break. The great majority of unqualified coatings that can be categorized as indeterminate would probably also go through a LOCA event without failure except for the areas in the immediate vicinity of a line break. The remainder of the unqualified coatings may fail in a manner that will produce debris.

In the case of a line break, the coatings very close (probably within a radius of a few pipe diameters) to the break that are exposed to the initial shock wave and the very high pressure, very hot jet impingement would probably fail in the first few seconds of the LOCA event. The initial thermal shock and high steam flow would likely remove loosely bonded coating at some distance from the break and could be expected to break off blisters already present in a coating film fairly close to the line break. According to Figure 4-3 in the proposed NUREG/CR-6224, the steam pressure and temperature would drop off fairly rapidly as the distance away from the line break increases. The jet impingement effects upon the coating will depend on pressure, temperature, flow rates, enthalpy, line size, type of break, etc.

High pressure water blasting is often used to clean coatings for maintenance coating work. A 3000 to 5000 psi water jet will not remove sound well bonded coatings of the types expected to be found inside a BWR containment. These types of coatings can be steam cleaned to remove oil, grease and other contaminants or to sanitize coated surfaces in food processing facilities. Hot water pressure washing is often carried out at pressures exceeding 2500 psi and at temperature of 200°F to 250°F. These cleaning methods will not remove sound IOZ or epoxy coatings. Some alkyds could be affected by the hot pressure wash but only after the coating has softened from the heat.

Coatings not in the immediate vicinity of a line break would not be expected to fail by disbondment, delamination or detachment until the LOCA event is well underway. As previously discussed, with minor exceptions, the dynamic forces are what will cause a coating to detach and form debris if a coating does fail due to the LOCA. It takes time for vapor pressure to build up within the coating film or at the film and substrate interface. It also takes time for the hot moisture to soften and break the molecular bonds in alkyds or other such coatings. Even at the 340°F temperature that the drywell might see, the deterioration of the coating is a time dependent matter. Large pressure drops will not adversely affect the coating if there is no pressure differential across the film. The coating disbondment and detachment caused by pressure buildup in or behind the film is therefore time dependent.

The six-hour mark is used in the BWR umbrella DBA test to initiate the cold containment spray and cause the first big temperature and pressure drop. Actual plant conditions will vary from plant to plant. The main point is that the time of any coating failure in the form of debris generation is dependent on the dynamics of the LOCA event. The abrupt temperature changes and the abrupt pressure drops are the points in time when the coating may detach. Those abrupt changes are somewhat dependent on the use of the drywell and wetwell/torus sprays and the temperature differential between the containment spray and LOCA ambient conditions at the time the sprays are initiated or remain in use. It would be expected that coating failures which could generate debris would occur after six hours but within the first 4 days of the LOCA event. After 4 days into the event, there is a very gradual normalization of the containment pressure and temperature. Even in the unlikely event any coatings deteriorate and even blister or partially disbond after the first 4 days, they would not be expected to form any significant debris. Some alkyd coatings in a hot solution immersion phase may deteriorate to the point of film disintegration over time but the quantity of alkyds in containment would be small to begin with and the quantity subjected to immersion would probably by extremely small if not zero. Based on the above and previous discussions of the LOCA event, coating failure modes and the relationship between the two, it can readily be seen that sound coatings (not in the jet impingement zone) would not detach from the surface until some time into the LOCA, in the unlikely event they fail at all.

3.0 Coating Failure Modes

3.1 General

Protective coatings are consumable materials with finite lives. The protective coatings are expected to provide reliable service over some specific design life. The coating materials can be expected to deteriorate over time, but proper coating maintenance may extend the reliable coating life well beyond original design requirements. Eventually the coating may need to be replaced due to aging. The inability of a coating to perform as intended under the predicted service conditions is defined as a coating failure. Coating failures can be attributed to poor coating formulation, improper coating selection, improper surface preparation, coating application errors, improper design and exterior forces. Some of the coating failure modes are:

3.1.1 **Blistering** There are many different types of blisters which occur between the substrate and the coating. Those types of blisters include solvent blisters, osmotic blisters, pressure release blisters, and blisters caused by weak adhesion. It is also possible for intercoat blisters to form between coats. The intercoat blisters are usually caused by improper application or by surface contamination between coats. Blisters usually range in size from 1/16 inch to 1/2 inch in diameter but can be much larger.

- **Solvent Blisters** occur in coatings that contain solvent. The problem of solvent blistering is common in coatings applied too thick where the solvent is unable to pass through the thicker films before it surface cures. Solvent blisters also sometimes occur when coatings are applied on very hot days where the surface of the coating can dry quickly and trap solvents in the film.

- **Osmotic Blisters** are found most often in immersion applications and are the result of moisture passing through the coating film and reacting with a contaminant such as soluble salt on the substrate. When the moisture comes in contact with the surface contaminant, the contaminant absorbs the moisture and swells or the reaction produces a gas and a blister is formed. The torus in a BWR would be susceptible to osmotic blistering.

- **Pressure Release Blisters** are likely to be formed in a BWR when the coating is subjected to an external pressure drop followed by a cold spray solution. These conditions would be typical of what is expected during a LOCA event. The moisture permeates into the coating under pressure until the pressure across the film is equalized. When the pressure outside of the film is reduced quickly, the pressurized vapor inside the film expands and forms a blister if the vapor cannot pass back out of the film rapidly enough.

- **Blisters caused by weak adhesion** to the substrate can occur in any coating and in any service condition. Improper surface preparation is usually the cause of weak adhesion blisters.

- **Application Blisters** occur at the time of application. This can be due to entrapped air or solvent absorption into the substrate or undercoat. The solvent can then cause a blister from vapor pressure if it cannot be released. Entrapped air can expand from the heat of reaction during curing or later when exposed to elevated temperatures in service. The blistering that occurs when a full coat of organic coating is applied over an inorganic coating is a good example of application blistering.

- **Intercoat Blisters** are usually caused by contamination between coats. The contamination can be water, soluble salts, oil etc.

3.1.2 **Cracking** is a serious type of failure that results when stresses in the film exceed the tensile strength of the coating. Cracking is caused by film shrinkage, differential expansion and contraction of the coating and the substrate or exposure to elevated or very low temperatures. Cracking is usually a sign of film embrittlement. Cracking can also occur in blisters as they dry out and shrink. Cracking is most commonly seen after aging or outdoor exposure.

3.1.3 **Disbondment** is the loss of adhesion of a coating or a coating system to the substrate. Disbondment occurs when the tensile strength of the coating exceeds the bond strength of the coating and the coating is acted on by outside forces. Disbondment often results when the coating thickness is excessive or where there is inadequate surface preparation (surface cleanliness, profile, etc.). Lack of adhesion can contribute to the failure of a coating by blistering, peeling and flaking. Disbondment is a very serious coating failure inside a BWR drywell or wetwell / torus if the coating becomes dislodged from the substrate. Where the coating is used for corrosion control, the disbonded / dislodged coating leaves the substrate exposed to potential corrosive attack. In a BWR, the dislodged coating forms debris that can potentially adversely affect the operation of the ECCS during a LOCA.

3.1.4 **Delamination** is similar to disbondment / dislodgment except it is a loss of adhesion failure between coats or within the film instead of a loss of adhesion to the substrate. A delamination failure occurs when the coatings bond strength to the substrate exceeds the cohesive strength or the adhesive strength between coats and the coating is acted on by outside forces.

3.1.5 **Disintegration** is a loss of coating film integrity and occurs when the resin or other components of the film are deteriorated to the point the film looses its cohesive strength and comes apart. Disintegration of the coating film can result from heat or chemical or water attack. Protective coatings exposed to high heat can disintegrate when the coating film is carbonized or sintered to the point where the film's tensile strength is lost. Chemicals or water will attack the resin, fillers or pigments. High purity water at elevated temperature will dissolve the silicate binder in the IOZ and may destroy an alkyd coating.

3.1.6 <u>Debris Characteristics</u> When a coating fails and detaches / dislodges from the surface, it becomes debris. Coating debris can be in the form of large sheets, small sheets, chips or particles. There are no clear cut definitions of these terms but generally they can be described as follows:

1. Large sheets are generally described as being ≥ 1 square foot in area.
2. Small sheets can be described as having a surface area < 1 square foot and ≥ 1 square inch.
3. Chips can be considered as having a surface area < 1 square inch and ≥ 100µ.
4. Particles can be described as < 100µ in size.

- <u>Large sheets</u> are rarely seen as debris for the types of coatings used inside containment. They are more appropriately associated with elastomeric urethanes and similar systems. The thick film aggregate filled or reinforced floor coatings could detach in sheets > 1 square foot in area if the surface preparation is inadequate. This could also happen with the thicker film concrete wall coatings. However, the large sheets would quickly break up unless the coatings are fiber or otherwise reinforced.

- <u>Small sheets</u> would not be common debris as a result of the unlikely failure of unreinforced normal containment coatings. If failure in small sheets did occur, they would tend to break down into chips as they impacted objects or surfaces during free fall or transport.

- <u>Chips</u> are the most common debris generated by containment coatings that fail and especially by those that fail during a LOCA.

- <u>Particles</u> normally result from the disintegration of the coating film. The destruction of the resin or film matrix releases the fillers, pigments or other solid particles such as the zinc spheres in the case of IOZ coatings.

Blisters, if broken, would tend to be released as small chips 0.125" to 1.0" in diameter or width. As a blister grows the film over the blister thins and the thinning film looses its tensile properties. With most containment coatings the adhesion is high so the blister generally grows vertical to the surface rather than laterally with the plain of the surface. When the force behind the blister exceeds the tensile strength of the thinning coating film over the blister, the blister breaks and releases the pressure. That will normally happen while the blister is less than 1" in diameter with well bonded containment coatings. The cracked blister would still remain intact unless acted on by outside forces that are strong enough to break the cracked blister off the surface.

Generally, the more brittle the coating, the smaller the debris produced by a failure of the coating. IOZ would be expected to produce small chips or particles. The chips would quickly break down due to the relatively weak cohesive bonds of the IOZ film. Failed epoxy coating would be expected to produce chips or small sheets since epoxies have good tensile strength and are somewhat flexible during a LOCA event. Alkyds would also be expected to produce small debris unless over coated with a material having a high tensile strength such as an epoxy. When heated, the alkyd would become soft and weak so the epoxy could sheet off in small or even large sheets.

3.1.7 <u>Specific gravity (SG)</u> is another characteristic of coating generated debris that is as important as size when considering the effect debris will have in a BWR. The SG of a coating will depend on many factors including the base resin material, fillers, extenders and pigments. The lightest molecular weight resin normally found in a BWR containment is in the epoxy coating systems. The SGs that can be used as a minimum number for the various coatings inside containment are:

Resin	Specific Gravity
Epoxy	1.43
Alkyd	1.50+
IOZ	2.50+

Configuration and SG are the two main characteristics of containment coating debris that influence how the debris will be transported in the drywell and the wetwell / torus during a LOCA event. Debris transport is discussed in some detail in Section 5 of this report.

3.2 Probable Coating Failure Modes During LOCA

The following discussion addresses the probable failure modes of the coating systems described in 1.1 if they fail at all during a LOCA. Where appropriate, the failure of the coating is addressed for both the drywell and the wetwell/torus. Where appropriate, a differentiation is made between qualified and unqualified or indeterminate coatings. Expected debris characteristics are also discussed for each coating system.

- **Untopcoated IOZ**

  In the case of a line break, direct high pressure, high temperature steam or water impingement would be expected to disintegrate the IOZ by dissolving the water glass which forms the film matrix. This would be the case for a distance out from the break where the steam or water temperature exceeds 400°F. In addition to temperature, the impinging forces and the exposure time would be two factors affecting the rate of film failure. The debris generated would be in the form of spherical zinc particles having a specific gravity (SG) >2.5. The zinc particles would probably be in the 4 to 20μ size range. This initial loss of IOZ during a line break would occur during the first few seconds. Areas further away from the break would be affected to a lesser extent as the impingement temperature and force diminishes. There would probably be little or no jet impingement effect on the IOZ coating beyond the immediate pipe break area since the deterioration of the IOZ film matrix is time, temperature and force dependent The high temperature, high pressure and jet impingement is expected to last only for seconds and not minutes. Jet impingement would be of concern only in the drywell.

  The IOZ in the drywell would undergo very gradual surface erosion in areas subjected to low pressure steam or hot water sprays or flows. This gradual erosion in low temperature and low impingement force areas would only happen over days, weeks or months, and not in the very early stages of a LOCA event. This gradual erosion would produce a small amount of the zinc particles described above. This type of erosion could also be expected in the wetwell/torus in the later stages of the LOCA event. However, the much lower temperatures in the wetwell/torus would result in the gradual deterioration of the IOZ matrix well after the blowdown from the initial pressure surge. The turbulence in the wetwell/torus would not be high after blowdown.

The failure mode for the IOZ could include some small flakes that would very rapidly break up into particles or very small pieces. The size of the very small pieces would probably be much less than 0.060 inches across. The small chips or flakes would result only where the IOZ was disbonded, if such areas existed. A tightly bonded IOZ would erode by powdering and would not flake or chip off the surface.

- **IOZ Topcoated with Epoxy (IOZ/epoxy)**

When epoxy topcoats are used with an IOZ primer, they protect the IOZ from the erosive forces of the steam and water in the drywell and wetwell/torus during a LOCA. However, the topcoated system is more prone to fail by delamination because the tensile strength of the epoxy is so much greater than the tensile strength of the IOZ.

If the IOZ/epoxy system is used in the wetwell/torus, there could be blistering over time in normal service. Blistering due to the LOCA event could occur over a period of time in the vapor zone due to moisture permeation of the epoxy coating under pressure. Blisters formed during the LOCA event would be expected to remain intact on the surface since the dynamic forces in the vapor area are not great after the initial blowdown. Existing blisters could break and detach from the surface in the immersion zone by the force of the blowdown during the initial stages of a LOCA event. The debris would likely be in the form of epoxy paint chips with IOZ adhering to the back of the chips. The chips would probably initially be in the size range of 0.125" to 1.0 or 2.0" in width or diameter. It is not likely that the detached chips in the immersion zone of the wetwell/torus will be further broken down since they would be relatively flexible to begin with. The specific gravity (SG) of the IOZ/epoxy chips would probably be in the vicinity of 1.7 to 2.0 which is greater than the SG of epoxy and less than the SG of IOZ. The IOZ/epoxy system is not generally used in the wetwell/torus in the United States. We are aware that it has been used as a belly band coating in the water line area of a torus at one BWR power plant. That particular system was in excellent condition after the first 8 years of service including exposure to high pressure water washes for decontamination purposes.

The IOZ/epoxy coating in the drywell hit by high pressure/high temperature steam or water impingement during a line break may be affected to a greater extent than described above for the untopcoated IOZ. The epoxy does not have the heat resistance of the IOZ and is more susceptible to thermal shock. Due to the difference between the expansion and contraction properties of the epoxy and those of the IOZ, coupled with the lower tensile strength of the IOZ, the IOZ/epoxy system may possibly fail in larger sheets than would a straight epoxy system. The IOZ/epoxy system would likely fail through a splitting of the IOZ primer if a failure were to occur at all. This would leave zinc on the substrate and on the back of the epoxy coating. The lower resistance of the IOZ/epoxy system to impact and temperatures would probably cause the IOZ/epoxy system to become debris 50% further out from a line break than would be the case for an untopcoated IOZ. The initial form of the debris would likely be in chips or small sheets which would quickly break up from impacting against surfaces during the initial blow down. There would also be a small quantity of zinc particles generated. The debris reaching the wetwell/torus during the blowdown would probably be smaller than the debris created in the wetwell/torus and described above. The total quantity of the coating affected by the line break impingement would be small. The exact quantity will naturally depend on the condition of the IOZ/epoxy system at the initiation of the LOCA event and on the impingement forces involved.

An IOZ/epoxy system outside of the impingement zone in the drywell could develop blisters and could subsequently disbond. The blisters would form over time as the moisture permeates through the epoxy coating. Disbondment could occur during rapid pressure drops or temperature changes. The blisters would be larger than those that could be expected to appear in a straight epoxy system since the cohesive strength of the IOZ is weaker and the tensile strength of the epoxy will tend to hold the topcoat film together even though it is disbonded. The blisters would be expected to remain intact without detachment unless acted on by outside forces such as water spray impingement or fairly strong water flows. If the blisters were to detach from the surface, the size of the debris would likely be in chips greater than 0.125" to small sheets. It is unlikely that very small pieces of the coating film would break off unless the epoxy was deteriorated and embrittled prior to the LOCA event This is due to the normally high tensile strength of the epoxy and the flexibility provided by the moisture permeation.

The debris characteristics will certainly depend to a great extent on the condition of the coating at the time of a LOCA event. Normal aging and exposure to high heat or high radiation doses during plant operation would tend to make the epoxy topcoat brittle with time. The more brittle the epoxy topcoat, the smaller the size of the debris that is likely to be produced during a LOCA event. This is especially true at the initiation of the event before the heat and moisture have a chance to make the epoxy more flexible or pliable. Once heated and permeated by moisture, the epoxy coating would be less brittle and would tend to detach in larger pieces. Our assumption in most of the above discussions is that the coating is in good condition at the initiation of the LOCA event with no signs of major deterioration such as cracking, blistering, disbondment or delamination. The amount of debris created during the LOCA event can be expected to increase as the condition of the coating system deteriorates prior to the event.

There should be little or no failure of a qualified IOZ/epoxy system during a LOCA event except in the immediate area of a line break. An unqualified IOZ/epoxy system may or may not fail depending on what makes it unqualified and what condition it is in at the time of the initiation of the LOCA event. The probability of failure and the debris characteristics for an unqualified IOZ/epoxy system will also depend on the reason for the system being considered unqualified. For example, if the system is unqualified because of improper application and a lack of proper IOZ cure, the probability of failure by delamination during a LOCA event is high and the detached sheets could be large. In another case, a high quality epoxy that is not a qualified material may have been properly applied over a qualified, properly applied and cured IOZ primer. That second system, though not qualified, could be expected to act much in the same manner as a qualified system would during a LOCA event.

- **Epoxy**

In the event of a line break, the epoxy coatings will be more susceptible to impingement damage than an inorganic zinc. The thermal shock resistance of the epoxy is much lower than that of IOZ as is the peak temperature resistance. The thermal shock will have the most detrimental effect at the initiation of a line break. Within a few seconds after the initial thermal shock and pressure wave, the coating will heat up and be more susceptible to the effects of jet impingement. As the distance from the break increases, the adverse effects of the jet impingement will drop off fairly rapidly since temperature is the main deteriorating influence which makes the epoxy susceptible to the impingement process. The epoxy coatings qualified for use inside containment will withstand very erosive forces at ambient temperatures as can be proven with their resistance to high pressure water blasting. They even have good resistance to high pressure steam cleaning. The impingement from a line break would tear off the epoxy in small chips and pieces before large sheets can form. After the initial few seconds and the epoxy heats up, it will be eroded in areas of high pressure, high temperature jet impingement. The pieces would probably be less than 0.125" in size.

The epoxy has lower bond strength to concrete than to steel and the concrete has a relatively low tensile strength. The size of the debris would probably be larger for epoxies applied to concrete than for epoxies applied to steel. However, there is a high probability that many of the chips or pieces of coating would have concrete stuck to the back. This would increase the SG of the chips. Epoxy coatings on concrete walls in the drywell impacted by high pressure, high temperature jet impingement would probably produce smaller pieces of debris than floor coatings. The epoxy floor coatings are usually high build systems and can be fiber reinforced or aggregate filled. The floor coatings normally have high tensile strengths and would tend to hold together in larger pieces. These highly filled floor coating systems would require higher impingement pressures or more destructive outside forces in order to fail than would the thinner less filled wall coatings.

Epoxies inside the drywell that are in good condition at the initiation of a LOCA and that are not subjected to jet impingement, would be expected to survive a LOCA intact. This would be the case even if they blister or otherwise deteriorate during the LOCA event. If the epoxy systems do fail, they would be expected to produce chips. They would produce small sheets only if the coating adhesion to the substrate was poor to begin with. In order for the epoxy coating failure to produce debris it would have to loose its tensile strength. The containment spray system and resultant water flows over equipment , down walls or across floors will not be strong enough to overcome the tensile strength of the epoxy film unless the film has been weakened or is torn or otherwise damaged.

Deteriorated epoxy coating or unqualified epoxy coating systems having poor adhesion, could fail in sheets but, even then, large sheets would not be expected. Unless the coating was deteriorated prior to the LOCA or the coating is known to have poor adhesion or other problems, an unqualified epoxy coating can be expected to survive a LOCA intact without becoming debris.

Aged or embrittled epoxy coatings in the drywell may crack from thermal shock during a LOCA if the adhesion and film tensile strength have been significantly weakened. This could result in chips detaching from the surface.

Epoxy coatings in the wetwell / torus would be less effected by the initial blowdown than drywell coatings unless the coatings are already failing. The main effect of a line break on the coating in the wetwell / torus would come from the turbulence of the water during the pressure release blowdown. This would not be expected to have much effect on epoxy coatings that are in good condition at the time of the LOCA initiation. If already blistered, the coating blisters could be broken and detached, but even that is unlikely for small tight blisters. The blisters would have to be fairly large and therefore weak to be smashed by the water turbulence. This would produce large chips greater than 0.125" and probably greater than 0.5" in size. There would probably be little or no impact on the epoxy in the vapor phase of the wetwell / torus that could cause that coating to be released as debris. If the coating was badly deteriorated to begin with, the debris would probably be in the form of chips or small sheets. The adhesion of the epoxy coatings in the wetwell / torus can be expected to be lower because of the moisture permeation into the coating during normal plant operations. However, the decrease in bond strength is relative since qualified epoxies have extremely high initial bond strength. Also, the decrease in bond strength is compensated for by an increase in flexibility which will help the coating film resist breaking and subsequent detachment.

- **Alkyd**

The alkyd coatings can be expected to soften and deteriorate fairly rapidly from the high temperatures during a LOCA. The film will loose its tensile strength and the adhesive strength will be reduced. There should be no alkyd coatings inside the wetwell / torus. If used at all, alkyds are most likely to have been applied as manufacturer's standard coating systems on equipment such as motors, hoists, small valves and electrical parts such as solenoids.

Jet impingement is expected to be more destructive to the alkyds than to the epoxy or IOZ coatings discussed above. This is due to the destruction of the bond and even the resin matrix at high temperatures. The initial thermal shock and pressure wave would remove the alkyd paint further out from a line break than that noted above for epoxy. The zone affected would probably be twice as big as that for the IOZ. After the initial few seconds, the continued hot jet impingement would rapidly soften the alkyd film and erode the resin. The first shock would probably cause failure in small flakes. The continued erosion during the hot high pressure impingement could cause small soft pliable pieces to come off along with particles as the resin erodes.

In areas of the drywell not subjected to jet impingement, the alkyd would be expected to be severely deteriorated due to the high temperatures. The coating adhesion would become very weak and the resin would soften. The coating could be removed by even minor impingement from containment sprays. The radiation released during the LOCA would also contribute to the deterioration of the alkyd resin.

If alkyd coatings are used as primers for epoxy topcoats, the weakening of the alkyd primer adhesion along with its softening would probably cause the coating system to disbond in small or even large sheets. The epoxy would tend to hold the film together but the sheets would breakup readily after detachment and would probably end up as chips like those described above for the epoxy coatings.

All of the alkyd coatings are unqualified. However, as mentioned above, there should only be a small amount of alkyd coating in the containment of BWR power plants.

4.0 Failed Coating Quantities

4.1 Qualified Coatings

The quantity of debris from failed qualified coating is expected to be small during a LOCA at BWR power plants. The coatings that have been qualified through testing and are properly applied should not fail or create debris except for the small amount of coating removed by jet impingement during a line break.

The quantity of debris resulting from jet impingement of qualified containment coatings has been estimated for three representative coating systems. The jet impingement area has been bounded by assuming a 24 inch unrestrained pipe break removes 100% of the containment coating from the drywell wall at a distance of 20 feet (10 pipe diameters) from the break. The 10 pipe diameters is conservative. The model employed to calculate the jet structure is documented in Appendix C of ANSI/ANS 58.2-1988, "Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture". This is a conservative approach since the area affected for an unrestrained break (jet is a cone which expands at a 10 degree half-angle) is larger than that affected for a restrained break (jet is an expanding cylindrical section centered on the break and extending 360 degrees radially). For the assumed bounding parameters the diameter of the jet at a distance of 20 feet from the unrestrained break is 19.6 feet (302 square feet in area). Assuming that there could be pipe hangers, structural steel, valves or other coated items in the jet path within the cone, we have conservatively doubled the surface area of affected coating to 604 square feet. This very conservative figure of 604 square feet represents the worst case situation.

- For the IOZ coatings, the film thickness will average less than 5 mils (0.005 inch). A film thickness of 0.005 inch and a SG of 3.0 were conservatively used in quantifying the IOZ debris. Based on those parameters, the volume of the total failed IOZ would be 0.2516 cubic foot and the mass would be approximately 47 lbs. Some of the silicate binder will dissolve and not end up as debris but this has not been factored in to keep the quantity on the very conservative side. Debris could be expected to consist of zinc particles of 4-20μ size and some small chips that would probably be less than 0.060 inch across.

- For the IOZ top coated with epoxy, the film thickness will average less than 0.005 inch for the IOZ and less than 0.008 inch for the epoxy. In reality, the average thickness would probably be closer to 0.0035 inch for the IOZ and 0.006 inch for the epoxy. Using the conservative figures of 0.005 inch and 0.008 inch, the volume of the failed IOZ/epoxy coating system would be 0.65 cubic foot. Using a SG of 3.0 for the IOZ and a SG of 1.5 for the epoxy, the mass of the failed system would be approximately 85 lbs. The debris would likely consist of chips of epoxy paint ranging in size up to 0.125 inch across and some loose zinc particles. The chips of epoxy paint would probably have zinc adhered to the backside.

- For 100% epoxy coating, the maximum film thickness will average less than 0.015 inch for all but concrete floors. The concrete floors are located in the bottom of the drywell and should not be subjected to the jet impingement. Based on this film thickness, the volume of the failed epoxy will be 0.755 cubic feet. Using a SG of 1.5 for the epoxy coating system, the mass would be approximately 71 lbs. The debris would likely consist of small chips and pieces less than 0.125 inch across. Any chips from vertical concrete surfaces would be expected to have concrete stuck to the underside.

