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Docket Number 50-346

License Number NPF-3

Serial Number 2968

August 13, 2003

United States Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555-0001

Subject: Davis-Besse Nuclear Power Station
Transmittal of Final Report on Examination of the Reactor Vessel Head Degradation, and Safety Significance Assessment Update

Ladies and Gentlemen:

Enclosed for your information is BWX Technologies, Inc., BWXT Services, Inc. Report 1140-025-02-24, "Examination of the Reactor Vessel (RV) Head Degradation at Davis-Besse," June 2003, which was prepared for Framatome ANP, Inc. on behalf of the FirstEnergy Nuclear Operating Company (FENOC). This report describes the laboratory examinations performed on a portion of the degraded reactor vessel head and two control rod drive mechanism (CRDM) nozzles removed from the Davis-Besse Nuclear Power Station, Unit No. 1. The examinations included visual inspections, dye penetrant testing, scanning electron microscopy, energy dispersive spectroscopy, metallography, and Knoop microhardness. Detailed dimensional measurements of the exposed stainless steel cladding area in the Nozzle 3 cavity were also taken. These examinations were performed in accordance with a detailed plan that was approved by the NRC staff. Preliminary results of the examinations were previously shared with the NRC staff.

On April 8, 2002 (Serial Number 1-1268), FENOC submitted a safety significance assessment of the degraded reactor vessel head. On July 20, 2002 (Serial Number 1-1282), FENOC provided additional information regarding the estimated area sizes of exposed clad material that could potentially cause the cladding to fail at normal operating pressure. This information was presented in a calculation performed for FENOC by Structural Integrity Associates (SIA). The calculation was later updated and the results were provided to the NRC by letter dated November 18, 2002 (Serial Number 1-1290). The November 18, 2002 letter noted that the extracted cavity area of the reactor vessel head was under analysis at a laboratory, and that this analysis would provide additional data on clad thickness, exposed clad area, and cracking. The

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November 18 letter further noted that FENOC planned to submit a revision of the safety significance assessment based on the updated data, when available. As noted above, the BWXT report formally provides this data.

Based on discussions with SIA, the impact of the differences between the dimensional data utilized in the safety significance assessment calculation and the more exact data contained in the BWXT report, even considering the effect of cracking discovered in the exposed clad area (which was not considered in the calculation), should be minimal, and therefore, a revision to the safety significance assessment calculation is not warranted. The details of this qualitative assessment are provided in Enclosure 2.

No formal response to this letter is required or requested. However, should you have any questions or require additional information, please contact Mr. Kevin L. Ostrowski, Manager - Regulatory Affairs, at (419) 321-8450.

Very truly yours,

MKL

Enclosures

cc: Regional Administrator, NRC Region III
J. B. Hopkins, NRC/NRR Senior Project Manager
C. S. Thomas, NRC Region III, DB-1 Senior Resident Inspector
Utility Radiological Safety Board

Docket Number 50-346
License Number NPF-3
Serial Number 2968
Enclosure 1

EXAMINATION OF THE REACTOR VESSEL (RV) HEAD DEGRADATION AT DAVIS-BESSE

BWX TECHNOLOGIES, INC.
BWXT SERVICES, INC.
REPORT 1140-025-02-24

JUNE 2003

IMPACT OF LABORATORY EXAMINATION RESULTS ON THE SAFETY SIGNIFICANCE ASSESSMENT OF THE DEGRADED REACTOR VESSEL HEAD

Summary of Existing Analyses and Results

As part of the safety assessment evaluation of the wastage found in the vicinity of CRDM penetration No. 3, elastic-plastic finite element analyses were performed in References 1, 2, and 3 to determine the design margins in the exposed cladding.

