PSE&G

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Nuclear Business Unit

FEB 2 4 2000

LR-N990511
LCR H99-12

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Gentlemen:

REQUEST FOR APPROVAL OF UNREVIEWED SAFETY QUESTION
RADIOLOGICAL CONSEQUENCES OF A CONTROL ROD DROP ACCIDENT
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354

In accordance with 10CFR50.90, Public Service Electric & Gas (PSE&G) Company hereby requests approval of an unreviewed safety question (USQ) for the Hope Creek Generating Station. In accordance with 10CFR50.91(b)(1), a copy of this submittal has been sent to the State of New Jersey.

Implementation of the request contained in this submittal will incorporate different assumptions and a revised radiological analysis for the control rod drop accident (CRDA) into the design and licensing basis for Hope Creek. The revised analysis results in increased dose consequences for the CRDA; however, the results remain well within the 10CFR100 guidelines and meet Standard Review Plan (SRP) Section 15.4.9, Appendix A, acceptance criteria. Additionally, the radiological consequences remain within the GDC 19 guidelines for control room personnel and plant operators and remain bounded by the loss of coolant accident analysis for on-site personnel.

The proposed changes have been evaluated in accordance with 10CFR50.91(a)(1), using the criteria in 10CFR50.92(c), and a determination has been made that this request involves no significant hazards considerations. The basis for the requested change is provided in Attachment 1 to this letter. A 10CFR50.92 evaluation, with a determination of no significant hazards consideration, is provided in Attachment 2. The marked up UFSAR pages affected by the proposed changes are provided in Attachment 3.

A001

Upon NRC approval of this proposed change, PSE&G requests that the amendment be made effective on the date of issuance, but allow an implementation period of sixty days to provide sufficient time for associated administrative activities.

Should you have any questions regarding this request, please contact Mr. C. E. Manges, Jr. at 856-339-3234.

Sincerely,

Mark B. Bezilla
Vice President - Operations

Affidavit
Attachments (3)

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REF: LR-N990511
LCR H99-12

STATE OF NEW JERSEY)
) SS.
COUNTY OF SALEM)

Mark B. Bezilla, being duly sworn according to law deposes and says:

I am Vice President - Operations of Public Service Electric and Gas Company, and as such, I find the matters set forth in the above referenced letter, concerning Hope Creek Generating Station, Unit 1, are true to the best of my knowledge, information and belief.

Subscribed and Sworn to before me
this 24th day of Feb., 2000

Jennifer M. Turner
Notary Public of New Jersey

JENNIFER M. TURNER
NOTARY PUBLIC OF NEW JERSEY
My Commission expires on My Commission Expires July 25, 2000

ATTACHMENT 1
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354
REVISIONS TO HOPE CREEK UFSAR

BASIS FOR REQUESTED CHANGE:

Public Service Electric and Gas Company (PSE&G) requests that the Hope Creek Updated Final Safety Analysis Report (UFSAR) be modified as proposed herein to reflect use of the Mechanical Vacuum Pumps (MVPs) to evacuate the condenser during plant startup at power levels less than or equal to 5%. NRC approval of the proposed UFSAR changes is required, in accordance with 10CFR50.59, since the changes have been determined to involve an unreviewed safety question (USQ). An engineering calculation was performed to assess the impact of the use of the MVPs on the radiological consequences of a control rod drop accident (CRDA). The calculation demonstrated that the radiological consequences of a CRDA coincident with MVP operation increase but remain well within the 10CFR100 guidelines and meet SRP Section 15.4.9, Appendix A, acceptance criteria. Additionally, the calculation demonstrated that the radiological consequences are within the GDC 19 guidelines for control room personnel and plant operators and remain bounded by the loss of coolant accident analysis for on-site personnel. The proposed changes to the UFSAR are indicated on the marked-up UFSAR pages contained in Attachment 3 of this submittal.

REQUESTED CHANGE, PURPOSE AND BACKGROUND:

REQUESTED CHANGE

The proposal revises Hope Creek UFSAR Sections 6.4, 15.4, and 15.9 to be consistent with the radiological consequences analysis provided in the design calculation that accounts for use of the MVPs during startup at less than or equal to 5% power.

PURPOSE

The MVPs are used to evacuate the main condenser during startup or shutdown conditions at power levels less than or equal to 5% since the Off-Gas System is not capable of adequately evacuating the condenser at these power levels. The proposed changes to the UFSAR are required to reflect the analyzed dose consequences associated with a CRDA while in this mode of plant operation.

