FENOC

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Docket Number 50-346

License Number NPF-3

Serial Number 1-1277

June 12, 2002

Mr. James E. Dyer, Administrator
United States Nuclear Regulatory Commission
Region III
801 Warrenville Road
Lisle, IL 60532-4351

Subject: Confirmatory Action Letter: Response to Request for Additional Information Related to the Davis-Besse Nuclear Power Station Safety Significance Assessment

Dear Mr. Dyer:

The attachment to this letter responds to the NRC Region III Request for Additional Information (RAI) dated May 6, 2002, related to the Safety Significance Assessment of the Davis-Besse Nuclear Power Station Reactor Pressure Vessel Head as was submitted by FirstEnergy Nuclear Operating Company letter Serial Number 1-1268 on April 8, 2002. As communicated previously, the response to RAI 1.d will be provided on or before July 1, 2002.

If you have any questions or require further information, please contact Mr. Patrick J, McCloskey, Manager - Regulatory Affairs, at (419) 321-7148.

Sincerely yours,

HW Bergendahl

Attachments

cc: USNRC Document Control Desk
D.V. Pickett, DB-1 NRC/NRR Project Manager
S.P Sands, DB-1 NRC/NRR Backup Project Manager
C.S. Thomas, DB-1 Senior Resident Inspector
Utility Radiological Safety Board

Responses to Request for Additional Information Concerning the FENOC "Probabilistic Safety Assessment" for the Void in the RPV Head at Davis-Besse

Question 1

The probabilistic safety assessment does not address the probabilities that the cavity could have become larger before being detected or that the void could have formed at a location in the RPV head that had thinner cladding material. Please provide the following information to support the staff's estimation of the risk. Should the requested information be difficult to produce, provide the justification for your basis that the actual measurements are representative of the worst case values.

Question 1.a

All records of the clad thickness on the RPV head that were produced in the fabrication, quality control, and acceptance testing processes. The staff expects that some thickness measurements were made to verify that the cladding is within the design thickness of 1/16" to 3/8" in thickness.

Response 1.a

No records of measurements of the finished clad thickness could be located. In defining the minimum thickness of cladding on the underside of the Reactor Pressure Vessel Closure Head (RPVCH) center disc dome, the records of the fabrication of the head were reviewed. Discussions were held with individuals who were employed at the Babcock & Wilcox (B&W) Mount Vernon Works where the head was fabricated to determine what they recalled of the process. In addition, an ultrasonic testing (UT) examination of the cladding was recently performed by Framatome on the Midland half-head to provide comparative data on process consistency. The Midland half-head is located at the Framatome facility, Mill Ridge Road, located in Lynchburg, Virginia.

In general, the process used for cladding the underside of the RPVCH was multi-arc submerged arc welding (SAW). The Procedure Qualification Reports for this process were reviewed, but none documented clad thickness. Discussions with Framatome welding engineers have identified this process should apply a general thickness of 1/4 to 5/16 of an inch. A small diameter at the center of the dome was clad using the shielded metal arc welding (SMAW) process. Cladding repairs were generally performed using the SMAW process.

A review of the records did not identify any "as left" thickness measurements for the cladding. The review did find that eleven locations were repaired after liquid penetrant examination of the clad. The size of the defects ranged from ½ inch to 2 ½ inch in length, ½ inch to 1 ¼ inch in width, and 1/16 inch to 5/32 inch in depth. A review of the locations showed that they were in all four quadrants of the RVCH center disc dome. None of the defects were in the immediate vicinity of the cavity found at nozzle number 3. The repair process was to grind out the defect, weld repair to a thickness above the surrounding area, blend grind flush with the surrounding material, and perform liquid penetrant examination of the repair. The charted defects did not note that base metal was exposed after probe grinding. Therefore, it can be inferred that the minimum cladding thickness was greater than 5/32", since the defects were found to reach that depth without reaching base metal. Should any base metal have been exposed, it would have been noted such that American Welding Society classification E309 weld material would have been used for the first layer of the repair.