4.2 Unqualified Coatings

All coatings in the near proximity of a line break will fail regardless of their qualification status. The unqualified coatings affected by jet impingement could be expected to produce debris similar to that described above for qualified coatings. There could also be a small amount of alkyd paint involved. The alkyd would not add to the above figures since it would be in lieu of some of the other coatings and would have less volume and, therefore, a smaller mass than the epoxy. Therefore, the quantity and mass shown above for the qualified coatings represents a worse case situation for debris generated by jet impingement.

Unlike the properly applied qualified coatings which can all be expected to survive a LOCA intact beyond the jet impingement zone, some of the unqualified coating can be expected to fail outside of the jet impingement zone. The quantity of unqualified coating is plant specific. As previously discussed, the probability of failure will depend on what makes the coating system unqualified. Experience has shown that much of the coating that is called unqualified in most plants would better be categorized as indeterminate. Experience has also shown that much of that coating is called unqualified due to an inadequate paper work trail and not because the coating is known to have been misapplied or is actually expected to fail during a LOCA. Engineering evaluations at various nuclear power plants have shown that very little of the debris generated by failed coatings will ever reach the strainers or screens after the initial line break surges are equalized. The initial thermal shock and steam flow would likely remove loosely bonded unqualified coatings at some distance form the break, and could be expected to break off some blisters already present in the coating film fairly close to the line break. The amount of debris generated from these unqualified coatings in the early stages of a LOCA is expected to be small compared to the above bounding estimates for jet impingement debris.

Any other unqualified coatings that are not in the immediate vicinity of the line break would not be expected to come off the surface until many hours into the LOCA. It is very reasonable to assume that most of the unqualified coating will remain intact on the surface during a LOCA and will not form debris. In the unlikely event unqualified coatings fail in a manner to cause debris after the line break forces have dissipated, such failure would be expected to occur between 6 and 96 hours into the LOCA event. The coating damaging dynamic forces within the containment are essentially spent after the first 96 hours and it is very unlikely that coating debris would be generated after that time. Because of the low recirculation flow rate, most of the coating debris (chips, flakes and particles) generated after the first few minutes of the LOCA event would be expected to settle to the bottom of the drywell and would not be subject to transport to the suppression pool or torus. If coating debris reaches the suppression pool or torus, most would likely settle out on the bottom before reaching the suction strainers.

In order to be more specific on the quantity of failed unqualified coating that would result in debris during a LOCA after the initial jet impingement damage has occurred, a plant specific engineering evaluation would have to be carried out. Unqualified and indeterminate coatings would have to be identified and then evaluated. Only then can a reasonable prediction be made as to which of the unqualified coatings will probably fail and how. It is very unlikely that even the unqualified coatings will add appreciably to the debris reaching the suction strainers beyond that described for the line break event, which is extremely conservative to begin with.

5.0 Failed Coating Debris Transport

The failure modes of the various coating systems generally used inside a BWR containment are discussed in Sections 1.0, 2.0 and 3.0 above. The probable makeup of debris generated by a coating failure is also discussed in detail above and especially in Sections 3.1 and 3.2. The discussion in this section will concentrate on possible ways in which the failed coating debris might move around the drywell and wetwell / torus during different stages of a LOCA event. There have been a number of plant specific evaluations carried out that included coating debris transport. A lot of that data is generic. A paint chip, sheet or particle will be transported in a similar manner regardless of plant configuration as long as the air velocities, water velocities, etc. are equivalent. The transport mechanisms are the same. Only the transport parameters are different.

How much debris is transported and where it ends up during a LOCA at a BWR may vary from plant to plant. Debris transport depends on factors such as:

- Force of initial blowdown
- Presence and configuration of down comer screens, baffles and covers
- Timing of coating debris release
- Water flow rates on drywell floor
- Path between debris generation point and down comers or screens
- Water flow in wetwell / torus
- "Sail Area" of paint sheets or chips
- SG of paint particles
- Water temperature
- Water depth in wetwell / torus

The coating debris will be transported differently in the drywell as compared to the transport of coating debris in the wetwell. Therefore, the following discussion of coating debris transport is divided into two parts.

5.1 Debris Transport In The Drywell

Most coating debris generated in the initial stages of a LOCA during the blowdown will be carried with the general movement of the high velocity flow of steam and air toward the vent openings or down comers that feed the wetwell / torus. The quantity of coating debris would be relatively small at that stage of a LOCA and would be restricted to coatings that already failed prior to the initiation of the LOCA and coatings destroyed by the jet impingement and pressure wave. Sheets would tend to break up on grating and from impact with equipment. The flakes of paint could be expected to be carried downward in the drywell. Most of the chips created in the drywell could end up in the wetwell / torus during this initial blowdown.

Once the initial blowdown is over, the air movement in the drywell will subside and paint chips would fall due to gravity as they are produced from detachment of failed coating. Sheets would tend to float down like a piece of paper and would get hung up on equipment and grating. Coating chips from the vertical walls or liner plate may be carried down the surface by containment spray water or condensation flowing down the walls. The chips of coating will tend to get heavier as the time into the LOCA increases. This is due to moisture entering the coating which is time and pressure related. Coating chips that disbond due to substrate corrosion would probably also be heavier than the plain coating chips because they would probably include corrosion products adhered to the chips.

Once sheets, chips or particles of coating debris reach the drywell floor they must be carried over the floor to the down comers or vent openings in order to reach the wetwell / torus. This will require the coating chip to be carried in water flowing across the floor. If dry when it falls, the chip could float or sink when it hits the water. Some of the factors that dictate whether it sinks or floats are:

- Chip / sheet configuration and surface area
- SG of coating chips
- Orientation of chip / sheet when it lands
- Water turbulence or flow
- Water temperature (cooler water is more buoyant)

If the chip / sheet floats it will be carried along with the water flow until it breaks the water surface tension and starts to sink. Once it starts to sink a number of factors again come into play which dictate how far the chip will travel in the water. Those factors include:

- Water velocity
- "Sail Area" of paint chip / sheet
- SG or relative buoyancy
- Hydraulic drag
- Water absorption of paint
- Water temperature
- Water depth

A paint chip / sheet sinking in flowing water will accelerate until the frictional drag force equals the net gravitational and buoyancy forces. There are 4 different types of motion possible. The chip / sheet can "float", sail, tumble or slide. Of those, the most common transport method for coating chips in water is tumbling.

If the chip / sheet falls to the floor before the initiation of the containment sprays and the chip / sheet is resting on the floor when the water starts to flow, the water must overcome the initial inertia of the paint chip to start it moving. The density of the lightest coating normally found in containment is 90 lbs./ft^3 for epoxy The following is a table showing threshold slide velocities which are required for water to initiate the movement of paint chips of various sizes and densities.

SLIDE VELOCITIES IN FT / SECOND

CHIP SIZE	DENSITY				
	90 lbs./ft^3	100 lbs/ft^3	120 lbs./ft^3	150 lbs./ft^3	200 lbs./ft^3
2"	1.02	1.17	1.44	1.76	2.20
1"	0.72	0.83	1.02	1.25	1.56
1/2	0.51	0.59	0.72	0.88	1.10
1/4	0.36	0.42	0.51	0.62	0.78
1/8	0.25	0.29	0.36	0.44	0.55
1/16	0.18	0.21	0.25	0.31	0.39

Paint chips have a tendency to settle out in calm water. A paint chip 0.10" x 0.10" in surface area with a 6 mils thickness will settle vertically in calm water at approximately 0.064 feet per second. If the water is moving the chip will travel at a downward angle. The horizontal movement will depend on the water velocity and the water depth. These figures can be used to estimate how far away from the down comer / vent opening a chip must land in order to be transported to the wetwell / torus at various times into a LOCA. The 0.10" x 0.10" x 6 mil epoxy chip will travel 0.2 feet laterally and drop 0.064 ft. vertically each second in water moving at 0.2 feet per second. If the water is 2 feet deep, the paint chip will require approximately 31 seconds to hit bottom and will have traveled just over 6 feet horizontally. This is just an example of chip movement in the water. As stated above, water temperature and other factors come into play so that each case may be slightly different.

5.2 Debris Transport in the Wetwell / Torus

Any coating chips or particles that are in the wetwell / torus during the blowdown will be lifted and distributed throughout the water by the turbulence. Paint debris entering the wetwell / torus during the blowdown will also be distributed throughout the water. The movement of the coating chips will then depend on the flow of the water in the wetwell / torus. The ring girders and other obstructions in a torus and columns or other supports in a wetwell will influence the movement of coating chips as they settle. Movement of chips / sheets on the surface of the water or in the water will be governed by the same laws of physics and fluid dynamics that are at play in the drywell. The flow rates, water turbulence, chip / sheet size and SG, strainer / screen locations and other factors will help determine how much of the paint debris reaches the strainer / screens in the wetwell / torus. The heavy small spheres of IOZ would settle out very rapidly where as a 0.10" square chip of epoxy would settle relatively slowly by comparison.

6.0 Conclusion

This report summarizes the expected performance of typical containment coatings during a LOCA event including a line break. This engineering evaluation concludes that the quantity of coating debris is expected to be small during a LOCA at BWR power plants. This conclusion is drawn based on the following:

- Properly applied coatings that have been qualified through testing should not fail except in the close vicinity to a line break where the coating is subjected to jet impingement.
- The conservatively estimated bounding quantity of coating debris resulting from jet impingement ranges from 47 lbs for untopcoated IOZ coatings to 85 lbs for IOZ coatings top coated with epoxy.
- The amount of debris that might be generated by unqualified coatings that are close to a line break location is expected to be small compared to the bounding estimates for the jet impingement debris.
- Other unqualified coatings that are not in the immediate vicinity of the line break would not be expected to produce debris before several hours into the event in the unlikely event they fail by detaching from the surface. Because of the low recirculation flow rates at that time, that coating debris would be expected to settle to the bottom of the drywell and not be transported to the suppression pool or torus. Even if coating debris were transported to the suppression pool or torus at that time, it would most likely settle out before reaching the suction strainers.

Since the total quantity of the containment coating debris is expected to be relatively small, the safety significance of the coating debris with respect to ECCS pump suction strainer performance will be minimal when compared to the iron oxide sludge that is inherently present in the suppression pool or torus at BWRs in the United States.

TABLE I

COATING TEST REQUIREMENTS

TEST	ANSI	ASTM
Radiation Tolerance	N 512	D 4082
DBA	N 101.2	D 3911
Decontaminability	N 512	D 4256
Adhesion	N 512	D 4541
Flame Spread	N 101.2	E 84
Smoke Generation	N 101.2	E 84
Abrasion Resistance	N 512	D 4060
Chemical Resistance	N 512	D 3912
QA / QC	N 101.4	D 3843
Test Sample Preparation	N 512	D 5139

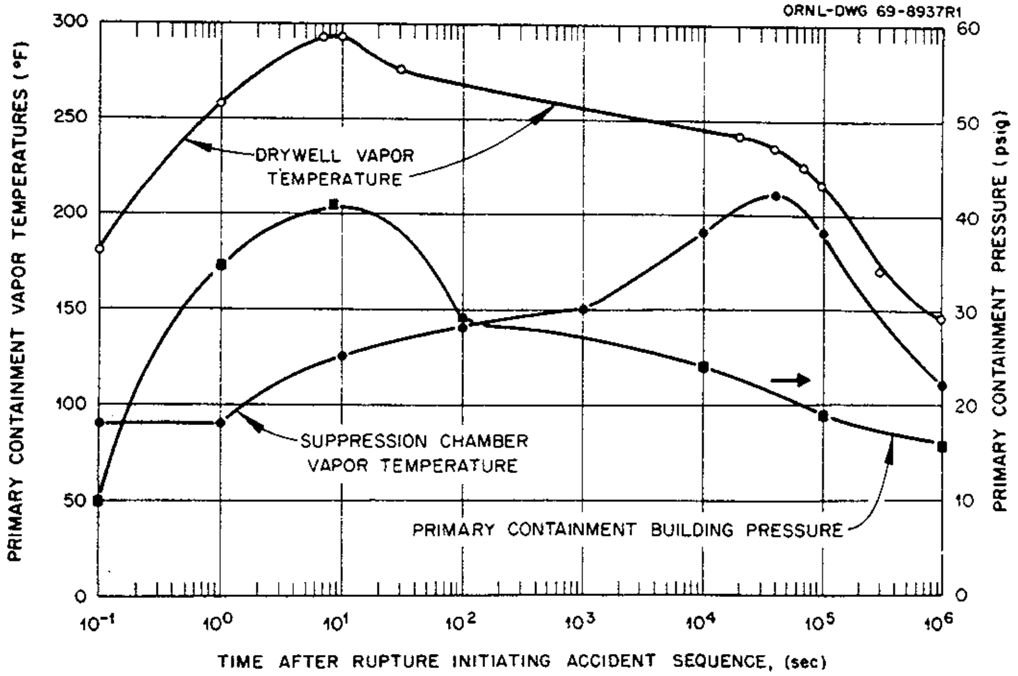

Fig. 2. Typical DBA curve for BWR containment facilities showing temperature and pressure vs. time.

During a DBA, protective coatings may be exposed to chemical spray solutions. Tests of the coatings for a specific application shall be conducted with the solution to be used in that application. Spray solutions may contain one or more of the chemical additives shown in Table 3. (Other chemical additives may be added in the future.) The chemical composition of spray solutions used in the coating test shall be determined before each test. The results shall be documented to verify the composition of the spray solution used.

2. FIRE EVALUATION TESTS

2.1 *Purpose and Scope.* This section provides a standard procedure for testing and quantitatively evaluating coating systems for fire-retardant properties. The scope includes

2.1.1 *Preparation of Test Specimens.*

2.1.2 *Flame Spread Test.*

2.2 *Definition of Terms.* See Section 8, Glossary of Terms.

2.3 *Preparation of Test Specimens.* All test substrates shall be noncombustible and shall be sized according to the requirements of the test methods in Subsection 2.4.

The protective coating shall be applied in the manner and at the recommended film thickness prescribed by the manufacturer. Following the application of the last coat, each specimen shall be cured for 30 days at 70° ± 5°F. at a relative humidity of between 50 and 69 before being tested.

2.4 *Flame-Spread Tests.* Flame-spread tests shall be conducted and evaluated in accordance with ASTM E 84, *Standard Method of Test for Surface Burning Characteristics of Building Materials*, Part 14, American Society for Testing and Materials, Philadelphia, Pennsylvania 19103, or with other methods which would give comparable results. The permissible flame-spread rating shall not exceed 50 as measured on the noncombustible substrate.

3. THERMAL CONDUCTIVITY DETERMINATION (See Appendix A)

Thermal conductivity for any coating system in question shall be determined by the methods described in the Appendix, either A.4 through A.7 or, in accordance with ASTM D 2214, *Tentative Method of Test for Thermal Conductivity Constant with the Cenco-Fitch Apparatus*, Part 15, American Society for Testing and Materials, Philadelphia, Pennsylvania 19103, A.8 through A.11.

Table 3. Typical Spray Solution Additives

Chemical compound	Concentration	Purpose
Sodium borate	2,000 to 4,000 p.p.m. boron	Reactivity control
Sodium thiosulfate *	≤ 2% by weight	Reactant for iodine
Sodium hydroxide†	≤ 0.2 normal	Alkaline reactant for iodine
Boric acid	2,000 to 4,000 p.p.m. boron	Reactivity control

*In conjunction with boric acid, with or without sodium hydroxide.
†In conjunction with boric acid, with or without sodium thiosulfate.

lowing a loss of coolant accident that would expose the coated surface of the containments of a light-water nuclear power facility to the temperature-pressure environmental parameters described.

3.1.14 *peeling*—separation of one or more coats or layers of a coating from the substrate.

3.1.15 *pressurized-water reactor (PWR)*—a nuclear power reactor design utilizing liquid water under high pressure as moderator-coolant.

3.1.16 *quality assurance*—the verification of the conformance of materials and methods of application to the governing specifications, in order to achieve the desired result.

3.1.17 *reactor containment (containment)*—the enclosure provided to protect the environment from the consequences of a nuclear incident.

4. Significance and Use

4.1 This test method is designed to provide a uniform test to determine the suitability of coatings used inside primary containment of light-water nuclear facilities under simulated DBA conditions. Variations in actual surface preparation and in application and curing of the coating materials may require additional testing as deemed necessary by the specifying or qualifying agency, or both, if it is anticipated that the variations may adversely affect the performance of the coating system during a DBA. This test method is intended only to demonstrate that under DBA conditions, the coatings will remain intact and not become debris which could compromise engineered safety systems.

5. Apparatus

5.1 *Environmental Test Chamber*, constructed of materials that are corrosion-resistant to the test solutions.

5.2 The equipment shall be capable of reproducing and continuously recording the temperature and pressure profiles of the DBA conditions.

5.3 A sufficient number of thermocouples shall be located in the test chamber to assure conformity to the test curve, and so that both the temperature of the vapor phase and of the liquid phase (if present) can be recorded.

6. Preparation of Test Specimens

6.1 Determine the appearance of the test panels prior to testing by photo documentation or equivalent methods in order to provide a basis for post-test comparison. The testing requirements should indicate if this assessment will be done prior to shipping to the test facility.

6.2 Unless otherwise specified, a minimum of four samples shall be required to establish conformance of a given coating system on a given substrate, with two of the four samples being irradiated prior to testing in accordance with Test Method D 4082. Typical laboratory test panels are 2 by 4 by ⅛ in. for steel panels and 2 by 2 by 4 in. for concrete panels.

6.2.1 *Steel Panels*—Prepare in accordance with ANSI N512 or as necessary to duplicate actual conditions.

6.2.2 *Concrete Blocks*—Prepare in accordance with ANSI N512 or as necessary to duplicate actual conditions.

7. Procedure

7.1 *Test Parameters:*

7.1.1 Test coatings using the applicable curves from the latest Safety Analysis Report (SAR) identified by the owner for the specific containment. Illustrations of time-temperature-pressure test curves that simulate primary containment atmospheres during a DBA are shown in Figs. 1 and 2.

7.1.2 The curves depicted in Figs. 1 or 2 may be used if they represent conditions equal to or more severe than those DBA conditions anticipated.

7.1.3 The parameters of the curves may be simulated during testing as continuous functions or as an enveloping stepwise function.

7.1.4 Steam is used initially to achieve the desired thermal shock and to raise the test chamber and its environment to the prescribed test conditions. After equilibrium is achieved, the temperature of the test chamber is maintained by means

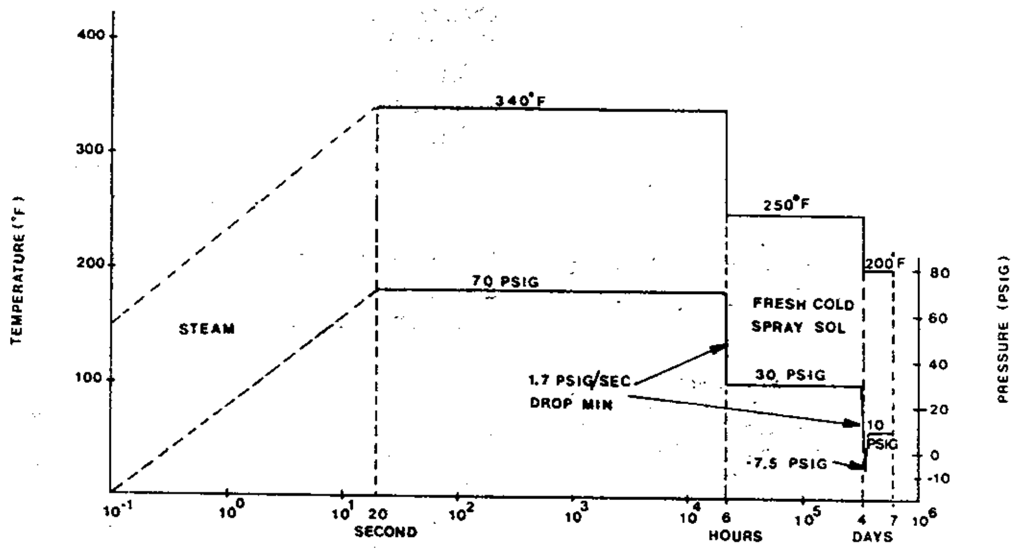

FIG. 1 Typical Design Basis Accident (DBA) Testing Parameters (Temperature-Time-Pressure)—BWR Drywell

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Evaluation for Existence of Blast Waves Following Licensing Basis Double-Ended Guillotine Pipe Breaks

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Evaluation for Existence of Blast Waves Following Licensing Basis Double-Ended Guillotine Pipe Breaks

Introduction

The BWR Owners' Group ECCS Suction Strainer Committee is developing alternate approaches for establishing an appropriate zone of destruction from which insulation debris may be generated following a double-ended guillotine pipe break.

The NRC has previously suggested that a blast wave may precede the saturated steam or water jet discharge. If a blast wave occurred that affected the generation of insulation debris at significant distances from the break location, it would preclude the use of the ANS/ANSI 58.2 model for determining the zone of destruction, and would also significantly increase the complexity for the alternate computation fluid dynamics (CFD) approach which will be used by the BWR Owners' Group to calculate the zone of destruction.

Test data are available on the fracture behavior of axial cracks in rolled and welded piping. Much of the work was done for pressurized gas pipelines. The piping was generally made of ferritic material (with less stringent requirements on toughness) operating at lower temperatures, and considered the effect of relatively long axial cracks. Test results from Batelle Columbus pipe fracture experiments with axial flaws included damage at significant distances from the test location that indicated the generation of atmospheric shock waves (Reference 1,2,3). This damage included glass breakage and cracking of building walls at several hundred feet from the test pipe. When axial (longitudinal) cracking occurs, large fishmouth type openings can form very rapidly and this has been shown to result in the formation of significant shock waves. More recent testing at Battelle in 1995 was focused on the pressure pulse caused by a circumferential rupture in the extrados of a stainless steel pipe elbow (16.1 inch diameter). These tests conducted as part of the IPIRG program were intended for application to nuclear power plant piping and simulated materials and component conditions (toughness property and test temperature) that were representative of nuclear piping. Although double-ended guillotine breaks did not occur during these experiments, no atmospheric shock waves were detected prior to the general exhausting pressure wave.

This BWROG technical position paper demonstrates that blast waves, characterized as moving shock waves, are not of significance following circumferential double-ended guillotine pipe breaks in boiling water reactor plant piping. Although longitudinal cracks are identified as potential high energy line breaks, cracking of these welds is of minimal concern since there has never been an axial pipe crack failure in the nuclear industry and axial failures are not expected to occur in the future. Axial welds are typically solution annealed in the factory, eliminating the residual stresses which play a dominant role in crack initiation in both ferritic and austenitic piping. In stainless steel piping, in addition to the residual stress benefit, solution heat treatment also eliminates weld sensitizing thereby eliminating the potential for intergranular stress corrosion cracking (IGSCC). It is of interest to note that since longitudinal weld failures do not occur, an ASME code change has been approved (1995 addenda) which eliminates longitudinal seam weld in-service inspections. The only code requirement is that when circumferential welds are inspected, a portion of the adjacent axial weld at its heat affected intersection with the circumferential weld is also inspected. Thus, for the purpose of evaluating the potential for blast waves, test data on fracture with axial weld cracks can be considered non-relevant. This leaves test results with defects in circumferential cracks (which showed no evidence of shock waves) as the basis for judging the potential for blast wave generation.

Criteria for Evaluating the Potential for Blast Wave Formation

The following discussion pertains to conditions which would have to be met in order to facilitate insulation debris generation due to blast wave propagation through the containment from a postulated pipe rupture. If a pipe rupture opens rapidly enough, the moving pressure gradient can steepen, forming a moving shock or blast wave. The movement of a blast wave past a structure can impose a resulting differential pressure which could lead to structural damage. Slower opening of a pipe rupture would transmit mild pressure waves, propagating on an average rising pressure throughout the containment without strong spatial pressure gradients. A blast wave would not be expected for a slower opening pipe rupture, and the fluid jet discharge would be more likely to create the insulation debris.

Arguments can be made to show that a double ended guillotine break is almost impossible in well supported piping made of ductile materials (such as those used in nuclear power plant piping) and designed to the ASME Code. In fact both the ASME and the US NRC has recognized the added margin from leak before break. Nevertheless, the potential for blast wave generation is evaluated *assuming an instantaneous guillotine break, which is a circumferential separation of two pipe sections*. The rate at which the two pipe ends separate from each other depends on both the closeby pipe layout geometry and the associated structural supports. That is, the pipes might separate axially, or bend off center at some rate. If a break opens slowly enough, blast waves would not be a concern, and fluid jet discharge would provide the only mechanism for creating insulation debris. A sudden break opening also would result in a discharging fluid jet, but preceded by a blast wave.

The criterion for evaluating the potential for blast wave formation is to compare the pipe rupture time, t_r , with the acoustic propagation time across the piping of interest, $t_a = L/C$; where L is a characteristic dimension for the pipe, and C is the speed of sound in the containment gas (approximately 1100 fps). Whether or not a pipe with insulation experiences a shock or uniform surrounding pressure rise depends on the rupture time relative to the propagation time of sonic waves past the pipe, where the appropriate characteristic length would be its diameter, that is, L = D. It follows that if the rupture opening time is long relative to the acoustic propagation time, then a blast wave would not be expected. Otherwise, blast waves (moving steep pressure gradients) can occur. That is,

If $t_r \gg L/C$, no blast waves are expected

The following table can be used to estimate whether the pipe rupture time is fast enough, assuming a factor of 5, that the pressure gradient moving past a pipe begins to resemble a blast wave. A factor of 5 might be too low if the expanding steam ball moves at a Mach number much greater than 1.0 relative to the undisturbed atmosphere.

Pipe Diameter	$t_a = L/C$	*Blast Waves Unlikely Conservative t_r ($5 \times t_a$)*
4"	0.0003 seconds	0.0015 seconds
12"	0.00091 seconds	0.005 seconds
24"	0.00182 seconds	0.009 seconds

The analysis (described in the next section) evaluates the potential for and significance of blast waves which may occur if a circumferential crack suddenly releases the two ends of a 24 inch pressurized pipe. Two extreme cases were considered: axial pipe separation and radial or bending separation. The axial separation case is characterized by the two ends moving away from each other on the axis, for which the pipe opening time from zero discharge flow to full double-ended blowdown flow is about 0.19 seconds. Based on the above criteria this is about twenty times too slow for a significant blast wave to form. The more likely radial or bending separation is characterized by a circumferential crack in an S-bend that joins two parallel, off-center 24 inch pipe segments which move away from each other by bending away from their initial axes. The bending separation case is shown to open faster than the axial separation case, having an opening time of about 0.014 seconds, which still appears to be long enough to preclude a significant blast wave concern.