BACKGROUND

During review of a revision to the condenser air removal system operating procedure, PSE&G discovered that the UFSAR accident analysis was not consistent with actual plant operation. Specifically, PSE&G identified that operating the MVPs to evacuate the condenser during startup at power levels less than or equal to 5% had not been assumed in the analysis of a CRDA.

The Hope Creek CRDA analysis assumes that the Off-Gas System is used to evacuate the condenser. The Off-Gas System is therefore credited to mitigate the fission product release during the CRDA. The trip/isolation of the MVPs and the associated pump suction isolation valves on a main steam line high radiation signal is not safety-related, and therefore cannot be credited in the accident analysis.

During plant startup, the MVPs are operating and bypassing the Off-Gas Treatment System. The MVPs are physically restrained to discharge directly and untreated to the South Plant Vent (SPV). Although, the SPV is monitored, the stack does not have gaseous effluent processing capability. Since this transport pathway is not evaluated in the CRDA analysis in Chapter 15.4.9, the current UFSAR analysis does not assume releases that would correspond to those that would exist with the MVPs in operation.

A calculation was prepared to assess the radiological consequences associated with the use of MVPs during startup at power levels $\leq$ 5% coincident with control rod drop accident. This calculation uses conservative assumptions identified in Standard Review Plan Section 15.4.9, Appendix A, for calculating off-site whole-body and thyroid doses. The calculation also assumes the following:

a) The MVPs do not trip automatically on high radiation monitor signal but are tripped manually one hour after a postulated CRDA,

b) The MVPs are secured from service prior to reactor power exceeding 5%,

c) Main condenser vacuum is established before the main steam isolation valves (MSIVs) are opened, and

d) The condenser evacuation rate using the MVPs is conservatively assumed to be 200 cfm after main condenser vacuum is established.

The basis for each of these assumptions is provided below.

Basis for Assumption a)

The assumption in the calculation that the MVPs would be tripped within one hour is conservative based upon the results of a plant walk-down and evaluation of the manual action in accordance with NRC Information Notice 97-78.

Equipment operators performed a walk-down and verified that the MVP breakers could be tripped in less than a half-hour. Therefore, the one-hour timeframe for taking manual operator action to trip the MVPs is a conservative assumption.

The evaluation of the manual action using the in NRC Information Notice 97-78 has been performed. The proposed manual action does not protect a Safety Limit. Other guidance questions are satisfied as discussed below.

1. Specific Operator Actions Required: Hope Creek Operating Procedure HC.OP-AB.ZZ-0203, Main Steam Line High Radiation, requires subsequent operator action to ensure the MVP is out of service within one hour following a main steam line high radiation alarm. An operator would be dispatched to the MVPs to manually trip each MVP breaker.
2. Potentially Harsh or Inhospitable Environmental Conditions Expected: Design Calculation H-1-CG-MDC-1795 determined that operator doses due to leaked condenser air and radiation sources in the condenser are acceptably within GDC 19 guidelines, thereby satisfying the requirements of NUREG-0737, Item II.B.2 for an area requiring infrequent access.
3. Discussion of Ingress/Egress Paths Taken to Accomplish Functions: Equipment operators walked down the path to trip each MVP breaker and verified it could be done less than a half-hour. Mechanical vacuum pump breakers are accessible to plant operators to manually trip the MVP.
4. Procedural Guidance for Required Actions: Hope Creek Operating Procedure HC.OP-AB.ZZ-0203, Main Steam Line High Radiation, requires subsequent operator action to ensure the mechanical vacuum pump is out of service within one hour following a main steam line high radiation alarm.
5. Operator Training Necessary to Carry Out Actions Including Operator Qualifications Required: No special training is required since the activities are normal actions routinely performed by plant operators.
6. Additional Support Personnel and/or Equipment Required to Carry Out Actions: No additional support personnel or equipment are required beyond the normally available.
7. Information Required to Determine if Operator Action is Required Including Qualified Instrumentation Used to Diagnose the Situation and Verify that Action Has Been Taken: The information required to determine if operator action is required and to verify that the action has been taken is available from several sources in the main control room. These sources included the following:
   - The MVP start and stop switches are backlit on Panel 10C651A
   - The high radiation trip of the MVPs is annunciated on Panel 10C651A
   - The MVP suction valves open/close status is indicated on backlit bezels on Panel 10C651A
   - The low main condenser vacuum is annunciated on the overhead panels
   - Condenser vacuum is indicated by three redundant pressure recorders on Panel 10C650

In addition to the above control room indication, local indication of MVP pump flow is available.