- Two bounding clad thicknesses (0.297 and 0.125 inch) were considered in this evaluation. The evaluation assumed a defect-free cladding.
- A very conservative stress-strain curve was used in the evaluation. A plot of the stress-strain curve is shown in Figure 1.
- The analysis was based on the geometry of the exposed cavity found by non-destructive examination (NDE) at the time of the analysis. As stated in Reference 2, in the initial evaluation in Reference 1, an exposed cladding area of 20.5 sq. inches was modeled. In the subsequent evaluation in Reference 2, an exposed cladding area twice this value was modeled. Finally in Reference 3, a range of cladding exposed areas from 20.5 to 82 square inches was considered.
- The results of the evaluation are summarized in Figures 2 and 3 for the two clad thickness cases considered. As can be seen from these Figures, significant margins exist for the exposed cladding area of 20.5 square inches even for the cladding thickness of 0.125 inches regardless of the failure criterion used. As shown in Figures 2 and 3, two failure criteria were used in the analyses; one based on plastic instability and the other based on the maximum strain limited to 11.15%.

Results of Destructive Examination

Pertinent information from the destructive examination performed by BWX Technologies Inc. in Reference 4 is summarized below.

- The actual exposed area of the cladding is approximately 16.5 sq. inches.
- The actual stress-strain curve of the cladding material is much tougher (measured by the area under the stress-strain curve) and more ductile (measured by the failure strain) than that used in the analysis. A comparison of the two curves is shown in Figure 4. It should be noted that the engineering stress-strain curves from Reference 4 have been converted to true stress-strain curves for the comparison in Figure 4.

- Seventy-eight thickness readings were taken of the exposed cladding cavity. The average thickness was 0.256 inches with a minimum reading of 0.202 inches and a maximum thickness of 0.314 inches.
- Cracking was identified on the exposed surface of the cladding, which followed a mixed interdendritic/intergranular path. The cracking occurred at the middle of the cavity and extended a maximum of 0.057 inches below the exposed cladding surface. At the location of the cracks, the minimum clad thickness is at least 0.236 inches. From the photograph shown in Figure 3.6.1 of Reference 4, the largest crack was estimated to be approximately 0.5 inches in length.

Assessment of the Effect of Observed Flaws On Existing Analyses

- Because of the ductile nature of the actual cladding material as demonstrated by its high toughness and ductility, the cladding is expected to fail by net section plastic collapse. As such, the crack tip stress intensity factor is not of essence and only the remaining ligament of the cladding is critical to the determination of the failure pressure.
- The cladding thickness at the location of the cracks is 0.236 inches and the maximum crack depth, at its present stage of development, is 0.057 inches, and therefore, the remaining ligament is 0.179 inches. This thickness is greater than the minimum clad thickness of 0.125 inches analyzed in support of the safety assessment. The results of the analysis of the 0.125 inches cladding should at least bound this actual remaining ligament of the cladding. This is even conservative considering the fact that the maximum crack length is only limited to about 0.5 inches in length and therefore covers only a fraction of the exposed area of the cladding.
- The area of the exposed cavity used in the analyses in References 1, 2 and 3 to support the safety evaluation (20.5 sq. inches) is greater than the actual measured area of 16.5 sq. inches in Reference 4. On an elastic basis, the failure pressure is expected to be inversely proportional to the square of the diameter of the exposed cladding and hence, the failure pressure for the actual exposed area is expected to be at least 20% greater than calculated with the larger area in the analyses.

Conclusion

The above observations support the conclusion that the observed cracks in the cladding should not have any major impact on the existing safety evaluation. Because of the ductile nature of the cladding material, it is expected to fail by net section plastic collapse. The failure pressure is expected to be higher than the results of the 0.125 inches cladding case considered in the existing safety evaluation.

References

1. Structural Integrity Associates Calculation No. W-DB-01Q-301, Rev. 1, "Elastic-Plastic Finite Element Analysis of Davis-Besse RPV Head Wastage Cavity."
2. Structural Integrity Associates Calculation No. W-DB-01Q-302, Rev. 0, "Elastic-Plastic Finite Element Analysis of Enlarged Davis-Besse RPV Head Wastage Cavity."
3. Structural Integrity Associates Calculation No. W-DB-01Q-305, Rev. 1, "Elastic-Plastic Finite Element Analysis of Davis-Besse RPV Head Wastage Cavity with Different Enlarged Areas and Thicknesses."
4. BWX Technologies Inc. Document No. 1140-025-02-24, "Examination of the Reactor Vessel Head Degradation at Davis-Besse," June 2003.

