UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	§	
	§	
HOUSTON LIGHTING AND POWER COMPANY	§	Docket No. 50-466
	§	
(Allens Creek Nuclear Generating Station, Unit 1)	§	
	§	

APPLICANT'S RESPONSE TO "JOHN F. DOHERTY'S AMENDMENTS TO CONTENTIONS NUMBERED: 26, 23 & 24

Houston Lighting & Power Company (Applicant) files this response to the supplemental pleading filed in this proceeding by Mr. Doherty on July 24, 1979.

Amended Contention No. 26

Although styled as an "amendment", Mr. Doherty has in fact raised a new contention concerning stud bolt integrity. The original contention 26 requested visual inspection of the stud bolts for reasons left unexplained. The "amendment" for the first time now raises a concern about the relationship of applied stress to yield strength for the reactor head stud bolts. As noted in a previous filing,*/ Applicant believes that the Board's June 25th order allowing amendments

*/ "Applicant's Response To TexPirg's Amended Contentions" dated July 10, 1979.

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is applicable only to the refinement of contentions filed on or before May 11, 1979. The Commission's regulations do not afford Mr. Doherty any further opportunity to file additional contentions without justification under the factors enumerated in 10 C.F.R. § 2.714(a). He has not attempted such a justification and, consequently this untimely additional contention should be dismissed.

Aside from the fatal untimeliness of this pleading, Mr. Doherty has also failed to provide any "reasonably specific" basis for the contention. 10 C.F.R. 2.714(b). The pleading mentions, without benefit of explanation, a perceived difference between (1) the magnitude of applied stresses (including or limited to "tensile," "ATWS" and "strain energy") and (2) the ultimate yield strength. Apparently, Mr. Doherty believes that the second compares unfavorably to the first. Simply stated, then, Mr. Doherty maintains that the stud bolts are weak; what is missing is a "reasonably specific" statement of why he believes the stud bolts do not have the strength to sustain any and all credible design loads. Absent this explanation, the contention fails to meet the requirements of the Commission's regulations. 10 C.F.R. 2.714(b).

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Applicant complies fully with the Commission's regulation in 10 C.F.R. §50.55a which incorporates ASME Boiler and Pressure Vessel Code Section III pertaining to vessels and closure bolts which are a part of the reactor coolant pressure boundary. Mr. Doherty offers nothing to suggest that this is not true. And, clearly, he does not show the special circumstances required to challenge a Commission's regulation. His untimely contention must be rejected for this additional reason.

Amended Contention No. 23

In this amendment, Mr. Doherty has attempted to detail a hypothetical pressure-surge induced LOCA. In so doing, he has again omitted any description of the initiating events and probable consequences. More importantly, however, Mr. Doherty still refuses to explain why the scenario he postulates--including the unidentified "power distribution shapes and peaking factors" he is concerned with--is any different from the pipe break or open steam relief valve accidents fully analyzed and designed for. See "NRC Staff Response to John F. Doherty's Additional Contentions," at 10 (June 27, 1979). Without this minimal explanation there is no basis for the contention set forth with "reasonable specificity." 10 C.F.R. 2.714(b). According, the contention should be dismissed.

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Amended Contention No. 24

This amendment condenses to a single disputed fac.: will the peak energy of the fuel exceed the limit of 280 cal/gm if a droppped rod is worth 1.4% or more? Mr. Doherty alleges that it will. The sole basis for this assertion are the "calculations in the PSAR of the Montague Nuclear Plant." Attached to this response is a full summary of the results of the control rod drop accident analysis contained in the Montague PSAR (Table 15.1.38-7). This table clearly shows, using Mr. Doherty's own evidence, that a rod worth in excess of 1.4% will result in a peak enthalpy well below the limit. The additional pages referenced by Mr. Doherty (also enclosed) show as well that the pivotal statements which form the basis of this contention are wrong in every detail. This contention, as "amended" should be dismissed.

Respectfully submitted,

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	§	
	§	
HOUSTON LIGHTING & POWER COMPANY	§	Docket No. 50466
	§	
(Allens Creek Nuclear Generating	§	
Station, Unit 1)	§	

CERTIFICATE OF SERVICE

I hereby certify that copies of the foregoing Applicant's Response to "John F. Doherty's Amendments to Contentions Numbered: 26, 23 & 24 in the above-captioned proceeding were served on the following by deposit in the United States mail, postage prepaid, or by hand delivery this 8th day of August, 1979.

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Table 15.1.38-7

RESULTS OF DESIGN BASIS CONTROL ROD DROP ACCIDENT

Rod Worth	Core Exposure MWD/T	Rod Drop Velocity	Scram Time	Core Average Enthalpy	Global Peak Enthalpy	Power Peaking Factor	Peak Enthalpy (cal/gm)	Number of Failed Fuel Rods
.01435	Beginning	5 ft/sec	A	28.7608	202.9354	1.102	221.9564	444
	of	5 ft/sec	B	28.5543	200		218.772	
	Life	5 ft/sec	C	28.1874	196		214.314	
		5 ft/sec	D	27.1343	184		201.090	
		2.79 ft/sec	A	27.2850	186		203.294	
		2.79 ft/sec	B	27.0308	183		199.998	
		2.79 ft/sec	C	26.6027	178		194.478	
		2.79 ft/sec	D	25.4293	163		177.948	
.01142	3500	5 ft/sec	A	25.359	153.0929	1.087	164.981	0
		5 ft/sec	B	25.304	152		163.793	
		5 ft/sec	C	25.018	148		159.445	
		5 ft/sec	D	24.324	140.5		151.292	
		2.79 ft/sec	A	24.262	140		150.749	
		2.79 ft/sec	B	24.147	139		149.662	
		2.79 ft/sec	C	23.817	134.5		144.770	
		2.79 ft/sec	D	23.055	125		134.444	
.01130	End of	5 ft/sec	A	29.7848	182.7158	1.079	195.851	260
	Life	5 ft/sec	B	29.434	178		190.762	
		5 ft/sec	C	29.242	176		188.604	
		5 ft/sec	D	28.151	166		177.814	
		2.79 ft/sec	A	28.635	170.5		182.670	
		2.79 ft/sec	B	28.193	166.3		178.138	
		2.79 ft/sec	C	27.968	164		175.656	
		2.79 ft/sec	D	26.755	152		162.708	

A = 5 sec. tech spec.
B = 4 sec. tech spec.
C = 4 sec. average
D = measured average

Reference 1(b)