Discussions with personnel employed at B&W's Mount Vernon Works at the time of manufacturing of Davis-Besse Nuclear Power Station (DBNPS) RPVCH provided the following additional information. The B&W drawings identified a minimum thickness for acceptance, no maximum. The practice was to assure that the thickness of metal deposited by the SAW process was greater than the specified minimum. This practice reduced the possibility of having to add additional cladding weld layers and to account for any future machining operations. The thickness of the as-deposited clad was hand measured during the cladding process. The shop process designated the supervisor to inspect and prepare the cladding to be suitable for non-destructive examination (NDE). Discussions with personnel working there indicated a limited amount of grinding was done, and this was generally on cladding starts and stops and other areas as necessary for proper NDE to be performed. Therefore, the majority of the as-deposited surface was not ground or machined. UT was performed for bond only and was on an approximately 12-inch square grid.

The two Midland heads were fabricated at the Mount Vernon Works facility in the same general time frame as the DBNPS head using the same processes. UT examination was performed on the Midland half-head cladding consisting of readings taken at 39 points over all portions of the partial head. The table below compares the results of that inspection with readings taken of the DBNPS cladding. The columns of DBNPS readings represent a summary of the cladding readings taken and a subset of those from the center of the exposed cladding. The subset of readings was used to calculate the average and minimum values used in the stress analysis of the exposed cladding.

	Midland half-head cladding	Davis-Besse total cladding readings	Davis-Besse exposed cladding readings
Number of readings	39	44	25
Average Thickness	0.307	0.308	0.297
Maximum Thickness	0.340	0.380	0.365
Minimum Thickness	0.280	0.240	0.240

Two additional UT readings were taken on the Midland half-head. These were at the very top of the center disc dome around nozzle number one. This is in the small area where the cladding was applied using the shielded metal arc welding process. These readings were 0.38 and 0.40 inches.

The data demonstrates that the cladding application process during fabrication resulted in a relatively consistent range of thickness and that the thickness measurements taken at the DBNPS in the vicinity of the cavity at nozzle 3 are representative of the range of clad thickness over the head. Therefore, the minimum measured clad thickness is a reasonable estimate of the thinnest expected condition.

Question 1.b

All UT measurements that show clad thickness on the RPV head, including the head location coordinates for each of the measurements.

Response 1.b

Refer to Figure 1, which is a map of the UT measurements taken after discovery of the degraded condition.

Question 1.c

The estimated rate of growth of the cavity at the time just prior to plant shutdown on February 16, 2002. The average growth rate for the entire period of cavity development is not an appropriate response unless it is also demonstrated with appropriate evidence that the growth rate was constant over the period. Any discussion of assumed rates of cavity growth should address the difference between the aspect ratios of the cavities found at nozzles 2 and 3. Please provide growth rate estimates on terms of linear rate of cavity expansion in the directions perpendicular to the cavity walls. Volumetric estimates for growth rates are not useful for the intended analyses. Please provide an estimate of the uncertainty in the cavity growth rate at the end of the period, in a form suitable for use in probabilistic assessment.

Response 1.c

The Root Cause Analysis Report, submitted by FirstEnergy Nuclear Operating Company (FENOC) letter Serial Number 1-1270 dated April 18, 2002, provided a conservatively rounded estimated corrosion rate of approximately 2 inches per year at the time of shutdown. The growth rate estimate was based on the cavity dimensions and assuming that significant corrosion rates began approximately 4 years earlier. This assumption is consistent with the available collection of evidence from the DBNPS. It is also consistent with a number of corrosion tests presented in the EPRI Boric Acid Corrosion Guidebook, Revision 1.

The tests results in the Guidebook show that the maximum corrosion rates of tests 4.a, 5.a, and 6.a, using 0.01 gpm flow, had similar corrosion rates to tests 4.b and 6.b, which had a 0.1 gpm flow. The corrosion rates varied from 0.9 in/yr to 2.37 in/yr, with the lower corrosion rates associated with the higher flow rates. This shows that the corrosion rate is relatively independent of the leak rate, between 0.01 and 0.1 gpm for a given geometry. Hence, using an average value provides the DBNPS's best-estimate of the corrosion rate. The Root Cause Analysis Report conservatively estimated that the leakage occurred over a four year period, yielding a conservatively high corrosion rate. The first real evidence of leakage was at about 10RFO, representing 5.25 years of operation. Using this value and a 6.5 inches lateral length yields a 1.24 in/yr average rate, which is consistent with the Guidebook test data. With decreasing boric acid concentration during the operating cycle, the corrosion rate might possibly have decreased as the cycle progressed. Additionally, as the cavity dimensions grew, the effects of jet impingement would diminish. This would contribute to a decreasing advancement rate. These factors suggest that using an average value, rather than the instantaneous value at the time of shutdown yields conservative bounding results. Using a mean value of 2.0 in/yr, with upper and lower bounds of 3.0 in/yr and 1.0 in/yr respectively would encompass most of the applicable test data in the EPRI Boric Acid Corrosion Guidebook.