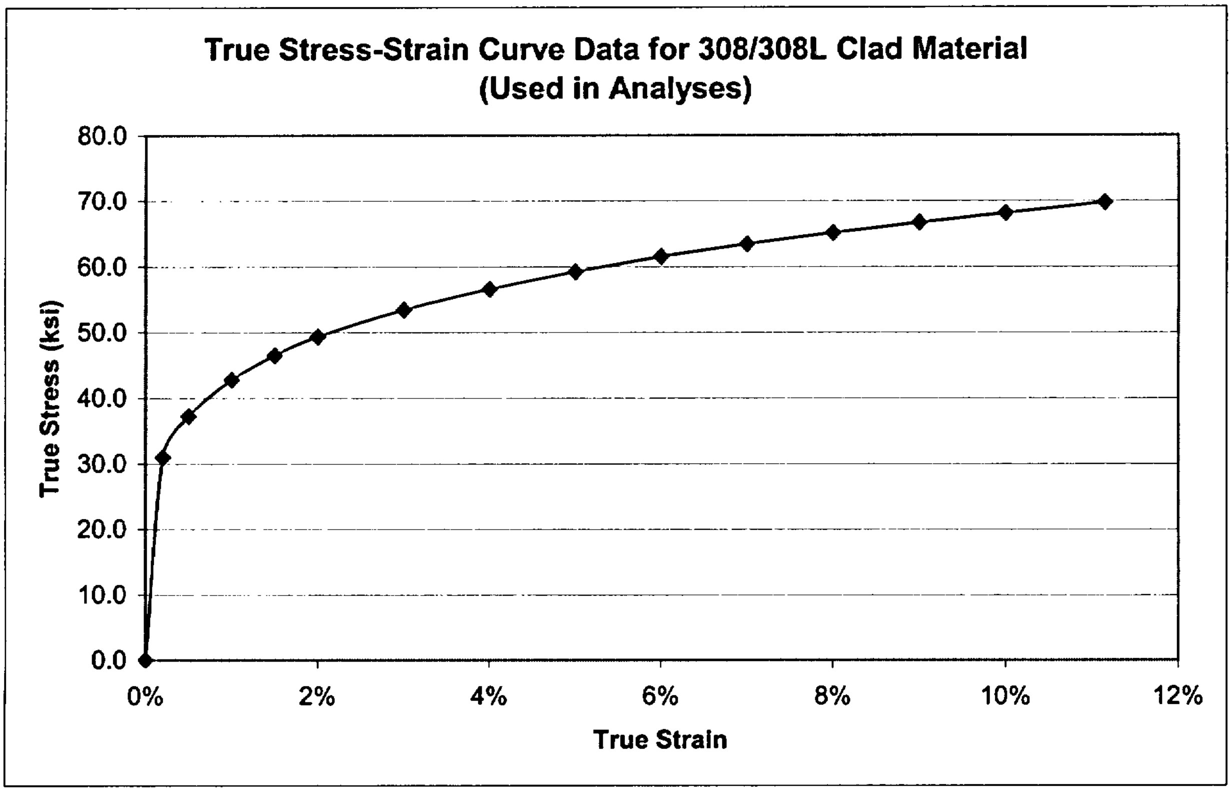

Figure 1. True Stress-Strain Curve for Stainless Steel Type 308/308L Clad Materials Used in Analyses