An additional analysis was included to estimate where and when a moving shock could form relative to the rupture, based on a conservative formulation of shock development. The analysis is based on one-dimensional flow rather than three-dimensional, with the fluid discharging into a ball which expands at ambient pressure. Actually, the expanding ball would have higher pressures near the break, higher densities, and consequently would expand slower than the conservative model described. The rapidly expanding ball velocity is treated as a one-dimensional piston, resulting in more rapid pressure gradient steepening than would be expected in spherical expansion and the associated radial attenuation that would more accurately characterize a pipe rupture in the containment. Results show that for both the axial and bending separation cases, an idealized shock forms instantly at the pipe rupture as it begins to open. However, the shock formed propagates radially from a relatively small initial spherical area. Such a shock becomes a simple sound wave within several pipe diameters of the rupture, and blast waves need not be considered further.

PIPE RUPTURE OPENING TIME FOR LONGITUDINAL SEPARATION

Fig. 1 shows a section of pipe, which is just undergoing a circumferential failure. This can represent a guillotine break where the failure is simultaneous around the circumference. Crack development in a circumferential weld could propagate almost completely, while the two ends are held together by a small ligament, which finally fails. Fluid pressure P inside the pipe exerts a force PA , which is balanced by the pipe longitudinal force $\sigma_i a$, where σ_i is the initial tensile stress and "a" is the pipe wall cross-sectional area. When the circumferential rupture is complete, the longitudinal stress becomes zero behind a stress relief wave, which travels away from the rupture at sound speed S in the pipe material, given by

$$S = \sqrt{\frac{g_o E_p}{\rho_p}} \qquad (1)$$

where E_p is Young's modulus and ρ_p the density for the pipe. Since E_p is the ratio of stress to strain δ_p , that is,

$$E_p = \frac{\sigma_p}{\delta_p} \qquad (2)$$

the changes in stress and strain across the stress relief wave can be expressed by

$$\Delta\delta_p = \frac{\Delta\sigma_p}{E_p} \qquad (3)$$

The strain which has been relieved behind the stress relief wave of Fig. 1 can also be written as

$$\Delta\delta_p = \frac{L_i - L}{L} = \frac{L_i - L}{St} \qquad (4)$$

But since $(L_i - L)/t = V_r$, the ruptured surface is moving at velocity

$$V_r = \Delta\delta_p S = \frac{\Delta\sigma_p}{E_p} S \qquad (5)$$

The change in pipe stress is given by the sudden removal of the fluid force PA , per unit pipe wall cross-sectional area a , or

$$\Delta\sigma_p = \frac{PA}{a} \qquad (6)$$

so that the ruptured pipe surface velocity is

$$V_r = \frac{PA}{a}\sqrt{\frac{g_o}{\rho_p E_p}} \qquad (7)$$

Since both ends of the ruptured interface are moving away from each other at the same velocity, the fluid discharge area is increasing at the rate of $2V_r\pi D_i$, where D_i is the pipe inside diameter. That is, the discharge area A_d can be written as

$$A_d = 2V_r \pi D_i t \qquad (8)$$

Eq. (8) applies only while the stress relief wave travels from the rupture to the anchored end of either pipe section, and returns to the rupture as a compression wave, but it describes the early break for a longitudinal separation.

Table 1 gives parameters for an example calculation. The calculated velocity of one ruptured pipe end is V_r = 2.33 fps .

TABLE 1
Parameters for Example Pipe Rupture Calclulation

P = 1000 psia, initial fluid pressure in the pipe
24 inch schedule 80 pipe
A = 2.536 ft2 , flow area
D_i = 1.797 ft , inside diameter
D_o = 2.00 ft, outside diameter
a = 0.605 ft^2 , pipe wall cross-section
E_p = 30×10^6 psi , Young's modulus for pipe
ρ_p = 500 lbm/ft^3 , Pipe density

The time for full pipe rupture opening t_o , is considered the time when the circumferential discharge area $\pi D_i 2V_r t_o$ is equal to the sum of two pipe flow areas for double-ended discharge, namely $2(\pi D_i^2/4)$, or

$$t_o = \frac{D_i}{4V_r} \qquad (9)$$

for which t_o = 0.19 sec , which is an order of magnitude slower than the pressure wave transmission time past a 2 ft diameter pipe.

PIPE RUPTURE OPENING TIME FOR RADIAL SEPARATION

Fig. 2 shows a postulated section of piping which is composed of two parallel off-center pipe segments, connected by an S-bend, which has just undergone a circumferential failure, and is in the process of separating. High pressure pipe fluid is exerting forces on the two elbows in opposite directions. Both pipe segments experience a fluid force F_L acting normal to their axis. One of these pipe segments is considered in Fig. 3, which shows the pipe bending into a time-dependent curve y(x,t), caused by the shearing force F at the pipe end x = L, where the displacement off the center line is given by $y_L(x,t)$. The pipe segment left end is treated as if it were cantilevered at x = 0 with a displacement of y(0,t) = 0, and a slope of $y_x(0,t) = 0$, indicating a rigid connection. Since only a shear is applied at x = L without a twisting moment, the second derivative of the displacement is zero, that is, $y_{xx}(L,t) = 0$, and the shear is expressed as the gradient of the moment such that $EIy_{xxx}(L,t) = -F_L$. The full problem is formulated with the uniform beam equation as,

$$\text{DE} \qquad y_{tt} + \alpha^2 y_{xxxx} = 0 \tag{10}$$

where

$$\alpha^2 = \frac{g_o EI}{m'} \tag{11}$$

Here, I is the beam section area moment of inertia,given by $\pi(R_o^4 - R_i^4)/4$ for the pipe of inner and outer radii R_i and R_o, and m' is the beam mass per unit length, given by the sum of $\rho_p a$ for the pipe wall and $\rho\pi R_i^2$ for the contained fluid. The initial conditions are specified as a stationary, undisturbed pipe segment, for which

$$y(x,0) = 0 \tag{12}$$

$$y_t(x,0) = 0 \tag{13}$$

and the boundary conditions are,

$$\text{BC's} \qquad y(0,t) = 0 \tag{14}$$

$$y_x(0,t) = 0 \tag{15}$$

$$y_{xx}(L,t) = 0 \tag{16}$$

$$y_{xxx}(L,t) = -\frac{F_L}{EI} \tag{17}$$

A solution is obtained by first transforming the full problem to make all the boundary conditions homogeneous. One such transformation is

$$y(x,t) = Z(x,t) + \frac{F_L L}{2EI}x^2 - \frac{F_L}{6EI}x^3 \tag{18}$$

for which the full problem for z(x,t) becomes

$$\text{DE} \qquad Z_{tt} + \alpha^2 Z_{xxxx} = 0 \tag{19}$$

$$\text{IC's} \qquad Z(x,0) = -\frac{F_L L}{2EI}x^2 + \frac{F_L}{6EI}x^3 \tag{20}$$

$$Z_t(x,0) = 0 \quad (21)$$

BC's $$Z(0,t) = 0 \quad (22)$$

$$Z_x(0,t) = 0 \quad (23)$$

$$Z_{xx}(L,t) = 0 \quad (24)$$

$$Z_{xxx}(L,t) = 0 \quad (25)$$

A product solution for Z(x,t) is expressed as

$$Z(x,t) = \chi(x)T(t) \quad (26)$$

for which the function $\chi(x)$ must satisfy the characteristic equation,

$$\chi'''' - \lambda^4 \chi = 0 \quad (27)$$

The orthogonality property of Eq. (27) is expressed in terms of solutions $\chi_n(x)$ and $\chi_m(x)$, for values of the separation constant λ_n and λ_m, respectively, as

$$(\lambda_n^4 - \lambda_m^4)\int_0^L \chi_n \chi_m dx = \left[\chi_m \chi_n''' - \chi_n \chi_m'''\right]_0^L - \left[\chi_m' \chi_n'' - \chi_n' \chi_m''\right]_0^L \quad (28)$$

It is seen that for the homogeneous boundary conditions of Eqs. (22) - (25) that the integral of Eq. (28) is zero for $n \neq m$, but otherwise,

$$\left.\int_0^L \chi_n \chi_m dx\right|_{n=m} = \int_0^L \chi_n^2 dx = I_n \quad (29)$$

A solution to Eq. (27) is given by

$$\chi(x) = C_1 e^{\lambda x} + C_2 e^{-\lambda x} + C_3 \cos\lambda x + C_4 \sin\lambda x \quad (30)$$

Eq. (30) is employed in the boundary conditions of Eqs. (22) - (25), which yields the eigenvalue equation,

$$1 + \cosh\lambda L \cos\lambda L = 0 \quad (31)$$

The first five eigenvalues are listed in Table 2.

TABLE 2

Eigenvalues, Pipe Bending Separation

n	$\lambda_n L$
0	1.8
1	4.7
2	7.8
3	11.0
4	14.1

The full solution for Z(x,t) is expressed as a sum of eigenfunctions in x , and time-dependent coefficients,

$$Z(x,t) = \sum_{0}^{\infty} T_n(t)\chi_n(x) \tag{32}$$

Substitution into Eqs. (19), (20), and (21), and making use of Eq. (27) yields the following differential equation for the time-dependent coefficients,

$$\text{DE} \qquad \sum_{0}^{\infty} [\ T_n'' + (a^2\lambda_n^{\ 4})T_n\ ]\chi_n(x) = 0 \tag{33}$$

with the initial conditions,

$$\text{IC's} \qquad \sum_{0}^{\infty} T_n(0)\chi_n(x) = -\frac{F_L L}{2EI}x^2 + \frac{F_L}{6EI}x^3 \tag{34}$$

$$\sum_{0}^{\infty} T_n'(0)\chi_n(x) = 0 \tag{35}$$

Use of the orthogonality property of Eqs. (28) and (29) with (18), (31), and (32) yields the full solution for y(x,t) in the form,

$$y(x,t) = \sum_{0}^{\infty} T_n(t)\chi_n(x) + \frac{F_L L^3}{EI}\left[\frac{1}{2}\left(\frac{x}{L}\right)^2 - \frac{1}{6}\left(\frac{x}{L}\right)^3\right] \tag{36}$$

where

$$\chi_n(x) = e^{\lambda_n x} + B_{2n}e^{-\lambda_n x} + B_{3n}\cos\lambda_n x + B_{4n}\sin\lambda_n x \tag{37}$$

and

$$B_{2n} = \frac{e^{\lambda_n L} - \sin\lambda_n L + \cos\lambda_n L}{e^{-\lambda_n L} + \sin\lambda_n L + \cos\lambda_n L} = \frac{e^{\lambda_n L} + \cos\lambda_n L}{\sin\lambda_n L} \tag{38}$$

$$B_{3n} = -\frac{e^{\lambda_n L} + e^{-\lambda_n L} + 2\cos\lambda_n L}{e^{-\lambda_n L} + \sin\lambda_n L + \cos\lambda_n L} = -(1 + B_{2n}) \tag{39}$$

$$B_{4n} = \frac{e^{\lambda_n L} - e^{-\lambda_n L} - 2\sin\lambda_n L}{e^{-\lambda_n L} + \sin\lambda_n L + \cos\lambda_n L} = -(1 - B_{2n}) \tag{40}$$

The time-dependent coefficients are given by

$$T_n(t) = T_{no}\cos[(\lambda_n L)^2(\alpha t / L^2)] \tag{41}$$

where

$$\frac{T_{no}}{(F_L L^3 / EI)} = \frac{-\frac{1}{2}\int_0^1 (x/L)^2 \chi_n(x/L)d(x/L) + \frac{1}{6}\int_0^1 (x/L)^3 \chi_n(x/L)d(x/L)}{\int_0^1 \chi_n^{\,2}(x/L)d(x/L)} \tag{42}$$

It was found that the first term in the series of Eq. (36) dominates, so that the pipe bending displacement is approximated within 1 percent by the equation,

$$\frac{y(x,t)}{(F_L L^3 / EI)} \cong -(0.0233)\chi_0(x/L)\cos[(\lambda_0 L)^2(\alpha t / L^2)] - \frac{1}{6}\left(\frac{x}{L}\right)^3 + \frac{1}{2}\left(\frac{x}{L}\right)^2 \tag{43}$$

Displacement of the discharge end at x = L is

$$y(L,t) \cong \frac{1}{3}\left(\frac{F_L L^3}{EI}\right)\left[1 - \cos\{(\lambda_0 L)^2(\alpha t / L^2)\}\right] \tag{44}$$

Based on the parameters of Table 1 for a 2 ft diameter pipe of length L = 20 ft , and the definition of α in Eq. (18),

$$\frac{F_L L^3}{EI} = 2.49 ft$$

$$\alpha = 9077 ft^2 / s$$

and the pipe end displacement is given by

$$y(L,t) = (0.82) ft (1 - \cos 77.7t) \tag{45}$$

Full break opening occurs when y(L,t) = R_i/2 = 0.899/2 ft = 0.45 ft, which from Eq. (45) corresponds to

$$t_{opening} = 0.014\,\text{sec}$$

If the the analysis is repeated for a 4 inch schedule 80 pipe,

$$\frac{F_L L^3}{EI} = 45.4\,ft$$

$$\alpha = 1008\,ft^2/s$$

$$y(L,t) = 15ft(1 - \cos 8.62t)$$

for which full opening of y(L,t) = R_i/2 = 0.16/2 = 0.08 ft occurs at

$$t_{opening} = 0.012\,\text{sec}$$

It is expected that the moment imposed by blowdown from a ruptured S-section could form a plastic hinge at the rigidly cantilevered end at $x = 0$. A 20 ft long section of 2 ft diameter pipe would have reached the full open position at an end deflection of 0.45 ft. Although it is doubtful that this deflection would have exceeded the yield stress to form a plastic hinge, a simple hinged beam analysis is included in Appendix A. A break opening time of 0.015 sec was predicted for the 20 ft long, 2 ft diameter pipe, which is close to the 0.014 sec predicted from the rigorous beam analysis.

EXPANDING STEAM VOLUME ANALYSIS FOR POSSIBLE SHOCK FORMATION

The blowdown of steam is chosen for this analysis because steam displaces more volume than the amount of steam formed from a liquid blowdown. Fig. 4 shows a steam region of volume $\forall$, which is expanding. The inflowing mass rate is $\dot{m}$, and the associated stagnation enthalpy is h_o. Conservation of energy is written as,

$$P\frac{d\forall}{dt} + \frac{dE}{dt} = \dot{m}h_o \qquad (46)$$

The contained energy for a perfect gas is written as

$$E = \frac{1}{k-1}P\forall \qquad (47)$$

and the stagnation enthalpy is,

$$h_o = \frac{k}{k-1}\frac{P_o}{\rho_o} \qquad (48)$$

If the discharging mass flow rate is written as the critical mass flux G_c, multiplied by the discharge area $A_d = 2\pi D_i y(L,t)$, formed as the two broken ends of the pipe move apart, the expanding volume is governed by

$$P\frac{d\forall}{dt} + \frac{\forall}{k}\frac{dP}{dt} = 2G_c\pi D_i \frac{P_o}{\rho_o} y(L,t) \tag{49}$$

The fastest expansion rate, which would result in the strongest moving shock, can be conservatively estimated for a case in which $P = P_\infty$ = constant, for which the second term of Eq. (49) becomes zero, and the predicted $d\forall/dt$ is larger than expected. For the initial condition of zero volume (when the pipe ends begin to separate), the solution of Eq. (49) gives the expanding volume as

$$\forall = \frac{2G_c\pi D_i P_o}{P_\infty \rho_o}\int_0^t y(L,t)dt \tag{50}$$

If the expanding volume is assumed to be spherical with the volume $\forall$ given by $(4/3)\pi R^3$, the radius grows with time as

$$R = \left(\beta\int_0^t y(L,t)dt\right)^{1/3} \tag{51}$$

where

$$\beta = 3\frac{P_0 R_i G_c}{P_\infty \rho_0} \tag{52}$$

The case of axial pipe separation with $y(L,t) = V_r t$, where V_r is obtained from Eq. (7), yields

$$R = \left(\frac{\beta V_r}{2}\right)^{1/3} t^{2/3} \quad \text{for axial separation} \tag{53}$$

whereas the case of bending separation with $y(L,t)$ from Eq. (44) yields

$$R = \left\{\frac{\beta}{3}\left(\frac{F_L L^3}{EI}\right)\frac{L^2}{\alpha(\lambda_0 L)^2}\left[\frac{(\lambda_0 L)^2 \alpha t}{L^2} - \sin(\lambda_o L)^2(\alpha t / L^2)\right]\right\}^{1/3} \tag{54}$$

The argument of the sine term is small enough to use the first two terms of the expansion, which reduces Eq. (54) to

$$R = \left[\frac{\beta}{18}\left(\frac{F_L L^3}{EI}\right)\frac{L^2}{\alpha(\lambda_0 L)^2}\right]^{1/3}\frac{(\lambda_0 L)^2 \alpha t}{L^2} \quad \text{for bending separation} \tag{55}$$

The expanding steam ball is next used to obtain the boundary of an expanding wall, which may cause the formation of a moving shock wave.

SHOCK WAVE FORMATION

The expanding wall of Eqs. (53) and (55) are employed in the idealized case of a one-dimensional gas cylinder, with a piston motion described by R(t) . The assumption of one-dimensional flow results in a stronger shock, since a spherical flow attenuates shock pressure at larger radii. The pressure and velocity in a compressible fluid are governed by the equations,

$$\frac{\partial P}{\partial t} + V\frac{\partial P}{\partial x} + \frac{\rho C^2}{g_o}\frac{\partial V}{\partial x} = 0 \qquad (56)$$

and

$$\frac{\partial V}{\partial t} + V\frac{\partial V}{\partial x} + \frac{g_o}{\rho}\frac{\partial P}{\partial x} = 0 \qquad (57)$$

where for isentropic compression of gas outside the expanding wall, the density is

$$\rho = \rho_i\left(\frac{P}{P_i}\right)^{1/k} \qquad (58)$$

and the sound speed is

$$C = \sqrt{\frac{kg_oP}{\rho}} = \sqrt{\frac{kg_oP_i}{\rho_i}}\sqrt{\left(\frac{P}{P_i}\right)^{(k-1)/k}} = C_i\left(\frac{P}{P_i}\right)^{(k-1)/2k} \qquad (59)$$

A solution for the outgoing pressure wave into stationary gas is given by[4, Chapter 8]]

$$V = \frac{2C_i}{k-1}\left[\left(\frac{P}{P_i}\right)^{(k-1)/2k} - 1\right] \qquad (60)$$

and

$$C = C_i + \frac{k-1}{2}V \qquad (61)$$

with

$$x = (V + C)t + f(V) \qquad (62)$$

Eqs. (59) - (62) relate either P or V to x and t . The function f(V) can be determined by a known boundary condition, like an accelerating wall. The accelerating steam ball gives the moving boundary of Eqs. (53) or (55). If this wall is idealized as a piston, moving into a one-dimensional gas, the function f(V) of Eq. (62) can be obtained by setting $x = R(t)$ and $V = dR/dt$. Employing Eq. (53) for R in the axial separation case, the sound speed C is eliminated with Eq. (61), giving

$$f(V) = \frac{2}{9}\beta V_r \left[\frac{1}{V^2} - \frac{2}{3}\frac{\left(C_i + \frac{k+1}{2}V\right)}{V^3} \right] \qquad (63)$$

Shock formation corresponds to two conditions, generally expressed as

$$\left(\frac{\partial x}{\partial V}\right)_t = 0 \qquad (64)$$

for the appearance of a discontinuous velocity (or pressure), or infinite pressure gradient, and

$$\left(\frac{\partial^2 x}{\partial V^2}\right)_t = 0 \qquad (65)$$

If either of Eqs. (64) or (65) gives incompatible results, shock formation occurs at the location where $V = 0$. However, for this case with $V = dR/dt$, Eq. (65) yields

$$V = \frac{4C_i}{2-k} \qquad (66)$$

for which Eq. (64) gives the time of shock formation as

$$t_{shock} = \frac{2(2-k)^4}{576C_i^3(k+1)}\frac{\beta V_r}{3} \qquad (67)$$

The calculated shock formation time, for $k = 1.4$, $\beta V_r/2 = 190{,}734\ ft^3/s^2$, $G_c = 2000\ lbm/ft^2\text{-}s$, Po = 1000 psia, $P_\infty = 14.7$ psia, $\rho_o = 2.24\ lbm/ft^3$, and $C_i = 1100$ fps in air, is $t_{shock} \approx 1.8\times10^{-8}$ seconds. This time implies that shock formation is essentially at the instant the pipe rupture begins to separate. The steam ball radius of Eq. (53) at the time of shock formation would be about $R_{shock} \approx 0.00037$ ft. The maximum pressure behind the shock could not exceed 1000 psia, which would attenuate to less than 1.0 psi in a distance of 1.0 ft of travel. This further supports the conclusion that blast waves for a circumferential pipe rupture and longitudinal separation are insignificant.

The case of pipe bending separation is obtained with the help of Eq. (55) for the expanding steam ball radius. It is seen that the steam ball velocity dR/dt is a constant,

$$\frac{dR}{dt} = \left[\frac{\beta}{18}\left(\frac{F_L L^3}{EI}\right)\frac{L^2}{\alpha(\lambda_0 L)^2}\right]^{1/3}\frac{(\lambda_0 L)^2\alpha}{L^2} \quad \text{for bending separation} \qquad (68)$$

which yields a velocity of

$$\frac{dR}{dt} \approx 500 fps$$

for the parameters of Table 1 and a 20 ft long pipe segment. For this case, it is easier to employ the normal shock equations [2]. A piston which starts sudden motion of 500 fps velocity into stationary ambient air creates a shock pressure of less than 12 psi immediately upon the start of pipe separation. Even a modest 1/r attenuation (associated with spherical waterhammer) would reduce this magnitude substantially before it reached a pipe several diameters away. The attenuation of a spherical shock is closer to $1/r^2$, since this describes the corresponding energy attenuation. Again, it is concluded for a bounding case that shock waves produce relatively small pressure gradients which would not be expected to remove insulation.

It should be noted that the calculated steam ball velocity for the axial pipe separation decreases with time, whereas for bending separation, the steam ball velocity is constant.

SUMMARY

It has been determined that the generation of insulation debris from blast waves caused by a postulated pipe rupture in a BWR containment should not be a concern if the rupture opening time is long, relative to the propagation time of a pressure wave past a pipe wrapped with insulation. That is, for a sound speed in air of 1100 fps, the pressure wave transmission time past a 2 ft diameter pipe would be about 0.002 sec. A pipe rupture which takes five or more times as long as the pressure wave transmission time would not be expected to generate a pressure gradient and associated forces large enough to create debris as it moved past a pipe wrapped with insulation. Nevertheless, analyses were formulated to estimate pipe rupture opening times, and the nature of shocks which might form if the spherical expansion of a pressure wave was treated as a one-dimensional expansion.

Pipe rupture opening analyses for two limiting cases in which pressurized pipe sections separate at a circumferential crack, either in the axial or radial (bending) directions, have been described. If the two ends of a pressurized pipe separate axially, the pipe opening time from zero discharge flow to full double-ended blowdown flow is about 0.2 sec, which is about 100 times slower than wave propagation past a 2 ft diameter pipe. A second postulated geometry consists of two straight, parallel, but offset pipe sections, joined by a connecting S-bend, which ruptures so that double-ended discharge creates equal and opposite shear forces normal to the axes of each straight pipe section, requiring the pipes to bend in order to separate. The predicted full opening time for this geometry was about 0.014 sec., which is 7 times the sonic pressure wave transmission time past a 2 ft diameter pipe.

Since it was recognized that the expanding steam ball and pressure waves moving into the containment atmosphere could exceed a Mach number of 1.0, it was assumed that a shock would eventually form. Therefore, an additional analysis was done to estimate where and when a moving shock would form, relative to the postulated pipe ruptures, based on a more rigorous formulation of shock development. The shock analysis is based on one-dimensional flow rather than three-dimensional, and is therefore conservative. It shows that for the case of axial pipe separation, a shock forms instantly at the rupture as it begins to open, having a small surface area. This shock would become a simple sound wave within less than one pipe diameter of the rupture, which shows that blast waves need not be considered for the axial pipe separation described . A similar analysis for the S-bend pipe rupture yields the instantaneous formation of a small shock wave, which also attenuates with distance to a level which is expected to be incapable of damaging insulation. A shock pressure attenuation proportional to 1/r was assumed, corresponding to linearized waterhammer. The actual attenuation is closer to $1/r^2$, which makes the analysis even more conservative.

APPENDIX A , HINGED PIPE ANALYSIS

An idealized pipe hinge was assumed to replace the cantilevered end of the section shown in Fig. 3. Newton's law in the form of torque and angular acceleration was employed to estimate the time for full opening of the ruptured end. The angular motion is governed, therefore, by the equation,

$$Torque = P_0 A L = \frac{I}{g_0}\frac{d^2\theta}{dt^2} \qquad \text{(A-1)}$$

The pipe moment of inertia about the hinge at x = 0 is given by

$$I = \int_0^L r^2 dm = \frac{\pi}{4}\left[\left(D_0^{\ 2} - D_i^{\ 2}\right)\rho_p + D_i^{\ 2}\rho\right]\int_0^L r^2 dr \qquad \text{(A-2)}$$

The integral is simply $L^3/3$, for which the moment of inertia is 1.23×10^6 lbm-ft^2 . It follows from the other data of Table 1 that

$$\theta = 95.8 (rad/s^2) t^2 (s^2) \qquad \text{(A-3)}$$

Since the end displacement is given by y = Lθ , it was found that for the 20 ft long, 2 ft diameter pipe, the opening time was 0.015 sec.

It can be shown that the opening time is proportional to √L , so that a 5 ft long, 2 ft diameter pipe hinged at one end would have an opening time of (0.015/2) = 0.0075 seconds. However, a cantilevered short pipe has a smaller moment from the blowdown thrust, and is less likely to yield. Furthermore, for sufficiently short pipes, the moment imposed by blowdown would not be enough to cause elastic response to the full open position.

