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May 31, 1984

Dr. Harold R. Denton
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U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, DC 20555

BEAVER VALLEY POWER STATION - UNIT NO. 2
Alternate Pipe Break Design Criteria and
Safety Balance for Elimination of Large
Primary Loop Pipe Ruptures

References:
1. NRC Generic Letter 84-04 dated 2/1/84.
2. DLC Letter 2NRC-4-017 dated 2/24/84.
3. NRC Letter to DLC dated 4/10/84.

Dear Dr. Denton:

In support of our request for exemption to General Design Criteria 4 (Reference 2), please find enclosed the Alternate Pipe Break Design Criteria and the Leak Before Break Safety Balance for Beaver Valley Power Station Unit #2. These two reports provide an alternative to postulating Double-Ended Guillotine Breaks (DEGB) which are required in General Design Criteria 4 of 10CFR50 Appendix A. The approach used in developing the BVPS-2 Alternate Pipe Break Criteria and the associated Safety Balance methodology is in accordance with the approach suggested by the NRC in Generic Letter 84-04 dated February 1, 1984.

Should you have any questions concerning this matter please contact this office.

Very truly yours,

E. J. Woolever

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Vice President
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ALTERNATE PIPE BREAK DESIGN CRITERIA FOR THE PRIMARY COOLANT LOOP AT BEAVER VALLEY UNIT #2

I. Introduction

Asymmetric blowdown loads on PWR primary systems results from postulated rapid-opening, double-ended guillotine breaks (DEGB) at specific locations of reactor coolant piping. This current pipe break criteria requires installation of pipe whip restraints and jet deflectors. Various restraints and deflectors have not yet been either designed, procured or installed at Beaver Valley Unit #2.

As an alternative to postulating DEGB (required in General Design Criteria - 4) the NRC has stated (Generic Letter 84-04) that demonstration of deterministic fracture mechanics analysis, as contained in References 1-4 is acceptable.

This report demonstrates the applicability of modeling and conclusions contained in References 1-4 to the reactor coolant primary piping in Beaver Valley Unit #2.

Conclusions are also provided regarding the susceptibility of the reactor coolant primary loop piping to failure from the effects of intergranular stress corrosion cracking, water hammer and fatigue.

II. Mechanistic Fracture Evaluation

The information contained in topical reports WCAP 9558 Rev. 2, and WCAP 9787 includes the following:

1. Definition of the primary piping loadings.

2. Analyses to define the potential for fracture from ductile rupture and unstable flaw extension.

3. Material tests to define the material tensile and toughness properties.

4. Predictions of leak rate from flaws that are postulated to occur in PWR primary system piping.

The following is a brief demonstration of the application of the deterministic fracture mechanics analysis as contained in WCAP 9558 and WCAP 9787. Input for this analysis was obtained from a review of the maximum dead weight, thermal and seismic load conditions on Beaver Valley Unit #2 primary loop piping.

A. Loads

Maximum loads in Beaver Valley Unit #2 have been determined to be enveloped by WCAP 9558 Rev. 2. Loads acting on the Reactor Coolant Pressure Boundary (RCPB) piping during various plant conditions include the following:

1) weight of piping and its contents, system pressure,

2) restraint of thermal expansion, operating transients in addition to start-up and shutdown, and

3) postulated seismic events.

The maximum combination of the axial and bending loads occurs at the crossover leg as shown below:

	Loading Conditions			
	Dead Weight	Thermal	Seismic	Total
Axial Load (Kips)	15.6	221.0	23.4	260.0
Bending Load (Ft - Kips)	28.2	1794.6	177.4	2000.2

These loads are enveloped by the loads in WCAP 9558 Rev. 2.

B. Fracture Mechanics Analysis

An elastic-plastic fracture mechanics analysis as performed in WCAP 9558 is performed to demonstrate that significant margins against double-ended pipe break would be maintained for PWR Stainless Steel primary piping that contains a large postulated crack and is subjected to large postulated loadings. The analysis includes the potential for growth of an existing crack due to the application of the load be determined.

1. Postulated Flaw

A throughwall crack 7.5 inches long is postulated along the circumference.