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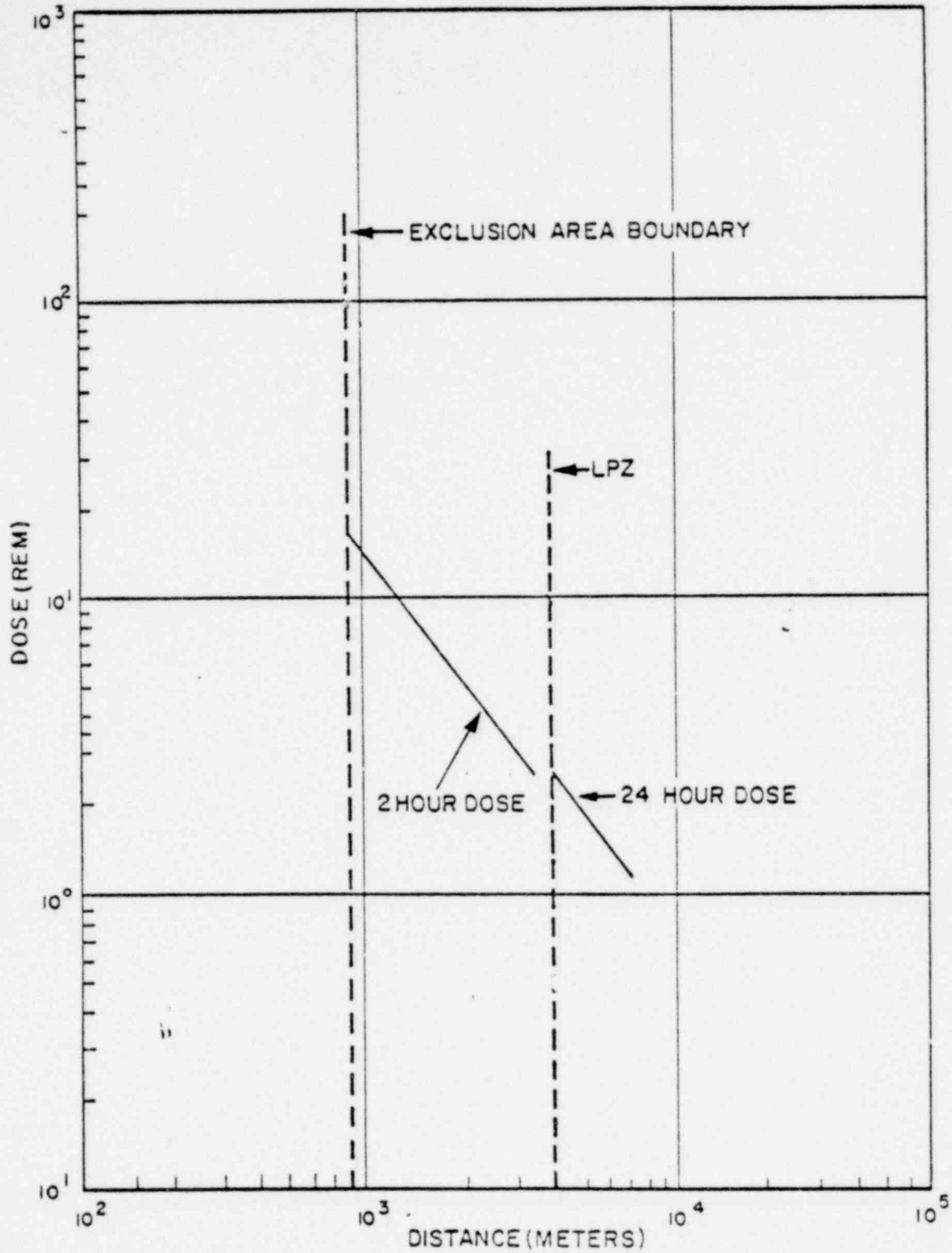

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FIG. 15.1.38-1
CONTROL ROD DROP ACCIDENT NRC-DRL ASSUMPTIONS INHALATION (THYROID) DOSE VERSUS DISTANCE
MONTAGUE NUCLEAR POWER STATION
UNITS 1 AND 2
PRELIMINARY SAFETY ANALYSIS REPORT

to the actual performance of operating reactors. Specific comparisons have been made for the Oyster Creek and Dresden 2 plants. The results of these comparisons show the calculated and actual results agree within experimental and manufacturing tolerances. The design methods have been shown to be able to compute local powers to within ±3% fuel assembly segment powers to within +10%. Pu/U ratios vs exposure to within ±3%, and core reactivities and cold shutdown margin to within 0.5% ΔK.

Experimental tests have also been used to verify the analytical calculations of both reactivity and isotopic composition. These tests give results nearly identical to the comparisons with the operating plants.

Reference 25 contains a complete discussion of errors, uncertainties, and calculations/data comparisons pertaining to the analytical methods used in the design of BWR cores. 4.1.4

4.3.3.2 Nuclear Evaluations

4.3.3.2.1 General

The analyses presented in Subsection 4.3.2 show that the safety design basis is satisfied in conjunction with the nuclear design requirements of Subsection 4.3.1. Adequate protection is provided for the cladding and nuclear system process barrier. The nuclear requirements for reactivity control systems and the settings of the reactor protection system are primarily associated with limitations on the levels and rates of change of reactivity, power, and temperature. Normal plant operation is conducted at rates and values of these parameters, such that reactor transients are readily observable and controllable by plant personnel.

4.3.3.2.2 Reactor Protection

The reactor protection system responds to some abnormal operational transients by initiating a scram. The reactor protection system and the CRD system act quickly enough to prevent the initiating disturbance from causing fuel damage. The scram reactivity curves used in the reactivity excursion analyses at hot startup at various exposures during the first fuel cycle are included as Figures 4.3-4, 4.3-5a and 4.3-5b. The scram reactivity for termination of these abnormal operational transients is shown in Figure 15.1.1-1. Abnormal operational transients are evaluated in Chapter 15. No fuel damage results from any abnormal operational transient. The full scram power decay curve is presented in Figure 4.3-12. This curve displays the relative core heat flux, including decay heat contribution, as a function of time after rod motion begins. The following conditions apply:

Full power scram

End of cycle conditions

Constant pressure

Constant flow

4.3.3.2.3 Control Rod Movement and Patterns

The specified rod withdrawal sequences and the Rod Pattern Control System maintain rod worth at acceptably low values to minimize the consequence of a rod reactivity accident. At any specified reactor state, peak enthalpies for rod removal accidents vary proportionately with rod worths. Peak enthalpy provides the best index for determining the consequences of a reactivity accident when correlated to experimental measurements. Analyses and a survey of pertinent experimental data (Reference 15) indicate that prompt dispersal of finely fragmented fuel into the coolant with subsequent large pressure rise rates does not occur at excursion energy densities below 425 cal/g. Energy densities above this level can cause pressure surges that may endanger the reactor coolant pressure boundary.

To provide a margin below the 425 cal/g, a design limit on peak fuel enthalpy of 280 cal/g is selected. This fuel enthalpy limit is supported by a careful study of all available SPERT, TREAT, KIWI, and PULSTAR tests (Reference 16). In addition, more than a thousand transient tests have been performed at the Capsule Drive Core (CDC) facility at SPERT during the past several years. A large majority of these tests were in the enthalpy range of 280 cal/g. To enumerate all the tests is not practical. However, typical tests are discussed in Reference 17. This is a report on fuel pin transient tests that were conducted in the CDC facility. The nuclear-to-mechanical energy conversion ratio was found to be essentially zero even for tests that resulted in peak fuel enthalpies as high as 338 cal/g.

The enthalpy of UO_2, as a function of fuel temperature, was obtained from experimental data taken by Hein and Flagella (Reference 18). These data indicated the fuel melting range to be 270 to 337 cal/g. In addition, more recent data support the UO_2 melting range to be 270 to 337 cal/g (References 19, 20, and 21). Hence, no phase change in UO_2 is expected until enthalpies in the range of 270 to 337 cal/g are achieved.

There are no experimental data to date that indicate a possibility of prompt fuel failure in the fuel enthalpy range discussed. Therefore, the peak fuel enthalpy and design limit of 280 cal/g is considered justifiable and conservative.

Specified control rod withdrawal sequences are designed to limit rod worth so that the drop of any control rod from the fully inserted position to the position of its drive results in a peak fuel enthalpy of not more than 280 cal/g. A velocity limiter limits the average measured rod velocity plus 3 standard deviations to less than 2.79 ft./sec. Control rod removal excursion analysis from the shutdown flux level (Reference 4) for a typical BWR using axial gadolinia indicates that peak fuel enthalpies of 280 cal/g result from rod worths of 0.0145 Δk (cold, critical) or 0.0145 Δk (hot, critical, zero voids) and removal rates of 2.33 and 2.79 ft/sec respectively. These analyses also show that for excursions initiated from flux levels corresponding to 20% power, the maximum possible control rod worth, 0.020 Δk, is insufficient to cause peak enthalpies of 280 cal/g. Preplanned rod patterns enforced to restrict incremental rod worth to approximately 0.01 Δk, although larger values are acceptable within the 280 cal/g limit. Therefore, no rod would have a worth high enough to produce peak enthalpy of 280 cal/g even if the rod were removed at 2.79 ft/sec.

