Westinghouse Electric Corporation

Water Reactor Divisions

Nuclear Technology Division

Box 355
Pittsburgh Pennsylvania 15230

March 31, 1981
AW-81-17

Mr. James R. Miller, Chief
Standardization and Special Projects Branch
Division of Licensing
U. S. Nuclear Regulatory Commission
Phillips Building
7920 Norfolk Avenue
Bethesda, Maryland 20014

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

SUBJECT: Proprietary Responses to "Request Number 5 for Additional Information on WCAP-9500," NRC letter from R. L. Tedesco to T. M. Anderson, March 24, 1981

REF: Westinghouse Letter No. NS-TMA-2409, Anderson to Miller, dated March 31, 1981

Dear Mr. Miller:

The proprietary material transmitted by the referenced letter supplements the proprietary material previously submitted concerning a request for additional information on WCAP-9500. Further, the affidavit submitted to justify the material previously submitted, AW-78-23, is equally applicable to this material.

Accordingly, withholding the subject information from public disclosure is requested in accordance with the previously submitted affidavit and application for withholding, AW-78-23, dated March 21, 1978, a copy of which is attached.

Correspondence with respect to this application for withholding or the accompanying affidavit should reference AW-81-17, and should be addressed to the undersigned.

Very truly yours,

Robert A. Wiesemann

Robert A. Wiesemann, Manager
Regulatory & Legislative Affairs

/bek
Attachment

cc: E. C. Shomaker, Esq.
Office of the Executive Legal Director, NRC

8104030538.

AFFIDAVIT

POOR ORIGINAL

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Robert A. Wiesemann, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Corporation ("Westinghouse") and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

Robert A. Wiesemann

Robert A. Wiesemann, Manager
Licensing Programs

Sworn to and subscribed
before me this 10 day
of March 1978.

Notary Public

POOR ORIGINAL

AW-78-23

(1) I am Manager, Licensing Programs, in the Pressurized Water Reactor Systems Division, of Westinghouse Electric Corporation and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing or rulemaking proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Water Reactor Divisions.

(2) I am making this affidavit in conformance with the provisions of 10 CFR Section 2.790 of the Commission's regulations and in conjunction with the Westinghouse application for withholding accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse Nuclear Energy Systems in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and

POOR ORIGINAL

whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Criteria and Standards Utilized

In determining whether information in a document or report is proprietary, the following criteria and standards are utilized by Westinghouse. Information is proprietary if any one of the following are met:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

POOR ORIGINAL

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(g) It is not the property of Westinghouse, but must be treated as proprietary by Westinghouse according to agreements with the owner.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.790, it is to be received in confidence by the Commission.

(iv) The information is not available in public sources to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal are the copies of slides utilized by Westinghouse in its presentation to the NRC at the March 21, 1978 meeting concerning the Westinghouse optimized fuel assembly. The letter and the copies of slides are being submitted in preliminary form to the Commission for review and comment on the Westinghouse optimized fuel assembly in advance of a formal submittal for NRC approval.

Public disclosure of this information is likely to cause substantial harm to the competitive position of Westinghouse as it would reveal the description of the approved design, the comparison of the improved design with the standard design, the nature of the tests conducted, the test conditions, the test results and the conclusions of the testing program, all of which is recognized by the Staff to be of competitive value and because of the large amount of effort and money expended by Westinghouse over a period of several years in carrying out this particular development program. Further, it would enable competitors to use the information for commercial purposes and also to meet NRC requirements for licensing documentation, each without purchasing the right from Westinghouse to use the information.

Information regarding its development programs is valuable to Westinghouse because:

(a) Information resulting from its development programs gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information which is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

POOR ORIGINAL

AW-78-23

(e) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

Being an innovative concept, this information might not be discovered by the competitors of Westinghouse independently. To duplicate this information, competitors would first have to be similarly inspired and would then have to expend an effort similar to that of Westinghouse to develop the design.

Further the deponent sayeth not.

REQUEST NUMBER 5 FOR ADDITIONAL INFORMATION ON WCAP-9500

212.1. Pages 15-6 thru 15-7 of the Regulatory Guide 1.70 (Rev. 3) requests specific parameters which should be provided in FSARs. These parameters are necessary for the reviewer to fully understand the behavior of the plant under review and perform independent analyses, should that be judged necessary. Based on the above needs, provide, where missing, time dependent graphical computer results of the following relevant parameters for each Section 15 transient:

(1) neutron power

(2) thermal power

(3) heat flux, average and maximum

(4) reactor coolant system pressure

(5) Departure from nuclear boiling ratio (DNBR)

(6) coolant conditions - core inlet temperature; core average temperature; average and hot channel exit temperatures, including void fractions

(7) Maximum fuel centerline temperature and maximum clad temperature

(8) Total reactor coolant inventory and mixture level in various locations in the system (i.e., loop seals, core, pressurizer, and steam generator secondary sides)

(9) Secondary system parameters - steam flow rate; steam pressure and temperature; feedwater flow rate and temperature; emergency feedwater flow rate and temperature; steam generator inventory

(10) Emergency core cooling system flow rates and pressure differentials across the core, as applicable;

(11) Containment pressure

(12) relief and/or safety valve flow rates.