The above indications are non safety-related; however, the instrumentation is considered to be adequate to determine the need for action and to verify its completion based on its diversity and redundancy.

8. Ability to Recover from Credible Errors in Performing the Actions and Expected Time Required to Make the Recovery: Equipment operators walked down the path to trip each MVP breaker and verified it could be done less than a half-hour (less than one half the time allowed for taking the action). Mechanical vacuum pump breakers are accessible to plant operators to manually trip the MVP. Control Room operators will be monitoring the performance of Procedure HC.OP-AB.ZZ-0203(Q). Based on these facts, the control room operators would be expected to identify any errors in performing the action in sufficient time to recover and ensure that the pumps are tripped within the required timeframe.
9. Consideration of Risk Significance of Actions: Based upon the information provided above, adequate controls are in place to ensure that the subject manual action is taken. In addition, the allowed time for performing the action is conservative with respect to the actual time required to take the action. The risk significance of the actions is judged to be low given the high probability of successfully completing the manual action within the required time.

Basis for Assumption b)

The assumption that the MVPs are secured from service prior to reactor power exceeding 5% is appropriate based on procedural requirements. Procedure HC.OP-SO.CG-0001(R), "Condenser Air Removal System Operation" limits use of the MVPs to less than or equal to 5% reactor thermal power.

Basis for Assumption c)

The assumption that main condenser vacuum is established before the MSIVs are opened is appropriate based on procedural requirements. Procedure HC.OP-SO.CG-0001(R) includes a prerequisite that specifies that the reactor vessel be isolated from the main condenser unless the MVPs are being used to "maintain" an established vacuum.

Basis for Assumption d)

The assumption of a condenser evacuation rate of 200 cfm using the MVPs after main condenser vacuum is established is a conservative value. The balance of the air flow through the MVPs is air drawn through the vacuum breakers at the MVP locations. The assumed air flow is conservative based on comparison with the 75 scfm capacity of a single 100% Steam Jet Air Ejector (SJAE) unit, which is sufficient to maintain main condenser vacuum.

The TACT5 computer code from the HABIT computer code package was used in the engineering calculation to determine off-site doses. The TACT5 code is a convenient analytical tool that provides the means to perform analyses using approved modeling methodologies such as those identified in SRP 15.4.9, Appendix A. The HABIT computer code package is a PSE&G approved critical software package. The CONHAB computer code from the HABIT computer code package was used in the engineering calculation to determine on-site doses.

The engineering calculation demonstrated that the radiological consequences of a CRDA coincident with MVP operation remain well within the 10 CFR 100 guidelines and meet SRP Section 15.4.9, Appendix A, acceptance criteria. Additionally, the calculation demonstrated that the radiological consequences of a CRDA coincident with MVP operation are within GDC 19 guidelines for control room personnel and plant operators and remain bounded by the loss of coolant accident analysis for on-site personnel.

JUSTIFICATION OF REQUESTED CHANGES:

An engineering calculation was performed that showed a site boundary two-hour whole-body dose of 1.760 rem and a site boundary two-hour thyroid dose of 19.53 rem. The radiological consequences of a CRDA coincident with MVP operation therefore remain well within the 10CFR100 exposure guidelines and meet SRP Section 15.4.9, Appendix A, acceptance criteria (i.e., 6 rem for whole-body doses and 75 rem for the thyroid doses). Additionally, the engineering calculation demonstrated that the radiological consequences of a CRDA coincident with MVP operation are within GDC 19 guidelines for control room personnel and plant operators and are bounded by the LOCA radiological consequences.

ENVIRONMENTAL IMPACT:

The proposed UFSAR changes were reviewed against the criteria of 10CFR51.22 for environmental considerations. The proposed changes do not involve a significant hazards consideration, a significant increase in the amounts of effluents that may be released offsite, or a significant increase in the individual or cumulative occupational radiation exposures. Based on the foregoing, PSE&G concludes that the proposed UFSAR changes meet the criteria given in 10CFR51.22(c)(9) for a categorical exclusion from the requirements for an Environmental Impact Statement.

