May 1974

U.S. ATOMIC ENERGY COMMISSION

REGULATORY GUIDE

DIRECTORATE OF REGULATORY STANDARDS

REGULATORY GUIDE 1.77

ASSUMPTIONS USED FOR EVALUATING A CONTROL ROD EJECTION ACCIDENT FOR PRESSURIZED WATER REACTORS

A. INTRODUCTION

Section 50.34, "Contents of applications: technical information," of 10 CFR Part 50, "Licensing of Production and Utilization Facilities," requires that each application for a construction permit or operating license provide an analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the potential risk to public health and safety resulting from operation of the facility. General Design Criterion 28, "Reactivity Limits," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, requires the reactivity control system to be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither result in damage to the reactor coolant pressure boundary greater than limited local yielding nor sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to impair significantly the capability to cool the core. General Design Criterion 28 also requires that these postulated reactivity accidents include consideration of the rod ejection accident unless such an accident is prevented by positive means.

This guide identifies acceptable analytical methods and assumptions that may be used in evaluating the consequences of a rod ejection accident in uranium oxide-fueled pressurized water reactors (PWRs). In some cases, unusual site characteristics, plant design features, or other factors may require different assumptions which will be considered on an individual basis. The Advisory Committee on Reactor Safeguards has been consulted concerning this guide and has concurred in the regulatory position.

B. DISCUSSION

The rate at which reactivity can be inserted into the core of a uranium oxide-fueled water-cooled power reactor is normally limited by the design of the control rod system to a value well below that which would result in serious damage to the reactor system. However, a postulated failure of the control rod system provides the potential for a relatively high rate of reactivity insertion which, if large enough, could cause a prompt power burst. For UO_2 fuel, a large fraction of this generated nuclear energy is stored momentarily in the fuel and then released to the rest of the system. If the fuel energy densities were high enough, there would exist the potential for prompt rupture of fuel pins and the consequent rapid heat transfer to the water from finely dispersed molten UO_2. Prompt fuel element rupture is defined herein as a rapid increase in internal fuel rod pressure due to extensive fuel melting, followed by rapid fragmentation and dispersal of fuel cladding into the coolant. This is accompanied by the conversion of nuclear energy, deposited as overpower heat in the fuel and in the coolant, to mechanical energy which, in sufficient quantity, could conceivably disarrange the reactor core or breach the primary system.

The Regulatory staff has reviewed the available experimental information concerning fuel failure thresholds. In general, failure consequences for UO_2 have been insignificant below 300 cal/g for both irradiated and unirradiated fuel rods. Therefore, a calculated radial average energy density of 280 cal/g at any axial fuel location in any fuel rod as a result of a postulated rod ejection accident provides a conservative maximum limit to ensure that core damage will be minimal and that both short-term and long-term core cooling capability will not be impaired.

For the postulated control rod ejection accident, a mechanical failure of a control rod mechanism housing is assumed such that the reactor coolant system pressure would eject the control rod and drive shaft to the fully withdrawn position.

A sufficient number of initial reactor states to completely bracket all possible operational conditions of interest should be analyzed to assure examination of upper bounds on ultimate damage. In areas of uncertainty, the appropriate minimum or maximum parameters relative to nominal or expected values should be used to assure a conservative evaluation. The initial reactor states should include consideration of at least the following:

Zero power (hot standby)	– Beginning of Life (BOL) and End of Life (EOL);
Low power	– BOL and EOL;
Full.power	– BOL and EOL.

The effects of the loss of primary system integrity as a result of the failed control rod housing should be included in the analysis. It should also be shown that failure of one control rod housing will not lead to failure of other control rod housings.

The approach that should be used in the radiological analysis of a control rod ejection accident is to determine the amount of each gaseous radionuclide released to the primary containment and, with this information in conjunction with the procedures set forth in Appendix B of this guide, to determine the radiological consequences of this accident for a pressurized water reactor.

C. REGULATORY POSITION

Acceptable assumptions and evaluation models for analyzing a rod ejection accident in PWRs are presented in Appendices A (Physics and Thermal-Hydraulics) and B (Radiological Assumptions) of this guide. By use of these appendices, it should be shown that:

1. Reactivity excursions will not result in a radial average fuel enthalpy greater than 280 cal/g at any axial location in any fuel rod.

2. Maximum reactor pressure during any portion of the assumed transient will be less than the value that will cause stresses to exceed the Emergency Condition stress limits as defined in Section III of the ASME Boiler and Pressure Vessel Code.[1]

3. Offsite dose consequences will be well within the guidelines of 10 CFR Part 100, "Reactor Site Criteria."

[1] Copies may be obtained from the American Society of Mechanical Engineers, United Engineering Center, 345 East 47th Street, New York, New York 10017.