2. J Integral

The maximum value of the J integral in the Beaver Valley Unit 2 primary loop piping is less than the maximum value of the J integral in WCAP 9558.

3. Material Properties

A comparison of Beaver Valley Unit #2 centrifugally cast stainless steel primary piping tensile and fracture toughness properties with those in the Westinghouse materials test program was made. The comparison showed that Beaver Valley Unit #2 properties were enveloped in Westinghouse's materials test program.

WCAP 10456 shows that thermal aging of piping will have end of life toughness properties with a tearing modulus at least a factor of 2 greater than the applied tearing load. The material chemistry examined in WCAP 10456 is worse than that expected to exist in Beaver Valley Unit #2.

4. Loads

As shown previously, the loads of WCAP 9558 envelope the loads in Beaver Valley Unit #2.

5. Leak Rate Calculations

Since the pressure and crack length are similar to those in WCAP 9558, specific leak rate calculatio..s are not required and a 10 gpm or greater leak rate is assumed. A leak rate of 10 gpm or larger is sufficient for detection during normal operation since the leak detection system at Beaver Valley Unit 2 has been designed to satisfy the requirements of Reg. Guide 1.45.

6. Summary

Based in items 1 through 5, the postulated 7.5 inch flaw is both locally and globally stable and will leak at a sufficiently high level to assure detectability.

III. Fatigue

Fatigue evaluation has shown that postulated surface flaws (inner diameter) will not grow significantly during the design life of Beaver Valley Unit #2. Thus fatigue effects are insignificant.

IV. Stress-Corrosion Cracking (SCC)

Intergranular stress-corrosion cracking (IGSCC) is not expected to occur because all the conditions required for IGSCC are not present. Contaminants in the water are controlled by the following:

1) The dissolved oxygen in the coolant is controlled to levels low enough to preclude IGSCC.

2) Hydrazine additions during start-up and hydrogen overpressure during operation control the dissolved oxygen. Thus, the levels are always below the technical specification limit for oxygen of 0.10 ppm when the coolant temperature is above 200° F.

3) Other impurities that might cause SCC such as halides and caustics are also rigidly controlled.

V. Water Hammer

Water hammer is defined as the pressure change in a closed pipe caused by sudden change in the fluid velocity or rate of flow. Water hammer transients are considered to include transients involving steam flow (steam hammer) and two-phase flow (e.g. water entrainment in steamlines, steam bubble collapse), in addition to the classical water hammer transients such as those involving valve closing and pump start-up in solid water systems.

In the primary loop the coolant travels at a constant rate of flow. Conditions for a sudden change in rate of flow would occur downstream of a safety valve or pilot operated relief valve. These valves are not located on the Primary Coolant System being analyzed and therefore, water hammer is not a concern.

VI. Conclusions

An evaluation of the Beaver Valley Unit #2 primary loop piping system shows that the loads and material properties (including the effects of thermal aging) are within the limit/criteria of References 1-4 and that IGSCC, water hammer and fatigue effects are insignificant and consequently the postulated reference flaw will be stable and will leak at a detectable rate.

REFERENCES

1. WCAP 9558, Revision 2 (May 1981) "Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Throughwall Crack"

2. WCAP 9787 (May 1981) "Tensile and Toughness Properties of Primary Piping Weld Metal for Use in Mechanistic Fracture Evaluation"

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

4. WCAP 10456 (November 1983) "The Effects of Thermal Ageing on the Structural Integrity of Cast Stainless Steel Piping for Westinghouse NSSS"

LEAK BEFORE BREAK SAFETY BALANCE

I. Introduction

Beaver Valley Unit #2 requests an exemption from General Design Criteria 4 (GDC-4) on the basis that the avoidance of operational occupational exposure associated with the use of pipe whip restraints and jet deflectors far outweighs the small increase in public risks and potential accident exposure. A safety balance was performed for sixteen Westinghouse A-2 plants as provided in an attachment to Enclosure 2 of Generic Letter 84-04. Using this methodology, a plant specific safety balance was performed for Beaver Valley Unit #2. This report provides a plant specific safety balance which estimates the public risk and the avoided occupational exposure for not installing pipe whip restraints and jet deflectors to mitigate the consequences of asymmetric blowdown loading in the primary system.

