Copy with Steve Long's personal notes

FENOC

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Contains Proprietary Information
Pursuant to 10 CFR 2.790

Docket Number 50-346

License Number NPF-3

Serial Number 1-1268

April 8, 2002

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

Subject: Safety Significance Assessment of the Davis-Besse Nuclear Power Station, Unit 1 Reactor Pressure Vessel Head Degradation

Ladies and Gentlemen:

As part of the investigation of the degradation of the Davis-Besse Nuclear Power Station, Unit 1 (DBNPS) Reactor Pressure Vessel (RPV) head, the safety significance assessment as outlined in FirstEnergy Nuclear Operating Company (FENOC) letter Serial Number 1-1267, dated March 27, 2002, has been completed.

Structural integrity analyses of the as-found RPV head were performed independently by Framatome ANP and Structural Integrity Associates, Inc (SIA). These analyses compared favorably with one another and showed that the structural integrity of the as-found RPV head, though degraded, would have functioned to maintain the DBNPS within its design basis during anticipated operational occurrences and postulated accidents. The more conservative of these analyses, the SIA analysis, was used as the basis for the probabilistic assessments and is provided in Attachment 1. This document is considered to be proprietary to Framatome ANP and is requested to be withheld from public disclosure. In accordance with 10 CFR 2.790, an affidavit providing the basis for withholding this information from public disclosure is provided in Attachment 4. A non-proprietary version of this document is provided as Attachment 2.

A deterministic safety assessment has concluded that in the unlikely event of RPV failure considering the as-found degraded RPV head condition: a) adequate core cooling could have been established and maintained for the long term, b) the reactor could have been placed and maintained in a safe shutdown condition, and c) the integrity of the

C/44

containment would not have been compromised. In addition, a probabilistic safety assessment concluded, per Regulatory Guide 1.174 guidelines, that there was a small increase in core damage frequency and a very small increase in large early release frequency. The safety assessments are provided in Attachment 3.

If you have any questions or require additional information, please contact Mr. David H. Lockwood, Manager – Regulatory Affairs, at (419) 321-8450.

Very truly yours,

Attachments

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

Docket Number 50-346
License Number NPF-3
Serial Number 1-1268
Attachment 4

Framatome ANP Affidavit for
Structural Integrity Associates, Inc. File W-DB-01Q-301 (Attachment 1)

(3 Pages Follow)

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Raymond W. Ganthner. I am Vice President, Engineering & Licensing for Framatome ANP ("FRA-ANP"), and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by FRA-ANP to determine whether certain FRA-ANP information is proprietary. I am familiar with the policies established by FRA-ANP to ensure the proper application of these criteria.

3. I am familiar with the information contained in the calculation package prepared by SIA (File No. W-DB-01Q-301) on the subject of "Elastic-Plastic Finite Element Stress Analysis of Davis Besse RPV Head Wastage Cavity." This package, known herein as "Document," contains information that is considered by Framatome ANP to be proprietary. Information contained in this Document has been classified by FRA-ANP as proprietary in accordance with the policies established by FRA-ANP for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by FRA-ANP and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure.

6. The following criteria are customarily applied by FRA-ANP to determine whether information should be classified as proprietary:

(a) The information reveals details of FRA-ANP's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for FRA-ANP.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for FRA-ANP in product optimization or marketability.

(e) The information is vital to a competitive advantage held by FRA-ANP, would be helpful to competitors to FRA-ANP, and would likely cause substantial harm to the competitive position of FRA-ANP.

7. In accordance with FRA-ANP's policies governing the protection and control of information, proprietary information contained in this Document has been made available, on a limited basis, to others outside FRA-ANP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. FRA-ANP policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

Raymond L. Gauthier

SUBSCRIBED before me this 5th day of April, 2002.

Brenda C. Maddox

Brenda C. Maddox
NOTARY PUBLIC, STATE OF VIRGINIA
MY COMMISSION EXPIRES: 07/31/03

I was commissioned a notary public as Brenda C. Cardona.

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 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 associated regulatory commitments.

COMMITMENTS	DUE DATE
None	

Docket Number 50-346
License Number NPF-3
Serial Number 1-1268
Attachment 2

Structural Integrity Associates, Inc.
File Number W-DB-01Q-301
"ELASTIC-PLASTIC FINITE ELEMENT STRESS ANALYSIS OF DAVIS-BESSE RPV HEAD WASTAGE CAVITY"

(23 Pages Follow)

NON-PROPRIETARY

Safety Assessment of the Davis-Besse Reactor Vessel Head Corrosion

Introduction

During the thirteenth refueling outage (13RFO), an inspection of the Davis-Besse reactor vessel head and its control rod drive mechanism (CRDM) nozzles was completed. The inspection identified several CRDM nozzles with cracking. While preparing to repair nozzle number 3, a region of substantial boric acid wastage adjacent to the nozzle was discovered. As a result of this as-found condition, an assessment of the plant's safety has been prepared. A deterministic assessment evaluates the potential consequences associated with a postulated pressure boundary breach from the boric acid wastage. In addition, a risk-based assessment examines the change in plant core damage frequency and large early release frequency. The purpose of this paper is to present the results of the assessments.

The deterministic assessment evaluates three critical aspects associated with a breach of the Reactor Coolant System (RCS) pressure boundary. These include the adequacy of the overall core cooling, the ability to establish and maintain safe reactor shutdown conditions, and the adequacy of the containment, the outer fission product barrier. The core cooling evaluation examines peak clad temperatures, inventory control, long term cooling, and boron concentration control. The reactor shutdown evaluation examines total core reactivity with consideration for potential loss of insertable reactivity. The potential failure mechanisms that could reduce the insertable reactivity are also discussed. Finally, the assessment of the integrity of the containment building evaluates the potential for damage due to generation of a missile and the potential for over-pressurization due to the RCS pressure boundary breach.

