Westinghouse Electric Corporation

Energy Systems

Box 355
Pittsburgh Pennsylvania 15230-0355

July 30, 1991
CAW-91-192

Document Control Desk
US Nuclear Regulatory Commission
Washington, DC 20555

Attention: Dr. Thomas Murley, Director

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: Westinghous ECCS Model for Analysis of CE-NSSS

Dear Dr. Murley:

The proprietary information for which withholding is being requested in the enclosed letter by Omaha Public Power District is further identified in Affidavit CAW-91-192 signed by the owner of the proprietary information, Westinghouse Electric Corporation. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.790 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying Affidavit by Omaha Public Power District.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-91-192, and should be addressed to the undersigned.

Very truly yours,

N J Hogarlo for

R. P. DiPiazza, Manager
Operating Plant Licensing Support

Enclosures

cc: M. P. Siemien, Esq.
Office of the General Counsel, NRC

Copyright Notice

Proprietary Information Notice

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.790 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (g) contained within parentheses located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(g) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.790(b)(1).

CAW-91-192

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

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COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared Robert W. Beer, 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 W. Beer

Robert W. Beer, Manager
Operations Engineering Technology

Sworn to and subscribed before me this 30th day of July, 1991.

Lorraine M. Piplica

Notary Public

NOTARIAL SEAL
LORRAINE M. PIPLICA, NOTARY PUBLIC
MONROEVILLE BORO, ALLEGHENY COUNTY
MY COMMISSION EXPIRES DEC. 14, 1991

Member, Pennsylvania Association of Notaries

(1) I am Manager, Operations Engineering Technology, in the Nuclear and Advanced Technology Division, of the Westinghouse Electric Corporation and as such, I am authorized to perform, on the behalf of Ronald P. DiPiazza, the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking proceedings, and am authorized to apply for its withholding on behalf of the Westinghouse Energy Systems Business Unit.

(2) I am making this Affidavit in conformance with the provisions of 10CFR 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 the Westinghouse Energy Systems Business Unit 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 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.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(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 a similar product.

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

(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.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse 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.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

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

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

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in "Westinghouse ECCS Evaluation Model For Analysis of CE-NSSS", WCAP-13027, (Proprietary), July 1991, for Fort Calhoun Station Unit 1, being transmitted by the Omaha Public Power District Company (OPPD) letter and Application for Withholding Proprietary Information from Public Disclosure, Mr. W. G. Gates, OPPD, to Document Control Desk. The proprietary information as submitted for use by Omaha Public Power District for the Fort Calhoun Station Unit 1 is expected to be applicable in other licensee submittals in response to certain NRC requirements for justification of Westinghouse evaluation models for Combustion Engineering NSSS.

This information is part of that which will enable Westinghouse to:

(a) Justify the application of the Westinghouse evaluation model to CE NSSS.

(b) Provide analysis methodology to perform large and small break LOCA analyses or CE NSSS.

(c) Assist the customer in obtain a licensee.NRC approval.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for purposes of satisfying NRC requirements for licensing documentation.

(b) Westinghouse can sell support and defense of this methodology to its customers in the licensing process.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar analytical methodologies and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended for the development of analytical methods and testing.

Further the deponent sayeth not.

FORT CALHOUN UNIT 1
CONTROL ELEMENT ASSEMBLY
EJECTION ANALYSIS REPORT

CONTROL ELEMENT ASSEMBLY EJECTION ACCIDENT

General

The CEA ejection accident is defined as the mechanical failure in the form of a complete circumferential rupture of a CEDM housing or nozzle on the reactor vessel head resulting in the ejection of a control rod. The consequence of this mechanical failure is a rapid reactivity insertion which when combined with an adverse power distribution may result in localized fuel damage.

In design and fabrication, the CEDM is considered to be an extension of the reactor coolant system boundary; hence the probability of such a failure is equivalent to any other rupture of the reactor coolant system and is considered highly unlikely. Further, even if the CEA nozzle should separate from the reactor vessel head, its potential vertical upward travel is limited by the missile shield blocks placed over the reactor head and drive mechanisms. The missile shield block placement will allow an upward movement of only 18 inches; therefore, an additional failure in the drive train must be postulated for the continued CEA ejection. In addition, if the ejection continues, it will do so at a substantially lower rate.