(13) Average and hot pin initial centerline temperature as a function of fuel pin elevation.

For those parameters remaining constant, graphical output is not required.

RESPONSE TO QUESTION 212.1

Regulatory Guide 1.70, Rev. 3, recommends that all of the time dependent parameters be given. The list of parameters given in the question is stated as a list of examples of the type of information considered to be relevant. For any given transient, many of the parameters listed in the table are not relevant to, not included in, or unnecessary for the analysis or results of that particular transient. Westinghouse has identified and organized relevant key parameters and results for each transient. Reporting of this data in Section 15 of WCAP-9500 meets the information requirements of R.G. 1.70 (Rev. 3) and provides sufficient relevant information to demonstrate safety of the plant with respect to each analyzed transient. However, the additional information required for each limiting steam line, feed line and loss-of-coolant break will be supplied by a separate cover as soon as possible.

212.2 The SRP recommends, in many cases, that the initial power for accident calculations be 102 percent of rated power. It is our understanding that your Improved Thermal Design Procedure does not require the assumption of 102 percent of power. Provide detailed justification for this position.

RESPONSE TO QUESTION 212.2

In WCAP-8568 (NRC approved) the improved thermal design procedure is detailed. In Section 5 of this report it explains how the additional 2% allowance for power has already been convoluted with other errors to derive the target value DNBR. Therefore, since the 2% has already been accounted for, it is not necessary that the initial power for accident calculations be 102% of rated power.

212.3 Section 15.2.3 does not address the assumption of the most reactive rod being in the stuck (withdrawn) position. Was the limiting rod assumed stuck and were the peaking factors correspondingly adjusted in the analyses? If not, provide analyses accounting for the stuck rod and the power distribution (including the peaking factors) for the .ot pin.

RESPONSE TO QUESTION 212.3

As discussed in Section 15.0.6, a total negative reactivity insert:on following a reactor trip of 4% ΔK is assumed in the transient analyses, unless stated otherwise. This trip worth is based on the conservative assumption that the highest worth RCCA is stuck in its fully withdrawn position. Some transients may assume some other trip worth, but the allowance for a stuck rod is made for every event in which a reactor trip is assumed.

In the determination of shutdown margin, a stuck rod is assumed for all ev:nts.

212.4 A significant difference between the optimized and the standard 17 x 17 fuel assemblies is that the optimized assemblies have smaller fuel rod diameters. This results in increased heat flux, decreased fuel rod heat capacity, and decreased coolant velocity. These effects could reduce the safety margins for postulated accidents and transients. Provide quantitative and qualitative differences in fuel behavior resulting from a limiting accident and transient for the standard and optimized 17 x 17 fuel assemblies (why does the optimized 17 x 17 fuel assembly result in higher peak clad temperature?).

RESPONSE TO QUESTION 212.4

The improved (optimized) 17x17 fuel is optimized from a fuel cost aspect only. Both the fuel pellet and clad outer diameters are smaller than the 17 x 17 standard fuel. This results in smaller fuel volume, less surface heat transfer area and larger core flow area plus a larger core fluid volume. By design the optimized fuel operates at the same average fuel temperature as the standard 17x17 fuel. This leads to the following fuel characteristics which effect LOCA.

1) Stored Energy - Less for the 17x17 optimized design and thus is a LOCA benefit.

2) Volumetric heat generation rate - Greater for the 17x17 optimized design as is a LOCA penalty.

3) Heat Transfer Area - Less for 17x17 optimized fuel and is therefore a LOCA penalty.

4) Core Flow Area and Core Fluid Volume - Larger for 17x17 optimized and is a LOCA penalty.

5) Hydraulic Diameter - Larger for 17x17 optimized fuel and is a LOCA penalty for single phase force convection heat transfer coefficients.

6) FLECHT Heat Transfer - Smaller 17x17 optimized fuel O.D. results in improved FLECHT heat transfer coefficient which is a LOCA benefit.