The risk assessment evaluates the change in Core Damage Frequency (CDF), the change in Large Early Release Frequency (LERF), and the effect on public health risk. These are based on calculations that consider the frequency of having a transient that could trigger the reactor vessel head to fail, the probability of the head failing, based on as-found conditions, and the conditional core damage probability, once a failure of the Reactor Coolant pressure boundary has occurred.

A. Deterministic Safety Assessment

A.1. Boundary Conditions and Assumptions

In evaluating the consequences of a breach of the pressure boundary, it is necessary to define the initial boundary conditions and assumptions included in the assessment. It is assumed that a small break LOCA is initiated from nominal plant conditions, i.e. an average coolant temperature of 582°F and a hot leg pressure of 2170 psia. The local fluid condition at the top of the reactor vessel (RV) head is approximately 2230 psia and 606.5°F. The nominal break area is based on the area of the exposed cladding material as depicted in Figure 1. The nominal break area, excluding CRDM# 3 is estimated to be approximately 25 in^2. However, a break area, ranging from a pinhole sized opening in the exposed stainless steel cladding up to an area 1.5 times the estimated area of exposed RV cladding material plus the area of a CRDM nozzle that may also have been lost, is considered in this evaluation. For the core reactivity aspects, it is assumed that the control rod located in CRDM #3 is not available for core reactivity shutdown. In addition it is assumed that the Technical Specification minimum Borated Water Storage Tank (BWST) and Core Flood Tank (CFT) volumes and boron concentrations are available.

A.2. Single Failure Considerations

Single failures do not play a particular role in this assessment. This assessment only evaluates the consequences of the initiating event. The failure effects analyses have been included in the various response analyses used to support this assessment. For example, the plant response due to a failure of one Emergency Core Cooling System (ECCS) train has already been evaluated and shown acceptable. This assessment does not alter that failure effects analysis. Where standard failure modes may not bound the as-found conditions, this assessment will include appropriate discussions. As an example, the potential for the failure of multiple CRDMs due to the postulated scenario is discussed within this assessment.

A.3. Capability to Maintain Adequate Core Cooling

A.3.1 Peak Clad Temperature

LOCA analyses performed to demonstrate compliance to 10 CFR 50.46 (Reference A.1) for Davis-Besse do not specifically include the potential failure of the reactor vessel or any of the attached CRDM nozzles. The analyses do include a spectrum of reactor coolant system (RCS) pipe break sizes from 0.01 ft^2 to 14.2 ft^2 in area, Reference A.2. The as-found boric acid wastage area on the top of the reactor vessel head exposed a portion of the stainless steel cladding on the inside of the vessel head that was estimated to be approximately 25 in^2, based on Figure 1. If the cladding had failed, the RCS break area could be anywhere from a pinhole up to the area of the exposed cladding. The smaller a break in the upper head, the less challenging to core cooling, since mass loss rates are lower. This keeps the core covered with froth, thereby limiting cladding temperature excursions. Because of the uncertainty in the size of the wastage area, a hole size up to 50 percent larger than the preliminarily estimated size will be considered (i.e.

37.5 in^2). The effects of a break of this size conservatively bounds the actual wastage area identified in References B.3 and B.4. It is also possible that a rupture near the CRDM #3 nozzle opening could result in that CRDM nozzle being dislodged from the head, thereby increasing the hole size by 12.5 in^2. This represents a maximum hole size of approximately 50 in^2 (~0.35 ft^2) to be considered. The consequences of any break size up to this maximum break area in the RV upper head has been reviewed and is concluded to be bounded by the consequences determined in the existing Davis-Besse LOCA analyses that demonstrate compliance to 10 CFR 50.46.

A break in the RV upper head is favorable from a core cooling standpoint because it is on the hot side of the core, such that all emergency core cooling system (ECCS) fluid is available for core cooling before it can be discharged out of the break. Consequently, the RV liquid inventory would be sufficient to keep the core covered throughout the transient with a two-phase mixture level that would prevent heat up of the fuel pins or cladding. Therefore, the maximum peak cladding temperature would be equal to the initial steady-state cladding temperature of roughly 715°F.

Davis-Besse also has a continuous RV head vent line from CRDM #14 to Steam Generator 1-2. The relative location of the continuous RV head vent line is shown in Figure 2. CRDM #14 is one nozzle removed from CRDM #3 as shown in Figure 1 and Figure 2. If ejection of a CRDM damages this line, the effective break area would not be significantly increased because the pipe area is less than 3.6 in^2 (~0.025 ft^2), Reference A.3, and the break is still within the phenomenological range being considered in this evaluation. There would be no increase in the expected peak clad temperature if the continuous head vent line were damaged.

A.3.2 Long Term Decay Heat Removal/Boron Concentration Control

The maximum break size is large enough to depressurize the RCS well within the low pressure injection (LPI) pressure range within the first six minutes, Reference A.2. Reduced break sizes would take longer to depressurize, but the RCS inventory loss would be lower such that HPI and the remaining RCS inventory are capable of supplying adequate core heat removal. Eventually, the RCS will depressurize to the point that the ECCS injection rate exceeds the leak rate allowing the vessel to begin to refill. The RCS and vessel will refill to the break elevation and the ECCS fluid that passes through the core and out the break can reestablish core exit subcooling.

Eventually, the ECCS injection will empty the borated water storage tank (BWST). If the break size is large, the RCS will depressurize below the core flood tank (CFT) pressure, allowing it to discharge. The containment spray flow, if actuated, and the RCS inventory that exits the break will flow through designed drain paths into the reactor building lower elevations and raise the containment water level until it spills into the emergency sump. The liquid in the emergency sump provides the inventory that the Containment Spray and ECCS pumps use to provide long-term recirculation injection. Consequently, an assured long-term core cooling liquid source from the emergency sump will be available. Switching the ECCS pump suction source from the BWST to the sump establishes the path for continuous long-term cooling.