List of References

Ref. 1. Letter report; M. Fleming and P. Scott, Batelle Columbus to M.E. Mayfield, US Nuclear Regulatory Commission documenting the external pressure data taken during IPIRG-2 Experiment 2-3; dated December 1, 1995

Ref. 2. K. Kussmaul, et al., *The Effect of Pipe Burst Investigations on the Postulation of Cracking in the Primary Cooling Systems of Light Water Reactors,* Circumferential Cracks in Pressure Vessels and Piping - Volume I. PVP - Vol. 94, G.M. Wilkowski, Ed., American Society of Mechanical Engineers, New York, 1983, pp. 27-63

Ref. 3. R.B. Duffey and H.T. Tang, *"Pipe Rupture and Depressurization Experiments,"* Circumferential Cracks in Pressure Vessels and Piping - Vol. II, PVP - Vol. 95, Ed. by G.M. Wilkowski, American Society of Mechanical Engineers, New York, 1983, pp. 271-278

Ref.4. Introduction to Unsteady Thermofluid Mechanics, F.J. Moody, Wiley Interscience, 1990

Figure 1

Circumferential Failure of Pipe Section

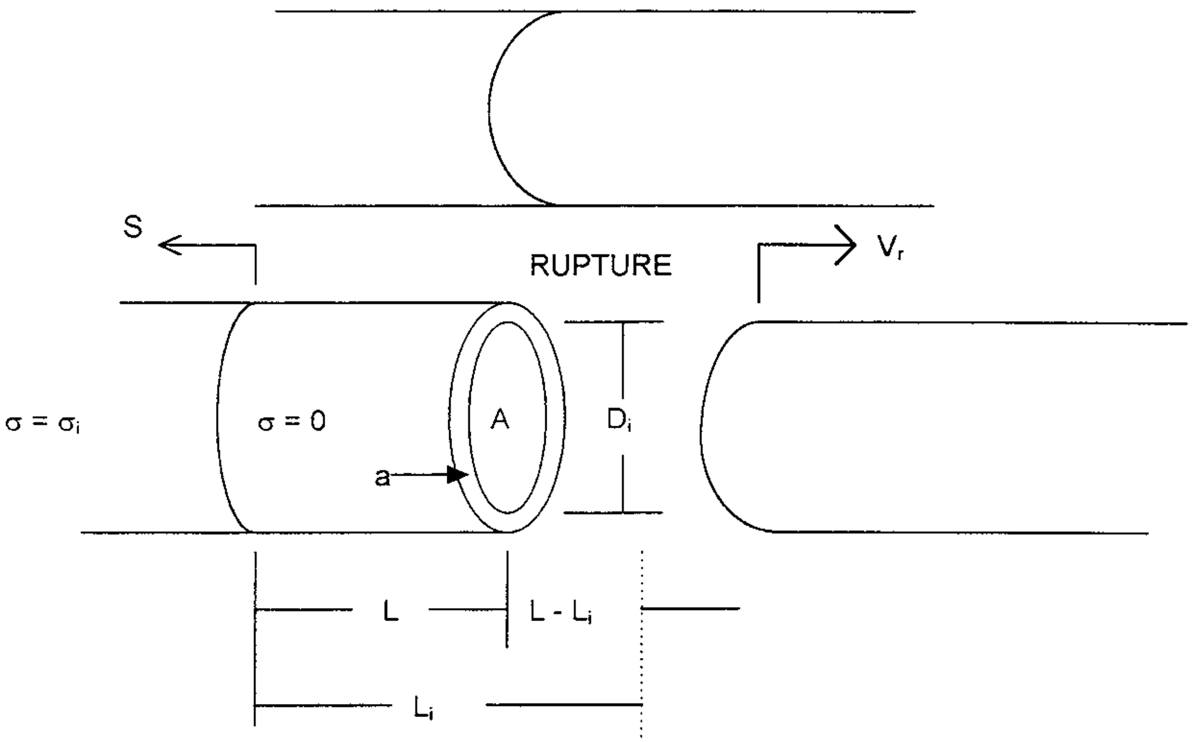


Figure 2

Rupture at S-Bend of Parallel Pipe Sections

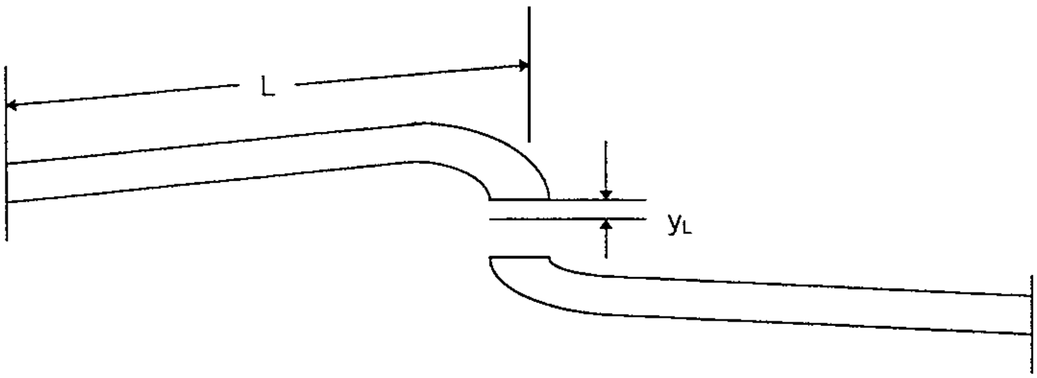


Figure 3

Bending Cantilevered Pipe Section, Shear Force at End

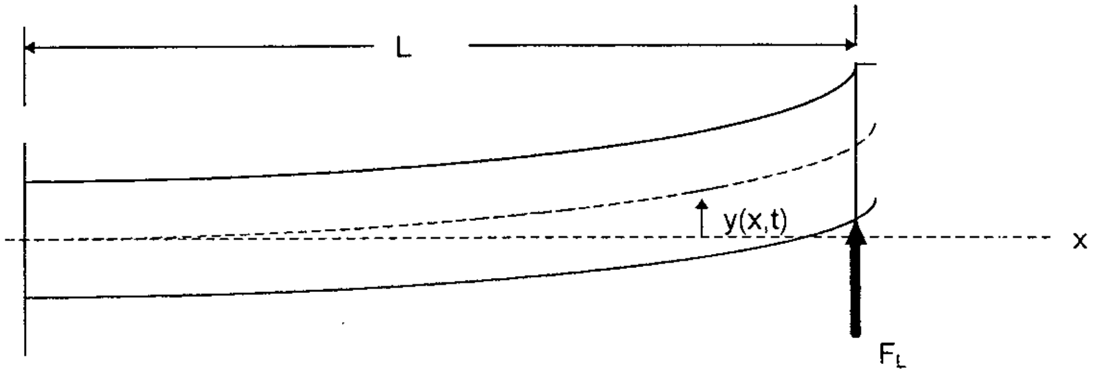
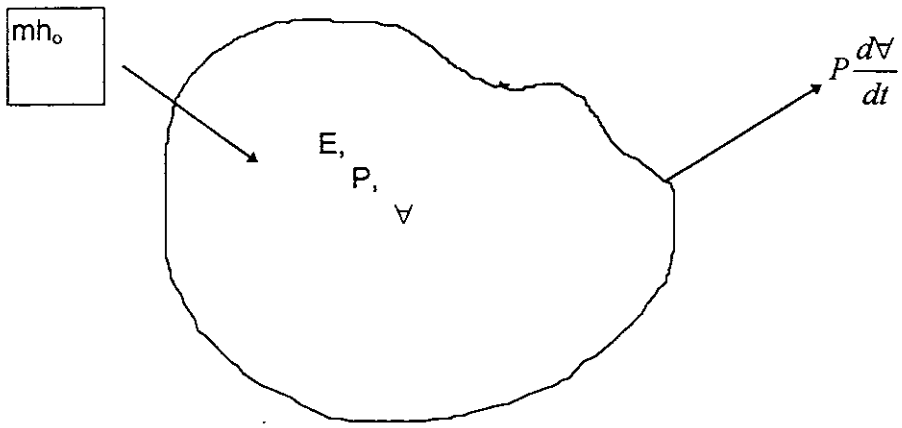


Fig. 4

Expanding Steam Ball



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Total Pressure Topography and Zone of Destruction for Steam and Mixture Discharge from Ruptured Pipes

Prepared

for

Boiling Water Reactor Owners' Group

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September 1996

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TOTAL PRESSURE TOPOGRAPHY AND ZONE OF DESTRUCTION FOR STEAM AND MIXTURE DISCHARGE FROM RUPTURED PIPES

ABSTRACT

The region inside which piping insulation can be removed by blowdown jet discharge from a pipe rupture is called the zone of destruction. It has been determined that insulation removal is likely to occur if the fluid jet stagnation pressure exceeds a certain value, which has been shown to typically range from several psi to about 50 psi for fibrous insulation materials. A rigorous computational fluid dynamics (CFD) program has been used to determine the steam-only discharge stagnation pressure topography in a three-dimensional region surrounding a double-ended pipe break in a BWR containment. This study shows that if the blowdown discharge was a steam/water mixture, the stagnation pressure along the jet centerline at any fixed location would be less than the stagnation pressure resulting from steam-only discharge. Furthermore, an expanding steam jet boundary is shown by model law dissimilarity not to be representative of an expanding steam/water mixture jet. However, a method is suggested for modifying the CFD steam pressure of destruction envelopes (calclulated volumes of destruction) to estimate a conservative topography for saturated water discharges.

BACKGROUND

A postulated pipe rupture inside containment produces a three-dimensional flow field, which can exert forces high enough to remove insulation from nearby pipes. It has been determined that if a fluid flow field can impose a sufficiently high stagnation or total pressure on the surface of insulation, the insulation is likely to be blown off the pipe.

Continuum Dynamics has applied a computational fluid dynamics program to determine the stagnation pressure topography for several pipe breaks, discharging saturated steam. That is, surfaces of constant stagnation pressure are profiled for a given pressure vessel state and double-ended pipe break geometry. The stagnation pressure that causes insulation removal is here referred to as the "critical stagnation pressure". Insulation lying inside the critical stagnation pressure surface is likely to be removed.

The computations were done for saturated steam discharge. However, there are pipe break conditions where steam/water mixture discharge can occur, which will lead to other stagnation pressure surfaces. Since the steam discharge surfaces are available, it is desirable to determine if the local stagnation pressure for steam/water discharge would be greater or smaller than it is for steam.

The purpose of this study is to provide a basis for determining whether local stagnation pressure caused by steam/water mixture discharge from a pipe rupture in the containment will be greater or smaller than the local stagnation pressure caused by steam discharge.

FREE JET IMPINGEMENT DATA

Steam/water blowdown discharge and jet impingement data was obtained from the Marviken test facility in the early 1980's [1]. Much of this data has been employed in the verification of an analytical model for predicting properties of the discharging jet ("Two-Phase Jet Modeling and Data Comparison," EPRI NP-4362, J. M. Healzer, & E. Elias, March, 1986).

One series of tests involved a measurement of the jet impingement force and centerline stagnation pressure for discharge against a flat plate. The experimental arrangement is shown in Fig. 1. Discharge from the pressure vessel through the nozzle was directed to the circular impingement plate, which was oriented in a plane perpendicular to the nozzle axis.

Since the stagnation pressure of jet impingement on a target determines if insulation would be blown off a pipe, the data of Fig. 2 was employed in this analysis. It is seen in Fig. 2 that three test traces were obtained for target plates at distances of L/D = 1.2 , 2.0, and 4 from the nozzle discharge, where the nozzle diameter in all cases was D = 509 mm. Blowdown conditions were essentially identical for each of the tests, namely tests labeled 7, 8, and 10, beginning with water at 35, 34, and 34 degrees subcooling, respectively. The initial water levels were 16.0, 16.4, and 16.4 m , respectively. Furthermore, the initial total masses of steam/water and initial vessel pressures were 262,000, 268,000, and 269,000 kg , and 5.01, 5.00, and 5.00 Mpa, respectively. It is therefore reasonable to assume that the three blowdowns of tests 7, 8, and 10 are essentially identical, and the only parameter distinguishing these tests is the target distance.

Fig. 2 shows a relatively high stagnation pressure ratio at the beginning of jet impingement during the subcooled portion of blowdown, for all three tests 7, 8, and 10. Once the subcooled water is discharged at about 20 seconds, saturated steam/water mixture begins to discharge, for which the stagnation pressure ratio is lower, but relatively constant for the next 25 seconds. The increase in stagnation pressure ratio when steam blowdown begins at about 45 seconds is a significant observation. The higher stagnation pressure ratio associated with steam discharge implies that mixture discharge results in a lower stagnation pressure. Rigorous computations were performed to obtain constant stagnation pressure surfaces for steam blowdown. It follows that the corresponding stagnation pressure for mixture blowdown appears to be less on any stagnation pressure surface obtained for steam. That is, steam blowdown bounds the zone of destruction for insulation on piping.

The Marviken data is for one nozzle and initial vessel state. A basis needs to be established which ensures that the apparent conclusion from Fig. 2, regarding jet stagnation pressure for steam/water mixture and steam-only jets, is valid. That is, it needs to be formally determined if the blowdown jet data of the Marviken tests is representative of expected reactor conditions. Furthermore, the vessel stagnation pressure decreases during blowdown. Yet, impingement pressure data of Fig. 2 has been normalized to vessel pressure. Assurance is needed that Fig. 2 data is not pressure scale dependent. Otherwise, interpretation of the results will be more difficult, involving a stagnation pressure dependence. Also, it is possible that compressible jets expand differently from large diameter nozzles than they do from small nozzles. The analysis that follows shows how the Marviken data can be employed to estimate jet centerline stagnation pressures for steam/water mixture and steam-only blowdowns.

GOVERNING EQUATIONS FOR JET EXPANSION

When a compressible fluid is discharged from a region of high pressure into a region of low pressure, it expands in both the foreward and radial directions. The expansion results in supersonic velocity in localized regions of the expanding jet. Overexpansion to local pressures below ambient results in oblique shocks, across which the flow adjusts to satisfy the mass, momentum, and energy equations, as well as the specific boundary conditions bounding the jet between the critical discharge plane and the ambient boundary pressure. Some parts of the expanding jet have continuous properties, which are separated by oblique shocks, or planes of property discontinuity.

In order to ensure that jet expansion and impingement data from the Marviken tests is representative of BWR blowdown conditions, the governing equations are written both for regions of isentropic flow, and shock discontinuities. Model, or similarity laws are then obtained from the governing equations to determine how the tests can be interpreted to predict BWR fluid jet properties.

Equations of mass conservation, momentum, and energy conservation for multidimensional flows with continuous properties are given by [2]

Mass:
$$\frac{\partial \rho}{\partial t} + \mathbf{V} \bullet \nabla \rho + \rho \nabla \bullet \mathbf{V} = 0 \qquad (1)$$

Momentum:
$$\frac{\partial \mathbf{V}}{\partial t} + \mathbf{V} \bullet \nabla \mathbf{V} + \frac{g_o}{\rho} \nabla P + g\mathbf{j} = \frac{g_o}{\rho} \nabla \bullet \Gamma \qquad (2)$$

Energy:
$$\frac{\partial T}{\partial t} + \mathbf{V} \bullet \nabla T = \frac{\beta T}{\rho c_p}\left(\frac{\partial P}{\partial t} + \mathbf{V} \bullet \nabla P\right) + \frac{1}{\rho c_p}\left[\nabla \bullet (\kappa \nabla T) + \Gamma \bullet \nabla \bullet \mathbf{V}\right] \qquad (3)$$

where $\mathbf{j}$ is the vertical unit vector, β is the thermal expansivity, and the viscous dissipation function $\nabla \bullet \Gamma$ is given by

$$\nabla \bullet \Gamma = \mu \nabla^2 \mathbf{V} + \left(\varsigma + \frac{1}{3}\mu \right) \nabla (\nabla \bullet \mathbf{V}) \tag{4}$$

Equations (1) - (4) apply to regions of an expanding fluid jet where properties are continuous.

If an expanding jet in one scale is to be representative at another scale, geometric similarity is implied. Therefore, oblique shocks in two different scales would have the same angles relative to the jet centerline. It follows that a normal shock discontinuity can be employed to obtain similarity laws introduced by oblique shocks. The normal shock equations are given by [3]

Mass: $$\rho_x V_x = \rho_y V_y \tag{5}$$

Momentum: $$\rho_y V_y^{\,2} - \rho_x V_x^{\,2} = g_o (P_x - P_y) \tag{6}$$

Energy: $$h_o = h_x + \frac{V_x^{\,2}}{2g_o} = h_y + \frac{V_y^{\,2}}{2g_o} \tag{7}$$

where subscripts x and y refer to fluid properties before (supersonic) and after (subsonic) a shock. Since enthalpy h can be expressed in terms of density and temperature, Eqs. (5), (6), and (7) introduce no new variables for describing compressible jet expansion properties.

Fluid state equations for perfect gas are used to represent steam-only jets, for which

$$h = c_p T = \frac{k}{k-1} \frac{P}{\rho} \tag{8}$$

$$C = \sqrt{k g_o P / \rho} \tag{9}$$

$$\rho = \frac{P}{RT} \tag{10}$$

and for isentropic state changes,

$$\frac{P}{\rho^k} = const. \tag{11}$$

The necessary state equations for a homogeneous equilibrium steam/water mixture are given by

$$h = h(P,v) = h_f(P) + \frac{h_{fg}(P)}{v_{fg}(P)} (v - v_f(P)) \tag{12}$$

$$v(P,s) = \frac{1}{\rho(P,s)} = v_f(P) + \frac{v_{fg}(P)}{s_{fg}(P)}(s - s_f(P)) \tag{13}$$

$$\frac{\beta}{c_p} = \rho \frac{v_{fg}(P)}{h_{fg}(P)} \quad ; \quad \beta = \frac{1}{v}\left(\frac{\partial v}{\partial T}\right)_p \quad ; \quad c_p = \left(\frac{\partial h}{\partial T}\right)_p \tag{14}$$

NORMALIZED VARIABLES

Variables in the governing equations are normalized so that their nondimensional values and derivatives have orders of magnitude of 1.0, that is, O(1.0). Since the jet expansion is the major focus, the normalizing pressure is the pressure difference between the discharging critical pressure and the ambient pressure, or

$$\Delta P = P_c - P_\infty \tag{15}$$

The normalizing velocity is the difference between the high velocity achieved during expansion from critical pressure P_c to ambient pressure P_∞ , and the discharge critical velocity, V_c . The simplified momentum diagram of Fig. 3 for a one-dimensional expanding jet without shocks yields

$$\Delta V = V_\infty - V_c = \frac{g_o}{\rho_c V_c}(P_c - P_\infty) \tag{16}$$

The normalizing density is approximated by the sound speed expression,

$$C = \sqrt{g_o \left(\frac{\partial P}{\partial \rho}\right)_s} \tag{17}$$

to give

$$\Delta\rho = -\frac{g_o}{C_c^2}(P_c - P_\infty) \tag{18}$$

The Claperyon equation,

$$\left(\frac{\partial P}{\partial T}\right)_v = \frac{1}{T}\frac{h_{fg}}{v_{fg}} \tag{19}$$

yields the normalizing temperature,

$$\Delta T = T \frac{v_{fg}}{h_{fg}} \Delta P \tag{20}$$

Distances are normalized with respect to a fixed length, selected as the discharge nozzle diameter, D, so that

$$\Delta x = D \tag{21}$$

Although the jet discharge is treated as quasi-steady, the trajectory of any fluid particle passes through a range of properties during its flight time. A reasonable time interval for normalizing time is the time required for a particle to travel a length of one nozzle diameter, or

$$\Delta t = \frac{D}{V_c} \tag{22}$$

The variables P , V , ρ , T , are normalized for the interval from when a fluid particle leaves the discharge plane till it reaches a steady state in the ambient environment. That is,

$$P^* = \frac{P - P_c}{\Delta P} \tag{23}$$

$$V^* = \frac{V - V_c}{\Delta V} \tag{24}$$

$$\rho^* = \frac{\rho - \rho_c}{\Delta \rho} \tag{25}$$

$$T^* = \frac{T - T_c}{\Delta T} \tag{26}$$

Space and time intervals are normalized as

$$x^*, y^*, z^* = \frac{x, y, z}{D} \tag{27}$$

and

$$t^* = \frac{t}{\Delta t} = \frac{V_c}{D} t \tag{28}$$

MODEL LAWS

When the normalized variables of Eqs. (23) - (28) are substituted into the governing equations, rearrangement yields nondimensional model coefficients for the various normalized variables and derivatives. The relative size of the model coefficients for a given application shows which terms dominate the process. The resulting model coefficients for an expanding jet are given by

$$\pi_1 = \frac{g_o \Delta P}{\rho_c C_c^2} = \frac{g_o \rho_c (P_c - P_\infty)}{G_c^2} \tag{29}$$

$$\pi_2 = \pi_3 = \pi_6 = \pi_1 \tag{30}$$

$$\pi_4 = \frac{\kappa_c \beta_c \Delta T \Delta t}{c_{pc} \rho_c \Delta x^2} = \kappa_c \left(\frac{v_{fg}}{h_{fg}} \right)^2 \frac{(P_c - P_\infty)}{C_c D} T_c \tag{31}$$

$$\pi_5 = \frac{\upsilon_c \beta_c (\Delta V)^2 \Delta t}{c_{pc} g_0 (\Delta x)^2} = \upsilon_c \left(\frac{v_{fg}}{h_{fg}} \right) \frac{g_o (P_c - P_\infty)^2}{\rho_c D C_c^3} \tag{32}$$

$$\pi_7 = \frac{g(\Delta t)^2}{\Delta x} = \frac{gD}{C_c^2} \tag{33}$$

$$\pi_8 = \frac{\upsilon_c \Delta V (\Delta t)^2}{(\Delta x)^3} = \frac{\upsilon_c g_o (P_c - P_\infty)}{\rho_c D C_c^3} \tag{34}$$

$$\pi_9 = \pi_{10} = \frac{\Delta T}{T} = \left(\frac{v_{fg}}{h_{fg}} \right)_c (P_c - P_\infty) \tag{35}$$

The critical discharge mass flux G_c has been used to replace $\rho_c V_c$.

Model coefficients π_1 through π_{10} are the nondimensional groups for multidimensional fluid jet expansion which determine whether two blowdown jets are similar, such that properties from one can be used to predict properties in the other. Generally, some of the π's are small relative to the others, and can be neglected, whereas one or two are large, and dominate the phenomena. It is necessary to preserve the dominant π groups to have similarity between two or more blowdown jets. When the dominant model coefficients are preserved, the resulting normalized variables P^*, V^*, ρ^*, and T^* should be the same, at various space points x^*, y^*, and z^*.

EVALUATION OF MODEL COEFFICIENTS

The model coefficients of Eqs. (29) - (35) are determined for both steam and steam/water mixture blowdowns in the full size system. The example data [2], [4] used for steam blowdown is given in Table 1.

Table 1
Example Data, Steam Blowdown

$P_c = 600$ psia
$P_\infty = 14.7$ psia
$C_c = 1600$ fps
$G_c = 2000$ lbm/s-ft^2
$\rho_c = G_c/C_c = 1.25$ lbm/ft^3
$v_{fg}(P_c) = 0.75$ ft^3/lbm
$h_{fg}(P_c) = 732$ B/lbm
$T_c = 486$ °F
$\kappa_c = 0.0228$ B/h-ft-°F
$\nu_c = 4.9 \times 10^{-4}$ ft^2/s
$D = 2$ ft

A steam/water mixture blowdown would have different values for P_c, C_c, G_c, ρ_c, T_c, κ_c, and ν_c. The values of G_c and P_c are obtained from [Ref. 2], for which T_c corresponds to the saturation pressure, and ρ_c can be obtained for isentropic flow from the vessel stagnation pressure P_o to the critical pressure, P_c, using state equation (13), with $P = P_c$, and the entropy $s = s_f(P_o)$. Table 2 gives example data for steam/water mixture blowdown. Only those parameters which change from values given for steam blowdown in Table 1 are listed.

Table 2
Example Data, Steam/Water Mixture Blowdown

$P_c = 800$ psia
$G_c = 5300$ lbm/s-ft^2
$\rho_c = 21.77$ lbm/ft^3
$C_c = G_c/\rho_c = 243$ fps
$T_c = 518$ °F
$\kappa_c = 0.3$ B/h-ft-°F (water), 0.0228 B/h-ft-°F (steam)
$\nu_c = 1.45 \times 10^{-6}$ ft^2/s (water), 4.9×10^{-4} ft^2/s (steam)

The model coefficients for steam-only blowdown and steam/water mixture blowdown were evaluated from data in Tables 1 and 2, and the results are given in Table 3.

Table 3
Model Coefficients, Steam and Steam/Water Mixture Jets

Coefficient	Steam Only	Steam/Water Mixture
$\pi_1 = \pi_2 = \pi_3 = \pi_6$	0.85	2.82
π_4	2×10^{-9}	2.6×10^{-8}
π_5	2.8×10^{-8}	3×10^{-7}
π_7	2.5×10^{-5}	1.1×10^{-3}
π_8	2.5×10^{-7}	2.1×10^{-6}
$\pi_9 = \pi_{10}$	0.11	0.15

It is seen that π_1 is the dominant model coefficient, with a smaller influence of π_9 or π_{10}. That is, a steam or steam/water mixture blowdown jet in one facility will be representative of a corresponding jet in another facility of the nondimensional group

$$\pi_1 = \frac{g_o \rho_c (P_c - P_\infty)}{G_c^2}$$

is the same in both facilities. The nondimensional group

$$\pi_9 = \pi_{10} = \left(\frac{v_{fg}}{h_{fg}}\right)_c (P_c - P_\infty)$$

is an order of magnitude smaller than π_1, having about a ten percent effect on similarity if it is not preserved. It is noted that both π_1 and π_9 are thermodynamic properties, which are functions of the vessel stagnation pressure, from which the fluid jet is discharged.

For the case of saturated steam discharge, the perfect gas state equations yield,

$$\pi_1 = \frac{1}{k}(1 - \frac{P_\infty}{P_c}) \tag{36}$$

Since the critical pressure P_c is about half of the stagnation pressure, the value of π_1 is essentially a constant, 1/k at all vessel pressures above approximately 200 psia.

For the case of saturated steam/water mixture blowdown, π_1 has a noticeable dependence on the vessel stagnation pressure, as seen in Table 4.