In the following analysis, it is assumed that a CEA is ejected instantaneously from the core, although no mechanism for such an event has been identified. The analytical results presented in this section deal with the nuclear portion of the transient, which is terminated within several seconds.

The analysis was performed for hot zero power and hot full power initial conditions assuming the most adverse initial CEA configurations which are determined from the Technical Specification on power dependent insertion limits (PDIL). Dual CEAs are not considered, because the PDIL prohibits their insertion when critical. At zero power Groups 1 and 2 must be totally withdrawn and Group 3 at least 20% withdrawn. At full power all Groups except Group 4 must be withdrawn, and the Group 4 insertion is limited to 75% withdrawn (see Figure 2-4 of Technical Specifications).

If the reactor is subcritical, Technical Specifications require all shutdown CEA's to be withdrawn before any regulating CEA's are withdrawn and all regulating CEA's to be inserted before any shutdown CEA's can be inserted. These specifications require that during shutdown dissolved boron concentration must be maintained such that all shutdown CEA's and Groups 1 and 2 regulating CEA's must be fully withdrawn and Group 3 regulating CEA's must be at least 20% withdrawn in order to achieve criticality. Ejection of any one dual CEA when the reactor is subcritical under the above conditions cannot result in criticality because the worth of any one dual CEA is less than the combined worth of all shutdown and regulating CEA's.

Following the rapid ejection of a CEA, either from full power or zero power (critical) initial conditions, the core power rises rapidly for a brief period until the increasing reactivity loss due to the widening absorption resonances (Doppler effect) in U-238 terminates and reversesthe increasing power transient. Increasing power will initiate a variable high power trip at 19% for the zero power case and a high power trip for the full power case, causing the CEA banks to insert which reduces the neutron power to negligible levels.

The loss of coolant resulting from the circumferential rupture of a CEDM housing or nozzle, and its consequences are bounded by the scope of the small break loss of coolant accident which is discussed in USAR Section 14.15.

Method of Analysis

The analysis of the CEA ejection accident is performed in two stages: (a) an average core nuclear power transient calculation and (b) a hot spot heat transfer calculation. The average core calculation is performed using spatial neutron kinetics methods to determine the average power generation with time including the various total core feedback effects, i.e., Doppler reactivity and moderator reactivity. Enthalpy and temperature transients in the hot spot are then determined by multiplying the average core energy generation by the hot channel factor and performing a fuel rod transient heat transfer calculation. The power distribution calculated without feedback is conservatively assumed to exist throughout the transient. A detailed discussion of the method of analysis can be found in Reference 1.

The spatial kinetics computer code, TWINKLE (Reference 2), is used for the average core transient analysis. This code solves the two group neutron diffusion theory kinetic equations in one, two, or three spatial dimensions (rectangular coordinates) for six delayed neutron groups and up to 2000 spatial points. The computer code includes a detailed multiregion, transient fuel-clad-coolant heat transfer model for calculating pointwise Doppler, and moderator feedback effects.

In this analysis, the code is used as a one-dimensional axial kinetics code since it allows a more realistic representation of the spatial effects of axial moderator feedback and CEA movement. However, since the radial dimension is missing, it is still necessary to employ very conservative methods (described below) of calculating the ejected rod worth and hot channel factor.

The average core energy addition, calculated as described above, is multiplied by the appropriate hot channel factors, and the hot spot analysis is performed using the detailed fuel and cladding transient heat transfer computer code, FACTRAN (Reference 3). This computer code calculates the transient temperature distribution in a cross section of a metal clad UO_2 fuel rod, and the heat flux at the surface of the rod, using as input the nuclear power versus time and the local coolant conditions. The zirconium-water reaction is explicitly represented, and all material properties are represented as functions of temperature. A conservative parabolic radial pellet power generation is used within the fuel rod.

FACTRAN uses the Dittus-Boelter (Reference 4) or Jens-Lottes (Reference 5) correlation to determine the film heat transfer before DNB, and the Bishop-Sandberg-Tong correlation (Reference 6) to determine the film boiling coefficient after DNB. The DNB heat flux is not calculated; instead the code is forced into DNB by specifying a conservative DNB heat flux. The gap heat transfer coefficient can be calculated by the code; however, it is adjusted in order to force the full power steady state pellet temperature distribution to agree with that predicted by design fuel heat transfer codes.