The break size and location are also favorable from a boron precipitation perspective. The specific break location will inherently provide adequate passive boron concentration control since flow will be maintained through the break. While not needed for this break location, one of the active boron dilution methods may also be initiated, if conditions specified by Reference A.4 are met. Both the passive dilution provided by the break and the active boron dilution control ensure that the post-LOCA core boron concentration will be maintained within acceptable limits.

A.4 Reactor Shutdown

A.4.1 Immediate Shutdown Margin

In addition to effective RCS inventory control to establish and maintain core cooling, adequate reactor shutdown capability must also be demonstrated for this event. During the early portion of a small break LOCA transient, shutdown is ensured by the insertion of the control rods. In the LOCA analyses, a minimum inserted control rod worth is credited. The analyses model a control rod worth that is sufficient to offset the moderator and Doppler deficits from hot full power conditions to hot zero power conditions plus 1-percent for shutdown margin. For Cycle 14 this is 2.3 %Δk/k (Reference A.5). [The calculation was based strictly on the Cycle 14 core design, with beginning of life core parameters, and the expected post-LOCA temperatures. The Cycle 14 core design will be more limiting than Cycle 13 because the Cycle 14 design includes four additional fresh fuel assemblies surrounding the interior of the core (i.e., core locations H07, G08, H09, and K08 in Figure 2). This will result in a relatively high worth for CRDMs #1, 3, 6, 7, and 11 when these control rods are not placed in the core, which results in a lower total inserted worth.] The minimum inserted rod worth modeled in the LOCA analyses is less than the total available, even after accounting for calculation uncertainties and a penalty for the maximum stuck control rod, Reference A.6. No credit is taken for other reactivity sources. While the total available rod worth would be somewhat less with the loss of CRDM #3, the shutdown margin modeled in the analyses would still be preserved. Evaluation shows that with the five rods around the corrosion area, and one other randomly stuck rod, there would be an insertable reactivity of 3.6 %Δk/k (Reference A.7). This exceeds the amount of reactivity needed to shut down the reactor to hot zero power conditions, as discussed above.

65

A.4.2 Potential for Collateral Damage to Adjacent CRDMs

In a previous evaluation, Reference A.8, performed for the B&W Owners Group (BWOG) it was assumed that an ejected CRDM would rise vertically. If there were an impact on an adjacent CRDM, the contact would not cause failure of those adjacent control rods because there was no indication of significant wastage near the adjacent CRDMs and the J-groove welds were not structurally impaired. Based on the as-found condition of the J-groove weld and the CRDM nozzle #3, this position is not necessarily invalidated but can not be rigorously demonstrated. Therefore, assessment of the potential for causing sufficient damage to prevent the adjacent rods from inserting is evaluated below. The BWOG study also evaluated the failure of an adjacent control rod to insert due to loose parts or metal fragments in the upper head region associated with a control rod ejection. This condition is also examined.

Ex 5

[redacted]

In order to address the potential damage to adjacent control rods, an assessment was performed (Reference A.9) to determine the effects of steam flowing out of a hypothesized rupture of the cladding at CRDM nozzle #3 into the service structure on the adjacent CRDMs. The break area, 37.5 in^2 assumed based upon the size of the exposed cladding with a 50% increase. The break area considered the cladding rupture along with the catastrophic loss of CRDM nozzle #3. There are no jet loads on the adjacent CRDMs since the CRDMs are parallel to the jet. As such, the loadings on the CRDMs have been treated as cavity pressures. The loadings are conservatively approximated based on ANS 58.2/1988, Reference A.10, by calculating the pressure as a function of the length from the RV head. The most conservative CRDM location is chosen to maximize the pressure loadings on the CRDM. The pressure loadings are applied to a structural model of the CRDM. The structural model is fixed at the RV Head and pinned at the service support structure (SSS) tie plate location. The bending moments are a maximum at the RV head location. Reviewing the section properties of the CRDM and housing, this is the critical location. Using the housing cross-section, just above the RV head, the bending stress is less than the yield strength at the normal full-power temperature. Therefore, it can be concluded that the adjacent CRDMs loaded from a hypothesized rupture of the as-found wastage area and the catastrophic loss of the CRDM nozzle stay elastic and the control rods can be inserted. Since the rods rapidly drop into the core, there is insufficient time for the ejected CRDM to leave the service structure, hit the missile shield, and fall back to the structure before rod insertion is complete. Therefore, subsequent CRDM damage would not affect the insertion of the rods.

Ex. 5

[redacted]

The "effective" break area considered in previous evaluations was small, such that it would take roughly 15 to 20 seconds for the RCS to depressurize to the low-pressure reactor trip setpoint. This timing could allow the loose parts that may be generated as the rod is ejected to enter the adjacent control rods and potentially impair their operation. It was postulated that a nozzle fragment could be generated if a circumferential crack joined up with axial cracks in the nozzle base metal below the weld. In the BWOG report (Reference A.9), it was noted that about 25 percent of the plenum cover area is comprised of control rod column weldments. If debris falls within a guide column weldment, to interfere with control rod movement, the debris would have to be trapped between a control rod guide brazement and the control rod spider. The area of the control rod guide brazement that debris must be trapped against to interfere with the control rod spider movement is a small portion of the column weldment area. The control rod guide brazements are at several elevations and blockage at each decreasing brazement elevation would permit an increasing length of control rod to enter the core. Consequently, the probability of impeding multiple control rods from being inserted is unlikely.

The indication surrounding Davis-Besse CRDM nozzle #3, however, is more severe than previously found for other CRDM nozzles or J-groove welds, in that the potential break area could be much larger than previously considered. Based on the as-found conditions, if the CRDM is lost and the stainless steel cladding gives way, a break size of approximately 0.35 ft^2 is postulated. The low RCS pressure reactor trip signal will be reached between 1 to 2 seconds. The control rods will begin falling, and will be fully inserted within two seconds after the trip signal. This timing will not allow enough time for the fragments from a potentially failed control rod assembly to inhibit the insertion of any adjacent control rods. As a result of the early reactor trip, fragments from a potentially failed control rod assembly would not inhibit the insertion of any adjacent control rods.