ATTACHMENT 2
HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354
SIGNIFICANT HAZARDS EVALUATION

10CFR50.92 EVALUATION

Public Service Electric & Gas has concluded that the proposed changes to the Hope Creek Generating Station Updated Final Safety Analysis Report do not involve a significant hazards consideration. In support of this determination, an evaluation of each of the three standards set forth in 10CFR50.92 is provided below.

REQUESTED CHANGE

The proposal involves the use of the Mechanical Vacuum Pumps (MVPs) to evacuate the main condenser during startup at power levels less than or equal to 5%.

BASIS

1. *The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.*

The Condenser Air Removal System has no safety-related function and its failure does not jeopardize the function of any safety-related system or component or prevent a safe shutdown of the plant. Neither the MVPs, nor other components associated with the Condenser Air Removal, Gaseous Radwaste Off-Gas, Process Radiation Monitoring, or Turbine Building HVAC systems or the South Plant Vent are design basis accident initiators. The operation of mechanical vacuum pump at power levels $\leq$ 5% will not increase the probability of occurrence of a main condenser air removal system leak or failure of the line leading to the steam jet air ejector (SJAE) near the main condenser. Additionally, the design and operation of the condenser off-gas system is not impacted. Moreover, MVP operation will not increase the probability of occurrence of a CRDA or any other design basis accident. Consequently, this proposal does not increase the probability of an accident previously evaluated.

The engineering calculation performed to assess the impact of the use of the MVPs demonstrated that the radiological consequences of a CRDA coincident with MVP operation increase but remain well within the 10CFR100 guidelines and meet SRP Section 15.4.9, Appendix A, acceptance criteria. Additionally, the calculation demonstrated that the radiological consequences of a CRDA coincident with MVP operation are within the GDC 19 guidelines for control room personnel and plant operators and remain bounded by the loss of coolant accident analysis for on-site personnel. Therefore, although the proposal does increase the consequences of a CRDA, the proposal does not significantly increase the consequences of an accident previously evaluated.

2. *The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.*

The proposal involves crediting manual action to trip the MVPs; however, PSE&G has evaluated this operator action against the criteria in NRC Information Notice 97-78 and has concluded that adequate controls are in place to ensure that the subject manual action is taken. In addition, the proposal does not change monitor setpoints, affect equipment qualification, or otherwise create an accident initiator not previously considered. Consequently, this proposal does not create the possibility of an accident of a different type from any previously evaluated.

3. *The proposed change does not involve a significant reduction in a margin of safety.*

The Condenser Air Removal System has no safety-related function. Failure of the system does not jeopardize the function of any safety-related system or component or prevent a safe shutdown of the plant.

The radiological activity evaluated in this proposal does not result in scenarios that could impact 10CFR50 Appendix I, 10CFR20, or 40CFR190 release criteria. Post-scram shutdown or startup condition MVP operation in accordance with plant operating procedures will not degrade the original design for the Condenser Air Removal System.

An engineering calculation was prepared that demonstrated that the radiological consequences of a CRDA coincident with MVP operation remain well within the 10CFR100 guidelines and that the consequences meet SRP Section 15.4.9, Appendix A, acceptance criteria. Additionally, the engineering calculation demonstrated that the radiological consequences of a CRDA coincident with MVP operation are within GDC 19 guidelines for control room personnel and plant operators and remain bounded by the loss of coolant accident analysis for on-site personnel.

Since no design bases are degraded, the Technical Specifications operating limits, that provide sufficient operating range such that the acceptance limits are not exceeded during plant operations and analyzed transients, are not be affected. Since the acceptance limits are not exceeded, implementation of this proposal does not reduce the margin of safety as described in the basis for any Technical Specifications.

CONCLUSION

Based on the above, PSE&G has determined that the proposed changes do not involve a significant hazards consideration.

HOPE CREEK GENERATING STATION
FACILITY OPERATING LICENSE NPF-57
DOCKET NO. 50-354
REVISIONS TO THE UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR)

UPDATED FINAL SAFETY ANALYSIS REPORT PAGES WITH PROPOSED CHANGES

The following Updated Final Safety Analysis Report pages are affected by this change request:

UFSAR SECTION	PAGE
6.4.4.1	6.4-10
15.4.9.5.1.2/15.4.10	15.4-20
15.9.6.5.3	15.9-66
Table 15.4-6	1 of 1
Table 15.4-10	1 of 1

of the accidents by the methodology described in Section 6.4.7 with the radiation source terms defined in the appropriate sections in Section 15 and the release locations relative to the control room intake shown in Figure 6.4-2 and Table 6.4-6.