Table 4
The Dependence of π_1 and π_9 on Vessel Pressure for Steam/Water Mixture Blowdown

Stagnation Pressure (psia)	π_1	π_9
1000	2.82	0.12
500	3.91	0.11
200	5.21	0.09

Although the model coefficient π_9 has a 30 percent variation between 200 and 1000 psia vessel pressure, its importance is only about 10 percent of the π_1 coefficient. It is desirable to estimate the effect of the π_1 variation on jet impingement pressure.

THE EFFECT OF MODEL COEFFICIENT DISTORTION

If the dominant model coefficient π_1 was independent of vessel pressure, there would be no question regarding how representative the Marviken steam/water jet pressure data is of BWR applications. That is, the normalized pressure within the expanding jet can be expressed as

$$P^* = \frac{P - P_c}{P_c - P_\infty} = P^*(x^*,\pi_1) \tag{37}$$

where the π_9 dependence is considered negligible. The critical pressure ratio P_c/P_o is a function of vessel stagnation pressure P_o and enthalpy h_o, or

$$\frac{P_c}{P_o} = f(P_o,h_o) \tag{38}$$

Since $P_\infty/P_o << 1.0$,

$$P^* \cong \frac{P}{P_o}\frac{1}{f} - 1 = P^*(x^*,\pi_1) \tag{39}$$

or, the Marviken jet pressure, as graphed in Fig. 2, can be expressed as

$$\frac{P}{P_o} = f + fP^*(x^*,\pi_1) \tag{40}$$

In order to estimate the effect of a distortion in π_1 on the jet pressure, Eq. (40) is written in difference form as

$$\Delta\left(\frac{P}{P_o}\right) \approx \left(\frac{\partial\left(\frac{P}{P_o}\right)}{\partial\pi_1}\right)\Delta\pi_1 \tag{41}$$

The derivative $\partial(P/P_0)/\partial\pi_1$ can be estimated from Fig. 2 in the mixture blowdown region, which lasts for about 20 seconds. It appears that during mixture blowdown, the jet pressure ratio P/P_0 is relatively constant in time. However, vessel pressure must be decreasing during this time interval, and the π_1 model coefficient would change with vessel pressure. Therefore, it is reasonable to conclude that even though the model coefficient π_1 distorts during vessel stagnation pressure changes, the jet pressure ratio P/P_0 is relatively insensitive to the distortion in π_1. This implies that even though the Marviken jet pressures during steam/water mixture blowdown apply to a range of decreasing stagnation pressure in the vessel, the jet pressure ratio P/P_0 will not be strongly affected by vessel stagnation pressure. Moreover, the Marviken jet pressure data can be used to predict jet pressure in BWR blowdown jets from bigger or smaller pipes and higher or lower vessel pressures.

STEAM AND STEAM/WATER MIXTURE JET PRESSURE PROFILES

The Marviken jet stagnation pressure ratio P/P_0 on the jet centerline was obtained from the plate pressures of Fig. 2, and is plotted in Fig. 4 for the measured target distances of $x^* = 1.2$, 2.0, and 4.0. It is seen that at any location where the steam-only jet pressure is known, the jet pressure resulting from a steam/water mixture at the same vessel pressure would be less, except at distances approaching $x^* \to 0$. The ratio of local-to-vessel jet stagnation pressure for a steam/water mixture is between 70 and 80 percent of that for steam only. That is, if a steam only blowdown jet stagnation pressure is compared with the stagnation pressure of a saturated water blowdown, the saturated water blowdown pressure is smaller on the jet centerline.

Although a series of oblique shocks and rarefaction waves occur in an expanding compressible jet, a jet centerline analysis was performed for which the jet stagnation enthalpy would remain constant, the pressure would reach sufficiently low values from the rarefactions to eventually "shock up" to ambient pressure. Assuming that the last shock was a normal shock, the shock relationships described by Eqs. (5), (6), and (7) were employed with steam/water state equations to estimate the final jet stagnation pressure ratio. It was found that for steam-only discharge, the final stagnation pressure ratio was

$$\frac{P}{P_o} \approx 0.0200 \qquad x^* \text{ large} \quad ; \quad \text{steam-only}$$

and for steam/water mixture discharge,

$$\frac{P}{P_o} \approx 0.016 \qquad x^* \text{ large} \quad ; \quad \text{steam/water mixture}$$

JET EXPANSION GEOMETRY AND VOLUME

Although stagnation pressures on the jet centerline are smaller for steam/water discharge than for steam only, there are remaining questions about the differences in shape and volume of the expanding jet boundaries for mixture and steam blowdowns. That is, the three dimensional calculations performed for steam blowdown jets have resulted in uniform stagnation pressure surfaces in the containment. Since mixture blowdown has different expansion characteristics than steam only blowdown, it is expected that the actual jet boundaries for mixture and steam blowdowns are not similar. Even though jet centerline pressures lead to the conclusion that mixture jets have smaller local stagnation pressures than steam jets, mixture jet boundaries are likely to be altogether different than steam jet boundaries.

Continuum Dynamics Inc. (see Appendix) has computed the ratio of [Total volume of an expanded saturated water break] to [Total volume of an expanded saturated steam break] for the bounding ANSI 58.2 unrestrained pipe breaks. Results show that the saturated water jet volume is approximately 70 % of the volume of a saturated steam break when the local jet stagnation pressure is 42 psi. For local jet stagnation pressures less than 20 psi, the volume ratios are within 10%, and it is recommended that no correction be applied for these cases.

SUMMARY

The results of this study show that the local stagnation pressure of a steam/water mixture discharge along the jet centerline is less than the stagnation pressure caused by steam-only discharge. Close to the rupture, the stagnation pressure for steam/water mixture and steam discharges are the same. Beyond one diameter, the jet centerline stagnation pressure for mixture discharge is between 70 and 80 percent of the steam jet stagnation pressure.

The expanding jet boundaries for steam jets are not representative of mixture jets, as indicated by dissimilarity of the dominant modeling parameter. However, based on the Continuum Dynamics Inc. (see Appendix) computation of the ratio of [Total volume of an expanding saturated water break] to [Total volume of an expanded saturated steam break], the following correction factors are recommended for saturated liquid zone of destruction volumes:

Insulation Destruction Pressure (psi)	Saturated Water Volume / Saturated Steam Volume
0 - 20	1.0
20 - 30	0.9
30 - 40	0.8
40 - 50	0.7
50 - 60	0.5
> 60	0.4

REFERENCES

[1] "The Marviken Full-Scale Jet Impingement Tests, Fourth Series, Report Series MXD-200 and 300, Joint Reactor Safety Experiments in the Marviken Power Station Sweden," 1980, 1981.

[2] Moody, F. J., Introduction to Unsteady Thermofluid Mechanics, Wiley, 1990.

[3] Anderson, J. D., Modern Compressible flow, McGraw-Hill, 1982.

[4] ASME 1967 Steam Tables

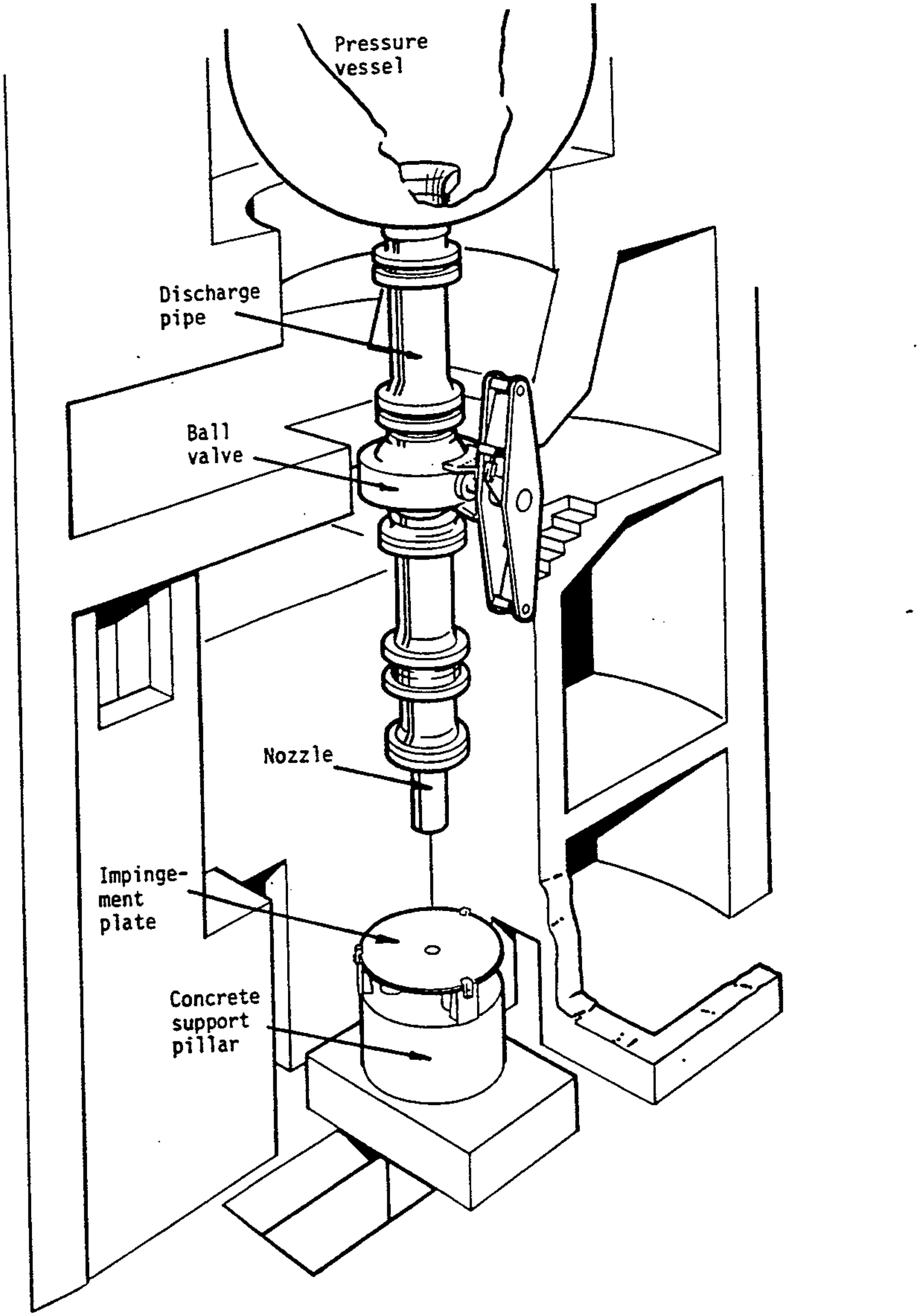

FIG. 1 Arrangement for the Jet Impingement Load Tests, MARVIKEN

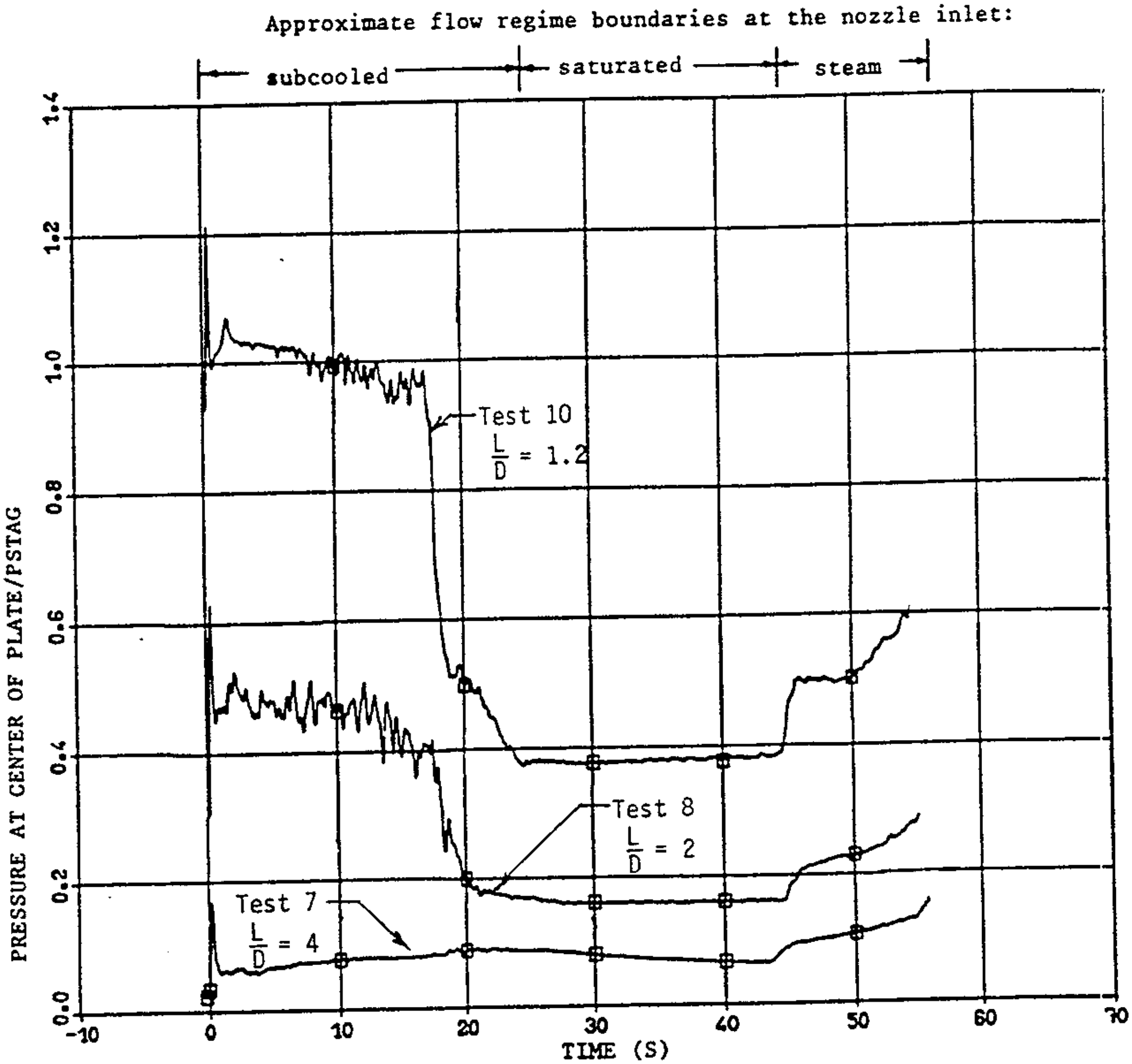

FIG. 2 Stagnation Pressure at Impingement Plate Center for Tests 7, 8 and 10

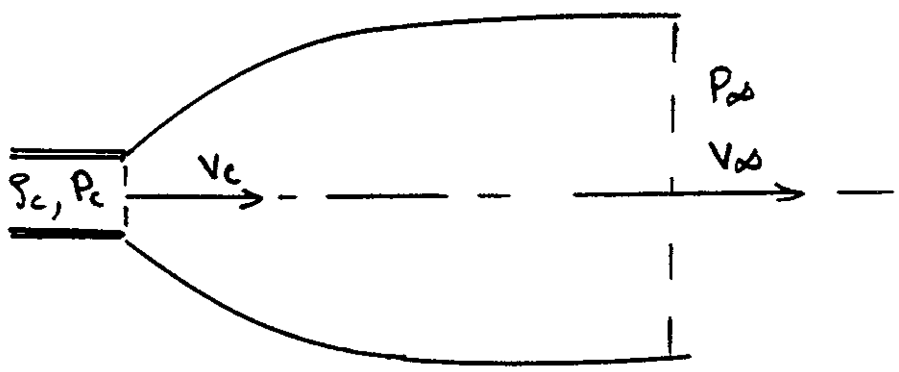

FIG. 3 SIMPLIFIED ONE-DIMENSIONAL FLUID JET

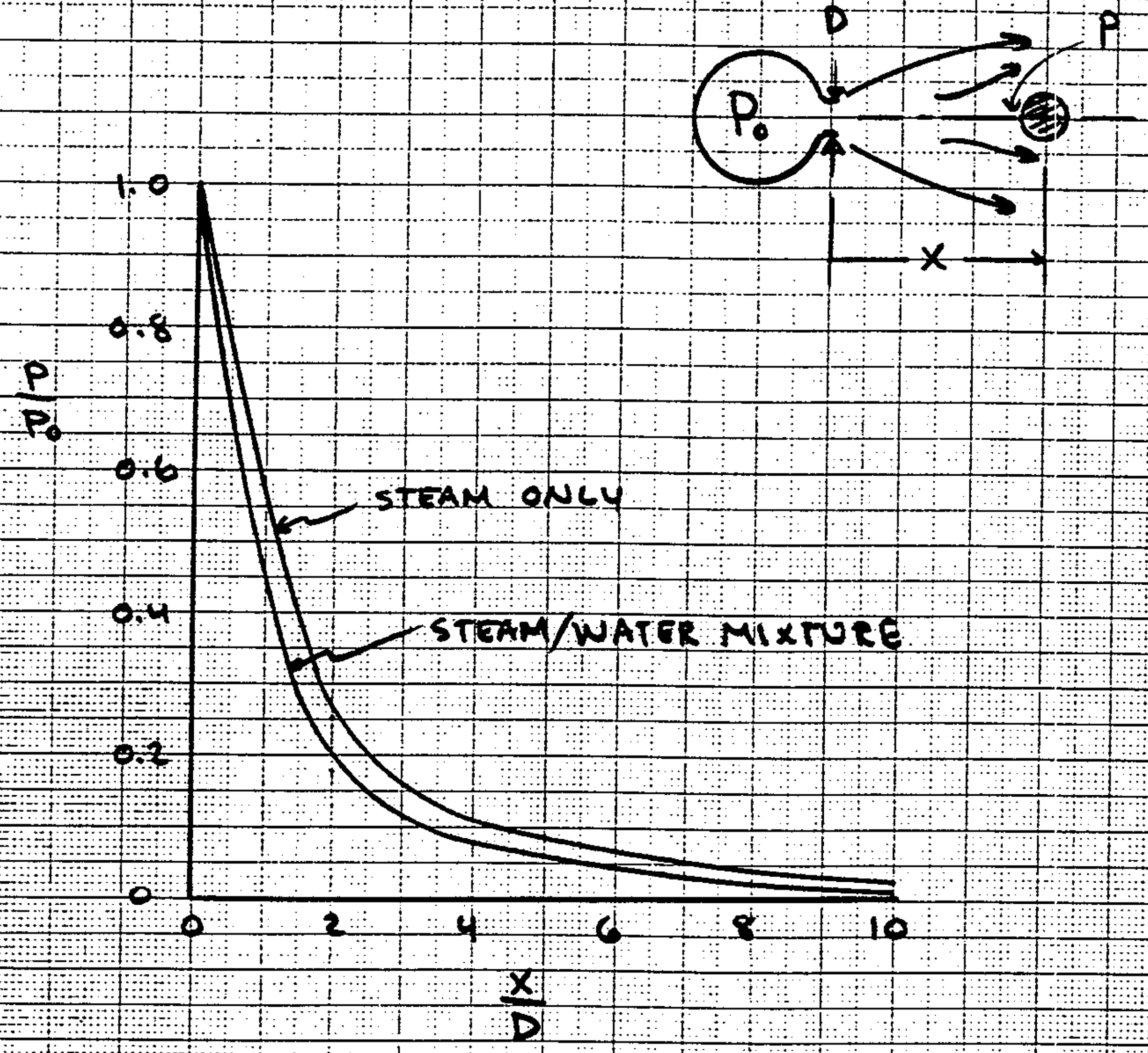

FIG. 4 JET STAGNATION PRESSURE RATIO ON CENTERLINE

APPENDIX

C.D.I. TECHNICAL MEMORANDUM NO. 96-15

VOLUME OF INFLUENCE OF SATURATED STEAM VERSUS SATURATED WATER

Revision B

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1. INTRODUCTION

The BWROG has undertaken an effort to compute volumes about a steam line break where the stagnation pressure exceeds a prescribed amount. This effort is reported in Continuum Dynamics, Inc. Report No. 96-01 entitled "Zone of Influence as Defined by Computational Fluid Dynamics," Revision 1 (Ref. 1). In this report, flow fields about pipe breaks discharging high quality steam were computed using a computational fluid dynamics code, NPARC. The calculations did not simulate pipe break flows which would result in saturated water discharge, since it is known that steam jets at the same stagnation pressures are bounding. This note computes a factor for using the zone of influence as computed from a steam jet to estimate the zone of influence for a jet discharge of saturated water. Specifically, the ANSI/ANS-58.2-1988 jet model is compared under conditions of saturated water and saturated steam from a circular break discharging into an unbounded volume. The volume of the jet flow having an ambient pressure greater than or equal to a prescribed value is computed for saturated water and saturated steam discharges.

2. FORMULATION

A jet exiting from a high pressure reservoir to a low pressure volume is under-expanded and immediately expands to an area A_a in distance L_a (see Figure 1). At distances from the break greater than or equal to $x = 0$, the radial and axial pressure P_j in the jet is given by

$$\frac{P_j}{P_{jc}(x)} = \frac{D_j(x) - 2r}{D_j(x)} \tag{1}$$

where

D_j = jet diameter
P_{jc} = jet centerline pressure

The jet centerline pressure for $x > 0$ is given by

$$P_{jc}(x) = \frac{3C_T P_o A_e}{A_j(x)} \tag{2}$$

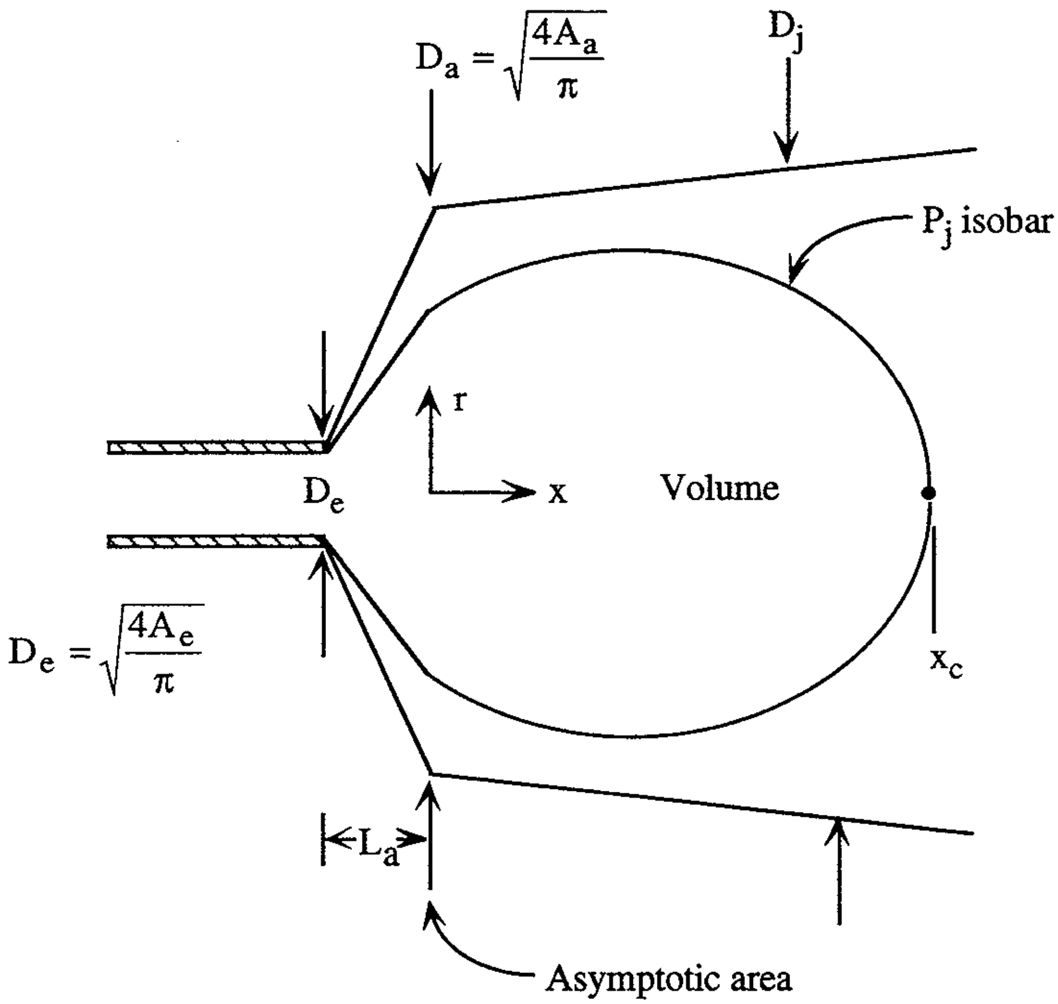
where

C_T = thrust coefficient
P_o = vessel stagnation pressure
A_e = break area = $\pi D_e^2 / 4$
A_j = jet downstream area

$$= A_a\left(1 + \frac{2x}{D_e}\sqrt{\frac{A_e}{A_a}}\tan 10^\circ\right)^2$$

The fluid above or equal to pressure P_j in the length of jet L_a from the break is contained in a volume that may be approximated by a right frustum of a cone. This volume is

$$V = \frac{\pi}{3} r^2 L_a\left(1 + \frac{D_e}{2r} + \frac{D_e^2}{4r^2}\right) \tag{3}$$



Nomenclature that is used in Reference 2 - ANSI/ANS-58.2-1988

Figure 1. Nomenclature used for volume computations.

where r is computed from Equation (1) with $x = 0$ and

$$\frac{L_a}{D_e} = \frac{1}{2}\left(\sqrt{\frac{A_a}{A_e}} - 1\right) \tag{4}$$

A_a/A_e is given in Figure C-4 of Reference 2 as a function of reservoir properties and piping losses.

For $x > 0$, the following integral is computed numerically

$$V = \int_0^{x_c} A(x)dx \tag{5}$$

where

$$A(x) = \left(1 - \frac{P_j}{P_{jc}(x)}\right)^2 A_j(x) \tag{6}$$

where x_c is the value of x where $P_j = P_{jc}(x)$. The volume enclosed by this jet with pressure greater than or equal to P_j is the sum of the volumes in Equations (3) and (5).

3. RESULTS

Assuming no frictional piping losses, the asymptotic area for a saturated steam and water jet for Figure C-4 of Reference 2 is

	Saturated Steam	Saturated Water
$\frac{A_a}{A_e}$ =	43.0	52.0

Assuming negligible losses in discharge, a thrust coefficient of 1.26 is utilized for both saturated steam and water. The results of these volume calculations are shown in Figure 2 for a jet pressure range 0 to 60 psig. Over this range it may be seen that the volume of saturated liquid jet is on the average about 70% of the volume of the saturated steam jet. However, at lower pressures the volumes are comparable.