A.4.3 Maintaining Safe Shutdown Conditions

As discussed above in Section 4.1, sufficient negative reactivity exists to immediately shutdown the reactor. Early in the transient, voiding in the core region and xenon buildup augments the inserted control rod worth. Because of this specific break location, core subcooling may be regained and over time, xenon will decay, eliminating these additional sources of negative reactivity. Depending on the inserted control rod worth and the moderator temperature coefficient, the real transient core power may become somewhat cyclic. If the power does increase, subcooling margin will again be lost, localized boiling may be initiated, and the power will decrease. This cycle will continue until the mixed-mean boron concentration has been increased sufficiently to suppress the core reactivity.

Post-LOCA long-term shutdown of the core is provided by the soluble boron in the ECCS fluid and total inserted worth. The required boron concentration in the ECCS fluid is based on only having 50 percent of the control rod worth available. As a result, even with a stuck rod penalty and the loss of a single control rod (CRDM nozzle #3), adequate long-term shutdown would be demonstrated.

Notwithstanding the conclusions developed by the CRDM loading evaluation, a determination of the overall available control rod worth was made assuming that CRDMs #1, 3, 6, 7, and 11 fail to insert. (See Figure 2 for details on the CRDM locations.) The calculation was based strictly on the Cycle 14 core design, with beginning of life core parameters, and the expected post-LOCA temperatures. The Cycle 14 core design will be more limiting than Cycle 13 because the Cycle 14 design includes four additional fresh fuel assemblies surrounding the interior of the core (i.e., core locations H07, G08, H09, and K08 in Figure 2). This will result in a relatively high worth for CRDMs #1, 3, 6, 7, and 11 when these control rods are not placed in the core, which results in a lower total inserted worth. Using the smaller inserted rod worth from the Cycle 14 core design and applying it to the boron requirements for both Cycle 13 and Cycle 14 ensures that a conservative evaluation is performed. The results of the calculation (Reference A.11) concluded that the minimum total inserted worth of the remaining control rods, less an additional penalty for the highest worth remaining rod to be stuck out of the core, was found to be approximately 3.1% $\Delta k/k$.

In parallel with the evaluation contained in Reference A.11, a calculation was performed to determine the control rod worth required to maintain adequate shutdown margin after considering the negative reactivity addition from the boron in the RCS, BWST, CFTs, and makeup tank for the Cycle 13 and 14 core designs, Reference A.7. Using the minimum Technical Specification BWST and CFT volumes and boron concentration of 2600 ppm, Reference A.12, the required control rod worth for long term shutdown, including a 1-percent shutdown margin, was between approximately 1.0% to 1.9% $\Delta k/k$, depending on the fuel cycle and fluid temperature being evaluated.

Based on a simple comparison of the available control rod worth versus the required worth, there would have been sufficient negative reactivity in the inserted control rods and the effective mixed-mean boron concentration to adequately shutdown the core following this postulated LOCA scenario. The negative reactivity would maintain the core in a subcritical state even if CRDMs #1, 3, 6, 7, and 11 did not insert along with the highest remaining worth control rod stuck out of the core. In other words, there would be sufficient boron in the available fluid systems and the control rods that were inserted for long-term core reactivity control following this location-specific small break LOCA.

A.5 Containment Integrity

The containment integrity must also be considered. Davis-Besse has a reactor vessel missile shield installed above the reactor vessel that would preclude an ejected CRDM from hitting the containment structure and impairing its fission product barrier function.

The mass and energy release to the containment associated with the spectrum of LOCA break sizes has been evaluated from a peak pressure and temperature perspective, Reference A.13. The rapid blowdown from a double-ended hot leg pipe break (14.14 ft^2) is limiting for Davis-Besse and the consequences bound this smaller break size because the estimated area is more than an order of magnitude smaller. The resulting rate of energy release to the reactor building will be much slower, such that the peak allowable containment pressure and temperature will not be challenged for this postulated reactor vessel break.

This size break may or may not actuate the Containment Spray system. That system is set to start automatically if the containment pressure limit is approached. The system can be started manually if deemed appropriate by plant operators. This may be done to speed depressurization of containment and to scrub radioactive releases from the atmosphere, however, manual actuation will not affect the consequences of the accident.

B. Risk Based Safety Assessment

B.1 Boundary Conditions and Assumptions

An assessment of the change in core damage frequency was made due to the degradation of the reactor vessel head. This evaluation is based, in part, on the deterministic evaluation described above and Reference B.1. This includes assuming sufficient negative reactivity is available to shut down the reactor and maintain it shut down throughout the event. The basis for this assumption is described in the Section A.4, above. It is also assumed that no impairment to containment integrity occurs as a result of the head failure, including ejection of a CRDM. The basis of this assumption is discussed in the Section A.5, above.

It is not necessary to postulate specific equipment failures in this probabilistic portion of the Safety Assessment. Equipment failures are addressed in the core damage probability calculation though the normal failure rates included in the PSA evaluation model.

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B.2 Event Initiators

The only events that could have challenged the degraded RCS pressure boundary include overpressure transients and seismic events. The Davis-Besse seismic PSA analysis (Reference B.2) was reviewed to estimate the frequency of pressure boundary failure due to a seismic event. Based on this review it was determined that a seismic event of sufficient magnitude to initiate failure of the degraded RCS pressure boundary would have a frequency sufficiently low to justify neglecting seismic events in this analysis.

It is therefore postulated that only an overpressure condition initiated by a plant transient could have challenged the RCS integrity. This assessment evaluates the risk contribution from a LOCA due an overpressurization transient causing failure of the wastage area in the head.

B.3 Event Frequency

The frequency of a transient that actually causes failure of the reactor vessel head was calculated. This frequency is based on the probability of head failure and the frequency of the overpressure transient. The probability of head failure and the frequency of the over pressurization transient are discussed below.