4.3.3.2.4 Xenon Transients. The maximum xenon reactivity buildup on shutdown from full power and the rate of xenon reactivity burnout on return to full power when the maximum shutdown xenon buildup occurs, are calculated for both the beginning-of-life and the end-of-cycle reactor conditions. The maximum rate of reactivity change is obtained by assuming an instantaneous return to full power. The results of these calculations are shown in Figure 4.3-13 for the beginning-of-life condition. From this analysis it was determined that the maximum reactivity addition caused by burnup of xenon was +0.00010 $(\Delta k/k)$/minute. Assuming a control rod worth of 0.001 $\Delta k/k$ with an insertion rate of 3 in./sec, the reactivity addition by the control rod insertion is -0.00125 $(\Delta k/k)$/minute. Therefore, a very weak control rod can easily compensate for a xenon-burnup reactivity addition. The standby liquid control system, used for emergency shutdown only, is more than adequate to compensate for the reactivity added by xenon decay. With a boron injection rate of 6 ppm/minute, the reactivity insertion of the liquid control system is -0.0013 $(\Delta k/k)$/minute. The design injection rate of 6 to 25 ppm/minute.

Boiling water reactors do not have instability problems due to xenon. This has been demonstrated by operating BWR's for which xenon instabilities have never been observed (such instabilities would readily be detected by the LPRM's), by special tests which have been conducted on operating BWR's in an

attempt to force the reactor into xenon instability, and by calculations. All of these indicators have proven that xenon transients are highly damped in a BWR due to the large negative power coefficient.

The most recent analysis and experiments conducted in this area are reported in Reference 22.

Yields of I-135 and Xe-135 for the various fissionable isotopes are provided in Table 4.3.12. 4.62

4.3.3.2.5 Variation of Nuclear Parameters. The BWR nuclear fuel design incorporates sufficient conservatism to allow for minor variations in the nuclear parameters such as excess reactivity, reactivity coefficients and reactivity insertion rates. These parameters are individually analyzed for each fuel design, with small reliance being placed on previous designs or prior practice.

Excessive reactivity, for example, is a matter of primary design effort. Extensive analyses are performed to establish suitable amounts and locations of the burnable poison (Gd_2O_3) such that the specified reactivity margins are obtained. These margins are bounded by shutdown limits on one end, and performance considerations on the other.

The reactivity coefficients, as already described, provide the negative feedback necessary for normal and accident reactor control. Small variations in the magnitude of these coefficients are not significant due to realtive abundance of negative reactivity feedback in the BWR. Large unexpected variations are precluded by the extensive design evaluation performed for each fuel design.

Reasonable variations in the controlled reactivity insertion rates have very small effect in the BWR. Control rods are operated one at a time resulting in very low reactivity addition rates which are well below the limiting criteria. Considerable variations could be tolerated without major concern.

Finally, the in-core instrumentation system provides an important safeguard against the effects of unexpected or unusual nuclear parameters. Together with the process computer, the instrumentation system provides prompt and reliable data which can be used to identify, analyze and formulate appropriate action as needed and mitigate the effects of undesirable variations in the nuclear parameters. For a further discussion of in-core instrumentation, see Subsection 4.3.6.

4.3.3.2.6 Scram Function Curves. Both the total scram reactivity worth and shape function are strongly dependent on the fuel design and loading pattern (e.g., reload fuel, axial gadolinium distribution, etc.), and for this reason

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15.1.38 Control Rod Drop Accident

15.1.38.1 Identification of Causes. There are many ways of inserting reactivity into a boiling water reactor. However, most of them result in a relatively slow rate of reactivity insertion and therefore pose no threat to the system. It is possible, however, that a rapid removal of a high worth control rod could result in a potentially significant excursion. Therefore, the accident which has been chosen to encompass the consequences of a reactivity excursion is the control rod drop accident.

15.1.38.2 Starting Conditions and Assumptions. Before the control rod drop accident is possible, the following sequence of events must occur:

(1) The complete rupture, breakage or disconnection of a fully inserted control rod drive from its cruciform control blade at or near the coupling.

(2) The sticking of the blade in the fully inserted position as the rod drive is withdrawn.

(3) The falling of the control rod to the rod drive position.

This unlikely set of circumstances makes possible the rapid removal of a control rod. The dropping of the rod results in a high local k_∞ in a small region of the core. For large, loosely couples cores, this would result in a highly peaked power distribution and subsequent shutdown mechanisms. Significant shifts in the spatial power generation would occur during the course of the excursion. Therefore, the method of analysis must be capable of accounting for any possible effects of the power distribution shifts.

The Rod Pattern Control System limits the worth of the rod which could be dropped. This system prevents the movement of an out of sequence rod or rod gang in the 100% to 50% rod density range and from the 50% rod density point to the preset power level, the RPCS will only allow group notch mode or gang rod withdrawal or insertion. The 50% rod density configuration corresponds to the condition in which 50% of the rods are fully inserted in the core and 50% are fully withdrawn. With the stipulation that no out of sequence rod may be moved prior to 50% rod density, the postulated rod drop accident cannot result in peak enthalpies in excess of 280 cal/gm for any possible plant operation or core exposure.

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The envelope of the maximum worth control rods with the RPCS operational is given by Figures 4.3-2a and 4.3-2b. The analysis performed for the control rod drop accident assumes that the maximum worth control rod which exists at the 50% control rod density pattern has its drive fully withdrawn and drops from the core. It shall be emphasized that the RPCS would prevent this from occurring; however, this is a convenient analytical procedure for establishing an upper branch on the rod worth which would result in a peak fuel enthalpy which approaches the design limit of 280 cal/gm. Using this approach, it is demonstrated that a rod drop accident involving an in-sequence rod enforced by the RPCS will result in in peak enthalpies less than 280 cal/gm. The reactivity function for the design basis control rod drop accident at various first cycle exposures are given in Figures 15.1.38-3 through 15.1.38-5. The corresponding scram reactivities used for these analyses are given in Figures 4.3-4, 4.3-5a, and 4.3-5b.

15.1.38.3 Accident Description. The accident is defined as:

(1) The RPCS is functioning.

(2) The highest worth rod that can be developed at any time in core life under any operating conditions drops from fully inserted position to the control rod drive position.

(3) The rod drops.

(4) The scram is that defined in the technical specifications.

The detailed analysis of this accident is discussed in Reference 1, 1a and 1b. A continuing effort is being made in the area of analytical methods to assure that nuclear excursion calculations reflect the latest "state-of-the art."

The sequence of events and the approximate times of occurrence are as follows:

Event	Approximate Elapsed Time
(1) Reactor is operating at 50% control rod density pattern.	
(2) Maximum worth control blade becomes decoupled.	
(3) Operator selects and withdraws the control rod drive of the decoupled maximum worth rod along with the other control rods assigned to that RPCS notch group or gang to the fully withdrawn position.	
(4) Blade sticks in the fully inserted position.	

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Event	Approximate Elapsed Time
(5) Blade becomes unstuck and drops at the nominal measured velocity plus 3 standard deviations.	0
(6) Reactor goes prompt critical and initial power burst is terminated by the Doppler Reactivity Feedback.	<1 sec
(7) APRM 120% power signal scrams reactor.	
(8) Scram terminates accident.	<5 sec

15.1.38.4 Identification of Operator Actions. The termination of this excursion is accomplished by automatic safety features or inherent shutdown mechanisms. Therefore, no operator action during the excursion is required.

15.1.38.5 Analysis of Effects and Consequences

15.1.38.5.1 Realistic Evaluation Methods. The analytical methods and associated assumptions which are used in evaluating this accident are considered to provide a realistic, yet conservative assessment of the consequences.

15.1.38.5.1.1 Methods, Assumptions and Conditions. The methods, assumptions, and conditions for evaluating the excursion aspects of the control rod drop accident are described in detail in References 1, 1a and 1b.