It is tedious but straight forward to show from Equation (5) that

$$\frac{V}{D_e^3} \cong 0.83\left(\frac{C_T P_o}{P_j}\right)^{3/2} \qquad (7)$$

as

$$\frac{P_j}{P_o} \rightarrow 0$$

and in this limit to the order of this analysis the volume is independent of whether the jet exits from a saturated steam or saturated liquid reservoir.

The pressure P_j can be equated to the pressure measured in the CEESI facility (Ref. 3) at which a given insulation type was observed to be damaged by the CEESI air jet. In this manner, it is possible to determine the volume in which a given insulation type is expected to be damaged should that insulation type be located in this volume.

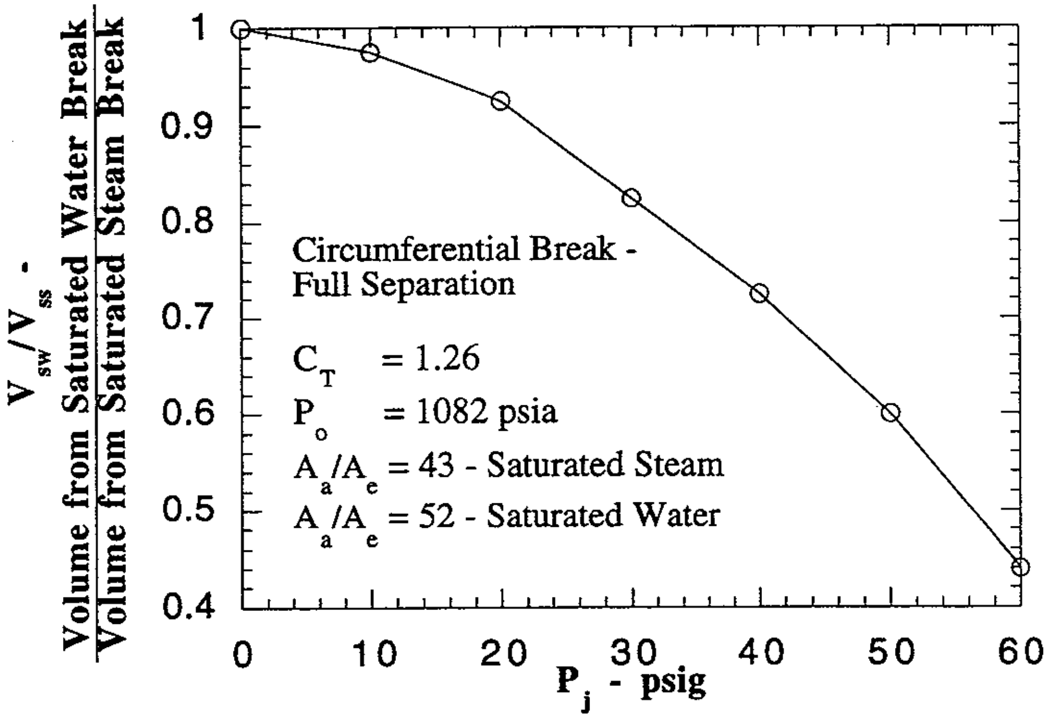


Figure 2. Comparison of the volume of a jet with pressure greater than or equal to P_j from saturated water and steam breaks.

4. REFERENCES

1. Teske, M.E., Boschitsch, A.H., and Curbishley, T.B. 1996. Zone of Influence as Defined by Computational Fluid Dynamics, Revision 1. Continuum Dynamics, Inc. Report No. 96-01, September.

2. American National Standard. 1988. Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture. Report No. ANSI/ANS-58.2-1988. American Nuclear Society: La Grange Park, IL.

3. Munchausen, J.H. 1996. Air Jet Impact Testing of Fibrous and Reflective Metallic Insulation, Draft. Continuum Dynamics, Inc. Final Report prepared for the Boiling Water Reactor Owners' Group Emergency Core Cooling System (ECCS) Suction Strainer Committee, 13 September.

C.D.I. TECHNICAL MEMORANDUM NO. 96-13

EFFECT OF TEMPERATURE ON NET POSITIVE SUCTION HEAD INCLUDING STRAINER HEAD LOSS

Revision A

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September 1996

The pressure at the inlet of an ECCS pump, P_{in}, is a function of the containment pressure, P_c, the water head on the pump, ρgH, the piping losses ΔP_{pipe} and strainer losses, ΔP_s. The schematic in Figure 1 illustrates a simplified ECCS suction strainer connected to pump and these variables. The inlet pressure, P_{in}, can be expressed as:

$$P_{in} = P_c + \rho gH - \Delta P_{pipe} - \Delta P_s \tag{1}$$

where the piping losses, ΔP_{pipe}, include any velocity squared terms associated with the strainer, H is positive when the pump is below the water surface, ρ is the density of water, g is the gravitational constant, and the head loss across the strainer debris bed, ΔP_s, can be expressed as:

$$\Delta P_s = \frac{k\mu(T)tU}{d^2} \tag{2}$$

where k is a constant, μ is the viscosity of water which is a function of temperature (T), t is the bed thickness on the strainer, U is the approach velocity and d is the interfiber spacing for the debris bed.

For the ECCS pump to operate the pump inlet pressure must be above some critical value

$$P_{in} > P_{crit}(T) \tag{3}$$

where P_{crit} is the minimum suction pressure required for the pump. P_{crit} can be expressed as:

$$P_{crit} = P_{vap}(T) + C \tag{4}$$

where P_{vap} is the vapor pressure of water and C is a constant that depends on the pump. This constant can be determined from the specific pump NPSH curves. The difference between P_{in} and P_{crit} is the net positive suction head margin (when the pressures are expressed in terms of head). Combining these four equations yields: Rev. A

$$P_c + \rho gH - \Delta P_{pipe} - C > P_{vap}(T) + \frac{ktU}{d^2}\mu(T) \tag{5}$$

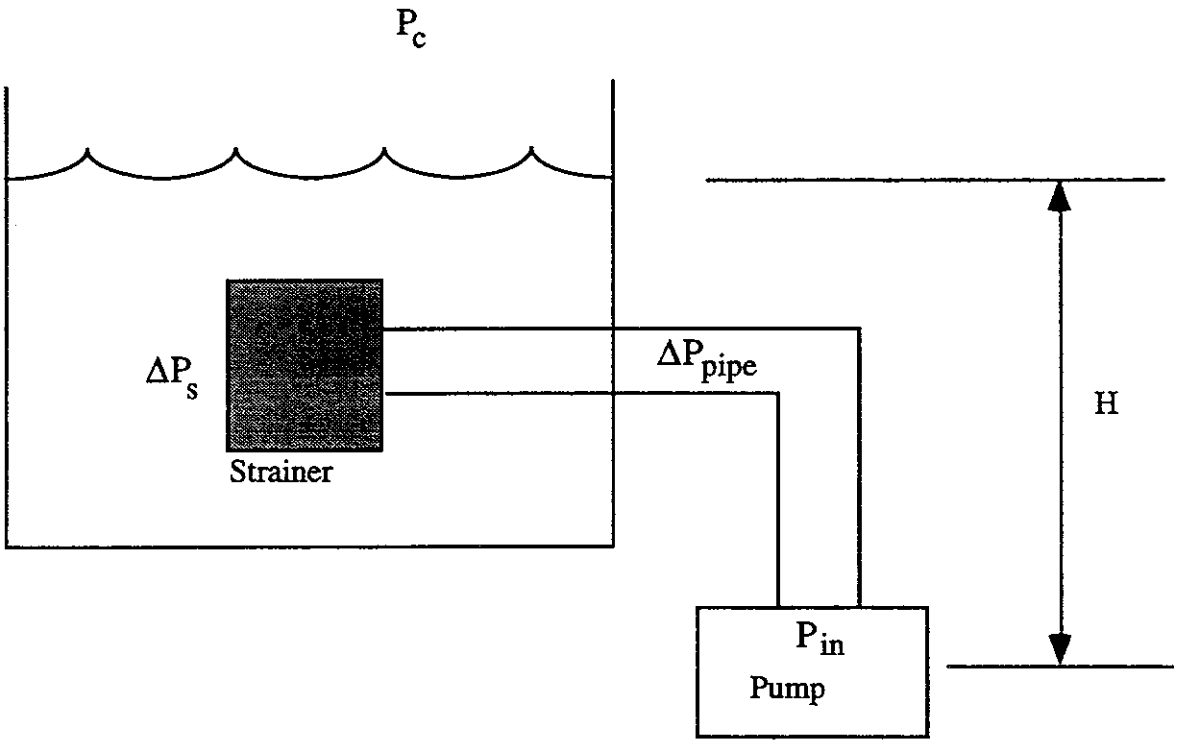


Figure 1. Simplified ECCS schematic.

When the left hand side is greater than the right hand side then there is a net positive suction head margin and the pump will operate. Only two terms are temperature dependent (neglecting the small change in density); the strainer head loss and water vapor pressure. These terms appear on the right hand side of the inequality and they act in opposition, i.e., as the temperature increases the strainer head loss decreases and the vapor pressure increases. Note that the rate of change of strainer head loss depends not only on the temperature, but on the debris loading (or strainer head loss) and is greater when the strainer head loss is higher.

As shown in Eq.(5), the minimum suction head margin occurs when the sum of the vapor pressure and strainer head loss is a maximum. The temperature at which this occurs depends on the temperature range considered and the strainer head loss. The following plots illustrate this point.

Figure 2 plots the vapor pressure and the strainer head loss against temperature. Four strainer head loss curves are shown (head loss of 1, 5, 10, and 20 feet of water at 180°F). Figure 3 plots the sum of the strainer head loss and vapor pressure for these 4 cases. The minimum suction head margin occurs at the maximum of these curves. The maximum margin occurs at the minimum of these curves. Note that as strainer head loss increases the maximum margin occurs at higher temperatures. Rev. A

Figure 4 plots the temperature where the maximum NPSH margin occurs against strainer head loss. If the temperature range of interest is always above the temperature at which the maximum margin occurs, then the minimum margin occurs at the maximum temperature. Conversely, if the temperature range of interest is always below the temperature at which the maximum margin occurs, then the minimum margin occurs at the minimum temperature. If the temperature range of interest includes the temperature at which the maximum margin occurs then the temperature at which there is a minimum suction head margin depends on the temperature range. Rev. A

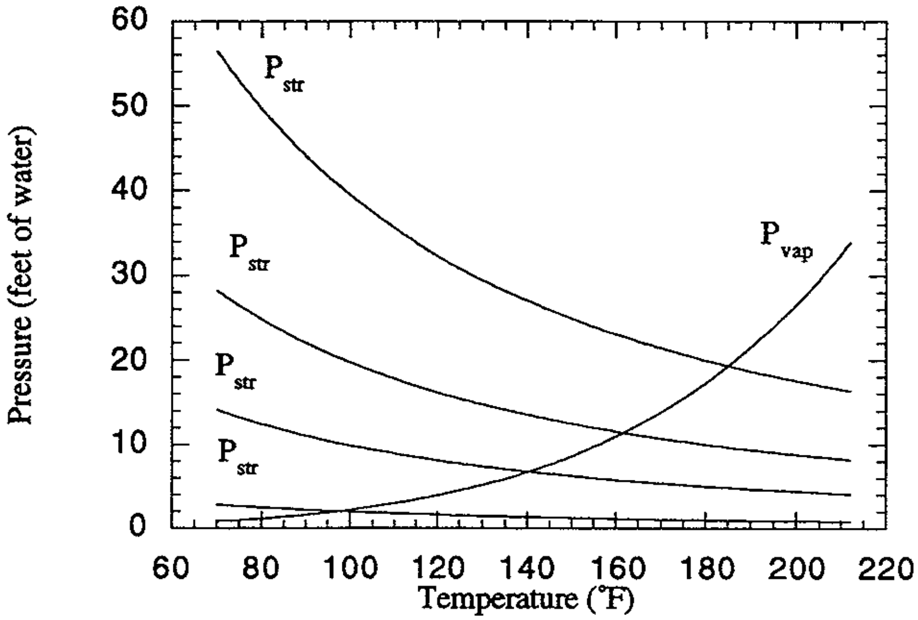


Figure 2. Water vapor pressure and strainer head loss plotted against temperature. Strainer head loss plotted for 1, 5, 10, 20 feet of water @ 180 °F.

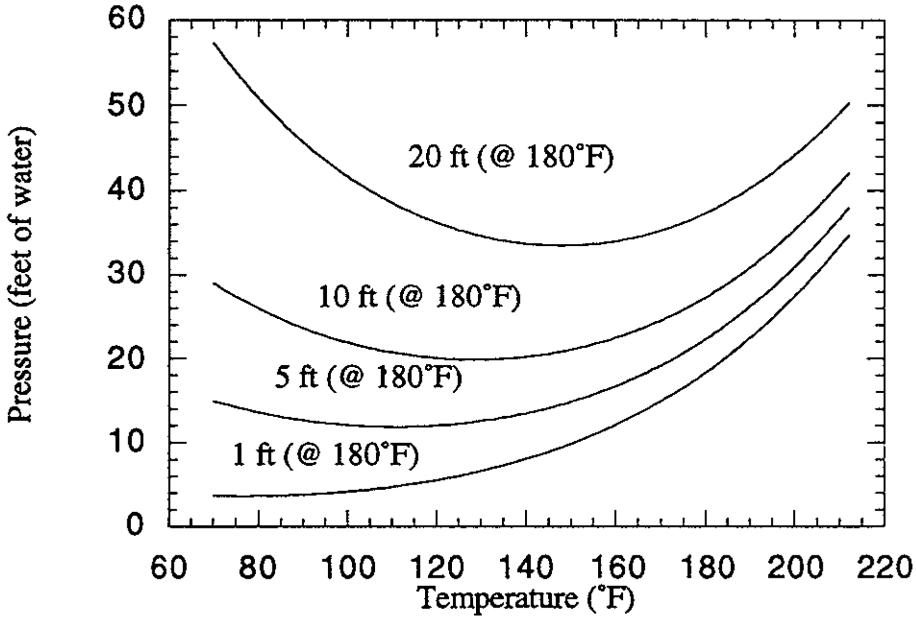
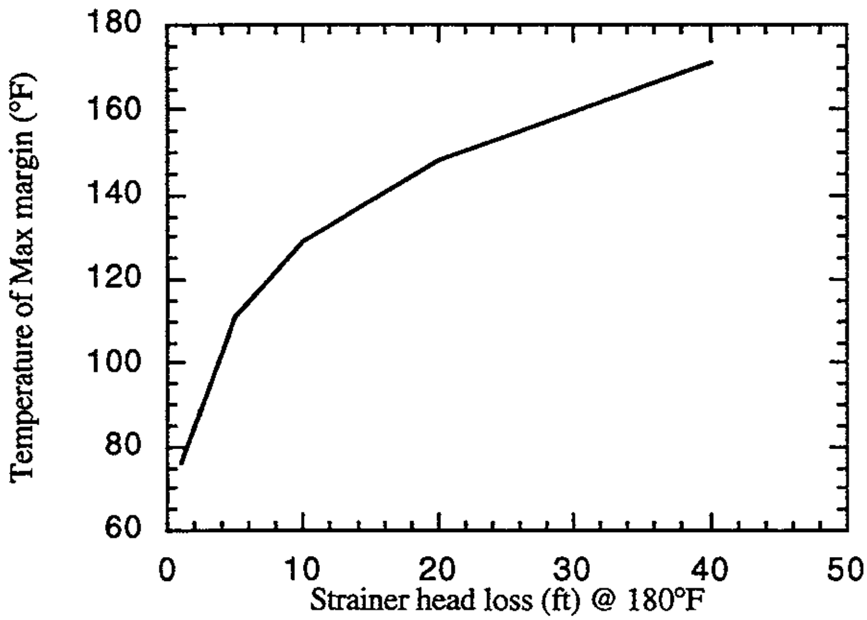


Figure 3. Sum of water vapor pressure and strainer head loss plotted against temperature. Strainer head loss plotted for 1, 5, 10, 20 feet of water @ 180 °F. Minimum suction head margin occurs at the maximum value for a curve. Curves are plotted from 70°F to 212°F.



Rev. A

Figure 4. Temperature where the maximum net positive suction head margin occurs versus strainer head loss. Strainer head loss values are in feet of water at 180°F.

The following steps can be followed to determine the minimum suction head margin. Rev. A

(1) Determine the temperature range of interest. The lowest expected temperature exists early in the event after all of the operational and LOCA generated debris reaches the suppression pool and is processed by the suction strainer. The maximum suppression pool temperature is determined from the plant technical specifications. The maximum temperature is typically between 185° and 212°F. Rev. A

(2) Using data from the Alternate Strainer Test Report (or strainer vendor provided data), calculate the strainer head loss at any selected temperature and at the minimum and maximum temperatures. Rev. A Given the strainer head loss at a temperature (T_1), the strainer head loss at any other temperature (T_2) can be determined by multiplying the strainer head loss by the water viscosity of the desired temperature (T_2) divided by the viscosity at the temperature for which the head loss was known (T_1).

$$\Delta P_s(T_2) = \Delta P_s(T_1)\frac{\mu(T_2)}{\mu(T_1)} \qquad (6)$$

where T_1 is the temperature for which the strainer head loss is known and T_2 is the temperature for which the strainer head loss is to be calculated. Figure 5 shows the viscosity of water as a function of temperature. Rev. A

(3) Determine the vapor pressure at the minimum and maximum temperature from Figure 2 (or a thermodynamic table or chart).

(4) Add the vapor pressure and corresponding strainer head loss together. The temperature at which the this sum is larger is the temperature at which the net positive suction head is a minimum.

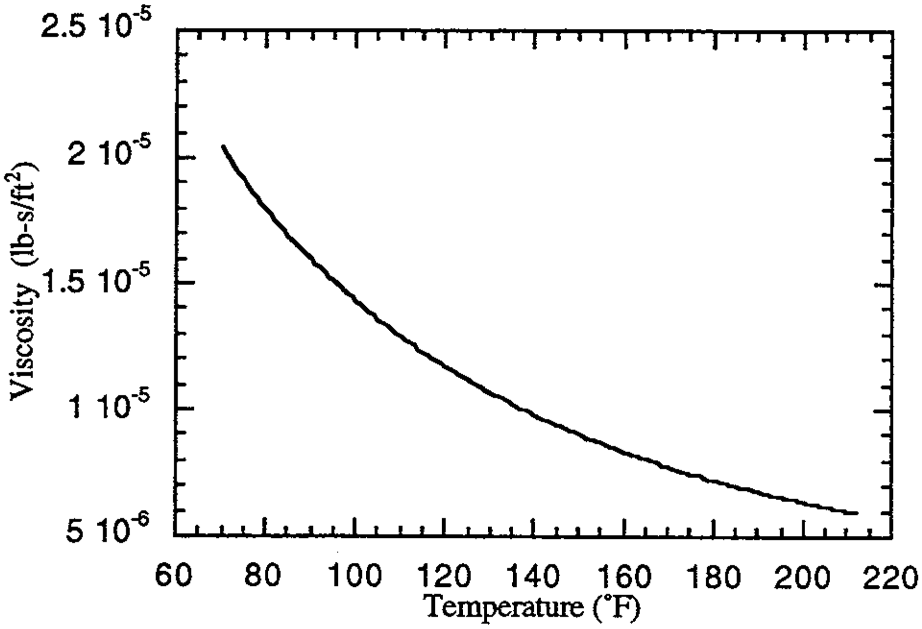


Figure 5. Viscosity of water as a function of temperature.

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PIPE BREAK PROBABILITIES IN BOILING WATER REACTORS

NOVEMBER 1993

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1.0 INTRODUCTION

This paper provides information on the probability of pipe breaks in boiling water reactors (BWRs). It is intended for consideration of the NRC and its contractor, Science and Engineering Associates, Inc. (SEA) in preparation of a regulatory analysis of the potential for BWR emergency core cooling system suction strainer blockage during loss of coolant accidents (LOCAs). NRC's planned approach to this analysis was presented in a public meeting at NRC's Bethesda Maryland offices on September 15, 1993.

2.0 ANALYSIS APPROACH PRESENTED AT SEPTEMBER 15, 1993 MEETING

As documented in Reference 1, NRC indicated that they intend to follow the analysis methodology used in Appendix D of NUREG-0869, Rev. 1 (Reference 2). A key input to the NUREG-0869 analysis was Table 1 (Attachment A to this paper) which listed PWR piping failure probability as a function of pipe size and weld type.

NRC intends to develop a similar table of pipe break probabilities which are representative of BWRs, and then implement an analysis approach similar to that performed in NUREG-0869. At the September 15 meeting, SEA indicated it intended to use the BWR piping failure probabilities reported in NUREG/CR-4792, Volume 1 (Reference 3) as a fundamental input to development of a BWR pipe failure rate table similar to that used in the NUREG-0869 analysis.

3.0 REVIEW OF THE NUREG/CR-4792 ANALYSIS

NUREG/CR-4792 reports on the results of an analysis of BWR reactor coolant piping performed by Lawrence Livermore National Laboratory for the NRC. This analysis used a probabilistic fracture mechanics methodology to estimate lifetime system leak and double ended guillotine break (DEGB) probabilities for a representative BWR (Brunswick). The PRAISE computer code was used to perform this analysis. The analysis included evaluation of main steam, feedwater, and recirculation system piping. Results indicated that, other than for Intergranular Stress Corrosion Cracking (IGSCC) related failures, the probability of leak ranged from 6.0E-8 to 1.0E-6 per reactor year, and the probability of DEGB ranged from 1.0E-12 to 3.8E-12 per reactor year.

The NUREG/CR-4792 analysis also includes an estimated DEGB break probability for 304 stainless steel recirculation systems due to IGSCC of 1E-3 per reactor loop-year. The IGSCC analysis is discussed in detail in section 3.1.

The balance of this paper addresses the applicability of the NUREG/CR-4792 IGSCC analysis, and identifies other sources of BWR recirculation system piping failure rate estimates.

3.1 NUREG/CR-4792 IGSCC Analysis

Lawrence Livermore used the PRAISE code to perform an analysis of the probability of pipe leakage and pipe rupture due to IGSCC for a selected BWR (Brunswick). Two recirculation system configurations were evaluated: 1) the original Brunswick system configuration as constructed with 304 stainless steel, and 2) a proposed replacement system configuration made with 316NG stainless steel. The primary purpose of the NUREG/CR-4792 IGSCC analysis was to evaluate the relative resistance to IGSCC of the replacement 316NG material as compared to the original 304 stainless steel. As such, a number of simplifying assumptions were used which introduce conservatism. These included:

- No credit was taken for the probability that cracks would be detected through in-service inspection prior to failure;
- The analysis of the 304 stainless steel system did not include analysis of the potential benefits of alternate mitigative actions which could have been taken, such as weld overlay, stress improvement, or use of hydrogen water chemistry control;
- Conservative assumptions regarding stress conditions.

The results of the analysis indicated that the original, as constructed Brunswick 304 stainless steel recirculation system would be expected to have a pipe break probability of approximately 1E-3 per year due to IGSCC.

Use of this estimated pipe failure probability of 1E-3 per year as input to the regulatory analysis of the potential for BWR ECCS strainer blockage is inappropriate for the following reasons:

1) The NUREG/CR-4792 analysis calculates break probabilities for a 304 stainless steel recirculation system with no actions taken to mitigate IGSCC, and with no credit for inspection, which detects cracks prior to pipe failure.

   This condition does not exist at any operating BWR in the United States. All operating BWRs have taken IGSCC mitigative actions in response to NRC Generic Letter 88-01 and NUREG-0313 Revision 2 (Reference 4). A summary of the mitigative actions taken in the industry as of March, 1988 is presented in Attachment B. This summary was excerpted from Reference 5.

2) The IGSCC mitigative actions taken by the industry have been demonstrated to be effective through an extensive testing program performed by the BWROG/EPRI/GE in the 1980s. NRC recognized the efficacy of these corrective actions in NUREG-0313, Rev. 2.

3) The IGSCC analysis performed in NUREG/CR-4792 was performed primarily to compare the relative performance of 304 and 316NG material, not to provide a definitive estimate of actual pipe failure probability due to IGSCC.

As discussed in Appendix D of NUREG-1061 Volume 1 (Reference 6), consideration of actual plant operating history indicates that there is little statistical support for the hypothesis that piping failure rates typically used in industry probabilistic risk assessments were significantly in error due to IGSCC, even though the presence of IGSCC may point to a greater failure precursor population. As discussed in section 4.0 of this paper, studies of pipe failure based on review of operational experience indicate that "upper bound" failure rates for BWR recirculation system piping are significantly lower than the value of 1E-3 per year calculated in NUREG/CR-4792.

4.0 Alternate BWR Recirculation System Pipe Failure Rate Estimates

BWR recirculation piping systems are, by design, robust structures. Due to the robustness of design, and the use of inservice inspection methods which detect small flaws before larger failures occur, there is no failure history to allow calculation of actual failure probabilities using operational experience. This leaves three options in preparing estimated pipe failure rates:

1) Use of analytical approaches which estimate pipe failure rates by modeling flaw initiation and growth to failure;

2) Use of operational experience data to generate "upper bound" failure rates; or

3) Use of a combination of operational experience and analytical results.

Each of these approaches is discussed in sections 4.1 through 4.3 below.

4.1 Analytical Estimates of Pipe Failure Probability

NUREG/CR-4792 was identified above as one source of information regarding analytically derived failure estimates. This analysis concluded that probabilities of double ended guillotine breaks are extremely low in the absence of active IGSCC. Table 3.6 from Reference 3 provides best estimate DEGB probabilities for recirculation (absent IGSCC), feedwater, and main steam system piping on the order of 1E-10 to 1E-11 <u>over the forty year life of the plant.</u>

The fact that these probabilities are much lower than those commonly used in probabilistic risk assessments should not be taken as evidence that they are incorrect. Numbers typically used in PRAs are very conservative bounds, as shall be shown in section 4.2.

NUREG-1061 Volume 3 (Reference 11) also includes results of probabilistic fracture mechanics analyses performed for the NRC. These analyses resulted in similarly low estimates of pipe failure probability.