B.3.1 Probability of Pressure Boundary Failure due to an Overpressure Transient

A finite element stress analysis of the Davis-Besse head wastage cavity was performed by Framatome ANP (Reference B.3) and independently by Structural Integrity Associates (Reference B.4). These analyses determined a maximum pressure for the cladding without loss of pressure boundary function. The definition of maximum pressure was chosen to be the pressure at which the strain value reaches 11% or more through the full thickness of the cladding. A maximum pressure of 5600 psig was assumed for the risk analysis based on the more limiting average of the analytical results.

Exceeding the maximum pressure of 5600 psig is not considered credible. Overpressure protection is provided by the Pilot Operated Relief Valve (PORV) and the two Code Safety Valves, with setpoints of 2450 psig and 2500 psig, respectively. Furthermore, the pressure sufficient to lift the reactor vessel head from the vessel and break the seal between the head and the vessel flanges has been estimated to be in the range of 3000 psig to 4000 psig. Consequently, the head would lift and relieve pressure through the flanges before exceeding the capability of the wastage area. However, to calculate the risk of the as-found condition, it is recognized that the clad in the wastage area could fail before the calculated maximum pressure. The probability of failure before reaching the maximum calculated pressure was estimated by applying the method taken from

NUREG/CR-5603, Reference B.5. This approach is also described in NUREG/CR-5604, Reference B.6 and NUREG/CR-2300, Reference B.7. Based on these references, the failure probability of the cladding on the wasted area of the reactor head can be expressed as follows:

$$f' = \Phi\left[\frac{\ln\left(\frac{p}{\hat{P}}\right)}{\beta_c}\right]$$

Where

f' = Probability that failure occurs at pressure p

$\hat{P}$ = Median pressure capability (psig)

p = random pressure capacity (psig)

βc = Composite logarithmic standard deviation for randomness

ϕ = Standard gaussian cumulative distribution function

The median pressure capacity represents the internal pressure level for which there is a 50% probability of failure. This value is assumed to be 5600 psig based on the results in Reference B.3 and B.4.

The logarithmic standard deviation β_c is a function of various uncertainties. Uncertainties can exist due to difference in modeling and material properties. Modeling uncertainties are associated with the assumptions used to develop the analytical models and differences between analytical idealization and the real conditions. Material uncertainties involve variability in material properties. Reference B.5 provides values of logarithmic standard deviations for various materials and applications. For this analysis, a very conservative logarithmic standard deviation of 0.33 was used based on the values reported in References B.5 and B.6 for a temperature of 600 degrees.

Applying the assumed median pressure of 5600 psig and logarithmic standard deviation of 0.33, the equation can be solved for probability of failure as a function of pressure. The results of this analysis are as shown in Table 1.

[redacted]

Table 1 – Failure Probabilities as a Function of Pressure

Pressure (psig)	Failure Probability
2250	2.86E-3
2300	3.50E-3
2350	4.25E-3
2400	5.12E-3
2450	6.12E-3
2500	7.27E-3

B.3.2 Transient Frequencies

The transient history as documented in Davis-Besse Transient Assessment Program Reports was reviewed to collect data on pressure excursions at Davis-Besse from July 1979 through February 2002. The yearly frequency of pressure excursions was calculated based on total operating hours from July 1979 to February 2002. The number of pressure transients and the yearly frequencies are shown in Table 2. No pressure excursions of 2450 psig or greater have occurred in the Davis-Besse operating history. Therefore, the frequency of these events was estimated using a Bayesian update with a non-informative prior as described in NUREG/CR-2300, Reference B.7. The frequency for events greater than 2450 psig was further divided into two categories representing events with pressures greater than 2450 psig and greater than 2500 psig. Reference B.8 provides the details of this calculation.

Table 2 – Frequency of Pressure Transients

Pressure (psig)	Number of Events	Frequency (yr^{-1})
> 2250	4	2.54E-1
> 2300	8	5.08E-1
> 2350	2	1.27E-1
> 2400	1	6.35E-2
> 2450	0	1.59E-2
> 2500	0	1.59E-2

B.3.3 Frequency of Pressure Boundary Failure

The frequency of failure for the affected area of the reactor vessel head was calculated using the information in Tables 1 and 2. The frequency of each category of overpressure transient was multiplied by the probability of failure of the head area for that pressure. The results were then summed to calculate a total failure frequency.

Table 3 – Frequency of Pressure Boundary Failure

Pressure (psig)	Overpressure Frequency (yr^{-1})	Head Failure Probability	Failure Frequency (yr^{-1})
> 2250	2.54E-1	2.86E-3	7.27E-4
> 2300	5.08E-1	3.50E-3	1.78E-3
> 2350	1.27E-1	4.25E-3	5.40E-4
> 2400	6.35E-2	5.12E-3	3.25E-4
> 2450	1.59E-2	6.12E-3	9.71E-5
> 2500	1.59E-2	7.27E-3	1.15E-4
			3.58E-3

B.4 Conditional Core Damage Probability

Possible LOCAs that could result from a failure of the wasted area range from very small to 0.35 ft^2. This includes breaks that would be categorized as small or medium in the Davis-Besse PSA. The conditional core damage probability for small LOCAs is smaller than for medium LOCAs. Therefore, for this analysis, all failures will be assumed to be in the medium LOCA range.
Medium LOCAs, as defined in the PSA, cover a range of break areas of 0.02 ft^2 to 0.5 ft^2. The largest LOCA size, postulated for failure of the head wastage area, is less than the maximum of the medium LOCA range. Therefore, this analysis will evaluate the conditional core damage probability for a 0.35 ft^2 LOCA. The largest LOCA within the postulated range 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. A LOCA of 0.1 ft^2 represents the upper range of the LOCAs that require high pressure injection so this size LOCA wa[redacted] also evaluated.