The peak clad temperature impact of these LOCA parameters has not been determined on an individual basis but the overall peak clad temperature (PCT) impact from the combination of items 1 through 6 results in a PCT penalty of between [ ].+ Items 1 through 6 will now be discussed in more detail. +(a,c

Stored Energy

Stored energy has only a small effect on PCT since PCT occurs during the reflood period, which is later in time, and long after the stored energy was removed during blowdown. Thus stored energy is a negligible benefit for 17x17 optimized fuel.

Volumetric Heat Generation Rate

The larger volumetric heat generation rate results in a larger adiabatic heat up rate resulting in a penalty during the refill part of the transient when only radiation heat transfer is taken into account.

Surface Heat Transfer Area

The smaller 17x17 optimized clad outer diameter leads to a smaller surface area for heat transfer and thus higher clad temperatures are needed to remove the decay heat.

Core Flow Area and Fluid Volume

The smaller optimized fuel rod O.D. leads to larger core flow areas and fluid volumes. This has the largest impact on core reflood velocities. The larger flow area and volume reduces the inlet flooding velocity and this reduces the FLECHT heat transfer coefficients.

Hydraulic Diameter

The larger hydraulic diameter will reduce forced convection heat transfer coefficients, for the same mass flux, by about [ ].+ This is another penalty for 17x17 optimized fuel. +(a,

FLECHT Heat Transfer Coefficients

FLECHT data shows a benefit for smaller rod diameters. The FLECHT heat transfer benefit for optimized fuel when compared to standard fuel is about [ ]+ based on approved methods for extrapolating the FLECHT heat transfer correlation. +(a,c) This results in a PCT benefit for the non-UHI analysis given in WCAP-9500 of about [ ].+ +(a,c)

212.5 As requested in Regulatory Guide 1.70, provide the following parameters in Table 15.0-3: hot channel coolant exit temperature, maximum centerline fuel temperature, reactor coolant system inventory, feedwater flow rate, CVCS flow and boron concentration, and control rod worth.

RESPONSE TO QUESTION 212.5

Table 15.0-3 is to be used for an overview of initial input conditions and not meant to be all inclusive, Each Chapter 15 transient list relevant input parameters in the form of accident assumptions. An all inclusive list of initial input parameters contains items which are not relevant to only specific event. Therefore these assumptions are handled on a case by case basis for all Chapter 15 analyses. These relevant assumptions/inputs are provided in the text.

212.6 Section 16, page 2-1, states: "The following core design criteria shall be met during normal plant operation: a. DNBR shall be greater than or equal to 1.17." This technical specification permits plant operation at conditions for which a DNBR of 1.17 could exit. However, all Chapter 15 transient analyses assumed a minimum DNBR of at least 1.8. Please explain this apparent discrepancy since the safety analyses are inconsistent with the stated operation.

RESPONSE TO QUESTION 212.6

Safety Limits, as defined by 10CFR 50.36 (c) (1) (i) (A) are, ". . . limits upon important process variables which are found to be necessary to reasonably protect the integrity of certain of the physical barriers which guard against the uncontrolled release of radioactivity. If any safety limit is exceeded, the reactor shall be shutdown. The licensee shall notify the commission, review the matter and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude reoccurrence. Operation shall not be resumed until authorized by the Commission. "In compliance with this requirement Westinghouse has historically provided (and the NRC historically approved) limits on two basic parameters, DNBR and RCS pressure. As noted in NUREG-0452 Rev. 3, for typical Westinghouse Designs, the limits to prevent DNB are provided as figures and as noted in the bases for this requirement operation within these limits precludes DNBR values of less than 1.30. WCAP-9500 section 16.2.1.1 is intended to satisfy the same requirements by providing limits on equivalent parameters. To further clarify this intent the wording to 16.2.1.1 has been modified as follows: "The following core design criteria shall be met during (n) loop operation: "where (n) loop operation is defined as 4 loop operation for a four loop plant, 3 loop operation for a three loop plant, etc. It is believed that this change removes the ambiguity noted in the question and the additional information requested is no longer necessary.

212.7 Loss of power to the reactor coolant system (RCS) pumps can occur during the course of an accident. The standard review plan for Chapter 15 accidents and transients typically requires the assessment of plant responses to transients and accidents with and without offsite power available. We interpret this to include losing offsite power at any time during the transient. For each of the transients analyzed in Chapter 15, justify that losing offsite power during the course of the transient will not provide more limiting consequences than those already analyzed.