15.1.38.5.1.2 Results and Consequences

15.1.38.5.1.2.1 Fuel Damage. The fuel damage thresholds are based on both experimental and theoretical data. This information was discussed previously in Section 4.2 of this document, with additional detailed information presented in Section 5 of Reference 2.

The results of the rod drop accident analysis are presented in Table 15.1.28.8. The peak enthalpy results of the design basis control rod drop accident (Reference 1b) are less than the 280 cal/gm design limit for all exposures. The number of failed fuel rods due to the design basis control rod drop accident is less than 770 rods for all plant operating conditions and exposures. However, the radiological exposure calculations have been performed on the assumed failure of 770 fuel rods.

15.1.38.5.1.2.2 Fission Product Release from Fuel. The following assumptions are used in calculating fission product activity release from the fuel:

(1) The reactor has been operating at 3758 MWt until 30 min prior to the accident. When translated into actual plant operation, this assumption means that the reactor was shut down from design power, taken critical, and brought to the initial temperature conditions within 30 min of the departure from design power. The 30-min time represents a conservative estimate of the shortest period in which the required plant changes could be accomplished and defines the decay time to be applied to the fission product inventory calculations.

(2) An average of 1.8% of the noble gas activity and 0.32% of the halogen activity in a perforated fuel rod is assumed to be released. These percentages are consistent with actual measurements made during defective fuel experiments (Reference 3).

(3) For this plant, the following fission product activities are contained in the core, at the time the accident occurs:

Noble gases	4.8×10^8 Ci
Iodine	8.7×10^8 Ci

(4) The fraction of solid fission product activity available for release from the fuel is negligible.

(5) The fission products produced during the nuclear excursion are neglected. The excursion is of such short duration that the fission products generated are negligible in comparison with the fission products already present in the fuel.

Using the above assumptions, the following amounts of fission product activity are released from the failed fuel rods to the reactor coolant:

Noble gases (Xe, Kr)	1.3×10^5 Ci
Iodine	4.6×10^4 Ci

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15.1.38.5.1.2.3 Condenser Activity

The following assumptions are used in calculating the amount of fission product activity transported from the reactor vessel to the main condenser:

a. The recirculation flow rate is 25 percent of rated, and the steam flow to the condenser is 5 percent of rated. The 25 percent recirculation flow and 5 percent steam flow are the maximum flow rates compatible with the maximum fuel damage. The 5 percent steam flow rate is greater than that which would be in effect at the reactor power level assumed in the initial conditions for the accident. This assumption is conservative because it results in the transport of more fission products through the steam lines than would be expected. Because of the relatively long fuel-to-coolant heat transfer time constant, steam flow is not significantly affected by the increased core heat generation within

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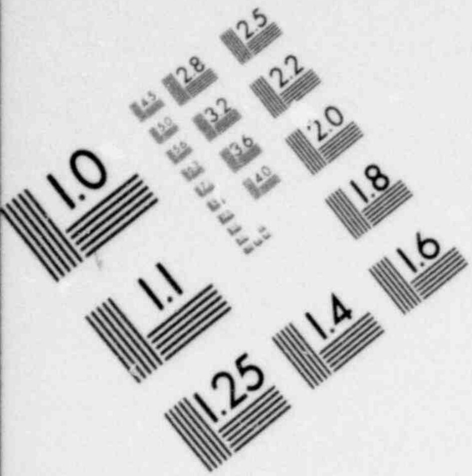

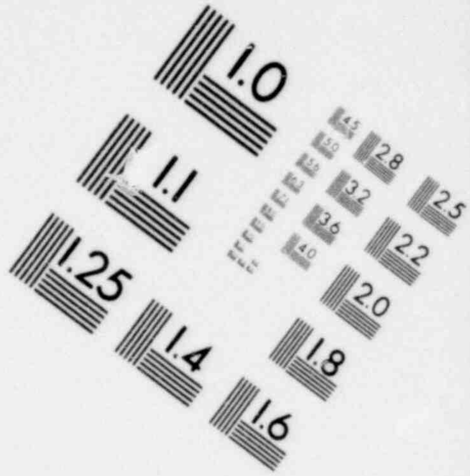

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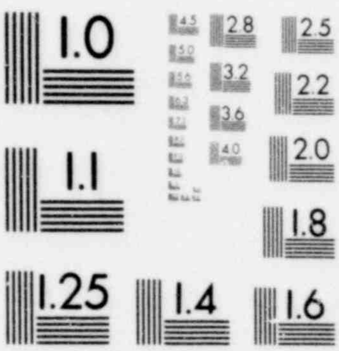

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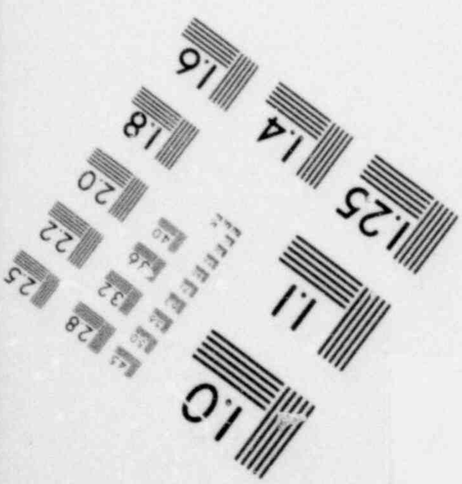

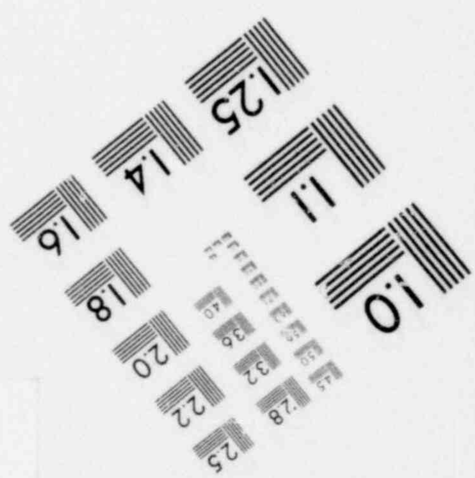

the time required for the main steam line isolation valves to achieve full closure.

b. The main steam line isolation valves are assumed to receive an automatic closure signal 0.5 sec after detection of high radiation in the main steam lines and to be fully closed at 5 sec from the receipt of the closure signal. The automatic closure signal originates from the main steam line radiation monitors. The total time required to isolate the main steam lines (5.5 sec) combined with the assumptions in (1), dictates the total amount of fission product activity transported to the condenser before the steam lines are isolated.

c. All of the noble gas activity is assumed to be released to the steam space of the reactor vessel.

d. The mass ratio of the halogen concentration in steam to that of the water is assumed to be 2 percent.

e. Fission product plate-out is neglected in the reactor vessel, main steam lines, and condenser.

Of those fission products released from the fuel and transferred to the condenser, it is assumed that 100 percent of the noble gases are airborne in the condenser. The iodine activity airborne in the condenser is a function of the partition factor. The partition factor assumed applicable is 100. Based on the above condition, the activity airborne in the condenser is presented in Table 15.1.38-1.

15.1.38.5.1.2.4 Activity Released to Environment

The fission product activity released to the environment is a function of the total amount of activity airborne in the condenser and the condenser leak rate. For the purpose of this analysis it is assumed that:

a. 100 percent of the noble gas activity transferred to the condenser is airborne and available for release to the environment.

b. The iodine activity airborne is in proportion to the waterborne activity, the partition factor, and the volumes of air and water.

c. The condenser leak rate is 0.5 percent of the combined condenser and turbine-free volume per day.

d. The activity released from the condenser becomes airborne in the turbine building and is released to the environment at a rate of 8.4 air changes per day.

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Based on the above assumptions, the fission product release rate to the environment is presented in Table 15.1.38-2.