The results of the analysis documented in Reference B.9 demonstrates that the highest conditional core damage probability (CCDP) of any LOCA in the range from very small through 0.35 ft^2 occurs for a break of about 0.1 ft^2. The conditional core damage probability of this size break is 2.9E-3. This result differs from the conditional core damage probability in the Davis-Besse Individual Plant Examination (IPE) based on the following:

Ex 5 [redacted]

- The CCDP in the IPE represents the worst case conditions for any medium LOCA and combines the human error probability of the largest possible medium LOCA with the more restrictive success criteria of a smaller LOCA.
- The human error probability (HEP) for the action required to transfer to recirculation was based on a specific calculation for a 0.1 ft^2 LOCA. Additionally, the methods used for timing calculations have been improved by the use of the Modular Accident Analysis Program (MAAP) which replaces more simple calculations based on rated pump flows.
- The Davis-Besse Emergency Procedure, Reference A.4 was improved based on feedback from the PSA human reliability analysis. This improvement involved adding the procedural requirement to initiate outside of the control room actions necessary to initiate low pressure recirculation prior to reaching low level in the BWST. This change greatly expands the time window for outside the control room actions. Additionally, the actions have been included on a check off list requiring a specific written entry.

B.5 Core Damage Frequency

The more limiting core damage probability was evaluated to be associated with the 0.1 ft^2 LOCA. Therefore, the result for this smaller LOCA will be used in this analysis. Based on the conditional core damage probability of 2.9E-3 and the failure frequency of 3.6E-3/yr, the core damage frequency can be calculated as follows:

CDF = 2.9E-3 x 3.6E-3/yr
CDF = 1.0E-5 /yr.

B.6 Core Damage Probability Evaluation

The calculation of a core damage probability requires information concerning the duration of the head wastage condition, which is difficult to assess. However, to estimate a result, the vessel head condition can be assumed to have existed at the as-found condition for one year. The total core incremental damage probability for this time would be 1.0E-5. Based on the guidance in the EPRI PSA Procedures Guide ,Reference B.10, this core damage probability is conservatively estimated at the value of 1E-5, which is considered the lower threshold of being risk significant for a temporary change.

Ex 5 [redacted]

B.7 Large Early Release Frequency Contribution

For the assessment of large early release frequency (LERF), it is conservatively assumed that all pressure boundary failures result in a medium LOCA. The conditional large early release probability (CLERP) for a medium LOCA is about 4.0E-6 based on the results of the Davis-Besse PSA (Reference 11).

Using the frequency of failure calculated for the reactor vessel head from Table 3 and the conditional large early release probability, LERF for this condition is calculated as follows:

LERF = 3.6E-3 /yr. x 4.0E-6
LERF = 1.43E-8 /yr.

The CLERP for a medium LOCA is very small resulting in a LERF contribution from nozzle failures that is much less significant than the core damage frequency contribution. However, this small LERF contribution is expected for a medium LOCA due to specific-plant features. Davis-Besse has a large dry containment with a large containment to core power ratio. Consequently, the most significant contributors to release from containment for Davis-Besse are containment bypass events, such as steam generator tube ruptures or interfacing system LOCAs. Other significant contributors to LERF include sequences where core damage results from loss of support systems that also support containment cooling. These sequences include loss of cooling water systems or station blackout. Medium LOCAs are not representative of either of these dominant LERF sequences. Additionally, LOCAs depressurize the reactor coolant system prior to core damage, which reduces the possibility of high pressure melt ejection.

B.8 Public Health Risk

The Davis-Besse Level 3 PSA, Reference B.12, was used to estimate the impact on the public, in terms of person-REM, from the condition of the head. Based on the results of the Level 3 PSA analysis, a conditional population dose given a medium LOCA is 1.57E2 person-REM.

Using the frequency of failure calculated for the reactor vessel head from Table 3 and the conditional population dose, the health consequences for this condition are calculated as follows:

Person-REM = 3.6E-3 /yr. x 1.57E2 Person-REM
Person-REM = 0.56 Person-REM/yr.

Conclusions

Based on the deterministic safety assessment, it is concluded that

- Adequate core cooling can be established and maintained for the long term.
- The reactor can be placed and maintained in a safe shutdown condition.
- The integrity of the containment will not be compromised as a result of the damage found on the reactor vessel head.

The results of the risk based safety assessment show that:

- The core damage frequency (CDF) contribution of the as-found condition is conservatively estimated to be 1.0E-5 / yr. or less. This is a value that would be considered small per Regulatory Guide 1.174 (Reference B.13).
- The large early release frequency (LERF) contribution of the as-found condition is estimated to be 1.4E-8 / yr. This is a value that would be considered very small per Regulatory Guide 1.174 (Reference B.13).
- The public health risk was calculated to be 0.56 person-REM/yr which is a very small contribution to the total risk for Davis-Besse.
- Core damage probability is difficult to assess since the duration of the condition is not well known. However, it is expected to be approximately 1E-5. This is the lower threshold that is considered risk significant per the EPRI PSA Applications Guide (Reference B.10).

References

A.1 Title 10 of the Code of Federal Regulations, Part 50.46.

A.2 Framatome ANP Document 86-5006232-00, "DB-1 LOCA Summary Report."

A.3 Davis-Besse System Description SD-039A, "Reactor Coolant System."

A.4 Davis-Besse Operating Procedure DB-OP-02000, "RPS, SFAS, SFRCS Trip or Steam Generator Tube Rupture."

A.5 Framatome ANP Document No. 86-5006232, "DB-1 LOCA Summary Report."

A.6 Framatome ANP Document No. 103-2368-00, "Davis-Besse Nuclear Power Station Unit 1, Cycle 13 Reload Report," BAW-2368.

A.7 Framatome ANP Document No. 86-5017531-00, "D-B Post LOCA Sump Support From Task 11."

A.8 Framatome ANP Document 51-5011603-01, "RV Head Nozzle and Weld Safety Assessment."

A.9 Framatome ANP Document No. 32-5017526-00, "CRDM Integrity Due to RV Head Rupture."

A.10 American National Standard, "Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture, ANSI/ANS-58.2-1988."

A.11 Framatome ANP Document No. 32-5017530-00, "Davis-Besse RV Head SBLOCA Required Rod Worth."