The differences in aspect ratios between nozzles 2 and 3 are probably the result of earlier initiation of crack leakage in nozzle 3, but most importantly, a greater leakage rate from the nozzle 3 crack(s). The greater leakage promoted greater cooling of the head surface and potential persistence of moisture outside the nozzle annulus. This promotes aerated boric acid corrosive attack from the top down as well as laterally. Nozzle 2 cracks did not provide sufficient moisture to advance the corrosion beyond the annulus. Please refer to the Root Cause Analysis Report for additional discussion of this topic.

Question 1.d

The estimated areas of exposed clad material that would cause the cladding to fail at normal operating pressure for clad thicknesses of 0.297" and 0.125"

Response 1.d

This question requires preparation of a detailed calculation. The request is being analyzed and results will be submitted on or before July 1, 2002.

Question 2

The probabilistic safety assessment uses a log-normal equation to represent the probability distribution for the strength of the clad material. Please provide the following information:

Question 2.a

The value of the constant, β_c, used to represent the randomness of the material strength parameter.

Response 2.a

The value of the constant β_c used in the safety assessment to represent the randomness of the material strength parameter was 0.33. The value came from NUREG/CR-5603, *Pressure-Dependent Fragilities for Piping Components*, and NUREG/CR-5604, *Assessment of ISLOCA Risk-Methodology and Application to a Babcock and Wilcox Nuclear Power Plant.* The values for the constant β_c ranged from 0.06 to 0.39 in those reports. The value of the constant β_c used in the safety assessment was not based upon specific strength properties of stainless steel, but it does reflect a representative value for stainless steel in similar configuration and at head operating temperature.

Question 2.b

Any data on the strength properties of stainless steel alloy 308 that demonstrate the degree of randomness exhibited by that material.

Response 2.b

The reference document NUREG/CR-5603, *Pressure-Dependent Fragilities for Piping Components,* does not provide any data on the degree of strength randomness but does state, "At higher temperatures, increased variability in the yield strength coefficient of variation (Reference 2) results in increased overall variability. Table 2-8 shows the log-normal standard deviations developed over the temperature range of interest."

The document being cited as Reference 2 is: Weiss, V. and J. G. Sessler, Eds., **Aerospace Structure Metals Handbook, Vol. I: Ferrous Alloys,** ASD-TDR-67-741, Air Force Materials Laboratory, Wright-Patterson Air Force Base, Ohio, March 1963.

Question 2.c

The mathematical relationship between the data and the value of β_c used in the safety assessment.

Response 2.c

The mathematical relationship between the data and the value of β_c used in the safety assessment is from NUREG/CR-5603. The typical relationship is provided in the following example:

$(0.1^2 + 0.12^2 + 0.11^2)^{1/2} = 0.19$ (Stainless Steel at Room Temperature)

Where:
0.1 is the coefficient of variation for plastic collapse for semi-ellipsoidal heads.
0.12 is the typical coefficient of variation on the yield strength of stainless steel at room temperature.
0.11 is the coefficient of variation for buckling capacity developed from limited test results.

At higher temperatures, increased variability in the yield strength coefficient of variation (Reference 2 above) results in increased overall variability. As discussed in paragraph (a) above the value increases to 0.33 for stainless steel vessel at 600°F.

Question 3

In Table 2 in Section B.3.2, the probabilistic safety assessment provides a set of RCS pressure ranges and the corresponding values for the number of events experienced in those ranges at Davis-Besse and the estimated frequency for experiencing events in those ranges. Please clarify the following information:

Question 3.a

The pressure ranges are all shown as greater than a specific numerical value, indicating a cumulative distribution, but the number of events experienced at ">2300 psig" is larger than the number shown as ">2250 psig," which indicates that the distribution is not cumulative with respect to the number of events experienced. Is the distribution for the number of events cumulative, or should the table indicate pressure ranges? For the last pressure ">2500 psig," is the frequency value intended to be cumulative for all pressures above 2500 psig, or does it apply to a pressure interval limited by an upper bound? If an upper bound is applicable, what is it?