The estimated reduction in public risk for installing pipe whip restraints and jet deflectors as necessary to mitigate or withstand asymmetric pressure blowdown loads is very small, 0.84 man-rem total for the nominal case. The estimated reduction in accidental occupational exposure due to installation is 0.12 man-rem total for the nominal case. These small changes result from the estimated small reduction in core-melt frequency of 1.4×10^{-7} events/reactor year that would result from installation of the protective structures. On the other hand, the occupational exposure estimated for maintaining the protective structures would increase by 80 man-rems. Consequently, the savings in exposure by granting the exemption far exceed the potentially small increase in public risk and avoided accident exposure associated with pipe restraint devices.

II. Scope

A. Jet Impingement Protection

The essential targets of reactor coolant line breaks can be divided into three categories:

1. Structural,
2. Westinghouse piping scope, and
3. Stone & Webster (S&W) piping scope.

Among the many assumptions in this analysis, it is first assumed that the breaks would not be shielded at their source. Therefore, any target requiring protection would be shielded at the target itself. Secondly, it is assumed that after the stress analyses on structural targets are completed the result would be that there are no unacceptable structural interactions. Therefore, no shields are postulated to protect structures from the reactor coolant line breaks.

As for the above Westinghouse scope piping targets, S&W has completed the stress analysis for the primary loop. Loads must be analyzed to determine if jet load protection is required. It will be assumed for this analysis that no jet bumpers will be required. These assumptions have a conservative effect on both the value and impact results.

The S&W target/break interactions are identified below:

Target Line No.	Break No.
2CHS-002-97-1	3,5,8,12
2RCS-002-045-1	3,4,5,8,12
2SIS-006-266-1	4,5,8,12
2SIS-006-270-1	4,5,8,12
2BDG-025-10-2	5,8
2RCS-008-40-1	5,8
2RCS-008-41-1	5,8,12
2SIS-012-71-1	5,8
2CHS-002-141-1	12
2CC-002-PB3	12
2CC-002-PB4	12

8 Total S&W Scope Break-Target Interactions have been identified for Loop B.

It is assumed that these eight S&W scope piping targets for this loop are typical for the other two loops. Therefore, there would be approximately 24 shields to be added. All shields would require periodic inspection and maintenance.

B. For RC Equipment Restraints

This analysis applied to the following restraints which are designed to prevent damage to RC loop equipment after a postulated DEGB: for each of 3 loops there are two restraints on the crossover leg elbows, 1 restraint on the hot leg, and 1 restraint on the cold leg.

III. Assumptions

The above estimated changes in public risk and avoided accident exposure were obtained by utilizing applicable portions of the plant risk model developed for the calculation of severe reactor accident risks provided in the Beaver Valley Unit 2 Environmental Report.

The following major assumptions were utilized:

1. A Double-Ended Guillotine Break (DEGB) was assumed. This assumption is very conservative in light of the mechanistic fracture analysis which has shown that the postulated reference flaw is stable and will leak at a detectable rate.

2. The installation of pipe whip restraints is assumed to eliminate the possibility that a DEGB inside the reactor cavity would lead directly to a large LOCA proceeding to an early core melt. This also is a conservative assumption.

3. If a DEGB were to occur outside the reactor cavity, it could lead to core melt through the additional failure of emergency core cooling.

4. Estimates of DEGB frequencies for large primary system piping were developed from two sources of data.

   a. The upper estimate of 10^{-5} per reactor-year is based on a paper on nuclear and non-nuclear pipe reliability data "Reliability of Piping in Light Water Reactors" dated 1977 by S. H. Bush as quoted in Generic Letter 84-04.

   b. The nominal estimate of 6×10^{-7} per reactor-year for primary system piping <u>outside</u> the reactor cavity and 1.4×10^{-7} per reactor-year inside the reactor cavity are the values for a three loop plant derived from Report SAI-001-PA dated June 1976 prepared by D. O. Harris and R. R. Fullwood as quoted in Generic Letter 84-04.

   c. Both the upper and nominal estimated DEGB frequencies are less than the WASH-1400 large LOCA median frequency of 1×10^{-4} per reactor-year. The following table identifies several factors associated with Beaver Valley Unit #2, Westinghouse A-2 Plants, and the data base used for WASH 1400 that support the use of a lower pipe break frequency.