A.12 Davis-Besse Nuclear Power Station, Unit No. 1, Docket No. 50-346, Technical Specifications, Through Amendment 255, Limiting Condition for Operation 3.5.1 and 3.5.4.

A.13 Davis-Besse Calculation C-NSA-60.05-009, "Containment Analyses for Caldon Power Uprate."

B.1 Framatome ANP, 51-5017529-00, Davis-Besse RV Head SBLOCA Assessment."

B.2 Davis-Besse Nuclear Power Station Seismic Risk Calculation

B.3 Framatome ANP, 32-5017543-00, DB1- Damage Assessment of CRDM 3

B.4 Structural Integrity Associates Calculation W-DB-01Q-301, Elastic Plastic Finite Element Stress Analysis of Davis-Besse Head Wastage Cavity.

B.5 NUREG/CR-5603, Pressure-Dependent Fragilities for Piping Components, October 1990.

B.6 NUREG/CR-5604, Assessment of ISLOCA Risk-Methodology and Application to a Babcock and Wilcox Nuclear Power Plant, April 1992.

B.7 NUREG/CR-2300, PSA Procedures Guide, January 1983.

B.8 Calculation C-NSA-99.16-49, Revision 0, Success Criteria and Human Action Timing for Rx Head LOCA, March 2002.

B.9 Calculation C-NSA-99.16-050, Rev. 0, Risk Assessment for Reactor Head Wastage, April 2002.

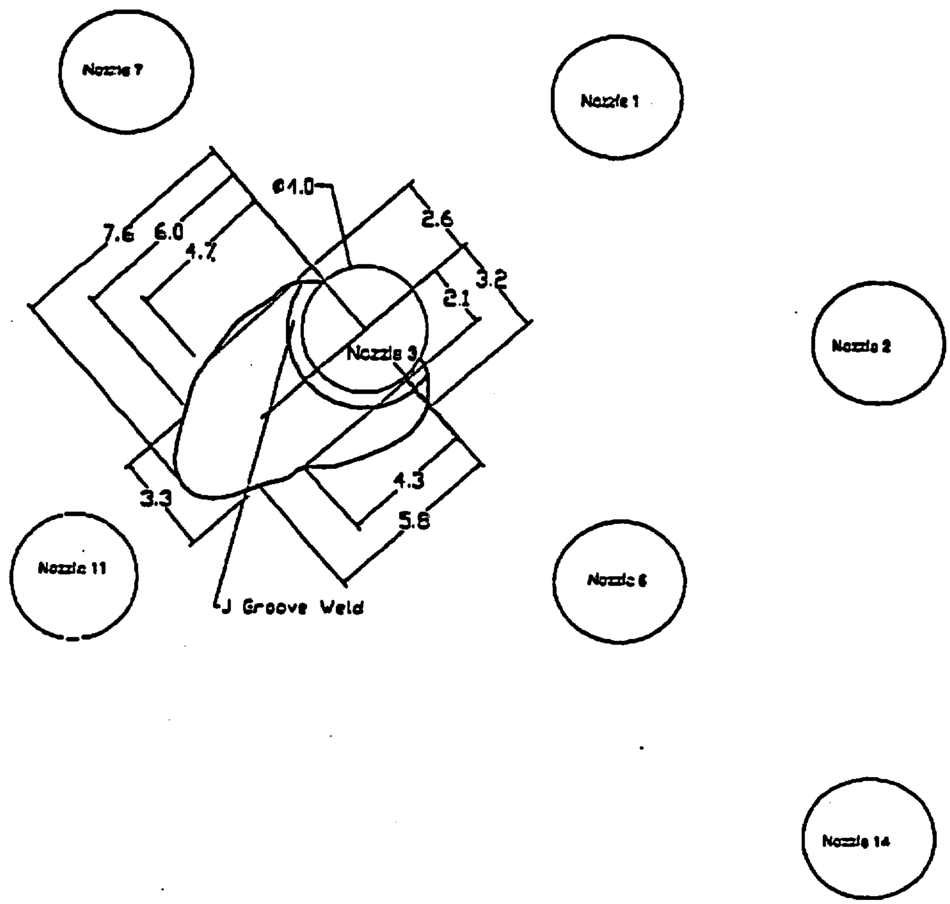
B.10 Electric Power Research Institute (EPRI) Report TR-105396, PSA Applications Guide.

B.11 Davis-Besse PSA, Rev. 3.

B.12 BAW-2382, Level 3 PRA for Davis-Besse, January 2001.