15.1.38.5.1.2.5 Radiological Effects

Based on the release rates presented in Table 15.1.38-2, the resultant radiological exposures are presented in Table 15.1.38-3. It should be noted that all of the exposures are orders of magnitude below the guidelines set forth in 10CFR100.

15.1.38.5.1.3 Consideration of Uncertainties

Consideration of uncertainties with regards to the core physics calculations have been reported previously in Ref. 2. In addition, Ref. 1 presents a sensitivity analysis of the rod drop accident with regards to rod drop velocities, scram-insertion rates, and control rod worth for a wide spectrum of operating conditions. This approach has been taken to demonstrate the comparison between a realistic and a worst case condition.

15.1.38.5.2 Conservative (AEC) Licensing Basis Evaluation Methods

15.1.38.5.2.1 Methods, Assumptions and Conditions

While the AEC has not published an official guide for the control rod drop accident, the assumptions, methods, and conditions used in this report are typical of those used by the AEC in past licensing efforts.

15.1.38.5.2.2 Results and Consequences

15.1.38.5.2.2.1 Fuel Damage

As noted in the previous sections, the exact extent of fuel damage has not been established for this accident. However, for the purpose of providing a relative dose effect, it is assumed that 770 fuel rods experience cladding damage.

15.1.38.5.2.2.2 Fission Product Release from Fuel

It is assumed that 50 percent of the halogens and 100 percent of the noble gases contained in those rods which experience cladding damage are released from the fuel. Those rods which experience cladding damage are assumed to have a peaking factor of 1.5. Therefore, the activity released from these rods is 75 percent and 150 percent, respectively, of the halogen and noble gas activity contained in the average fuel rod. Of those fission products released from the fuel, 90 percent of the halogens and 0 percent of the noble gases are absorbed by the reactor water. The remaining activity is released to the condenser prior to isolation valve closure.

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15.1.38.5.2.2.3 Condenser Activity

Based on the failure and transport mechanisms defined above, and assuming a plate-out factor of 2 in the condenser for iodines, the activity airborne in the condenser is presented in Table 15.1.38-4.

15.1.38.5.2.2.4 Activity Released to Environment

The fission product activity released to the environment is dependent upon the activity airborne in the condenser, the condenser leak rate, and the turbine building leak rate. For the purpose of this analysis it is assumed that the condenser leak rate is 0.5 percent per day and the turbine building leak rate is infinite. Based on the airborne activity presented in the previous subsections and the above leakage rates, the noble gas and iodine release rates to the environment are presented in Table 15.1.38-5.

15.1.38.5.2.2.5 Radiological Effects

a. Offsite. On-site meteorology for a ground level release, is assumed for this event. Consideration of the fission product release rates in Table 15.1.38-5 and the above meteorology results in the radiological exposures presented in Figures 15.1.38-1 and 15.1.38-2. It should be noted that these exposures are well below the guidelines set forth in 10CFR100. 6

b. Control Room. Based on the control room assumptions listed in Section 15.1.39, the whole body dose in the control room due to this accident is 26 mrem.

15.1.38.5.3 Comparison of Realistic and Conservative Parameters

As mentioned previously, the basis for the conservative calculation for this accident is past AEC practice in licensing BWRs. The comparison of realistic and conservative parameters is made in Table 15.1.38-6.

15.1.38.6 References

1. R.C. Stirn et al., "Rod Drop Accident Analysis for Large Boiling Water Reactors," NEDO-10527, March 1972.

1a. NEDO-10527, Supplement No. 1, Aug 1972.

1b. NEDO-10527, Supplement No. 2, March 1973.

2. J.E. Boyden et al., "Summary Memorandum of Excursion Analysis Uncertainties," Dresden Nuclear Power Station Unit 3 Amendment No. 3. 5

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3. N. R. Horton, W. A. Williams, J. W. Holtzclaw, "Analytical Methods for Evaluating the Radiological Aspects of General Electric Boiling Water Reactors," APED-5756, March 1969. 5

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TABLE 15.1.38-1

CONTROL ROD DROP ACCIDENT REACTIVITY AIRBORNE IN THE CONDENSER (REALISTIC ANALYSIS)

	Airborne Activity (Ci) vs Decay Time						
Isotope	1 min	1 hr	2 hr	8 hr	1 day	4 days	30 days
I-131	2.5+0	2.5+0	2.5+0	2.4+0	2.3+0	1.7+0	1.6-1
I-132	3.8-1	3.8-1	3.7-1	3.5-1	3.1-1	1.6-1	5.4-4
I-133	1.3+0	1.3+0	1.2+0	1.0+0	5.9-1	5.4-2	0
I-134	3.0-1	1.4-1	6.2-2	5.3-4	1.7-9	0	0
I-135	7.3-1	6.6-1	5.9-1	3.2-1	6.1-2	3.5-5	0
Total Iodine	5.2+0	5.0+0	4.7+0	4.1+0	[illegible].3+0	1.9+0	1.6-1
Kr-83m	5.1+1	3.5+1	2.5+1	2.8+0	8.2-3	0	0
Kr-85m	2.7+2	2.3+2	2.0+2	7.6+1	6.1+0	7.1-5	0
Kr-85	4.7+2	4.7+2	4.7+2	4.7+2	4.7+2	4.6+2	4.0+2
Kr-87	2.0+2	1.2+2	6.9+[illegible]	2.8+0	5.6-4	0	0
Kr-88	4.5+2	3.5+2	2.7+[illegible]	6.2+1	1.2+0	2.1-8	0
Kr-89	1.2-1	3.4-7	7.7-[illegible]	0	0	0	0
Xe-131m	3.6+1	3.6+1	3.6+1	3.5+1	3.4+1	2.8+1	5.5+0
Xe-133m	1.5+2	1.5+2	1.5+2	1.4+2	1.1+2	4.4+1	1.5-2
Xe-133	7.4+3	7.4+3	7.3+3	7.1+3	6.5+3	4.3+3	1.3+2
Xe-135m	1.8+1	1.5+0	2.8-1	9.9-2	1.9-2	1.1-5	0
Xe-135	1.8+3	1.7+3	1.5-3	9.8+2	2.9+2	1.2+0	0
Xe-137	4.4-1	1.4-5	3.8-10	0	0	0	0
Xe-138	6.5+1	5.9+0	5.1-1	2.1-7	0	0	0
Total NG	1.1+4	1.0+4	1.0+4	8.9+3	7.1+3	4.8+3	5.4+2

Note: 2.5+0 = 2.5×10^{0}

TABLE 15.1.38-2

CONTROL ROD DROP ACCIDENT FISSION PRODUCT RELEASE RATE TO ENVIRONMENT (REALISTIC ANALYSIS)

	Release Rate (Ci/sec) vs Time						
Isotope	1 min	1 hr	2 hr	8 hr	1 day	4 days	30 days
I-131	8.4-10	4.3-8	7.2-8	1.3-7	1.3-7	1.0-7	9.4-9
I-132	1.3-10	6.4-9	1.1-8	1.9-8	1.8-8	9.2-9	3.1-11
I-133	4.4-10	2.1-8	3.5-8	5.4-8	3.4-8	3.1-9	0
I-134	1.0-10	2.3-9	1.8-9	2.9-11	0	0	0
I-135	2.5-10	1.1-8	1.7-8	1.7-8	3.5-9	2.0-12	0
Total Iodine	1.8-9	8.4-8	1.4-7	2.2-7	1.9-7	1.1-7	9.4-9
Kr-83m	1.7-8	6.1-7	7.2-7	1.5-7	4.8-10	0	0
Kr-85m	9.1-6	3.9-6	5.7-6	4.2-6	3.5-7	4.1-12	0
Kr-85	1.6-7	8.0-6	1.4-5	2.6-5	2.7-5	2.7-5	2.3-5
Kr-87	6.7-8	2.0-6	2.0-6	1.5-7	3.2-11	0	0
Kr-88	1.5-7	6.0-6	8.0-6	3.4-6	6.8-8	1.2-15	0
Kr-89	4.1-11	5.8-15	0	0	0	0	0
Xe-131m	1.2-8	6.1-7	1.0-6	1.9-6	2.0-6	1.6-6	3.2-7
Xe-133m	5.1-8	2.5-6	4.3-6	7.4-6	6.4-6	2.5-6	8.8-10
Xe-133	2.5-6	1.3-4	2.1-4	3.9-4	3.8-4	2.5-4	7.5-6
Xe-135m	6.1-9	2.6-8	8.0-9	5.4-9	1.1-9	6.4-13	0
Xe-135	6.1-7	2.9-5	4.5-5	5.3-5	1.7-5	6.7-9	0
Xe-137	1.5-10	2.4-13	0	0	0	0	0
Xe-138	2.2-8	1.0-7	1.5-8	1.2-14	0	0	0
Total NG	3.7-6	1.8-4	2.9-4	4.9-4	4.3-4	2.8-4	3.1-5