For full power cases, the design initial hot channel peaking factor is input to the code. The hot channel factor during the transient is assumed to increase linearly from the initial steady state design value to the maximum transient value in 0.05 seconds, and remain at the maximum for the duration of the transient. The values for ejected rod worths and peaking factors are calculated using multi-dimensional calculations. No credit is taken for the flux-flattening effects of reactivity feedback. This is conservative, since detailed spatial kinetics models show that the hot channel factor decreases shortly after the nuclear power peak due to power flattening caused by preferential feedback in the hot channel. Appropriate margins are added to the results to allow for calculational uncertainties.

Results

The magnitude of fuel failure can be determined by the following limits:

(1) The average fuel pellet deposited energy at the hot spot is no greater than 200 cal/gram (clad damage threshold).

(2) The centerline enthalpy threshold for incipient melting is no greater than 250 cal/gram.

(3) The centerline enthalpy threshold for the fully molten condition is no greater than 310 cal/gram.

The criterion for determining the fraction of fuel rods that will release their radioactive fission products during the CEA ejection is the same as item (1) above for determining clad damage. Thus, it is assumed that any fuel rod that exceeds a total average enthalpy of 200 cal/gram releases all of its gap activity. The gap activity corresponding to the most limiting fuel rod during the cycle is conservatively assumed for each rod that suffers clad damage.

Table 1 lists the significant input variables for the limiting analyses at full power and zero power. All of the ejected CEA worths and radial peaking factors include appropriate allowances for calculation uncertainties. In all cases analyzed, a conservative value of 0.05 seconds was assumed for the total ejection time. For the full power and zero power cases, a Variable Overpower trip is conservatively assumed to initiate at 112% and 29.1% (19.1% + 10% uncertainty) of full power, respectively. The initial conditions assume the core was operating at 102% of full power for the full power cases while an initial power of 10^{-9} of nominal was assumed for the zero power case.

TABLE 1
CEA EJECTION ACCIDENT ASSUMPTIONS

Parameter	Units	Analysis Value
Full Power		
Moderator Temperature Coefficient	$10^{-4}\Delta\rho/^{0}F$	+0.5
Doppler Defect	$\%\Delta\rho$	-1.25
Ejected CEA worth	$\%\Delta\rho$	0.36
Delayed Neutron Fraction, b		0.0061
Pre-ejected Rod Hot Spot Peaking Factor		2.52
Post-ejected Rod Hot Spot Peaking Factor		6.93
CEA Worth at Trip	$\%\Delta\rho$	4.2
Zero Power		
Nominal Core Power Fraction		10^{-9}
Ejected CEA worth	$\%\Delta\rho$	0.69
Delayed Neutron Fraction, b		0.0061
Post-ejected Rod Hot Spot Power		10.51
CEA Worth at Trip	$\%\Delta\rho$	1.5

The results of the full and zero power CEA ejection events may be found in Table 2. This analysis was assessed against the Regulatory Guide 1.77 criteria (Reference 7) which limits the average hot pellet enthalpy to less than 280 cal/gram. The previous acceptance criteria of 200 cal/gram is more conservative with respect to the Regulatory Guide '.mit. The centerline melt criterion was not assessed in this analysis since the Regulatory Guide does not require it.

TABLE 2
CEA EJECTION ACCIDENT RESULTS

Parameter	Analysis Value
Full Power	
Total Average Enthalpy of Hottest Fuel Pellet (cal/gram)	182.1
Total Centerline Enthalpy of Hottest Fuel Pellet (cal/gram)	286.6
Fraction of Rods That Suffer Clad Damage (Average Enthalpy ≥ 200 cal/gram)	0.0
Fraction of Pellet at Hot Spot Having at Least Incipient Centerline Melting (Centerline Enthalpy ≥ 250 cal/gram)	0.09
Fraction of Fuel Having a Fully Molten Centerline Condition (Centerline Enthalpy ≥ 310 cal/gram)	0.0
Zero Power	
Total Average Enthalpy of Hottest Fuel Pellet (cal/gram)	60.6
Total Centerline Enthalpy of Hottest Fuel Pellet (cal/gram)	71.8
Fraction of Rods That Suffer Clad Damage (Average Enthalpy ≥ 200 cal/gram)	0.0
Fraction of Pellet at Hot Spot Having at Least Incipient Centerline Melting (Centerline Enthalpy ≥ 250 cal/gram)	0.0
Fraction of Fuel Having a Fully Molten Centerline Condition (Centerline Enthalpy ≥ 310 cal/gram)	0.0

Radiological Consequences

The analysis of radiological consequences of a CEA ejection accident considers the release of secondary coolant activity as well as the reactor coolant activity released through the ruptured CEDM housing. The major assumptions used in the analysis are:

1. CEA ejection occurs while the reactor is operating at 102% of 1500 MWt with 1% failed fuel and a 1.0 gpm primary-to-secondary leak.