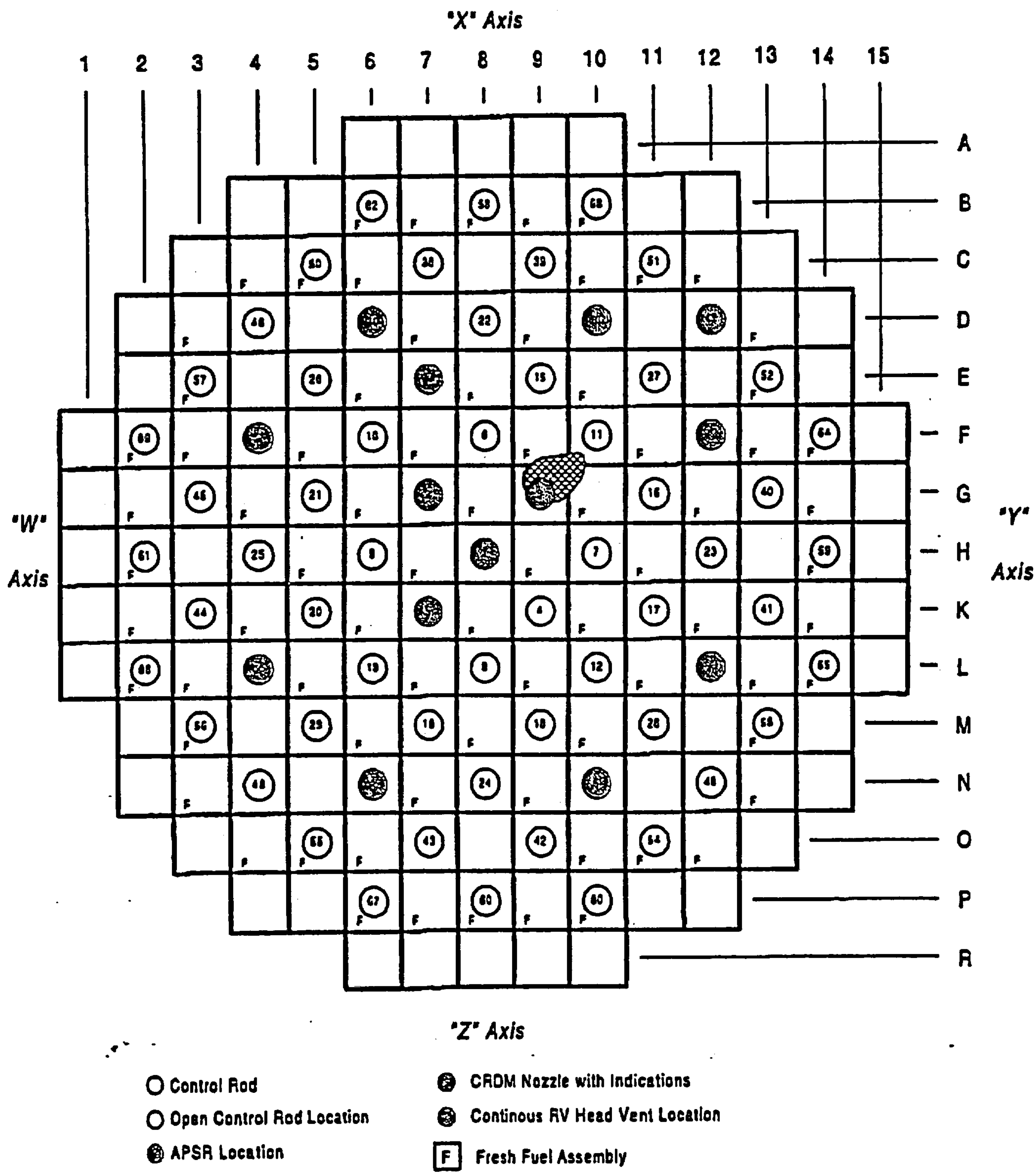
B.13 Regulatory Guide 1.174, An Approach for Using Probabilistic Risk Assessment in Risk Informed Decisions on Plant Specific Changes to the Licensing Basis, July 1998.

Figure 1 – Exposed RV Cladding Area at CRDM Nozzle #3



Note: This drawing was based on a preliminary investigation of the exposed reactor vessel cladding area. For the small break LOCA evaluation, break areas anywhere from a pinhole sized opening in the exposed stainless steel cladding up to an area that is 1.5 times larger than the estimated area, plus the area of a CRDM nozzle, was considered.

Figure 2 - Davis-Besse Fuel and CRDM Locations (Cycle 14)



June 11, 2002

Mr. Howard Bergendahl
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SUBJECT: DAVIS-BESSE NUCLEAR POWER PLANT - REQUEST FOR WITHHOLDING INFORMATION FROM PUBLIC DISCLOSURE (TAC MB4799)

Dear Mr. Bergendahl:

By letter from FirstEnergy dated April 8, 2002, and Framatome ANP (FRA-ANP)'s affidavit, executed by Raymond W. Ganthner, dated April 5, 2002, the following proprietary document was submitted:

> "Elastic-Plastic Finite Element Stress Analysis of Davis-Besse RPV Head Wastage Cavity and Safety Assessment of the Davis-Besse Reactor Vessel Head Corrosion."

FRA-ANP requested that the proprietary information be withheld from public disclosure pursuant to 10 CFR 2.790. Also, a nonproprietary version of this document has been placed in the Nuclear Regulatory Commission (NRC) public document room and added to the Agencywide Documents Access and Management System Public Electronic Reading Room located on the NRC internet website at www.NRC.gov. FRA-ANP stated that the information should be considered exempt from mandatory public disclosure for the following reasons:

(6)a The information reveals details of FRA-ANP's research and development plans and programs or their results.

(6)b Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(6)c The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage of FRA-ANP.

(6)d The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for FRA-ANP in product optimization or marketability.

(6)e The information is vital to a competitive advantage held by FRA-ANP, would be helpful to competitors to FRA-ANP, and would likely cause substantial harm to the competitive position of FRA-ANP.

We have reviewed your submittal and the material in accordance with the requirements of 10 CFR 2.790 and, on the basis of FRA-ANP's statements, have determined that the submitted information sought to be withheld contains trade secrets or proprietary commercial information. Therefore, the information marked as proprietary in your submittal of April 8, 2002, will be withheld from public disclosure pursuant to 10 CFR 2.790(b)(5) and Section 103(b) of the Atomic Energy Act of 1954, as amended.

Withholding from public inspection shall not affect the right, if any, of persons properly and directly concerned to inspect the document. If the need arises, we may send copies of this information to our consultants working in this area. We will, of course, ensure that the consultants have signed the appropriate agreements for handling proprietary information.

If the basis for withholding this information from public inspection should change in the future such that the information could then be made available for public inspection, you should promptly notify the NRC. You should also understand that the NRC may have cause to review this determination in the future, for example, if the scope of a Freedom of Information Act request includes your information. In all review situations, if the NRC needs additional information from you or makes a determination adverse to the above, you will be notified in advance of any public disclosure.

If you have any questions regarding this matter, I may be reached at 301-415-1364.

Sincerely,

/RA/

Douglas V. Pickett, Senior Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-346

cc: See next page

(6)e The information is vital to a competitive advantage held by FRA-ANP, would be helpful to competitors to FRA-ANP, and would likely cause substantial harm to the competitive position of FRA-ANP.

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