Note: 8.4-10 = 8.4×10^{-10}

TABLE 15.1.38-3

CONTROL ROD DROP ACCIDENT
(REALISTIC ANALYSIS)

Radiological Effect

Location	Dose Duration	Dose (rem) Thyroid	Beta	Gamma
Exclusion Area (815 m)	2 hr	$6.6x10^{-5}$	$3.4x10^{-5}$	$3.2x10^{-5}$
Low Population Zone (4,023 m)	24 hr	$5.4x10^{-6}$	$1.6x10^{-6}$	$9.6x10^{-6}$

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TABLE 15.1.38-4

CONTROL ROD DROP ACCIDENT ACTIVITY AIRBORNE IN THE CONDENSER
CONSERVATIVE (NRC) ANALYSIS

Isotope	Airborne Activity (Ci) vs Decay Time				
	1 min	1 hr	2 hr	8 hr	1 day
I-131	6.9+4	6.9+4	6.9+4	6.7+4	6.3+4
I-132	1.0+5	7.4+4	5.5+4	8.9+3	6.9+1
I-133	1.1+5	1.1+5	1.0+5	8.5+4	5.0+4
I-134	1.3+5	5.7+4	2.6+4	2.2+2	7.0-4
I-135	1.1+5	9.7+4	8.7+4	4.7+4	9.0+3
Total Iodine	5.2+5	4.1+5	3.4+5	2.1+5	1.2+5
Kr-83m	2.7+5	1.9+5	1.3+5	1.5+4	4.4+1
Kr-85m	5.5+5	4.7+5	4.1+5	1.6+5	1.3+4
Kr-85	2.8+4	2.8+4	2.8+4	2.8+4	2.7+4
Kr-87	1.7+6	9.8+5	5.8+5	2.4+4	4.7+0
Kr-88	2.6+6	2.1+6	1.6+6	3.7+5	6.9+3
Kr-89	2.2+3	6.2-3	1.4-8	1.9-42	0.0
Xe-131m	2.0+4	2.0+4	2.0+4	2.0+4	2.0+4
Xe-133m	1.9+5	1.9+5	1.9+5	1.8+5	1.6+5
Xe-133	5.3+6	5.3+6	5.2+6	6.2+6	5.0+6
Xe-135m	1.4+6	1.2+6	1.1+6	6.0+5	1.1+5
Xe-135	5.0+6	4.9+6	4.8+6	4.0+6	1.8+6
Xe-137	1.6+4	5.3-1	1.5-5	5.9-33	0.0
Xe-138	1.0+6	9.0+4	7.8+3	3.3-3	3.2-20
Total NG	1.8+7	1.5+7	1.4+7	1.1+7	7.1+6

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Note: 6.9+4 = 6.9×10^4

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TABLE 15.1.38-5

CONTROL ROD DROP ACCIDENT FISSION PRODUCT RELEASE RATE TO ENVIRONMENT CONSERVATIVE (NRC) ANALYSIS

Isotope	Release Rate (Ci/sec) vs Time				
	1 min	1 hr	2 hr	8 hr	1 day
I-131	4.0-3	4.0-3	4.0-3	3.9-3	3.6-3
I-132	5.8-3	4.3-3	3.2-3	5.1-4	4.0-6
I-133	6.4-3	6.2-3	6.0-3	4.9-3	2.9-3
I-134	7.2-3	3.3-3	1.5-3	1.3-5	4.0-11
I-135	6.2-3	5.6-3	5.1-3	2.7-3	5.2-4
Total Iodine	2.8-2	2.3-2	2.0-2	1.2-2	7.0-3
Kr-83m	1.6-2	1.1-2	7.6-3	8.6-4	2.5-6
Kr-85m	3.2-2	2.7-2	2.3-2	9.1-3	7.3-4
Kr-85	1.6-3	1.6-3	1.6-3	1.6-3	1.6-3
Kr-87	9.6-2	5.7-2	3.3-2	1.4-3	2.7-7
Kr-88	1.5-1	1.2-1	9.3-2	2.1-2	4.0-4
Kr-89	1.3-4	3.6-10	8.1-16	0	0
Xe-131m	1.2-3	1.2-3	1.2-3	1.2-3	1.2-3
Xe-133m	1.1-2	1.1-2	1.1-2	1.0-2	9.4-3
Xe-133	3.0-1	3.0-1	3.0-1	3.0-1	2.9-1
Xe-135m	8.0-2	7.1-2	6.4-2	3.5-2	6.6-3
Xe-135	2.9-1	2.8-1	2.8-1	2.3-1	1.0-1
Xe-137	9.5-4	3.1-8	8.4-13	0	0
Xe-138	5.8-2	5.2-3	4.5-4	1.9 10	1.8-27
Total NG	1.0+1	8.9-1	8.2-2	6.1-1	4.1-1

Note 4.0-3 = 4.0×10^{-3}

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TABLE 15.1.38-6

CONTROL ROD DROP ACCIDENT - PARAMETERS TO BE TABULATED FOR POSTULATED ACCIDENT ANALYSES

	Conservative (NRC) Assumptions	Realistic Assumptions
I. Data and assumptions used to estimate radioactive source from postulated accidents		
A. Power level	3,758 MWt	3,758 MWt
B. Fuel damaged	770 Rods	770 Rods
C. Release of activity by nuclide	Table 15.1.38.5	Table 15.1.38.2
D. Iodine fractions		
1. Organic	0	0
2. Elemental	1	1
3. Particulate	0	0
E. Reactor coolant activity before the accident	15.1.39.5.1.2	15.1.39.5.1.2
II. Data and assumptions used to estimate activity released		
A. Condenser leak rate (%/day)	0.5	0.5
B. Turbine building leak rate (%/day)	-	840
III. Dispersion Data		
A. Boundary and LPZ distances (m)	815/4,023	815/4,023
B. χ/Qs for time intervals of:		
1. 0-2 hr - SB	6.87x10-[illegible]	6.87x10-[illegible]
2. 0-8 hr - LPZ	2.29x10-[illegible]	2.29x10-[illegible]
3. 8-24 hr - LPZ	7.85x10-[illegible]	7.85x10-[illegible]
IV. Dose Data		
A. Method of dose calculation	Regulatory Guide 1.3	Regulatory Guide 1.3
B. Dose conversion assumptions	Regulatory Guide 1.3	Regulatory Guide 1.3
C. Peak activity concentrations in containment	Table 15.1.38.4	Table 15.1.38.1
D. Doses	Fig. 15.1.38-1	Table 15.1.38.3