2. The steam generator equilibrium activity for both steam generators is assumed to be 0.1 iCi/gm DEC I-131.

3. Offsite power is lost; the main condenser is not available for steam relief via the turbine bypass system.

4. The activity available for leakage from containment is based on the equilibrium reactor coolant activity. The activity instantaneously available for release from the containment is 100% of the noble gases and 25% of the halogens.

5. The containment leakage rate is assumed to be 0.2 volume percent per day for the first 24 hours and 0.1 volume percent per day for the duration of the accident (1-30 days).

6. A post-accident decontamination factor of 10 was used in the steam generator between the water and steam phases.

7. The total activity released from the secondary system is presented in Table 3.

TABLE 3
ACTIVITIES RELEASED FROM THE SECONDARY SYSTEM

Nuclide	Activity (Ci)
Kr-83m	5.0 E-02
Kr-85m	2.6 E-01
Kr-85	4.4 E+01
Kr-87	1.4 E-01
Kr-88	4.8 E-01
Xe-131m	3.6 E-01
Xe-133m	5.5 E-01
Xe-133	5.0 E+01
Xe-135m	3.1 E-02
Xe-135	8.5 E-01
Xe-138	1.1 E-01
I-131	2.9 E+00
I-132	2.2 E-01
I-133	1.3 E+00
I-134	3.7 E-02
I-135	4.9 E-01

8. The total activity released from the containment, 0-2 hours and for 0-30 days, is presented in Table 4.

TABLE 4
ACTIVITIES RELEASED FROM THE CONTAINMENT

Nuclide	Activity (Ci) 0-2 hrs	Activity (Ci) 0-30 days
Kr-83m	2.23 E-03	2.23 E-03
Kr-85m	1.40 E-02	1.46 E-02
Kr-85	2.66 E+00	4.57 E+02
Kr-87	5.37 E-03	5.37 E-03
Kr-88	2.36 E-02	2.37 E-02
Xe-131m	2.16 E-02	1.72 E+00
Xe-133m	3.46 E-02	5.34 E-01
Xe-133	3.15 E+00	1.26 E+02
Xe-135m	3.75 E-04	3.75 E-04
Xe-135	4.95 E-02	7.80 E-02
Xe-138	1.11 E-03	1.11 E-03
I-131	1.99 E-02	1.18 E+00
I-132	3.75 E-03	3.76 E-03
I-133	1.94 E-02	8.75 E-02
I-134	1.25 E-03	1.25 E-03
I-135	9.93 E-03	1.21 E-02

9. The dispersion factors for the EAB and the LPZ outer boundary are 2.55 E-04 sec/m^3 and 4.53 E-06 sec/m^3, respectively (Reference 8).

10. The adult breathing rate for the EAB and LPZ is assumed to be 3.47 E-04 m^3/sec.

Based on these assumptions, the results doses are as follows:

	Thyroid (Rems)	Whole Bod (Rems)
EAB	4.4 E-01	1.8 E-03
LPZ	9.7 E-03	1.7 E-04

Conclusions

The analysis of the CEA ejection accident shows that the energy increase at the hot spot is limited and that no fuel rods suffer any significant damage following a CEA ejection from full or zero power at beginning or end of cycle.

The results of radiological consequences of a CEA ejection accident are presented above. The calculated values for thyroid dose and whole body dose show that the doses based on conservative assumptions are well within the limits specified in 10CFR, Part 100.

References

1. D. H. Risher, Jr., An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods, WCAP-7588, Revision 1-A, January 1975.

2. D. H. Risher, Jr., and R. F. Barry, TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code, WCAP-7979-P-A (Proprietary), WCAP-8028-A (Non-Proprietary), January 1975.