Response 3.a

Table 2 in Section B.3.2 is not meant to show cumulative pressure ranges. A revised table is provided below:

Pressure (psig)	Number of Events	Frequency (yr-1)
2250 - 2300	4	2.54E-01
2300 - 2350	8	5.08E-01
2350 - 2400	2	1.27E-01
2400 - 2450	1	6.35E-02
2450 - 2500	0	1.59E-02
2500 - 2550	0	1.59E-02

With respect to an upper bound value, pressures in excess of the safety valve setpoint of 2500 psig are very unlikely to occur. Consequently, the highest pressure range considered is 2500 to 2550 psig. This includes all predicted pressures for transients analyzed in the DBNPS Updated Safety Analysis Report (USAR).

Question 3.b

The text states that the frequency column entries for RCS pressures above 2405 psig were based on "a Bayesian update with a non-informative prior..." Please describe the shape of the prior as a function of pressure, including any limits used on the pressures to which the prior distribution is assumed to be applicable. Please provide statistical information used to perform the update, in sufficient detail for the staff to duplicate the computation.

Response 3.b

The largest over pressure transient ever experienced at the DBNPS was less than 2450 psig. However, because some transients analyzed in the USAR exceed that value, values for occurrence were estimated. This was accomplished by means of a Bayesian update, assuming a non-informed prior distribution. The process used is described on page 5-50 of NUREG/CR-2300, *PRA Procedures Guide*. The posterior mean and the variance for the case (zero events in T years) was calculated as:

$$\bar{\lambda}\text{ (0 failures)} = \frac{1}{2T}$$

$$\beta^2\text{ (0 failures)} = \frac{1}{2T^2}$$

The inputs and results for each pressure range are provided below:

Inputs			Outputs					
Press (psig)	Failures	Time (yr)	mean	Error factor	Variance	5% -tile	Median	95% -tile
2300≥ Press >2250	4	15.754	2.54E-01	2.77	3.02E-02	7.55E-02	2.09E-01	5.81E-01
2350≥ Press >2300	8	15.754	5.08E-01	1.96	4.72E-02	2.38E-02	4.67E-02	9.16E-01
2400≥ Press >2350	2	15.754	1.27E-01	5.20	2.79E-02	1.48E-02	7.68E-02	4.00E-01
2450≥ Press >2400	1	15.754	6.35E-02	3.65	3.46E-03	1.27E-02	4.65E-02	1.70E-01
Press >2450	0	15.754	3.17E-02	3.93	1.01E-03	5.71E-03	2.24E-02	8.83E-2

The last distribution interval of greater than 2450 psig was then divided into two with one interval being pressure greater than 2450 psig and less than or equal to 2500 psig. The second interval is the pressure greater than 2500 psig as listed in the table in the Response to 3.a above.

Question 4

In the Davis-Besse IPE submittal dated February 1993, it is stated in the description of a large LOCA: "A large LOCA is, by definition, sufficient to depressurize the RCS to the point at which reflooding of the core would be required by the core flood tanks, with makeup in the longer term by decay heat removal (DHR) system operating in the low pressure injection (LPI) mode... It is assumed that rate of loss from the RCS would be large enough that the high pressure injection (HPI) and makeup pumps would not be capable of providing sufficient flow to keep the core covered without running out... The break size that defines the large LOCA therefore ranges from the smallest break that could be accommodated solely by the LPI and the core flood tanks, up to a double ended rupture of a reactor coolant hot or cold leg. The large LOCA ... is therefore any break whose equivalent flow area exceeds 0.5 ft^2."

The description of a medium LOCA in the IPE submittal includes: "For Davis-Besse, this corresponds to a range of equivalent break areas of 0.02 to 0.5 ft^2. ...It should be noted that, at the lower end of this range (approximately 0.02 to 0.1 ft^2).... Only HPI is needed to provide adequate makeup to the RCS. ... As a practical matter, the frequency of a medium LOCA is estimated in part that there have been no initiating breaks in this range. Hence, it is reasonable to define one event that covers the full range to simplify the analysis..."

This seems to indicate that the medium LOCA category should be considered to be two classes of LOCAs, which we will designate "big-medium" and "little-medium" to avoid nomenclature confusion. The "big-medium LOCA" appears to be break sizes between 0.1 ft^2 and 0.5 ft^2, and require success of only core flood tanks and LPI (injection and recirculation modes) to prevent core damage. The "little-medium LOCA" appears to be break sizes between 0.02 ft^2 and 0.1 ft^2, and require success of at least HPI (injection mode) to prevent core damage.