RESPONSE TO QUESTION 212.7

The Standard Review Plant specifically has chosen several transients which must be analyzed with and without offsite power available. Two examples of these specifically noted transients are the steamline break and the feedline break. Other transients such as complete Loss of Flow and Station Blackout also include effect of Loss of Power to the reactor coolant pumps. Loss of Offsite Power is considered in Section 15 for all transients where specifically required. This is responsive to the SRP and acknowledges that assumed Loss of Offsite Power is not required for transients for which the consequences are less limiting than those discussed above.

212.8 Table 15.1-2 lists a series of equipment required to function in the event of a steam line break. Verify that every equipment required to operate is safety grade or that its failure would not result in unacceptable consequences.

RESPONSE TO QUESTION 212.8

Of the pieces of equipment or systems noted in Table 15.1-2 all are required to be safety grade with the exception of the service water and component cooling water systems. While these two systems are not specifically noted as being safety grade there is sufficient redundancy in the systems to assure proper operation as required, i.e., both systems are set up such that there are at least three 50% of full design load capability pumps and heat exchangers for each system with the pumps supplied by safety grade power supplies (loaded on emergency diesels if necessary). In the event of a rupture of a main steamline and assuming the plant is not on RHR, one component cooling water pump and heat exchanger and one service water pump are sufficient to carry the heat loads generated by operating safety grade equipment.

212.9 Section 15.4.8.3 addresses the radiological consequence of a postulated rod ejection accident. In this section is is stated that the fraction of fuel melting was conservatively assessed by assuming a conservative upper limit of 50 percent of the rods experiencing cladding damage may experience centerline melting. Justify and quantify the conservatism of this assumption.

RESPONSE TO QUESTION 212.9

For the radiological consequences of a postulated rod ejection accident (Section 15.4.8.3), it is stated that the fraction of fuel melting was conservatively assessed by assuming a conservative upper limit of 50 percent of the rods experiencing cladding damage may experience centerline melting. Justify and quantify the conservatism of this assumption.

A complete statement of the assumption is that less then 50% of the fuel rods experiencing cladding damage (perforation) due to overheating (caused by DNB) experiences centerline fuel melting. This is a result of the inherent rapid fall-off of the peaking factor in fuel rods in assemblies away from the ejected RCCA location, and the short length of time that the fuel rods are in DNB. The most limiting conditions typically occur in the BOL HFP case, which generally results in maximum fuel centerline melting and has a flatter power distribution than the zero power cases. A detailed analysis of the BOL HFP case has been performed for a similar plant, assuming a peaking factor which resulted in exactly [ ]+ fuel centerline melting at the hot spot. Taking +(a,c
into account the power distribution in the surrounding fuel rods, it was found that [ ]+ of the fuel rods in the core reached centerline +(a,c
melt at the axial hot spot. This substantiates the conservatism of the assumption that no more than 5% of the fuel rods experience melting.

212.10 Describe how the systems characteristics for Plant A and B were derived. Why are the characteristics assumed for Plant B often less conservative than those used for Plant A? Compare these characteristics to actual plant data.

RESPONSE TO QUESTION 212.10

Plant "B" (blue pages) systems characteristics were derived from an actual plant which is going to use optimized fuel. Plant "A" (white pages) systems characteristics were chosen to be similar to a proposed plant but in most cases a bounding value for systems data was chosen. This conservative approach for plant "A" was taken since it incorporates the new protection system and it was felt that bounding values should be chosen to prove the adequacy of this system.

212.11 Verify our understanding that WCAP-9500 is not intended to result in a generic approval for N-1 loop operation.

RESPONSE TO QUESTION 212.11

WCAP-9500 is not intended to be an N-1 loop topical. It is not meant to replace or enhance an existing N-1 loop topical.

212.12 Why is surveillance of the Pressurizer Pressure and the Low Reactor Trip System Instrumentation not required duirng Mode 2 (as specified in NUREG-0452)?

RESPONSE TO QUESTION 212.12

Surveillance of Pressurizer Pressure - Low, Pressurizer Water Level - High, and Reactor Coolant Pump Underspeed Reactor Trips is not required during operation in Mode 2 because these trip functions are blocked below P-10 ($\sim$10% of Rated Thermal Power). The analyses performed assuming reactor power below P-10 also assume that these trip functions are inoperable.

212.13 Section 15.3, page 19: Describe the relationship amoung hydrogen bubble size, amount of clad reaction, and the reactor coolant system pressure.

RESPONSE TO QUESTION 212.13

Using the conservative assumptions in Section 15.3.3 the H_2 released in calculated to be less than [ ].+ +(a,c)

Even if all of this non-condensable gas were to "migrate" and collect at the top of the U-tubes (volume of this is ~ 70 ft^3/SG) this would not be enough to disrupt core cooling. This is substantiated by the natural circulation work done in [ ].+ +(a,c)

212.14 Verify our understanding that all the computer codes utilized in Section 15, with the exception of the LOCTA code, are identical to those used for the standard fuel designs.