3. H. G. Hargrove, FACTRAN - A FORTRAN IV Code for Thermal Transients in a UO2 Fuel Rod, WCAP-7908-A, December 1989.

4. F. W. Dittus and L. M. K. Boelter, University of California (Berkeley), Publs. Eng., 2, 433, 1930.

5. W. H. Jens and P. A. Lottes, Analysis of Heat Transfer, Burnout, Pressure Drop, and Density Data for High Pressure Water, USAEC Report ANL-4627, 1951.

6. A. A. Bishop, et al., "Forced Convection Heat Transfer at High Pressure After the Critical Heat Flux," ASME 65-HT-31, August 1965.

7. Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors", U. S. Nuclear Regulatory Commission, May, 1974.

8. Gebers, S., "Radiological Services, Atmospheric Dispersion: USAR Calculations", October 31, 1990.

9. OPPD Engineering Analysis EA-FC-91-001, "1% Failed Fuel" Rev. 0.

10. OPPD calculation PED-FC-91-1357, "Atmospheric Dispersion: USAR Calculations".

FORT CALHOUN UNIT 1
CONTROL ELEMENT ASSEMBLY
EJECTION ANALYSIS REPORT

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(3) The centerline enthalpy threshold for the fully molten condition is no greater than 310 cal/gram.

The criterion for determining the fraction of fuel rods that will release their radioactive fission products during the CEA ejection is the same as item (1) above for determining clad damage. Thus, it is assumed that any fuel rod that exceeds a total average enthalpy of 200 cal/gram releases all of its gap activity. The gap activity corresponding to the most limiting fuel rod during the cycle is conservatively assumed for each rod that suffers clad damage.

Table 1 lists the significant input variables for the limiting analyses at full power and zero power. All of the ejected CEA worths and radial peaking factors include appropriate allowances for calculation uncertainties. In all cases analyzed, a conservative value of 0.05 seconds was assumed for the total ejection time. For the full power and zero power cases, a Variable Overpower trip is conservatively assumed to initiate at 112% and 29.1% (19.1% + 10% uncertainty) of full power, respectively. The initial conditions assume the core was operating at 102% of full power for the full power cases while an initial power of 10^{-9} of nominal was assumed for the zero power case.

TABLE 1
CEA EJECTION ACCIDENT ASSUMPTIONS

Parameter	Units	Analysis Value
Full Power		
Moderator Temperature Coefficient	$10^{-4}\Delta\rho/^0F$	+0.5
Doppler Defect	$\%\Delta\rho$	-1.25
Ejected CEA worth	$\%\Delta\rho$	0.36
Delayed Neutron Fraction, b		0.0061
Pre-ejected Rod Hot Spot Peaking Factor		2.52
Post-ejected Rod Hot Spot Peaking Factor		6.93
CEA Worth at Trip	$\%\Delta\rho$	4.2
Zero Power		
Nominal Core Power Fraction		10^{-9}
Ejected CEA worth	$\%\Delta\rho$	0.69
Delayed Neutron Fraction, b		0.0061
Post-ejected Rod Hot Spot Power		10.51
CEA Worth at Trip	$\%\Delta\rho$	1.5

The results of the full and zero power CEA ejection events may be found in Table 2. This analysis was assessed against the Regulatory Guide 1.77 criteria (Reference 7) which limits the average hot pellet enthalpy to less than 280 cal/gram. The previous acceptance criteria of 200 cal/gram is more conservative with respect to the Regulatory Guide limit. The centerline melt criterion was not assessed in this analysis since the Regulatory Guide does not require it.

TABLE 2
CEA EJECTION ACCIDENT RESULTS

Parameter	Analysis Value
Full Power	
Total Average Enthalpy of Hottest Fuel Pellet (cal/gram)	182.1
Total Centerline Enthalpy of Hottest Fuel Pellet (cal/gram)	286.6
Fraction of Rods That Suffer Clad Damage (Average Enthalpy ≥ 200 cal/gram)	0.0
Fraction of Pellet at Hot Spot Having at Least Incipient Centerline Melting (Centerline Enthalpy ≥ 250 cal/gram)	0.09
Fraction of Fuel Having a Fully Molten Centerline Condition (Centerline Enthalpy ≥ 310 cal/gram)	0.0
Zero Power	
Total Average Enthalpy of Hottest Fuel Pellet (cal/gram)	60.6
Total Centerline Enthalpy of Hottest Fuel Pellet (cal/gram)	71.8
Fraction of Rods That Suffer Clad Damage (Average Enthalpy ≥ 200 cal/gram)	0.0
Fraction of Pellet at Hot Spot Having at Least Incipient Centerline Melting (Centerline Enthalpy ≥ 250 cal/gram)	0.0
Fraction of Fuel Having a Fully Molten Centerline Condition (Centerline Enthalpy ≥ 310 cal/gram)	0.0