The release location for the DBA LOCA, fuel handling accident, main steam line break accident and instrument line break accident which occur inside primary containment and the Reactor Building is the FRVS exhaust vent at the top of the Reactor Building. The release location for the control rod drop accident with mechanical vacuum pumps in operation, main steam line break accident, and off-gas system failure which occur in the Turbine and Radwaste Buildings is the south plant vent. The control rod drop accident could also result in releases from leakage from the condenser through the Turbine Building or the north plant vent if the off-gas system is available. The off-gas system failure could also result in releases from the north plant vent depending upon the actual accident location in the building. The release location for the main steam line break accident which occur in the main steam tunnel is the blowout panels located between the reactor building and the south plant vent.

The release location for the HPCI steam supply line break accident, which occurs in the Reactor Building at Elevation 63 feet-0 inches, is the Reactor Building blowout panels located in the west wall of the Reactor Building. The radiation source term for this accident can be conservatively assumed to be equivalent to the main steam line break for purposes of this evaluation.

Of all the accidents releasing from the FRVS exhaust vent, the highest source term results from the DBA LOCA. Releases from the south plant vent occur from accidents which have lower source terms~~ and a smaller atmospheric dispersion factor with respect to control room intake LOCA~~. The main steam line break and HPCI line break have smaller source terms in comparison to the DBA LOCA. Therefore, the consequences are less. Consequently, the DBA LOCA has been determined to result in the controlling accident conditions and has been designated the worst case accident scenario for control room habitability design purposes. The resulting calculated doses for control room occupancy on a rotating shift basis, for the Reactor Building design basis inleakage rate, are

Of the activity reaching the condenser, 100 percent of the noble gases and 10 percent of the iodines remain airborne and available for leakage. Iodine removal is due to partitioning and plateout. 100 percent of the noble gases are assumed to enter the GWMS, and are released to the environment via the normal offgas release point after holdup in the system. All of the iodine which enters the offgas treatment system is retained indefinitely and does not contribute to the offsite doses.

If the GWMS is unavailable, the transport pathway reduces to leakage of the airborne activity from the condenser.

In the first case, Iit is assumed that the condenser is isolated, and that the activity airborne in the condenser leaks from the Turbine Building at ground level directly to the environment at a rate of 1.0 percent a day. No credit is taken for holdup and decay in the Turbine Building. Radioactive decay is accounted for during residence in the condenser, and is neglected after release to the environment. The release continue for 24 hours and then terminates.

In the second case, if mechanical vacuum pumps are operating, it is assumed that the activity airborne in the condenser is removed at a rate of 200 cfm and released at the South plant vent until the pumps are manually tripped one hour after the accident is initiated. Then, it is assumed that the condenser is isolated, and that the activity airborne in the condenser leaks from the Turbine Building at ground level directly to the environment at a rate of 1.0 percent a day. No credit is taken for holdup and decay in the Turbine Building. Radioactive decay is accounted for during residence in the condenser, and is neglected after release to the environment. The release continues until terminating 24 hours after the accident is initiated.

15.4.9.5.1.3 Radiological Results

Site boundary doses based on a Hope Creek specific atmospheric dispersion factor were calculated using the results presented in Reference 15.4-3.

The calculated doses from the design basis analysis are presented in Table 15.4-10.

15.4.9.5.1.4 Main Control Room

Main control room habitability for the CRDA is bounded by the analysis for the design basis loss-of-coolant accident (LOCA) and is addressed in Section 6.4.

15.4.10 References

15.4-1 C. J. Paone "Bank Position Withdrawal Sequence," NEDO-21231, September 1976.

15.4-2 General Electric, "General Electric Standard Application For Reactor Fuel," including the "United States Supplement," NEDE-240111-P-A-7, and NEDE-24011-P-A-7-US.

15.4-3 General Electric, "Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Isolation Valve Closure Function and SCRAM Function of the Main Steam Line Radiation Monitor," NEDO-31400A, May 1987.

15.4-4 "Computer Codes for Evaluation of Control Room Habitability (HABIT)", NUREG/CR-6210, June 1996.