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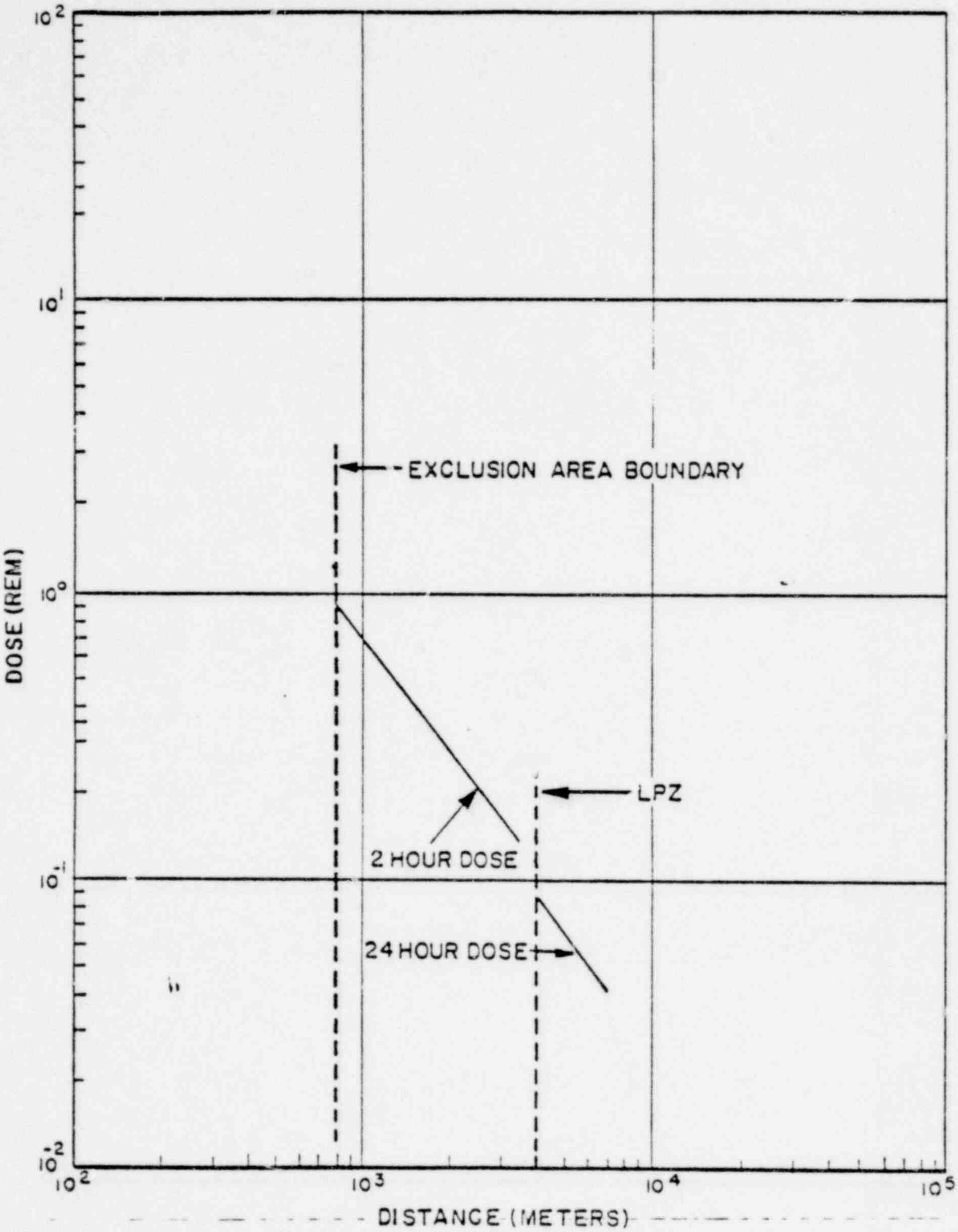

FIG. 15.1.38-2

CONTROL RCD DROP ACCIDENT NRC-DRL
WHOLE BODY DOSE VERSUS DISTANCE
MONTAGUE NUCLEAR POWER STATION
UNITS I AND 2
PRELIMINARY SAFETY ANALYSIS REPORT

1105 011

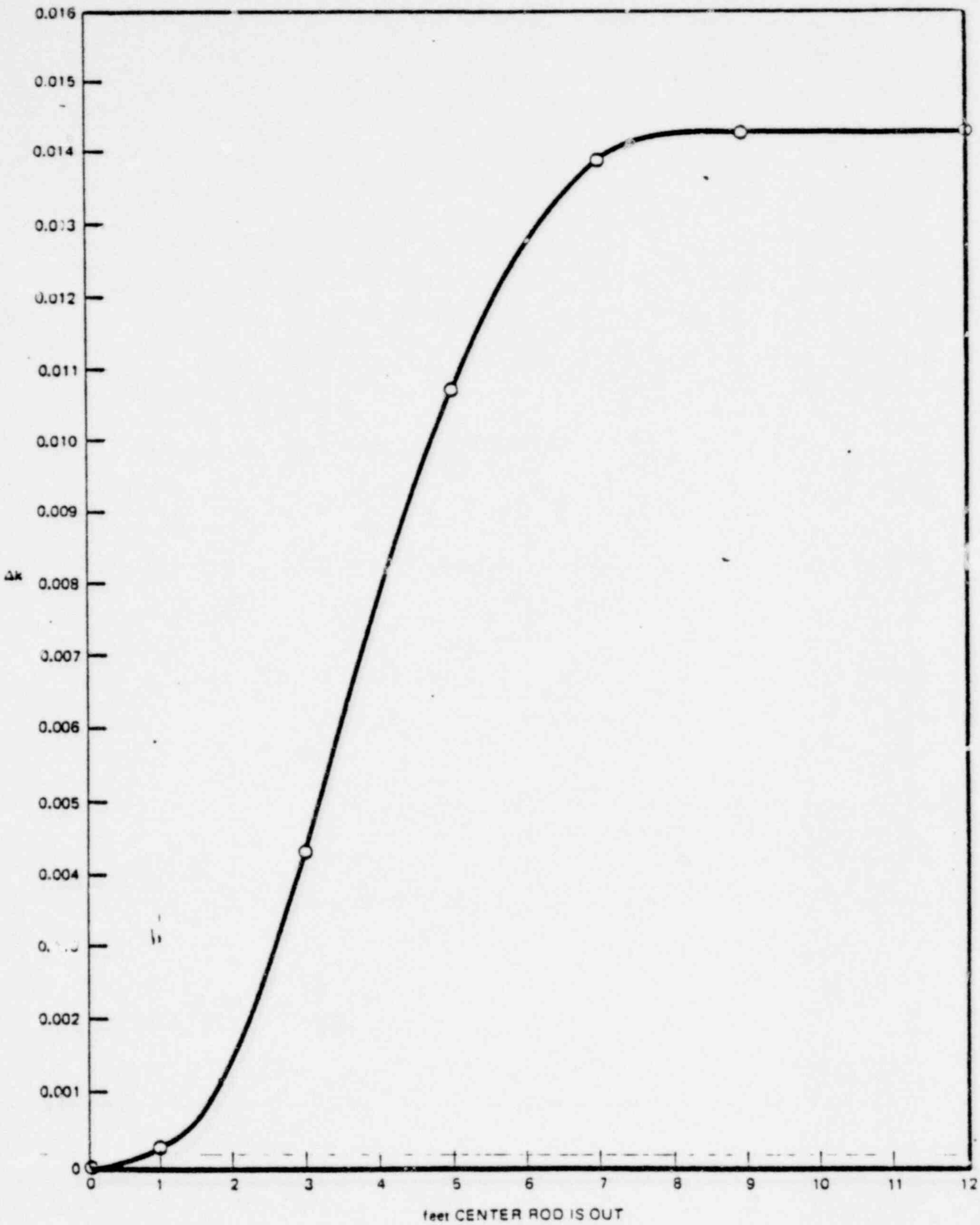

Figure 15.1.38- 3 Design Basis Rod Drop Accident Shape Function (Beginning of Life)