Docket Number 50-346
License Number NPF-3
Serial Number 2968
Enclosure 2
Page 5

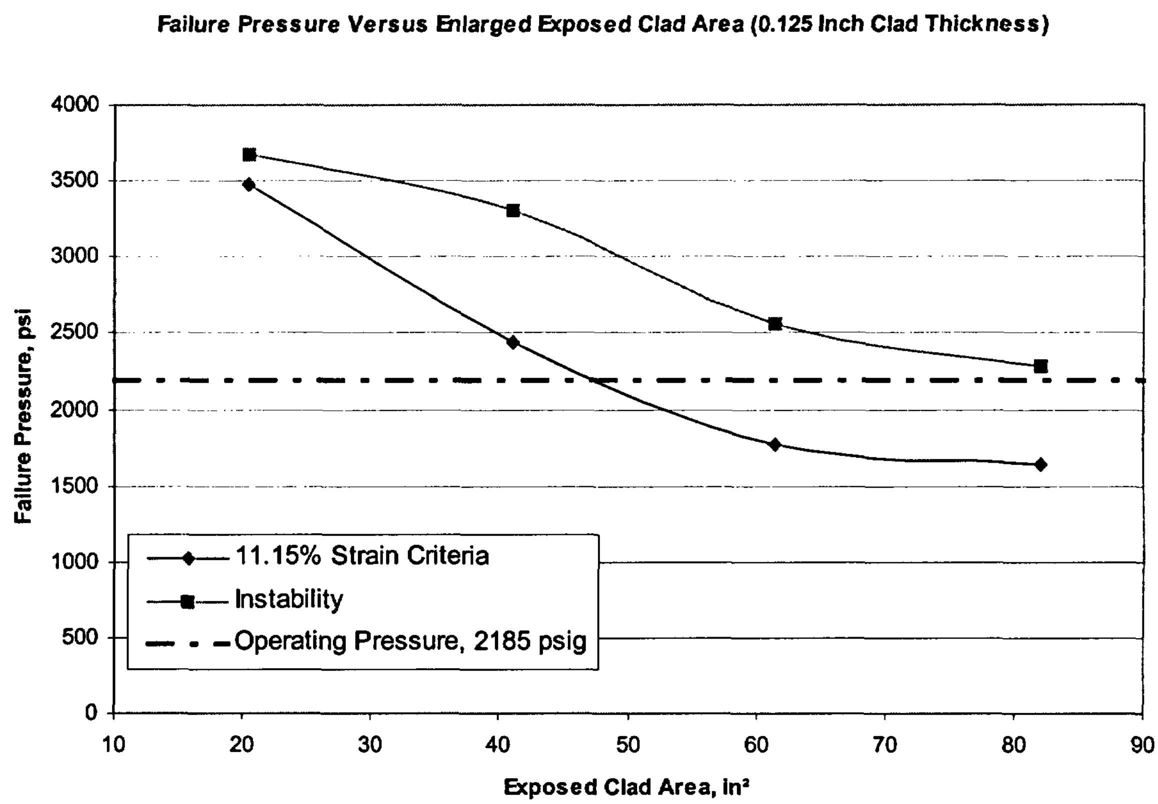

Figure 2. Failure Pressure Versus Cavity Exposed Area (Clad Thickness = 0.125 inches)