15.9.6.5.3 Event Definition and Operational Safety Evaluations

1. Event 40, control rod drop accident (CRDA) - The CRDA results from an assumed failure of the control rod to drive mechanism coupling after the control rod (very reactive rod) becomes stuck in its fully inserted position. It is assumed that the CRD is then fully withdrawn before the stuck rod falls out of the core. The control rod velocity limiter, an engineered safeguard, limits the control rod drop velocity. The resultant radioactive material release is maintained far below the guideline values of 10CFR100.

   The CRDA is applicable only in operating state D. The CRDA cannot occur in state B, because rod coupling integrity is checked on each rod to be withdrawn if more than one rod is to be withdrawn. No safety actions are required in states A or C where the plant is in a shutdown state by more than the reactivity worth of one control rod prior to the accident.

   Figure 15.9-41 presents the different protection sequences for the CRDA. The reactor is automatically tripped and isolated for all design basis cases except the MSLRM initiated case. The neutron monitoring, reactor protection, and CRD systems will provide a reactor trip from high neutron flux. The main steam line radiation monitoring system will initiate the isolation of the reactor water sample valves and a mechanical vacuum pump trip on high high radiation in the main steam lines. However, the radiological consequences analysis does not credit the automatic mechanical vacuum pump trip, but credits a manual trip one hour after the accident commences. Following a valid high-high MSLRM signal indicating high MSL radiation the reactor will be manually scrammed and the MSIV's will be manually closed in that order. Scramming the reactor first prevents further fuel damage due to the reactor pressure spike that occurs if the MSIV's are manually closed without scramming the reactor.

   After the reactor has been tripped and isolated, the RPV pressure relief system allows the steam, produced by decay heat, to be directed to the suppression pool. Initial core cooling is accomplished by the RCIC, HPCI, or normal feedwater system. With prolonged isolation, as indicated on Figure 15.9-41, the reactor operator initiates the RHR/suppression pool cooling mode and

TABLE 15.4-6

CONTROL ROD DROP ACCIDENT EVALUATION PARAMETERS

1.	Data and Assumptions Used to Estimate Radioactive Source from Postulated Accident		Assumptions
	a.	105% Core Power level, MWt	3458
	b.	Number of fuel rods damaged	850
	c.	Total number of fuel bundles in core	764
	d.	Number of rods per bundle	60
	e.	Peaking factor	1.59
	f.	Fission product released from failed fuel rods	
		melted	100% NG/50% I
		non-melted	10% NG/10% I
	g.	Mass fraction of fuel that reaches or exceeds the initiation temperatures for melting (2842°C)	0.0077
2.	Data and Assumptions Used to Estimate Activity Released		
	a.	Fraction of fission products transported to main condenser	100% NG/10% I
	b.	Fraction of fission products airborne in main condenser	100% NG/10% I
	c.	Condenser leak rate, percent/day	1.0
	d.	Gaseous Waste Management System specifications (per Sec. 11.3.2.1.2.1)	
		1) Mass of charcoal lbs	322,000
		2) holdup times	
		Normal Operation (65°F Temp/40°F dewpoint) Kr = 35.5 h, Xe = 34.1 d	
		Ambient Operation (77°F Temp/45°F dewpoint) Kr = 20.7 h, Xe = 15.3 d	
		3) air/noble gas flow rate, scfm	75
	e.	Assumed mechanical vacuum pump (MVP) condenser air removal rate, scfm	200
	f.	Assumed MVP release duration, hours	1
3.	Dispersion Data (~~x~~X/Q calculated by methodology in Section 2.3.4.2.1)		
	a.	Site boundary distance, m	901
	b.	~~x~~X/Q (s/mE3) for time interval - 0 to 2 h	1.9E-4

TABLE 15.4-10

CONTROL ROD DROP ACCIDENT (DESIGN BASIS ANALYSIS)
RADIOLOGICAL EFFECTS

	Description	Site Boundary (2-hour dose) Whole Body Rem	Site Boundary (2-hour dose) Thyroid Rem
1)	Release via GWMS at normal operating conditions (65°F)	2.03E-2	N/A
2)	Release via GWMS at ambient operating conditions 77°F	3.50 E-1	N/A
3)	Release via isolated condenser	2.50E-2	3.50E-1
4)	Release with mechanical vacuum pump operation	1.760	1.953E+1