Radiological Consequences

The analysis of radiological consequences of a CEA ejection accident considers the release of secondary coolant activity as well as the reactor coolant activity released through the ruptured CEDM housing. The major assumptions used in the analysis are:

1. CEA ejection occurs while the reactor is operating at 102% of 1500 MWt with 1% failed fuel and a 1.0 gpm primary-to-secondary leak.

TABLE 4
ACTIVITIES RELEASED FROM THE CONTAINMENT

Nuclide	Activity (Ci)	
	0-2 hrs	0-30 days
Kr-83m	2.23 E-03	2.23 E-03
Kr-85m	1.40 E-02	1.46 E-02
Kr-85	2.66 E+00	4.57 E+02
Kr-87	5.37 E-03	5.37 E-03
Kr-88	2.36 E-02	2.37 E-02
Xe-131m	2.16 E-02	1.72 E+00
Xe-133m	3.46 E-02	5.34 E-01
Xe-133	3.15 E+00	1.26 E+02
Xe-135m	3.75 E-04	3.75 E-04
Xe-135	4.95 E-02	7.80 E-02
Xe-138	1.11 E-03	1.11 E-03
I-131	1.99 E-02	1.18 E+00
I-132	3.75 E-03	3.76 E-03
I-133	1.94 E-02	8.75 E-02
I-134	1.25 E-03	1.25 E-03
I-135	9.93 E-03	1.21 E-02

9. The dispersion factors for the EAB and the LPZ outer boundary are 2.55 E-04 sec/m^3 and 4.53 E-06 sec/m^3, respectively (Reference 8).

10. The adult breathing rate for the EAB and LPZ is assumed to be 3.47 E-04 m^3/sec.

Based on these assumptions, the results doses are as follows:

	Thyroid (Rems)	Whole Bod (Rems)
EAB	4.4 E-01	1.8 E-03
LPZ	9.7 E-03	1.7 E-04

Conclusions

The analysis of the CEA ejection accident shows that the energy increase at the hot spot is limited and that no fuel rods suffer any significant damage following a CEA ejection from full or zero power at beginning or end of cycle.

The results of radiological consequences of a CEA ejection accident are presented above. The calculated values for thyroid dose and whole body dose show that the doses based on conservative assumptions are well within the limits specified in 10CFR, Part 100.

2. The steam generator equilibrium activity for both steam generators is assumed to be 0.1 μCi/gm DEC I-131.

3. Offsite power is lost; the main condenser is not available for steam relief via the turbine bypass system.

4. The activity available for leakage from containment is based on the equilibrium reactor coolant activity. The activity instantaneously available for release from the containment is 100% of the noble gases and 25% of the halogens.

5. The containment leakage rate is assumed to be 0.2 volume percent per day for the first 24 hours and 0.1 volume percent per day for the duration of the accident (1-30 days).

6. A post-accident decontamination factor of 10 was used in the steam generator between the water and steam phases.

7. The total activity released from the secondary system is presented in Table 3.

TABLE 3
ACTIVITIES RELEASED FROM THE SECONDARY SYSTEM

Nuclide	Activity (Ci)
Kr-83m	5.0 E-02
Kr-85m	2.6 E-01
Kr-85	4.4 E+01
Kr-87	1.4 E-01
Kr-88	4.8 E-01
Xe-131m	3.6 E-01
Xe-133m	5.5 E-01
Xe-133	5.0 E+01
Xe-135m	3.1 E-02
Xe-135	[illegible].5 E-01
Xe-138	[illegible].1 E-01
I-131	[illegible].9 E+00
I-132	[illegible].2 E-01
I-133	[illegible].3 E+00
I-134	3.7 E-02
I-135	4.9 E-01

8. The total activity released from the containment, 0-2 hours and for 0-30 days, is presented in Table 4.

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