RESPONSE TO QUESTION 212.14

In WCAP-9500, Westinghouse has not employed the use of any new codes not previously seen by the NRC. Listed below are several examples of "not yet" approved codes which have received sufficient NRC review to warrant their use.

1. WCAP-7907, "LOFTRAN Code Description"
2. WCAP-7908, "FRACTRAN"

The following extract from McGuire 1/2 SER, ". . . our reviews (of FRACTRAN. . .and LOFTRAN) have progressed to the point that there is reasonable assurance that results of analyses dependent on the codes will not be appreciably altered by any change in method that may be required by the staff".

212.15 Why aren't the N-16 trip channels recalibrated immediately after removal of leaky fuel? (Such removal causes the N-16 signal to decrease.) - Chapter 16.3/4.3.

RESPONSE TO QUESTION 212.15

It is Westinghouse's intent to recalibrate the N-16 Nuclear Power High kw/ft and N-16 Nuclear Power Low DNBR anytime fuel is removed and new fuel inserted. This cannot be accomplished without placing the plant in the Refueling mode. It is believed that current technical specification requirements and administrative controls provide sufficient assurance that trip functions will be properly calibrated. It should be noted that the technical specifications also require a daily channel check against a power calorimetric which also provides assurance that the N-16 detectors are in calibration.

212.16 The technical specifications in Chapter 16 (page 3/4 3-19) states that the delay time for ECCS injection may be as high as 27 seconds from time of of ECC signal initiation (assuming loss-of-offsite power). However, chapter 15 ECCS analyses assumed as 25 second delay in ECCS initiation. Either reduce the technical specifications to be less than 25 seconds or reanalyze the chapter 15 accidents utilizing an ECCS delay time equal to or greater than 27 seconds.

RESPONSE TO QUESTION 212.16

The SI total response time of 27 seconds noted in Table 3.3-4 is based on sequential loading of High Head SI/Charging pumps, Intermediate Head SI pumps, and Low Head SI/RHR pumps on the diesels. In this loading sequence the High Head SI/Charging pumps are up to speed and rated discharge pressure providing flow before the Intermediate Head SI pumps are loaded on the diesels. the Intermediate Head SI pumps are up to speed and discharge pressure before the Low Head SI/RHR pumps are up to speed and discharge pressure by 27 seconds after initiation at the sensor. This loading sequence results in small step increases in flow, i.e., the High Head SI/Charging pumps ramp up to speed and are providing flow, then the Intermediate Head SI pumps ramp to speed, and finally the Low Head/RHR ramp up to speed. Thus flow is being provided by the High Head SI/Charging pumps 17 seconds into the event.

The LOCA analyses assume no flow until 25 seconds into the event when full flow is assumed from one train of pumps. This assumption is conservative with respect to the actual ECCS flow delivered as a result of the diesel loading sequence.

212.17 Section 16, page 2-12, states that the feedwater isolation system is activated by coincident signals from high feedwater flow and a low cold leg temperature (Low-1). The Low-1 cold leg temperature set point is only 2.5 ^{O}F lower than nominal. Accounting for practical limitations of instrumentation and control equipment, we believe that the Low-1 level would result in unnecessary feedwater isolation. Justify why this set-point would not result in excessive trips or revise the Low-1 temperature set point to a more-realistic value.

RESPONSE TO QUESTION 212.17

The Low-1 T-cold setpoint will be changed to a nominal setpoint of $\geq$ 550^{O}F and an allowable value of $\geq$ 547^{O}F. This results in the nominal setpoint being 7^{O}F below the no load T-cold value of 557^{O}F, which is adequate to allow for instrumentation and controller errors and in the approximate temperature range of Tavg-Low-Low for a non-Integrated Protection System plant.

212.18 Verify that the W-3 DNBR correlation was utilized for all LOCA analyses and that the WRB-1 DNBR correlation was utilized for all non-LOCA transients.

RESPONSE TO QUESTION 212.18

The DNBR correlations utilized in the non-LOCA analyses are verified in column No. 6 of Table 15.0-3 of WCAP-9500. This indicates that the W-3 correlation was utilized for analysis of the accidental depressurization of the main steam system and for the steam system piping failure. The WRB-1 correlation was utilized for all other applicable non-LOCA transient analyses. The WRB-1 correlation was not utilized for LOCA analyses. The DNBR correlations utilized for LOCA analyses are stated in the NRC reviewed and approved Westinghouse Appendix K methodology reports, referenced in WCAP-9500.