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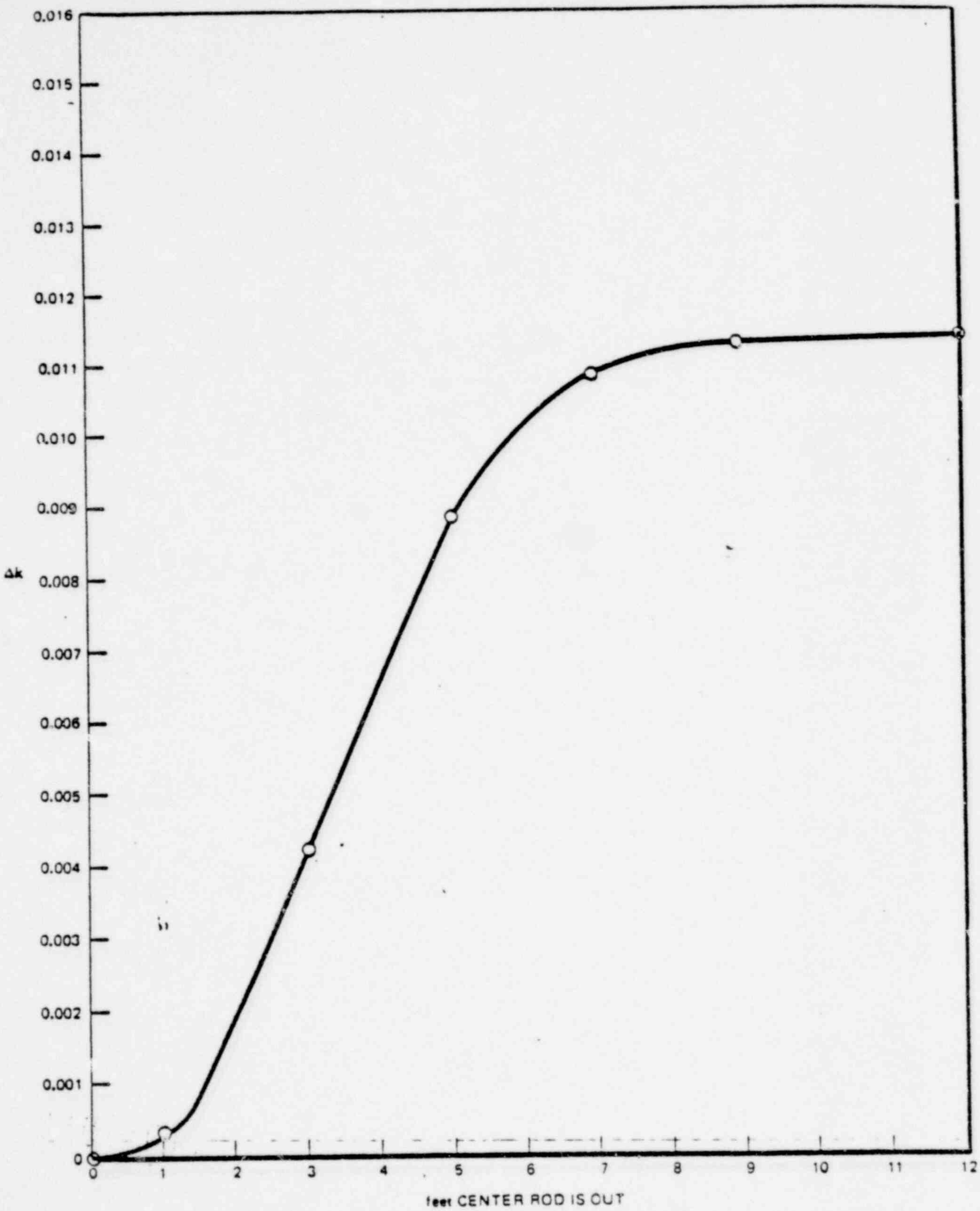

Figure 15.1.38-4. Design Basis Rod Drop Accident Shape Function (3.5 GWd/t)

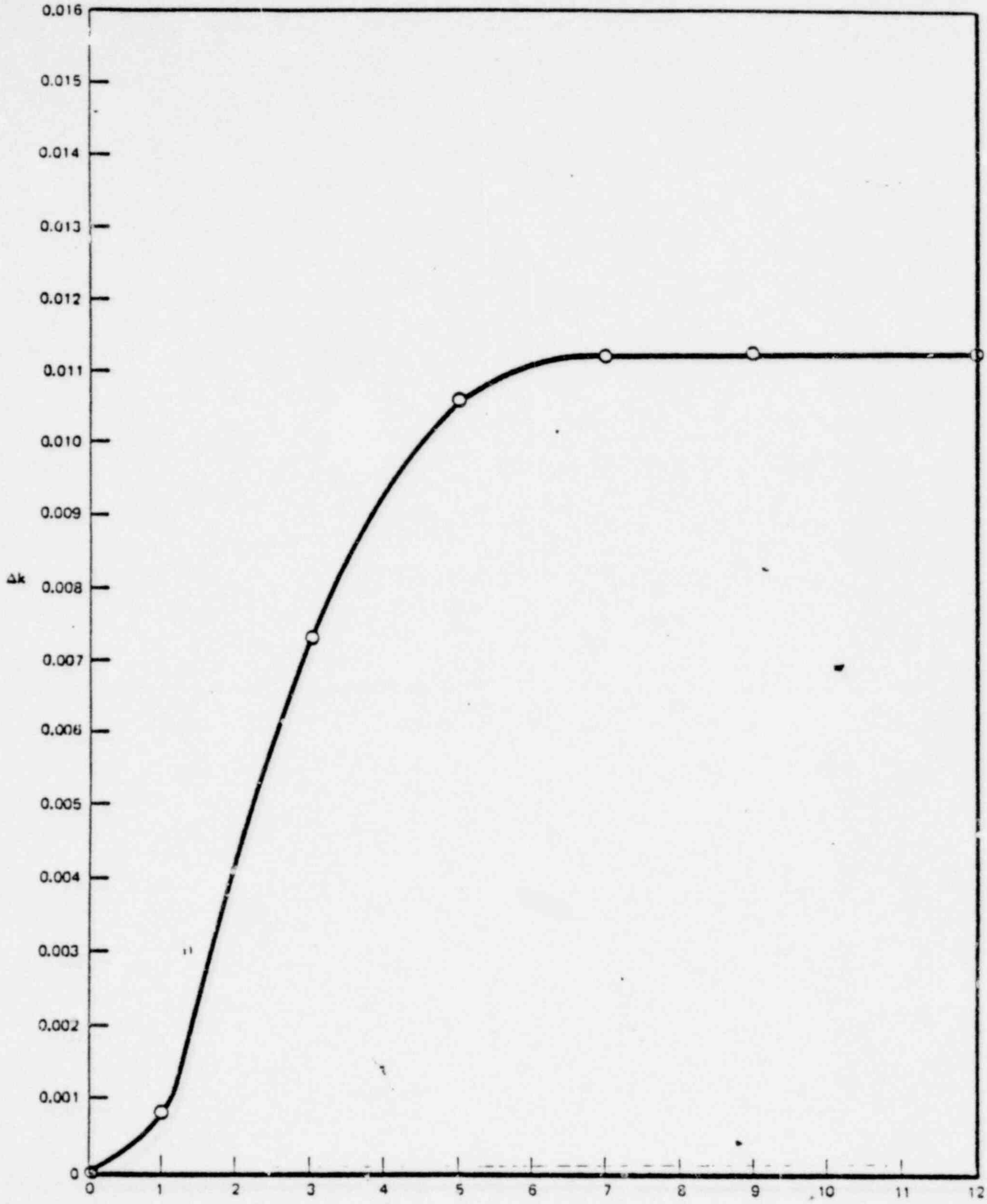

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Figure 15.1.38-5. Design Basis Rod Drop Accident Shape Function (⁻.4 GWd/t)