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

5. Public dose estimate for the release categories were derived using the CRAC-2 code. Inputs to the code include the release categories used in the Beaver Valley Unit 2 (BV-2) severe accident impact evaluation, site specific population and meteorology, and 10 mile population evacuation. These are based on a 350 mile radius release model.

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

7. The estimated occupational exposure associated with maintaining the protective structures is also considered.

8. Financial considerations have not been addressed in this report.

IV. Results

A. Public Health

A.1 Description of Methodology

The Value-Impact Analysis attached to NRC generic letter 84-04 contains analysis for 16 Westinghouse PWR plants in assessing the average man-rem release from DEGB LOCA events. The analysis is carried out for plants which do not have pipe whip restraints. In the analysis initiating events are divided into two categories:

Category I. - DEGB LOCA occurs in the reactor cavity and leads to early core melt.

Category II. - DEGB LOCA occurs outside the reactor cavity and leads to challenges to the ECCS and containment safeguards.

The nominal and upperbound initiating event frequencies for the above events are given below:

I. nominal frequency = 9.0E-08/py. (for two-loop plants)

upperbound frequency = 2.0E-06/py.

II. nominal frequency = 4.0E-07/py. (for two-loop plants)

upperbound frequency = 8.0E-06/py.

An updated WASH-1400 analysis is used to calculate the average man-rem exposure from the above events.

In the following analysis for Beaver Valley Unit 2, the same initiating event frequencies are used and the same major modeling assumptions are utilized. The severe accident risk analysis carried out for the Beaver Valley Unit 2 environmental report is used to estimate the average man-rem exposure from the above events. This analysis is based on a 350 mile radius around the plant, whereas a 50 mile radius is considered in the attachment to Generic Letter 84-04. Thus, the results of the present analysis tend to be conservative.

A.2 DEGB LOCA Incremental Risk Analysis.

The initiating event frequencies used in this analysis are given below:

I. Breaks in reactor cavity: 1.4E-07/py (nominal)
2.0E-06/py (upperbound)

II. Breaks outside reactor cavity: 6.0E-07/py (nominal)
8.0E-06/py (upperbound)

The nominal frequencies have been adjusted by a factor of 3/2 to account for the three loops of the present plant.

The seven release categories associated with DEGB LOCAs are defined in Table 1. The mean value for the total whole body man-rem for each release category is given in Table 2.

The incremental risk is calculated by using the same model as was used in the attachment to Generic Letter 84-04. Namely the following equation is used:

$$dRISK = RISK_I + 0.2 * RISK_{II} \qquad \text{(EQUATION 1)}$$

where

$RISK_I$ = man-rem risk from in-cavity breaks,

$RISK_{II}$ = man-rem risk from breaks outside of the reactor cavity.

The factor of 0.2 is used to account for the effect of system interactions leading to additional risk without pipe whip restraints. This model conservatively assumes that the removal of pipe whip restraints results in asymmetric blowdown from in-cavity breaks causing all cases to result in core melt.

RISK is calculated as

$$RISK = \sum_{i=1}^{7} F_i * R_i \qquad \text{(EQUATION 2)}$$

where

F_i = frequency of being in the i^{th} release category (summarized in Table 3);

R_i = average man-rem associated with the i^{th} release category (Table 2)

A.2.1 Calculation of Nominal Risk from DEGB LOCAs

a. Initiating event Category I.

In this category, the ECCS is modeled to be insufficient to cool the core and early core melt is assumed. The frequencies of the release categories have been calculated and are displayed in column 1 of Table 3.

Using Table 2, column 1 of Table 3 and Equation 2, the man-rem risk is calculated to be

$RISK_I$ = 2.1E-02 man-rem/py

b. Initiating event Category II.