With respect to the conditional core damage probability for these two parts of the medium LOCA spectrum, there seems to be a discrepancy between the IPE submittal and the "probabilistic safety assessment" for the RPV head cavity. The IPE submittal states "It should be noted that, at the lower end of this range (approximately 0.02 to 0.1 ft^2), the success criteria are actually substantially less restrictive than those applied later for the full range of medium breaks.... From a qualitative perspective, therefore, it is conservative to include these smaller breaks in the medium LOCA category." However in Section B.4, on page 12 of 19 in the safety assessment it is stated that "The largest LOCA within the postulated range [for cavity failure] allows the shortest time to transfer to recirculation, but exceeds the LOCA size that would require high pressure injection. Therefore, a smaller LOCA that requires high pressure injection could be more limiting."

In order to clarify the risk analyses, please provide the following information:

Question 4.a

For the Davis-Besse PSA, what systems/modes of operation are required to perform successfully to prevent core damage for the "little-medium" LOCAs? Can the need for ECCS recirculation mode be avoided? If ECCS recirculation mode is not avoided, is recirculation required in the high, low or both pressure ranges?

Response 4.a

Medium LOCAs at the DBNPS in the lower end ("little-medium") of the range defined in the DBNPS IPE require injection by only one high pressure injection pump (HPI) followed by injection into the RCS by one of two decay heat removal pumps in the low pressure recirculation mode with the suction source switched to the containment emergency sump. The PSA and the IPE success criteria conservatively assume that one of two LPI pumps and one of two core flood tanks are also required for the injection phase for this range of LOCA. This somewhat conservative success criteria permits the inclusion of medium LOCAs at the higher end (large break) of the range and core flood line break LOCAs into the same grouping called medium LOCAs. This avoids the need for yet another category of LOCAs, without resulting in an undue contribution to the core damage frequency. All medium LOCAs require ECCS recirculation, both the high and low pressure range. The significant difference is the amount of time available until recirculation is required, which will vary inversely with the break size. For the IPE and the PSA, this time was taken to that associated with a medium LOCA at the larger end of the spectrum.

Question 4.b

For the current Davis-Besse PSA, what is an appropriate CCDP for "big-medium LOCAs?"

Response 4.b

For a medium LOCA in the higher end of the range ("big-medium") the conditional core damage probability (CCDP) is 2.5E-3. This CCDP is a result of the specific analysis completed for the Reactor Vessel Head wastage concern. The Davis-Besse

PSA does not specifically address different ranges of breaks within the medium LOCA category.

Question 4.c

The core damage frequency contribution from medium LOCAs that is calculated in the Davis-Besse IPE submittal appears to be applicable to "big-medium LOCAs." What is the value of the CCDP for "big-medium LOCAs in the IPE submittal?" If it differs from the value in the current Davis-Besse PSA, is that due solely to requantification or were success criteria changed between the two PSA versions? If success criteria were changed, please clearly specify what changes were made.

Response 4.c

The CCDP for medium LOCAs in the DBNPS IPE submittal is based upon the success criteria for a medium LOCA in the lower end of range and the timing for the Borated Water Storage Tank (BWST) depletion for a medium LOCA in the upper end of the range. This approach combined the worse case attributes of both ends of the medium LOCA range. This provides a conservative CCDP and would identify any specific plant vulnerabilities. However, the method in the IPE does not correspond to any specific size LOCA. The CCDP for a LOCA in the upper end of the medium LOCA range was not calculated, but it would be less than reported in the IPE because high pressure injection pumps (HPI) would not be required for injection or recirculation.

The CCDP used in the "Risk Assessment for Reactor Head Wastage" for the "big-medium LOCA" used the sequences from the Large LOCA to eliminate the success criteria for HPI pumps to inject, since a LOCA in this range does not require HPI injection to maintain core cooling. However, the human actions for transfer to the emergency sump for recirculation mode were requantified to take into account the amount of time available until BWST depletion with a "big-medium" LOCA.