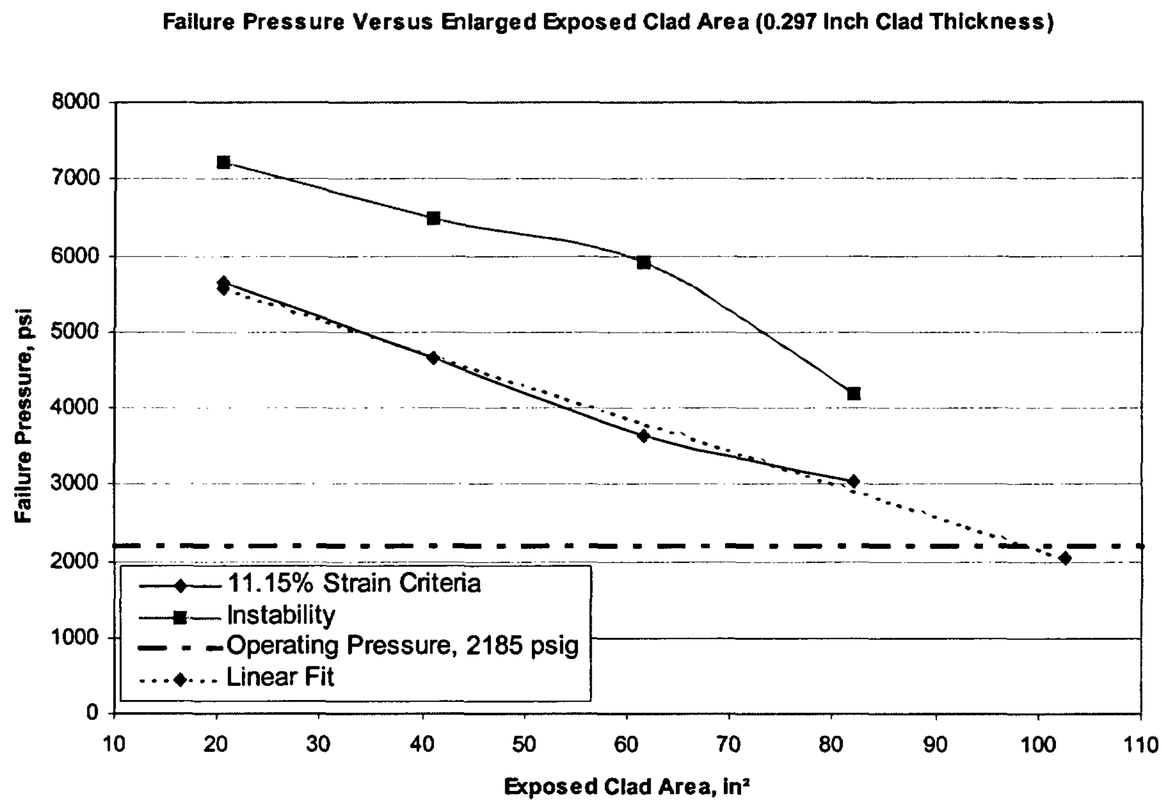

Figure 3. Failure Pressure Versus Cavity Exposed Area (Clad Thickness = 0.297 inches)

Docket Number 50-346
License Number NPF-3
Serial Number 2968
Enclosure 2
Page 7

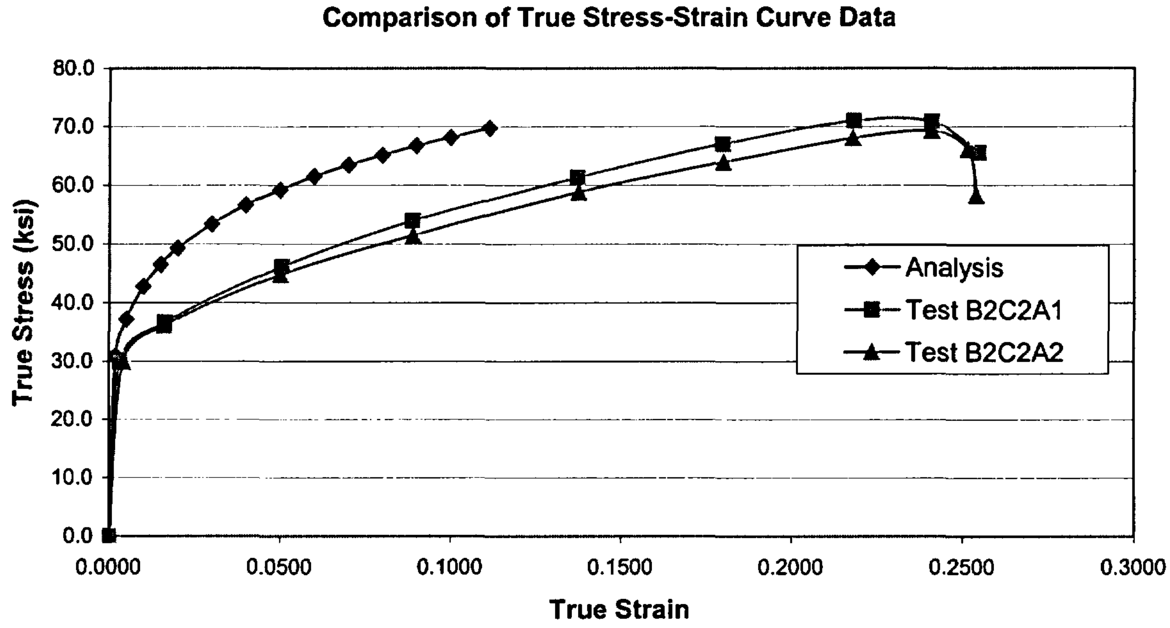

Figure 4. Comparison of Actual Stress-Strain Curve to That Used in the Analyses

Docket Number 50-346
License Number NPF-3
Serial Number 2968
Enclosure 3

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 (419-321-8450) AT THE DBNPS OF ANY QUESTIONS REGARDING THIS DOCUMENT OR ANY ASSOCIATED REGULATORY COMMITMENTS.

COMMITMENTS	**DUE DATE**
None	N/A