In this category, the already calculated LOCA event consequences have been modified to calculate the release category frequencies with the given initiating event frequency. The results are summarized in column 3 of Table 3 and the risk is calculated to be

$RISK_{II}$= 3.4E-04 man-rem/py.

Combining the two initiating events for inside and outside of the reactor cavity, the total risk is found (using Equation 1) as follows:

$dRISK_N$ = 2.1E-02 + 0.2 * 3.4E-04 = 2.1E-02 man-rem/year

$dRISK_N$ = 2.1E-02 + 0.2 * 3.4E-04 = 2.1E-02 man-rem/year

A.2.2 Calculation of Upperbound Risk from DEGB LOCA Events

The upperbound estimates are made using the upperbound initiating event frequencies given above.

a. Initiating event Category I.

The frequencies of the release categories have been calculated and are displayed in column 2 of Table 3. Using this and Table 2, the man-rem risk is calculated to be

$RISK_I$ = 0.31 man-rem/py.

b. Initiating event Category II.

The release category frequencies are summarized in column 4 of Table 3 and the risk is calculated to be

$RISK_{II}$ = 4.5E-03 man-rem/py.

Combining the two initiating events for inside and outside of the reactor cavity, the total risk is found as shown:

$dRISK_U$ = 0.31 + 0.2 * 4.5E-03 = 0.31 man-rem/py

A.3 Results

Taking the plant life to be 40 years, the nominal and upperbound risk estimates are calculated below.

$RISK_N$ = 40 X 0.021 = 0.84 man-rem

$RISK_U$ = 40 x 0.31 = 12.4 man-rem

B. Core Melt Frequency

The increase in core melt frequency is dominated by the in-cavity events. This occurs because the ECCS functions in most cases and only a small fraction of the LOCA events outside of the cavity proceed to core melt. The nominal and upperbound estimates of the change in core melt frequency are therefore given by the following expressions:

$dCM_N = 1.4 \times 10^{-7}/py$

$dCM_U = 2.0 \times 10^{-6}/py$

C. Occupational Exposure - Accidental

The increased occupational exposure from accidents can be estimated as the product of the change in total core melt frequency and the occupational exposure likely to occur in the event of a major accident. The change in core melt frequency was estimated as 1.4E-7 events/year. The occupational exposure in the event of a major accident has two components. The first is the "immediate" exposure to the personnel onsite during the span of the event and its short term control. The second is the longer term exposure associated with the cleanup and recovery from the accident.

The total avoided occupational exposure is calculated as follows:

$D_{TOA} = N*T*D_{OA}$, where $D_{OA} = P(D_{IO} + D_{LTO})$

where

D_{TOA} = Total avoided occupational dose

N = Number of affected facilities = 1

T = Average remaining lifetime = 40 years

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

P = Change in core-melt frequency

D_{IO} = "Immediate" occupational dose

D_{LTO} = Long-term occupational dose.

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

	Increase in Core Melt Frequency (events/ reactor-yr)	Immediate[a] Occupational Dose (man-rem/ event)	Long Term[a] Occupational Dose (man-rem/ event)	Total Avoided Occupational Exposure (man-rem)
Nominal Estimate	1.4E-7	1E3	2E4	.12
Upper Estimate	2E-6	4E3	3E4	2.72

(a) Based on cleanup and decommissioning estimates, NUREG/CR-2601 (Murphy 1982) as quoted in Generic Letter 84-04.

D. Occupational Exposure - Operational

By not installing the jet deflectors and pipe whip restraints, operational exposure dose is reduced. The amount of occupational exposure dose would be incurred due to the following:

1. Slowing down of normally anticipated work activities.

   a. Increased inaccessibility for personnel and equipment to and from primary system componets.

   b. Increased congestion between adjacent compartments.

   c. More difficult housekeeping in areas of high contamination potential.

   d. Occasional jet shield removal and replacement (often requiring polar crane time).

2. Additional routine maintence of jet deflectors.

   a. Periodic paint inspection.

   b. Maintenance of jet deflectors (requiring support from scaffold builders, laborers, inspectors and radiation protection personnel).

To estimate the potential exposure amount, it was assumed that two additional man-weeks per plant year would be spent inside containment if the jet deflectors and pipe whip restraints were installed.