The CCDP using the IPE submittal is 6.9E-3. In the risk assessment using the current DBNPS PSA is CCDP is 2.9E-3. The success criteria used in calculating the CCDP for the "Risk Assessment for Reactor Head Wastage" differ from the IPE only in that injection by HPI is not required for LOCAs greater than about 0.1 ft^2 which is the range of the "big-medium" LOCAs. There were no other changes in the success criteria between the two determinations. Other significant changes are listed below:

1. The DBNPS emergency operating procedure has been improved based upon inputs from the PSA human reliability analysis performed for the PSA. This improvement involved adding additional procedural requirements to initiate

outside the Control Room activities necessary for successful transfer to low pressure recirculation earlier in the procedure (as is already the case for Large LOCAs). This greatly expands the amount of time available for the actions outside the Control Room, which are the most time consuming and complex activities.

2. Additional emergency operating procedure improvements include adding place-keeping aids in the procedure. These two factors provide a significant improvement in the success rate for the action to transfer to the emergency sump.

3. Equipment performance improvements between the IPE and the PSA are significant. The most dramatic improvements has been the motor operated valves (MOV) reliability, which has significantly improved since the implementation of the NRC Generic Letter 89-10 program. The MOV reliability improvement has a significant effect on the Medium LOCA CCDP due to the mitigation systems being dependent upon proper response of motor operated valves. Additionally, other Emergency Core Cooling System (ECCS) mitigation systems components have had some improvements in the equipment reliability since the IPE submittal to the current DBNPS PSA.

Question 5

For the analysis provided in your April 8, 2002 submittal, please quantitatively describe (1) the uncertainty in the resulting value for the frequency of cavity rupture and (2) the uncertainty in the CCDP values used for the resulting medium LOCA. If the analysis for cavity rupture frequency is altered or augmented as a result of responding to the preceding questions, please provide a quantitative description of the uncertainty in that result, also.

Response 5 (1)

The uncertainty in the calculation of the clad rupture is highly dependent upon the selection of the β_c value. The uncertainty is difficult to quantify. Therefore, several conservative assumptions were made in order to provide a bounding result. Conservative assumptions include the following:

- Use of a high value of β_c. This number was used despite evidence from non-destructive testing that did not reveal any defects in the exposed clad.

- Inclusion of pressure ranges higher than ever experienced at DBNPS and higher than the design basis accident analysis peak pressures.

Response 5 (2)

The uncertainty in the CCDP value for the resulting LOCA was determined using a Monte Carlo routine with the previously determined error factors for equipment failures and human failure events. The propagation of uncertainty involved a number of runs to determine convergence. It was determined that after approximately 5000 iterations convergence was obtained. Then 7000 iterations were performed. The result shows a mean CCDP of 2.91 E-3 with the 5%, 95%, and median provided below:

5% confidence level	1.29E-3
50%:	2.17E-3
95% confidence level	6.07E-3

No changes to the cavity rupture frequency resulted from responding to the preceding questions.

Figure 1
Thickness Measurements
Plan View from Top of Head

180°

Nozzle 3

90° 270°

	A	B	C	D	E	F	G	H	I	J
1	6.5	2.6	3.4	0.344					NR	3.9
2	6.2	6.2	1.2	0.344					NR	NR
3	6.2	6.2	1.4	NR	0.346				1.2	6.2
4	5.9	NR	NR	0.309	0.260	0.365	0.305	0.313	NR	6.0
5	3.6	3.6	0.303	0.255	0.273	0.260	0.302	0.299	5.2	3.8
6	6.6	3.0	0.301	0.240	0.291	0.301	0.303	0.302	0.310	0.315
7	3.6	6.2	0.299	0.300	0.311	0.298	0.304	3.600	3.600	3.6
8	6.4	6.2	0.301	0.300	0.300	0.300	0.300	3.400	3.400	5.8
9	6.8	6.2	0.279	0.280	0.340	0.350	0.360	0.370	0.370	0.303
10	NR	NR	0.279	0.276						0.380
11	NR	NR	0.304							

J-groove Weld Area

0°

J-groove Weld Area

Nozzle 11

Approximate Location

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Attachment 2
Page 1

COMMITMENT LIST

The following list identifies those actions committed to by the Davis-Besse Nuclear Power Station (DBNPS) in this document. Any other actions discussed in the submittal represent intended or planned actions by the DBNPS. They are described only for information and are not regulatory commitments. Please notify the Manager - Regulatory Affairs (419-321-7148) at the DBNPS of any questions regarding this document or associated regulatory commitments.

COMMITMENTS	DUE DATE
The estimated areas of exposed clad material that would cause the cladding to fail at normal operating pressure for clad thickness of 0.297" and 0.125" is being analyzed and will be submitted following completion.	July 1, 2002