4.2 "Upper Bound" Pipe Break Estimates Based on Operational Experience

Several studies have been performed over the last fifteen years which develop bounding pipe failure rates based on available operational experience. An early study of this type was EPRI Report NP-438, "*Characteristics of Pipe System Failures in Light Water Reactors*", (Reference 9) published in August of 1977. A second example is NUREG/CR-4407, "*Pipe Break Frequency Estimation for Nuclear Power Plants*", (Reference 10) published in May 1987. This study was prepared by EG&G at the Idaho National Engineering Laboratory for the NRC under a DOE contract. EPRI has subsequently published EPRI Report TR-100380, "*Pipe Failures in US Commercial*

Nuclear Power Plants" (Reference 7) and an update to that study, EPRI Report TR-102266 (Reference 8), published in April of 1993.

Each of these studies includes an evaluation of actual operational experience to estimate an upper bound for break probabilities in major piping which has the potential for initiating a LOCA. While there are some differences in the methodologies used, the trend shows decreasing LOCA probabilities over time as additional operational experience is gained.

The most current of these studies is that reported in Reference 8. The methodology and results of that study are discussed in Section 4.2.1.

4.2.1 EPRI Pipe Failure Study (TR-102266)

The EPRI pipe failure study was undertaken to provide EPRI member utilities with an updated source of pipe failure rate data and methodology for use in preparing or updating individual plant examinations using probabilistic risk assessment methods.

Methodology

The methodology used in the EPRI Pipe Failure Study included review and analysis of failure data, primarily from Licensee Event Reports and from Nuclear Power Experience (published by the S. M. Stoller Corporation of Boulder, Colorado). Representative plants for each reactor type were selected to identify the number of "pipe sections" included in various plant systems as a function of pipe size. A "pipe section" is defined as a segment of piping, between major discontinuities such as valves, pumps, reducers, tees, etc. This definition is taken from WASH-1400. A pipe section is typically 10 to 100 feet long, and contains from four to eight welds. Each section can also contain several elbows and flanges. Instrumentation connections are not considered "major discontinuities". Use of pipe sections as the basis for population counts and probability of failure estimates was selected for the following reasons:

- As defined, numbers of pipe sections in any given plant or system can be relatively easily determined by review of plant piping and instrumentation drawings. Determination of actual piping lengths and actual numbers of welds would require much more exhaustive review of isometric drawings, or actual walkdowns.

- Pipe failure rates are dependent on pipe length and the number of welds, with the number of welds being significantly more important than the overall length of pipe. Use of pipe sections as the basis for normalization of the failure data provides a closer approximation of the actual population of interest than normalizing on the basis of the entire system or the entire plant, while still maintaining reasonable ease of application.

Failure reports were analyzed to identify the system within which each reported failure occurred, and events with greater than 50 GPM of leakage were considered "ruptures" (i.e., breaks). Estimated break probabilities per pipe section per hour were then calculated using the number of operating hours for each system in domestic nuclear power plants.

Estimated break probabilities for piping within specific systems as well as generic estimated break probabilities are provided in the report. The current concern is to identify an applicable, conservative estimate of BWR recirculation system pipe break probabilities due to all causes, including IGSCC. Specific results of the EPRI Pipe Failure Study which are applicable to estimation of pipe break probabilities in BWR recirculation systems are summarized below.

Results for BWR Recirculation Systems

No BWR recirculation system pipe breaks have occurred to date. A bounding estimate of the pipe break probabilities was made in the EPRI pipe failure study by assuming one break where in fact none had occurred, and calculating a probability based on the actual number of years of operational experience accumulated. The pipe failure study update (Reference 8, operational experience through August, 1991) resulted in the estimated "upper bound" break probabilities for BWR recirculation system piping listed below.

[It should be noted that the break probabilities listed below should not be interpreted as the "probability of a double ended guillotine rupture of a pipe of size x". They in fact represent the probability of a break in the size range listed occurring in any recirculation system piping. For example, the probability of a small break includes the probability that a small pipe (<2") will completely rupture, as well as the probability that larger pipes (up to the largest in the system) will experience a small break. Obviously, large breaks can only occur in large pipes. The probability estimated is that of a certain break size occurring, not the probability of complete rupture of a pipe of a certain size.]

Small Break	(0.5"≤ break size < 2")	1.60E-4/year
Intermediate Break	(2"≤ break size <6")	2.85E-5/year
Large Break	(6"≤ break size)	3.76E-5/year

A further update to this estimate using operational experience through September, 1993 and the Reference 8 methodology was prepared (Reference 12). The estimated break probabilities updated through September, 1993 are:

Small Break	(0.5" ≤ break size < 2")	1.38E-4/year
Intermediate Break	(2" ≤ break size <6")	2.46E-5/year
Large Break	(6" ≤ break size)	3.26E-5/year

Again, the decreasing value of the estimated upper bound is due to the increased number of reactor-years of operating experience without any incidence of reactor coolant system pipe breaks.

As noted above, the Reference 8 methodology included assuming one break in a population where in fact no breaks have occurred. An alternate (yet still conservative) statistical approach to developing a point estimate was used in NUREG/CR-4407 (Reference 10). This approach results in use of a numerator of 0.23 rather than 1 when calculating a "point estimate" break probability for a population in which no breaks have in fact occurred. Revised break probabilities using the NUREG/CR-4407 statistical approach and operational experience through September, 1993 were calculated in Reference 12. These revised BWR recirculation system pipe break probabilities are:

Small Break	(0.5" ≤ break size < 2")	3.17 E-5/year
Intermediate Break	(2" ≤ break size <6")	5.66E-6/year
Large Break	(6" ≤ break size)	7.51E-6/year

This last set of break probabilities are the most appropriate "upper bound" values currently available, for the following reasons:

- They incorporate the most current available operational experience; and
- They use a conservative statistical method of estimating break probabilities in a population in which no breaks have occurred.

It should be noted that the relative probability of small breaks, medium breaks, and large breaks listed above would not be expected to be a true representation of the relative likelihood of these events. No breaks of any size have actually occurred in BWR recirculation system piping. The difference in the estimated upper bound break probabilities for breaks of various sizes is an artifact of the methodology used to develop the estimates, and should not be interpreted as the actual relative probability of breaks of various sizes.

4.3 Combined Analytical and Operational Experience Based Estimate

Both the analytical method and the operational experience method have shortcomings when used alone. Reliance on analytical results alone is difficult, as they are open to criticism as to whether all potential failure mechanisms have been identified and conservatively modeled. Operational experience based estimates have the advantage of being independent of any assumptions about failure mechanisms or modeling methods, but are limited to being able to provide only conservative "upper bound" estimates in cases where no actual breaks have occurred in the population being studied.

An approach which combines the strengths of analytically based estimates with the conservative "upper bounds" developed by review of operational experience is presented in sections 4.3.1 through 4.3.2.

4.3.1 Analytical Results - Relative Probability of "Small" versus "Large" breaks

NUREG/CR-4792 calculated the relative probability of leaks and breaks in various BWR piping systems. Tables 3.5 and 3.6 of Reference 3 provide the results of these calculations. These tables indicate that the probability of a "leak" is approximately 10^5 times greater than the probability of a double ended guillotine break. Additional information on the relative probability of leaks and breaks is provided in NUREG-1061, Volume 3 (Reference 11).

From these analytical studies, one can infer that large diameter pipes are in fact much more likely to experience small breaks than large breaks.

As noted in section 4.2.1 of this paper, the "upper bound" pipe break probabilities calculated by reviewing actual operational experience do not reflect the relative frequency vs. break size which the analytical estimates would indicate are appropriate. Again, the primary reason is that no breaks of any size have occurred, and no meaningful statement about the relative probability of small versus large breaks can be made through review of operational experience.

An alternate estimate of the probability of large breaks can be developed based on the NUREG/CR-4792 analysis of the relative probability of leaks and breaks, and the conservative upper bound estimate of the probability of small breaks based on operational experience.

4.3.2 Probability of Small Breaks In Large Pipes - Operational Experience

A value of 3.17E-5 per reactor year is presented in section 4.2.1 as an "upper bound" for the frequency of a break size of between 0.5" and 2" in BWR recirculation system piping. This frequency is comprised of the probability of such a failure in piping 2" and less in size, as well as the probability of a break of between 0.5" and 2" occurring in piping greater than 2" in diameter. As documented in Reference 12, the individual contributions of each factor are a probability of 2.61E-5 per reactor year that a small break will occur in a pipe less than 2" in diameter, and a probability of 5.66E-6 per reactor year that a small break will occur in a pipe larger than 2".

4.3.3 Combined Analytical and Operational Experience Based Large Break Probability

An estimate of the probability of a large break can be made by applying a relative probability factor to the "upper bound" estimate of the probability of a large pipe experiencing a small break derived from operational experience. For illustration, a value of 5.66 E-6 per reactor year is used as the "upper bound" probability of a small break in a large pipe. The resultant large break probability estimates as a function of estimated probability relative to a small break are listed below:

"Upper Bound" Probability of a Small Break in a Large Pipe	Relative Probability of a DEGB as Compared to a Small Break	Estimated "Large Break" Probability
5.66E-6	1.0E-3	5.66E-9
5.66E-6	1.0E-4	5.66E-10
5.66E-6	1.0E-5	5.66E-11

The above results indicate that, even for relative probabilities of small versus large breaks significantly lower than those developed in NUREG/CR-4792 and NUREG-1061, the estimated probability of a large break would be significantly lower than the "upper bound" value presented in section 4.2 above.

5.0 Industry Individual Plant Examination Assumed Pipe Failure Rates

Plant specific probabilistic risk assessments have been performed for most operating plants in response to the Generic Letter 88-20 on Individual Plant Examinations. These analyses include assumed pipe failure rates as part of the evaluation of LOCA accident sequences. Pipe failure rates commonly used as input to BWR IPE analyses are shown below. The values from WASH-1400 are also provided for comparison:

	Typical BWR IPE Value	WASH-1400
Large Pipe (> 6 inches)	7.0E-4 per reactor year	1.0E-4 per reactor year
Medium Pipe (> 4 inches)	3.0E-3 per reactor year	3.0E-4 per reactor year
Small Pipe (> 1 inch)	8.0E-3 per reactor year	1.0E-3 per reactor year

The BWR IPE pipe failure rates listed above were initially developed by PRA analysts for use in the Shoreham PRA. A 1977 study of pipe failures performed by EPRI (Reference 9) was a primary source of data used by the analysts. This study evaluated operating experience available as of August 1976. That study included estimates of the probability per plant year of a break in any high pressure piping system. All of the high pressure pipe breaks identified in this report occurred in either feedwater, condensate, or steam piping systems. No breaks in primary plant high pressure systems were identified. The PRA analysts who developed the above estimates for use in the Shoreham PRA applied the failure history of balance of plant high pressure piping systems to primary systems, resulting in a conservative estimate of LOCA probability.

Using the failure rates listed above, the IPE analyses typically show LOCAs are not dominant contributors to the risk of core damage (less than 10% of the total core damage frequency.) There has been little incentive to use lower, updated values for the pipe failure probabilities in IPE models, as even with the relatively high values currently used, LOCA accident sequences are not large contributors to risk. Therefore, a common approach in developing IPEs has been to use the commonly accepted numbers, rather than using the lower values which are justified based on evaluation of operating experience since 1976.

6.0 CONCLUSIONS

The proposed BWR recirculation system pipe failure rate for IGSCC related failures of 1.0E-3 per reactor year developed in NUREG/CR-4792 is not appropriate for use in the NRC regulatory analysis of the ECCS strainer blockage issue. The 1.0E-3 per reactor year value was calculated for a 304SS recirculation system with no credit for inservice inspection, and with no credit for actions taken by the industry to mitigate the effects of IGSCC.

A conservative upper bound for the actual BWR recirculation system pipe failure probability is that derived from analysis of actual operating experience. Based on data through September, 1993, the "upper bound" for the probability of a large break LOCA (6" or greater) in BWR recirculation system piping is 7.51E-6 per reactor year.

The actual large break frequency for BWR recirculation system piping is most likely substantially (several orders of magnitude or more) below the upper bound calculated based on currently available operational experience. Application of the analytically derived relative probabilities of small and large breaks to the experience based probability of a small break would result in an estimated large break frequency several orders of magnitude lower than 7.51E-6. The NUREG/CR-4792 analysis calculated large pipe DEGB frequencies on the order of 1E-12 per reactor year, exclusive of IGSCC. With the effective actions taken by the industry to mitigate IGSCC, the actual large break failure rate lies at some intermediate point between 7.51E-6 and 1.0E-12 per reactor year.

Pipe failure rates currently included in BWR Individual Plant Examination risk assessments are significantly higher than those which can be justified using current plant operational experience. There has been little incentive for IPE analysts to use the more current numbers, as even with the older, higher failure rate numbers, LOCA has not been a dominant contributor to the risk of core damage.

7.0 REFERENCES

1. Memorandum, Aleck W. Serkiz, NRR, NRC, to Robert L. Baer. Division of Safety Issue Resolution, NRC, Meeting Minutes - September 15 1993 Meeting to discuss Potential for Loss of ECCS in BWRs Due to LOCA Generated Debris, dated September 21 1993

2. NUREG-0869, "USI A-43 Regulatory Analysis", October 1985

3. NUREG/CR-4792, Volume 1, "Probability of Failure in BWR Reactor Coolant Piping, Vol. 1: Summary Report, March 1989

4. NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping", Final Report, January 1988

5. EPRI Report NP-7154, "Summary of BWR Owners Group Intergranular Stress Corrosion Cracking Research Program, 1979 - 1988", Final Report, February 1991

6. NUREG-1061, Volume 1, "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, Investigation and Evaluation of Stress Corrosion Cracking in Piping of Boiling Water Reactor Plants", August 1984

7. EPRI Report TR-100380, "Pipe Failures in U. S. Commercial Nuclear Power Plants", Interim Report, July 1992

8. EPRI Report TR-102266, "Pipe Failure Study Update", Final Report, April 1993

9. EPRI Report NP-438, "Characteristics of Pipe System Failures in Light Water Reactors", Topical Report, August 1977

10. NUREG/CR-4407, "Pipe Break Frequency Estimation for Nuclear Power Plants", May 1987

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

12. EPRI Memo to File, "Pipe Failure Study Update with Operational Experience Through September 1993", November 5, 1993

ATTACHMENT A

EXCERPT FROM NUREG 0869, REVISION 1

TABLE 1
Piping failure probability estimates

Pipe size (inches)	Diameter class	Failure probability (1/Rx-Yr)	Weld failure probability distribution		
			Weld type 1	Weld type 2	Weld type 3
D	J	P_j	W_n	W_a	W_e
2 to <6	1	$3x10^{-4}$	0.7	0.15	0.15
6 to <10	2	$4x10^{-5}$	0.5	0.30	0.20
10 to <16	3	$3x10^{-5}$	0.5	0.30	0.20
16 to 28	4	$3x10^{-6}$	0.5	0.30	0.20
≥28	5	$3x10^{-6}$	0.5	0.30	0.20

$\Sigma P_j = 3.76x10^{-4}/Rx\text{-}yr$

Weld Type 1 = fabricated and non-standard joints
Weld Type 2 = high restraint joints and tees with joints
Weld Type 3 = elbows, reducers, and straight piping runs with jo

ATTACHMENT B

EXCERPT FROM EPRI REPORT NP-7154

Table 6-1

ALL U.S. OPERATING BWR'S
RESULTS FROM 37 PLANTS

PIPING MITIGATION STATUS SUMMARY

PRELIMINARY
RESULTS
3/8/88

(Welds >-4.0" dia. and >200F during power operation)

CATEGORY	FLEET TOTALS	LARGE DIA. RE-CIRC	RESD HEAT RMVL	CORE SPRAY (LP&HP)	RX WTR CLN UP	CNTR ROD DRV RTRN	RE-CIRC BY PASS	HEAD SPRY	ISOLA-TION COND-ENSOR	LPCI
SYSTEM MATERIAL(S)										
TOTAL ORIGINAL WELDS	10,056	4222	1189	1248	1019	353	671	776	408	170
ORIGINALLY IGSCC SUSCEPTIBLE	7053	3613	661	715	717	307	448	188	313	91
REMEDIES:										
1. Replaced With Resistant Matl	2057	1010	130	352	325	0	52	37	150	1
2. Stress Improvement	2155	1762	265	45	76	0	0	0	7	0
3. Solution Heat Treated	518	491	2	4	7	0	0	0	12	2
4. Corrosion Resistant Clad	227	214	0	4	0	0	0	0	9	0
5. Heat Sink Welded	8	0	2	0	6	0	0	0	0	0
6. Line Removed or Rerouted	695	16	0	0	2	237	292	134	14	0
7. HWC Protected	962	441	38	183	81	19	40	87	31	42
REMAINING SUSCEPTIBLE WELDS	1645	431	249	294	265	64	80	87	120	55
NON-SUSCEPTIBLE WELDS (%)	8411 84%	3791 90%	940 79%	954 76%	754 74%	289 82%	591 88%	689 89%	288 71%	115 68%
8. Weld Overlay Repair	279	197	30	9	9	0	10	0	24	0
SUSCEPTIBLE WELDS, EXCLUDING WELD OVERLAY REPAIRS	1366	234	219	285	256	64	70	87	96	55
NON-SUSCEPTIBLE WELDS, INCLUDING WELD OVERLAY REPAIRS (%)	8690 86%	3988 94%	970 82%	963 77%	763 75%	289 82%	601 90%	689 89%	312 76%	115 68%

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Summary of Debris Washdown Experience

Introduction

This paper provides a summary of available information regarding the washdown of debris from the drywell to the wetwell in BWRs. It includes washdown as a result of break leakage flow and from drywell spray flow. The washdown transport information from BWROG sponsored tests are not included herein but are reported separately in Reference 9.

Applicability/Significance

For many plants, debris transport as a result of washdown, may be a significant contributor of transport to the wetwell. The amount of debris carried over to the wetwell as a result of washdown will have a direct impact on the capability of any passive strainer to function during the postulated LOCA.

This issue is of significance to all plants either because they have fibrous insulation (including plants with predominantly RMI and small amounts of fibrous insulation) or because of other sources of drywell debris.

Summary of Applicable Test Results/Analyses

PP&L Tests

Testing sponsored by PP&L was conducted at Alden Labs to better understand material transport. The results are reported in a May 1994 PP&L report [1] which includes as a reference a report prepared by Alden [2]. The transport tests were designed to 1) determine whether fluid velocities on the order of those expected in the suppression pool would keep material suspended and 2) investigate the transport of material over a weir at various heads. The information from this second part is useful in assessing whether debris would remain on the drywell floor or be drawn over the drywell downcomer weir by leakage flow. Only the transport tests over the weir are discussed below.

The following materials were tested in the flume:

1. NUKON™ fiberglass insulation in fiber and clump forms
2. corrosion products (iron oxide rust)
3. inorganic zinc paint particles and flakes
4. iron oxide particulates (sludge)
5. Koolphen K insulation vapor barrier paper
6. stainless steel reflective metallic insulation foil (Transco). In 1" X 1", 2" X 2", and 3" X 3" squares. For each size, one specimen is flat, one is folded in the center to a 90° angle, and one is crumpled.

The transport tests were performed in a one-dimensional flow with water velocities at 0.2 ft/sec to 1.0 ft/sec. The weir test used a 12" high weir at one end of the flume. Flume flow was adjusted upward to change the head over the weir. Material was introduced to the flume at various locations and the influence of the weir flow was observed. Material which passed over the weir was collected on a screen at the end of the flume.

The results of the tests indicate that:

- for the range of flow velocities tested, it was difficult to uniformly mix the insulation debris in the water. Only the smallest particles (e.g., NUKON™ individual fibers) showed a tendency to be lifted off of the flume floor. Large NUKON™ clumps tended to roll along the flume bottom in tumble weed fashion

- most of the sludge was fully mixed at transport velocities higher than 0.3 ft/sec

- reflective metal pieces were rarely transported downstream or entrained for the range of transport flows tested. All the larger pieces settled on the flume bottom

- the influence of the weir in drawing debris over the edge increased with increasing head over the weir. At higher weir heads most of the individual NUKON™ fibers were drawn over the edge. NUKON™ clumps were drawn over from as far as 5 ft from the weir edge at a weir head of 0.25 ft.

- Most of the stainless steel reflective metallic pieces dropped to the floor and were not transported over the weir

- the transport characteristics of NUKON™ fibers were found to be independent of temperature for the range of temperatures tested (60° to 90° F).

The PP&L tests also considered the affects of a non-uniform distribution of flow into the downcomers. A range of "flow fractions" (the number of downcomers receiving flow divided by the total number of downcomers) was used to evaluate a change in flow velocity and which would in turn affect debris carryover into the suppression pool. Flow fractions from 0.2 to 1.0 were evaluated but the testing was not designed to determine which flow fraction would be appropriate. For a constant volume of break leakage flow, a lower flow fraction (i.e., fewer downcomers passing the flow) results in a higher head over the lip of the downcomer and a corresponding increase in flow velocity. Depending upon the debris type and the velocity required for transport, this higher flow velocity may increase debris transport to the suppression pool.

It is important to note that the PP&L tests were conservative. The model used at Alden had a straight weir and the actual downcomers are circular. The velocity field in a circular weir is inversely proportional to radial distance from the weir. The approach velocity drops rapidly for a circular weir with increasing distance, while it was constant for the flume tests at Alden. This means that the flow velocities at PP&L would actually be lower than indicated by the tests and that there would be less likelihood of transport over the weir and into the pool.

Another conservatism is that the Alden analysis for PP&L assumed a constant 58,000 gpm flow rate onto the drywell floor. For PP&L, this is the flow with all four RHR and two CS pumps running. During an actual event it is unlikely the containment spray pumps would be operating and contributing to the flow.

Evaluation of the PP&L Tests

The PP&L data is valuable in better understanding the transport of fibrous debris over a weir. It also shows that there is little likelihood of transport of stainless steel RMI debris over a weir even with conservative flow velocities.

The PP&L report conservatively assumes the full 58,000 GPM of flow available from all four RHR and two core spray pumps is available for washdown transport. It does not separate the break leakage flow component from the containment spray component and does not account for the likelihood that containment sprays would not be contributing to transport.

The PP&L tests provide some understanding of the application of "flow fractions" in evaluating the effects of non-uniform flow through a portion of the downcomers.. The testing was not intended to determine the applicable flow fraction.

NUREG/CR-3616 [3]

This January 1984 NUREG is part of the research conducted by Alden Laboratories to support resolution of the USI-A 43 issue. The RMI transport tests were conducted in a flume 6 ft wide, 6 ft deep and 40 ft long. No testing of transport over a weir was conducted. Various sizes/thickness of stainless steel RMI insulation provided by Diamond Power Specialty Company were tested. The report concluded:

- single sheets of thin stainless steel foil used in RMI (0.0025 and 0.0040 inch thick) can be transported by water flow velocities as low as 0.2 to 0.5 ft/sec. Single sheets of thicker foil (0.008 inch) required higher velocities for transport, about 0.4 to 0.8 ft/sec.

- crumpled foils tend to transport at lower velocities than uncrumpled foils

- transport velocity tends to increase with material thickness, except for easily flexible foils where thickness dependence is smaller

- in all cases the velocity of motion of the sample is much lower than that of the flow

- even with higher flow velocities (about 2 ft/sec) the samples were never observed to become "water borne," i.e., to lose contact with the bottom

- when several pieces of foil were released simultaneously, their interaction during the transport process often caused jamming and immobilization of the pieces. High flow velocities, up to 1.8 ft/sec, were then required to break up the jams and resume the transport

Evaluation of NUREG/CR-3616

This NUREG provides information which supports the position that it is difficult to transport stainless steel RMI debris. Although the main thrust of the NUREG was to evaluate transport across a PWR floor to sump screens, much of it is applicable to RMI transport across the drywell floor.

Oskarshamn Tests

Testing conducted at Oskarshamn was principally focused on the affects of containment spray on transport of insulation debris. However, the results provide information relative to the affect of break leakage flow on transport.

Tests of the ability of the containment spray system to transport insulation debris from the drywell to the suppression pool were conducted by ABB at Oskarshamm Unit 1 [4] and Unit 2 [5]. A brief description of these tests and the results obtained is provided here.

The purposes of the Oskarshamn tests were:

- To determine the amount of debris which will be transported by containment spray flow from the drywell to the suppression pool;

- To study the time required to transport debris from the drywell to the suppression pool with containment spray flow;

- To study the efficiency of shielding frames which had been installed at the blowdown pipes in preventing the carryover of insulation debris. These frames had been installed subsequent to the Barsebäck event.

For the Oskarshamn 1 tests, approximately 20 kg of new mineral wool insulation and 180 kg of aged mineral wool insulation were mechanically shredded. The size distribution which resulted was approximately 90% in the 1-5 cm size range, and 10% in the 5-30 cm size range. The shredded insulation material was then distributed in designated sections of one quadrant of the drywell floor. Containment spray was actuated and run for approximately 1 hour, and data was gathered on the amount of insulation debris which remained in the drywell, the amount of insulation debris transported to the suppression pool, and the amount collected on the containment spray system suction strainers.

Results of the tests indicate that approximately 5-6% of the insulation debris scattered on the drywell floor was transported to the suppression pool in these tests. Only very small amounts of debris were collected on the containment spray system suction strainers.

Evaluation of the Oskarshamn Tests

The results of the containment spray transport tests at Oskarshamn are markedly different than the results of the event at Barsebäck. The shielding frames installed around the vent pipes subsequent to the Barsebäck event are the primary reason for the difference. These frames were designed and installed for the explicit purpose of reducing the carryover of insulation debris into the vent pipes by containment spray and break leakage flow.

Two other variables may play a lesser role in explaining the difference in results. It is possible that the Barsebäck steam jet created a smaller average size distribution than the mechanical debris generation method used for the tests. With more small debris, greater entrainment and carryover might be expected. Secondly, the distribution of debris only on the drywell floor is not prototypical. At Barsebäck, the insulation debris was deposited on surfaces throughout the containment, with a much smaller initial concentration on the drywell floor than in the Oskarshamn tests. It is possible that the relatively high concentration of debris on the drywell floor resulted in an enhanced self-filtering effect, where small debris entrained in the containment spray flow is filtered out by passage through larger masses of insulation. With less debris initially on the drywell floor, an increase in carryover may initially be anticipated due to the small entrained debris having a freer path to the vent. As containment spray washes more debris to the drywell floor, the self-filtering effect would be expected to increase, reducing the ability to effectively transport small debris particles.

NUREG/CR-6224 [6]

The analysis in NUREG/CR-6224, Section 2.5, assigned a single drywell to suppression pool debris transport factor for the blowdown phase and the washdown phase based on break location in the drywell. Figure 2.1 shows the transport factors used. Note that the transport factor shown in Figure 2.1 combines the effects of blowdown and washdown. Only by using Table 4.1 can you see the individual contributions of each of these effects as used in the analysis.