It was assumed that all workers will receive an average 25 mr/hr. dose rate. Therefore the total dose is estimated below.

Operational dose averted = (80 man-hr/py)*
(40 plant-years) * (0.025 R/man-hr)
= 80 man-rem

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

$Dose_{upper} = 3X\ dose_{expected}$

Activity	Dose Avoided (man-rem)
Operation, Maintenance Nominal	80
Upper Estimate	240

V. Conclusions

The results of the safety balance are summarized below. The table clearly shows that there is a net safety gain as a result of not installing protection against asymmetric dynamic loads.

Leak Before Break Value Summary

Factors	Dose (man-rem) Nominal Estimate	Upper Estimate
Public Health	-.84	-12.4
Occupational Exposure (Accidental)	-.12	-2.72
Occupational Exposure (Operational)	80	240
Values Subtotal	79.04	224.88

As can be seen from the Values Subtotal a net safety gain of 79 to 225 man-rem is achieved by not installing the primary loop pipe whip restraints and the associated jet impingement shields.

TABLE 1. RELEASE CATEGORY DEFINITIONS

BV2-2	RELEASE CATEGORY FOR CORE MELT SEQUENCES WHICH COULD LEAD TO AN EARLY OVERPRESSURE OF THE CONTAINMENT WITH NO SPRAYS OPERATIONAL AND SHORT WARNING TIME FOR EVACUATION.
BV2-4	RELEASE CATEGORY FOR CORE MELT SEQUENCES WHICH CAN LEAD TO INTERMEDIATE CONTAINMENT FAILURE WITHOUT SPRAYS WITH LATE CORE MELT.
BV2-5	RELEASE CATEGORY FOR CORE MELT SEQUENCES WHICH CAN LEAD TO INTERMEDIATE CONTAINMENT FAILURE WITHOUT SPRAYS WITH EARLY CORE MELT.
BV2-6	RELEASE CATEGORY FOR CORE MELT SEQUENCES WHICH CAN LEAD TO LATE CONTAINMENT FAILURE AND NO SPRAYS OPERATIONAL.
BV2-7	RELEASE CATEGORY FOR CORE MELT SEQUENCES WHICH CAN LEAD TO INTERMEDIATE CONTAINMENT FAILURE WITH SPRAYS OPERATIONAL.
BV2-8	RELEASE CATEGORY FOR CORE MELT SEQUENCES WHICH CAN LEAD TO LATE CONTAINMENT FAILURE WITH SPRAYS OPERATIONAL.
BV2-9	RELEASE CATEGORY FOR CORE MELT SEQUENCES WHICH COULD LEAD TO BASEMAT MELT THROUGH.

TABLE 2. TOTAL WHOLE BODY MAN-REM FOR RELEASE CATEGORIES (R_i)

RELEASE CATEGORY	MAN-REM (MEAN VALUE)
BV2-2	4.6×10^7
BV2-4	3.7×10^7
BV2-5	3.7×10^7
BV2-6	2.9×10^7
BV2-7	8.9×10^4
BV2-8	3.8×10^4
BV2-9	4.0×10^3

TABLE 3. RELEASE CATEGORY FREQUENCIES FOR DEGB LOCAS (F_i)

RELEASE CATEGORY	COLUMN 1	COLUMN 2	COLUMN 3	COLUMN 4
	INSIDE CAVITY		OUTSIDE CAVITY	
	NOMINAL	UPPERBOUND	NOMINAL	UPPERBOUND
BV2-2	1.54E-10	2.20E-09	3.80E-12	5.07E-11
BV2-4	0.0	0.0	1.30E-13	1.73E-12
BV2-5	1.58E-10	2.26E-09	1.32E-12	1.76E-11
BV2-6	3.72E-11	5.32E-10	3.40E-13	4.54E-12
BV2-7	3.96E-08	5.65E-07	9.25E-10	1.23E-08
BV2-8	1.00E-07	1.43E-06	2.37E-10	3.16E-09
BV2-9	1.00E-10	1.43E-09	2.10E-09	2.80E-08