NUREG/CR-6224 Section 4.0 provides a more extensive discussion of drywell transport including equations for determination of the transport factor. However, because of very limited experimental data, the NUREG acknowledges that engineering judgement was used to establish drywell transport factors. The data from the Barsebäck event was used to estimate the drywell transport factors. According to Barsebäck plant estimates, >90% of the transport was due to washdown and only a small fraction was transported during blowdown. (Note that the contribution of the individual components in domestic BWRs is expected to be different during a LOCA event because domestic BWRs have raised downcomers unlike the flush mounted ones at Barsebäck and, unlike Barsebäck, typically do not use automatic containment sprays, and because the saturated steam and liquid flow as a result of a large LOCA would be greater than that experienced during the Barsebäck event.)

NUREG/CR-6224 Table 4-1 shows the following washdown transport factors for the reference plant, based on the break location within the drywell.

Drywell Break Location	Washdown Transport Fraction
High	0.10
Medium	0.15
Low	0.30

Excerpt From Table 4-1 of NUREG/CR-6224

NUREG/CR-6224 acknowledges that information from the HDR tests indicate that a fraction of the debris could be firmly attached to structures while the other fraction of the debris would be available for transport by the washdown water flows. Engineering judgement was used to estimate the fractions. However, the NUREG does not discuss what the relative fractions are or how they were applied in determination of the washdown transport.

In comments provided to the NRC [7], the BWROG noted that the draft NUREG analysis assumes transport of a fixed fraction of the total debris generated independent of break size. This means that the draft NUREG model predicts that the break leakage will transport the same amount of debris from a 2" line break as from a 24" line break even though the leakage from the 24" will be much greater and results in higher flow velocities across the drywell floor. The BWROG believes it is not realistic to use one fixed transport fraction, independent of break size.

In response to the BWROG comments, the final NUREG/CR-6224 states (see Appendix F, BWROG-B20) that it is acknowledged that the break size plays a vital role in debris transport during washdown but that conclusive experimental data is not available to quantify the dependence of washdown transport on break leakage flow from different sized breaks.

NUREG 0897 [8]

NUREG 0897, Appendix D provides a complex analytical methodology for determination of flow fields across PWR containment floors. Although complex, it appears that this methodology could be applied to determine flow across a BWR drywell floor.

Issues To Address

Comparison of the results from the Oskarshamn tests and the Barsebäck event make it clear that the transport fraction during the containment spray and break leakage flow phase of the event may vary dramatically dependent on the configuration of the drywell to suppression pool vents. Plants with flush mounted vent pipes will experience significantly higher transport than plants with vent pipe entrances which provide for separation of debris from the recirculation flow. This implies that the configuration of the vent pipes relative to the drywell floor must be considered in the analysis of drywell to suppression pool transport.

Additional effort is needed in order to be able to make realistic (as opposed to bounding) predictions on a plant specific basis of the amount of debris transported from the drywell to the wetwell as a result of washdown transport. Without this additional information there is not a sound technical base for a realistic estimate of the washdown transport fraction.

> **Note:** *The applicable test results reported in Reference 9 will be helpful in addressing some of the issues discussed below.*

Issues which need to be addressed include:

1. What is the expected distribution of insulation debris across the drywell floor? How does the break location affect the distribution? Does the type of debris

(e.g., fibrous insulation, RMI, other drywell debris) affect the distribution? Does the containment type affect the distribution?

2. Does the continued leakage flow result in further breakdown of insulation debris on the drywell floor into shreds and fines which are more readily transported? If so, how much?

3. What percentage of the fibrous insulation debris which initially adheres to a surface in the drywell may subsequently be washed down as a result of either break leakage flow or drywell spray flow? How is the washdown affected by different flows (i.e., by different break sizes and/or containment spray)? What affect do different containment types have?

4. What is the methodology to be used for determining the expected flow velocities across the drywell floor? Will the containment specific variations be considered in the methodology?

5. Additional settling tests (similar to those conducted by PP&L) or an engineering analysis are needed to determine at what drywell floor flow velocities other debris is expected to transport from the drywell to the wetwell. This information is needed for other debris which may be expected in a drywell but for which there is insufficient transport data available.

6. How effective are different designs of debris screens at reducing the washdown (not blowdown) carryover of various debris types? What design limitations (e.g., affect on blowdown/containment pressure) are there on use of such screens? At what location (entrance to the downcomer, internal to the downcomer, etc.) are such screens most effective?

7. What is the timing for introduction of debris transported from the drywell to the wetwell (i.e. latent debris)?

References

1. Pennsylvania Power & Light Company Report EC-059-1006, Rev. 0, "Results of Hydraulic Tests on ECCS Strainer Blockage and Material Transport in a BWR Suppression Pool", K. Brinckman, May 1994

2. Alden Research Laboratory Report 31-94/M216F "ECCS Strainer Model Study: Transport and Entrainment Studies in a Laboratory Flume", P. Murhty and M. Padmanabhan, May 1994

3. NUREG/CR-3616 "Transport and Screen Blockage Characteristics of Reflective Metallic Insulation Materials", D. Brocard, Alden Research Laboratory, January, 1984.

4. OKG Report 92-07528, Oskarshamn 1 - Transport of Insulation Material in the Reactor Containment

5. OKG Report 92-07635, Oskarshamn 2 - Transport of Insulation Material in the Reactor Containment

6. NUREG/CR-6224, "Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA Generated Debris", Final Report, October 1995

7. BWROG-94158, "BWR Owner's Group Comments on August 1994 Draft Report NUREG/CR-6224, 'Parametric Study of the Potential for BWR ECCS Strainer Blockage Due to LOCA Generated Debris,'" from R.A. Pinelli to Secretary of the Commission, December 15, 1994

8. NUREG-0897, Rev. 1, "Containment Emergency Sump Performance," October, 1985.

9. Continuum Dynamic Report 96-05 Rev. 0, "Testing of Debris Transport Through Downcomers/Vents Into the Wetwell," September 1996

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OMB No. 3150-0011
NRCB 95-02

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

October 17, 1995

NRC BULLETIN 95-02: UNEXPECTED CLOGGING OF A RESIDUAL HEAT REMOVAL (RHR) PUMP STRAINER WHILE OPERATING IN SUPPRESSION POOL COOLING MODE

Addressees

All holders of boiling-water reactor (BWR) operating licenses or construction permits for nuclear power reactors.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this bulletin to accomplish the following:

(1) Alert addressees to complications experienced during a recent event in which a licensee initiated suppression pool cooling in response to a stuck-open safety relief valve (SRV) and subsequently experienced clogging of one RHR pump suction strainer.

(2) Request addressees to review the operability of their emergency core cooling system (ECCS) and other pumps which draw suction from the suppression pool while performing their safety function. The addressee's evaluation should be based on suppression pool cleanliness, suction strainer cleanliness, and the effectiveness of their foreign material exclusion (FME) practices. In addition, addressees are requested to implement appropriate procedural modifications and other actions (e.g., suppression pool cleaning), as necessary, to minimize foreign material in the suppression pool, drywell and containment. Addressees are requested to verify their operability evaluation through appropriate testing and inspection.

(3) Require that addressees report to the NRC whether and to what extent they have complied with the requested actions. In addition, require a second report indicating completion of confirmatory test(s) and inspection(s) and providing the test results by addressees that have complied with the requested actions, or indicating completion of any proposed alternative course of action by addressees that have not complied with the requested actions.

Background

On September 11, 1995, Limerick Unit 1 was being operated at 100 percent power when control room personnel observed alarms and other indications that one safety relief valve ("M") was open. Emergency procedures were implemented. Attempts to close the valve were unsuccessful, and within 2 minutes a manual reactor scram was initiated. The main steam isolation valves were closed to reduce the cooldown rate of the reactor vessel. The maximum cooldown rate was 54° C/hr [130° F/hr].

Prior to the opening of the SRV, the licensee was running the "A" loop of suppression pool cooling to remove heat being released into the pool by leaking SRVs. Shortly after the manual scram, and with the SRV still open, the "B" loop of suppression pool cooling was started. Operators continued working to close the SRV and reduce the cooldown of the reactor vessel. Approximately 30 minutes later, fluctuating motor current and flow was observed on the "A" loop. Cavitation was believed to be the cause, and the loop was secured. After it was checked the "A" pump was restarted, but at a reduced flowrate of 8kl/m [2,000 gpm]. No problems were observed, so the flow rate was gradually increased back to 32kl/m [8,500 gpm], the full flowrate for the RHR pumps when operating in suppression pool cooling mode. Again, no problems were observed, so the pump continued to be operated at a constant flow. A pressure gauge located on the pump suction was observed to have a gradually lower reading, which was believed to be indicative of an increased pressure drop across the pump suction strainers located in the suppression pool. After about 30 minutes of additional operation, the suction pressure remained constant.

The rest of the reactor shutdown was routine, with no further complications.

Discussion

Limerick Unit 1 has been in commercial operation since 1986 without its suppression pool ever being cleaned. Cleaning was scheduled for the upcoming 1996 refueling outage. The pool of Unit 2 was cleaned during the last refueling outage in 1995.

At Limerick, each pump suction inlet is constructed in a "T" arrangement with two truncated cone-type strainers. The strainers are constructed of 0.95 cm [3/8 inch] thick perforated 304L stainless steel plate with 1.6 cm [5/8 inch] holes on 2.2 cm [7/8 inch] centers. All strainer surfaces are covered by a 12x12 316L stainless steel wire mesh. Because of the leaking SRVs, the "A" and "B" loops of RHR had typically been used for suppression pool cooling during the last few months before the event. Originally, the licensee only used the "A" loop for suppression pool cooling. Approximately 3 months before the event, the licensee changed its practice so that use of the "A" and "B" loops could be alternated.

After cooldown following the blowdown event, a diver was sent into the suppression pool at Unit 1 to inspect the condition of the strainers and the general cleanliness of the pool. Both suction strainers in the "A" loop of suppression pool cooling were found to be almost entirely covered with a thin "mat" of material, consisting mostly of fibers and sludge. The "B" loop suction strainers had a similar covering, but to a lesser extent. One of the "B" loop suction strainers was approximately 75% covered by the mat. The other had only limited coverage. The other strainers in the pool were covered with a dusting of corrosion products (sludge). Debris was subsequently removed from the strainers and the suppression pool floor, and the water was cleaned by use of a temporary filtration system. The strainers were easily cleaned by brushing the material off the surface.

It is believed that during operation of the suppression pool cooling system, the strainer filtered out fibers that were in the pool water. The resulting mat of fibers improved the filtering action of the strainers, thereby collecting sludge and other material on the surface of the strainer. The licensee has concluded that the blowdown caused by the SRV opening did not significantly increase the rate of debris accumulation on the strainer. Following the event, the licensee removed about 635kg [1400 pounds] of debris from the pool of Unit 1. A similar amount of material had previously been removed from the Unit 2 pool.

Analysis showed that the sludge was primarily iron oxides and the fibers were of a polymeric nature. The source of the fibers has not been positively identified, but the licensee has determined that the fibers did not originate within the suppression pool. There was no trace of either fiberglass or asbestos fibers. In addition, other foreign material was found in the pool, such as pieces of wood, nails, and hose. In light of these findings, the licensee decided to modify their FME procedures to specifically address material control in the suppression pool and drywell.

Section 50.46 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.46) requires that licensees design their ECCSs so that the calculated cooling performance following a loss-of-coolant accident (LOCA) meets five criteria, one of which is to provide long-term cooling capability of sufficient duration following a successful system initiation so that the 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 ECCS is designed to meet this criterion, assuming the worst single active failure and only partially obstructed flow through the strainer. Experience gained from the Limerick event demonstrates that inadequate suppression pool cleanliness can lead to unacceptable buildup of foreign material, debris and corrosion products on the strainers during normal operation, which could prevent the ECCS from providing long-term cooling following a LOCA. The staff concludes, therefore, that licensees should take the actions discussed below to ensure that debris which is located in the suppression pool, or will accumulate in the suppression pool during normal operation, does not adversely impact ECCS capability during normal or transient operations or following a LOCA.

Prior to the Limerick event, the staff had issued a draft bulletin for public comment entitled, "Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling Water Reactors." The draft bulletin and associated draft regulatory guide provide the staff's proposed resolution to the generic BWR strainer clogging issue. The issue covered by the draft bulletin, however, differs from the issue covered in this bulletin because the draft bulletin focuses on the potential for ECCS strainers to be clogged by debris generated by a LOCA. This bulletin has been issued to resolve a related issue, highlighted by the Limerick event, of the potential for ECCS suction strainers to be clogged during normal operations by debris which is presently in the suppression pool, or may accumulate in the suppression pool during normal operation. The draft bulletin was published in the *Federal Register* on July 31, 1995. The public comment period ended on October 2, 1995. The staff is currently involved in the review and disposition of the public comments as well as in resolving the open issues identified in the federal register notice.

Requested Actions

To ensure that unacceptable buildup of debris that could clog strainers does not occur during normal operation, all addressees are requested to take the following actions:

1) Verify the operability of all pumps which draw suction from the suppression pool when performing their safety functions (e.g., ECCS, containment spray, etc.), based on an evaluation of suppression pool and suction strainer cleanliness conditions. This evaluation should be based on the pool and strainer conditions during the last inspection or cleaning and an assessment of the potential for the introduction of debris or other materials that could clog the strainers since the pool was last cleaned.

2) The operability evaluation in requested action 1 above should be confirmed through appropriate test(s) and strainer inspection(s) within 120 days of the date of this bulletin.

3) Schedule a suppression pool cleaning. The schedule for cleaning the pool should be consistent with the operability evaluation in requested action 1 above. In addition, a program for periodic cleaning of the suppression pool should be established, including procedures for the cleaning of the pool, criteria for determining the appropriate cleaning frequency, and criteria for evaluating the adequacy of the pool cleanliness.

4) Review FME procedures and their implementation to determine whether adequate control of materials in the drywell, suppression pool, and systems that interface with the suppression pool exists. This review should determine if comprehensive FME controls have been established to prevent materials that could potentially impact ECCS operation from being introduced into the suppression pool, and whether workers are sufficiently aware of their responsibilities regarding FME. Any identified weaknesses should be corrected. In addition, the effectiveness of the FME controls since the last time the suppression pool was cleaned and the ECCS strainers inspected, and the impact that any weaknesses noted may have on the operability of the ECCS should be assessed.

5) Consider additional measures such as suppression pool water sampling and trending of pump suction pressure to detect clogging of ECCS suction strainers.

By letter dated September 29, 1995, (serial BWROG-95083), the BWR Owners Group (BWROG) Executive Oversight Committee (EOC) provided to the BWROG Executive Committee their recommended utility interim actions in response to the recent ECCS suction strainer plugging event at Limerick, Unit 1. The letter also provides additional guidance on the BWROG recommended method for evaluating pool cleanliness and on demonstrating adequate pool cleanliness.

Required Response

All addressees are required to submit the following written reports:

(1) Within 30 days of the date of this bulletin, a report indicating to what extent the licensee intends to comply with the requested actions in this bulletin. In the report, licensees that intend to comply should provide a detailed description of their actions, the results of their evaluations, any corrective actions they have taken, and a description of the licensee's planned test(s) and inspection(s) for confirming their operability evaluation. In addition, licensees should include their schedule for pool cleaning, the basis for the cleaning schedule, and a summary of any additional measures taken to detect and prevent clogging of the ECCS strainers. If a licensee does not intend to comply with these requested actions, its report should contain a detailed description of any proposed alternative course of action, its schedule for completing this alternative course of action, and the safety basis for its having determined the acceptability of the planned alternative course of action.

(2) If not addressed in the report discussed above by licensees that intend to comply with the requested actions, within 10 days of the completion of confirmatory tests and inspections or completion of proposed alternative actions, a second report confirming the completion of all pump operability testing and inspection and providing a description of the test/inspection results. Licensees who do not intend to comply with the requested actions should provide a second report indicating the completion of any proposed alternative actions within 10 days of completing the alternative actions.

Address the required written reports to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, D.C. 20555-0001, under oath or affirmation under the provisions of Section 182a, the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f). In addition, submit a copy of the reports to the appropriate regional administrator.

Related Generic Communications

Recent instances of problems with strainer clogging are described in the following generic communications:

- NRC Information Notice 95-47: "Unexpected Opening of a Safety/Relief Valve and Complications Involving Suppression Pool Cooling Strainer Blockage"
- NRC Information Notice 95-06: "Potential Blockage of Safety-Related Strainers by Material Brought Inside Containment"
- NRC Information Notice 93-34 and Supplement 1: "Potential for Loss of Emergency Core Cooling Function due to a Combination of Operational and Post-LOCA Debris in Containment"
- NRC Bulletin 93-02 and Supplement 1: "Debris Plugging of Emergency Core Cooling Suction Strainers"
- NRC Information Notice 92-85: "Potential Failures of Emergency Core Cooling Systems caused by Foreign Material Blockage"
- NRC Information Notice 92-71: "Partial Plugging of Suppression Pool Strainers at a Foreign BWR"

Backfit Discussion

The actions requested by this bulletin, if required, would be backfits in accordance with NRC procedures and are necessary to ensure that licensees are in compliance with existing NRC rules and regulations. Specifically, 10 CFR 50.46 requires that the ECCS be designed so that it is calculated to provide adequate flow capability to maintain the core temperature at an acceptably low value and to remove decay heat for the extended period of time required by the long-lived radioactivity remaining in the core following a LOCA. The Limerick event has demonstrated that suppression pool cleanliness can adversely impact ECCS performance and could prevent the ECCS from performing its safety function of long-term decay heat removal following a LOCA. Therefore, this bulletin is being issued as if the requested actions were compliance backfits under the terms of 10 CFR 50.109(a)(4)(i). A full backfit analysis was not performed. An evaluation was performed in accordance with NRC procedures. A statement of the objectives of and the reasons for the requested actions and the basis for invoking the compliance exception if the requested actions were to be required, has been included. A copy of this evaluation will be made available in the NRC Public Document Room.

Paperwork Reduction Act Statement

This Bulletin contains information collections that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et seq.). These information collections were approved by the Office of Management and Budget, approval number 3150-0011, which expires July 31, 1997.

The public reporting burden for this collection of information is estimated to average 240 hours per response, including the time for reviewing instructions, searching existing date sources, gathering and maintaining the data needed, and completing and reviewing the collection of information. The U.S. Nuclear Regulatory Commission is seeking public comment on the potential impact of the collection of information contained in the (Bulletin, etc.) and on the following issues:

1. Is the proposed collection of information necessary for the proper performance of the functions of the NRC, including whether the information will have practical utility?

2. Is the estimate of burden accurate?

3. Is there a way to enhance the quality, utility, and clarity of the information to be collected?

4. How can the burden of the collection of information be minimized, including the use of automated collection techniques?

Send comments on any aspect of this collection of information, including suggestions for reducing the burden, to the Information and Records

Management Branch (T-6 F33), U.S. Nuclear Regulatory Commission, Washington, DC 10555-0001, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0012), Office of Management and Budget, Washington, DC 20503.

The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless it displays a currently valid OMB control number.

If you have any questions about this matter, please contact the technical contact listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

Dennis M. Crutchfield
Dennis M. Crutchfield, Director
Division of Reactor Program Management
Office of Nuclear Reactor Regulation

Technical contact: Robert Elliott, NRR
(301) 415-1397

Lead project manager: Robert M. Latta, NRR
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Attachment:
List of Recently Issued NRC Bulletins

LIST OF RECENTLY ISSUED NRC BULLETINS

Bulletin No.	Subject	Date of Issuance	Issued to
95-01	Quality Assurance Program for Transportation of Radioactive Material	01/13/95	For Action - All radiography rahy licensees For Information - None
94-02	Corrosion Problems in Certain Stainless Steel Packagings Used to Transport Uranium Hexafluroide	11/14/94	For Action - Registered users of Model Nos. NCI-21PF-1 and GE-21PF-1 uranium hexafluoride transportation packages
94-01	Potential Fuel Pool Draindown Caused by Inadequate Maintenance Practices at Dresden Unit 1	04/1/94	For Action - All holders of licenses for nuclear power reactors that are permanently shut down with spent fuel in the spent fuel pool (except Shoreham). [Humboldt Bay, Indian Point 1, LaCrosse, Rancho Seco, San Onofre 1, Trojan, Yankee Rowe, and Dresden 1]. For Information - All holders of OLs or CPs for nuclear power reactors and all fuel cycle and materials licensees authorized to possess spent fuel.
93-02, Supp. 1	Debris Plugging of Emergency Core Cooling Suction Strainers	02/18/94	All holders of OLs or CPs for boiling-water reactors All holders of OLs or CPs for pressurized-water reactors

OL = Operating License
CP = Construction Permit

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION
WASHINGTON, D.C. 20555-0001

October 30, 1996

NRC INFORMATION NOTICE 96-59: POTENTIAL DEGRADATION OF POST LOSS-OF-

Addressees

All holders of operating licenses or construction permits for nuclear power reactors.

Purpose

The U.S. Nuclear Regulatory Commission (NRC) is issuing this information notice to alert addressees that the suppression pool and associated components of two boiling-water reactors (BWRs) have been found to contain foreign objects that could have impaired successful operation of emergency safety systems using water from the suppression pool. It is expected that recipients will review the information for applicability to their facilities and consider actions, as appropriate, to avoid similar problems. However, suggestions contained in this information notice are not NRC requirements; therefore, no specific action or written response is required.

Description of Circumstances

Niagara Mohawk Power Corporation, the licensee for Nine Mile Point Unit 2, reported on October 17, 1996, that a significant amount of debris was found during inspection of the drywell-to-suppression chamber downcomers. Most downcomers were clean or contained minimal debris. However, 17 downcomers contained debris, and 7 of the 8 downcomers located directly under the reactor vessel had cleanliness covers installed over the downcomer opening. Some debris was floating on the water inside the downcomers and consisted of foam rubber cleanliness covers, plastic bags, Tygon tubing, hard hats, and so on. The suppression pool had been cleaned during the previous refueling outage.

Commonwealth Edison Company reported on October 16, 1996, that during the first thorough cleaning of the LaSalle Unit 2 suppression pool, a significant amount of foreign material had been found under a layer of sludge. Sludge is a generic term for rust particles from the carbon steel piping connected to the suppression pool. Foreign material was also found in several downcomers. The foreign material included a rubber mat, a sheet of gasket material, a nylon bag, and a substantial amount of sludge. The licensee concluded that sufficient material was present to challenge the clogging limit for multiple emergency core cooling system (ECCS) strainers. The Unit 2 pool had been inspected previously to remove visible debris, and the strainers had been cleaned.

Discussion

Section 50.46 of Title 10 of the Code of Federal Regulations (10 CFR 50.46) requires that licensees design their ECCSs so that the calculated cooling performance following a loss-of-coolant accident (LOCA) meets five criteria, one of which is to provide long-term cooling capability of sufficient duration following a successful system initiation so that the core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period required by the long-lived radioactivity remaining in the core.

On October 17, 1995, the NRC issued Bulletin 95-02, "Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in Suppression Pool Cooling Mode," which requested BWR licensees to review the operability of their ECCS and other pumps that draw suction from the suppression pool while performing their safety function. The addressees' evaluations were to be based on suppression pool cleanliness, suction strainer cleanliness, and the effectiveness of their foreign material exclusion practices. In addition, licensees were to implement appropriate procedural modifications and other actions (e.g., suppression pool cleaning), as necessary, to minimize foreign material in the suppression pool, the drywell, and systems that interface with the suppression pool. Licensees were to verify their operability evaluation through appropriate testing and inspection.

The actions of both licensees were a consequence of the requested actions of Bulletin 95-02. The LaSalle Unit 2 suppression pool was being thoroughly cleaned as requested by the bulletin, and the Nine Mile Point Unit 2 suppression pool was being reinspected as part of the enhanced surveillance requested by the bulletin.

The NRC has issued a number of generic communications to describe aspects of the potential for loss of recirculation capability as a result of strainer clogging and debris blockage. While the past generic communications contain examples that focus on specific considerations that are most applicable to either pressurized-water reactors (PWRs) or BWRs, the basic safety concern applies to both BWRs and PWRs. These events as well as those in previous generic communications demonstrate the need for a thorough cleaning of all areas of PWRs and BWRs that may contain materials which could adversely affect LOCA recirculation. Visual inspection and spot cleaning cannot ensure that all undesirable and unanticipated foreign material will be eliminated.

Related Generic Communications

Recent instances of problems with strainer clogging are described in the following generic communications:

- NRC Generic Letter 85-22: "Potential for Loss of Post LOCA Recirculation Capability Due to Insulation Debris Blockage," dated November 22, 1985.

- NRC Information Notice 89-77: "Debris in Containment Emergency Sumps and Incorrect Screen Configuration," dated November 21, 1989.

- NRC Information Notice 92-71: "Partial Plugging of Suppression Pool Strainers at a Foreign BWR," dated September 30, 1992.

- NRC Information Notice 92-85: "Potential Failures of Emergency Core Cooling Systems Caused by Foreign Material Blockage," dated December 23, 1992.

- NRC Bulletin 93-02 and Supplement 1: "Debris Plugging of Emergency Core Cooling Suction Strainers," dated May 11, 1993 and February 18, 1994.

- NRC Information Notice 93-34 and Supplement 1: "Potential for Loss of Emergency Core Cooling Function Due to a Combination of Operational and Post-LOCA Debris in Containment," dated April 26, 1993 and May 6, 1993.

- NRC Information Notice 95-06: "Potential Blockage of Safety-Related Strainers by Material Brought Inside Containment," dated January 25, 1995.

- NRC Information Notice 95-47: "Unexpected Opening of a Safety/Relief Valve and Complications Involving Suppression Pool Cooling Strainer Blockage," dated October 4, 1995.

- NRC Bulletin 95-02: "Unexpected Clogging of a Residual Heat Removal (RHR) Pump Strainer While Operating in Suppression Pool Cooling Mode, dated October 13, 1995.

- NRC Bulletin 96-03: "Potential Plugging of Emergency Core Cooling Suction Strainers By Debris in Boiling-Water Reactors," dated May 6, 1996.

This information notice requires no specific action or written response. If you have any questions about the information in this notice, please contact one of the technical contacts listed below or the appropriate Office of Nuclear Reactor Regulation (NRR) project manager.

signed by

Thomas T. Martin, Director
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