APPENDIX A

PHYSICS AND THERMAL-HYDRAULICS

The assumptions described below should be applied in evaluating the physics and thermal-hydraulic behavior of the reactor system for a control rod ejection accident.

1. The ejected rod worth should be calculated based on the maximum worth rod resulting from the following conditions: (a) all control banks at positions corresponding to values for maximum allowable bank insertions at a given power level and (b) additional fully or partially inserted misaligned or inoperable rod or rods if allowed by operating procedures. Sufficient parametric studies should be performed to determine the worth of the most reactive control rod in each rod group for different control rod configurations, both expected and unexpected. The worth of single rods in rod groups should be evaluated during startup physics tests and compared with values used in the rod ejection analysis. The accident should be reanalyzed if the rod worths used in the initial analysis are found to be nonconservative. Calculated rod worths should be increased, if necessary, to account for calculational uncertainties in parameters such as neutron cross sections and power asymmetries due to xenon oscillations.

2. The reactivity insertion rate due to an ejected rod should be determined from differential control rod worth curves and calculated transient rod position versus time curves. If differential rod worth curves are not available for the reactor state of interest, conservatism should be included in the calculation of reactivity insertion through consideration of the nonlinearity in reactivity addition as the rod passes through the active core. The rate of ejection should be calculated based on the maximum pressure differential and the weight and cross-sectional area of the control rod and drive shaft, assuming no pressure barrier restriction.

3. The calculation of effective delayed neutron fraction (β_{eff}) and prompt neutron lifetime (ℓ^*) should be based on the well-known definitions resulting from perturbation theory, such as those described by Henry (Ref. 1), using available experimental delayed neutron data and averaging by the fraction of fission in the various fissionable materials. In cases where the accident is quite sensitive to β_{eff} (where the ejected rod worth $\geqslant \beta_{eff}$), the minimum calculated value for the given reactor state should be used. For smaller transients, conservatism in the value should include consideration of not only the initial power rise (which increases with decreasing β), but also the power reduction after the trip. Similar considerations should also be applied to determine an appropriately conservative value of ℓ^* to be used.

4. The initial reactor coolant pressure, core inlet temperature, and flow rate used in the analysis should be conservatively chosen with respect to their influence on the magnitude of the transient. Pressure and temperature are mainly significant with respect to their effect on the amount of reactivity inserted if there exists a positive moderator coefficient.

5. The fuel thermal properties such as fuel-clad gap heat transfer coefficient and fuel thermal conductivity should be conservatively chosen, depending upon the transient phenomenon being investigated. For conditions of a zero or positive moderator coefficient (usually at beginning of life), for example, high heat transfer parameters would reduce the Doppler feedback and increase any positive moderator feedback effects and hence tend to increase the magnitude of the reactivity transient. For a negative moderator coefficient, high heat transfer parameters could cause the magnitude of the transient to decrease if a given quantity of heat produces more feedback in the moderator than in the fuel. In the consideration of pressure pulses which may be generated, high moderator heating rates could cause significant pressure gradients to develop in the moderator channels. In computing the average enthalpy of the hottest fuel pellet during the excursion for power cases, low heat transfer would be conservative.

6. The specific heat of UO_2 has been determined experimentally and is a deterministic factor in the calculated amount of stored energy (enthalpy) in the fuel. Recommended values in the range of 25 to 902°C are the data reported by Moore and Kelly (Ref. 2). In the range of 900 to 2842°C, the data obtained by Hein and Flagella (Ref. 3), Leibowitz, Mishler, and Chasanov (Ref. 4), and Chasanov (Ref. 5) are recommended for the heat capacity of the fuel. These recommended values are for clean core conditions. Possible variation in the specific heat due to burnup should be investigated and appropriate values used, if necessary.

7. The moderator reactivity coefficients due to voids, coolant pressure changes, and coolant temperature changes should be calculated based on the various assumed conditions of the fuel and moderator using standard transport and diffusion theory codes. If no three-dimensional space-time kinetics calculation is performed, the reactivity feedback due to these coefficients should be conservatively weighted to account for the variation in their spatial importance in the missing dimension(s). If boric acid shim is used in the moderator, the highest boron concentration corresponding to the initial reactor state should be assumed.

8. The Doppler coefficient should be calculated based on the effective resonance integrals and should include corrections for pin shadowing (Dancoff correction). Calculations of the Doppler coefficient of reactivity should be based on and should compare conservatively

with available experimental data such as those of Hellstrand (Ref. 6). Since the Doppler coefficient reflects the change in reactivity as a function of fuel temperature, uncertainties in predicting fuel temperatures at different power levels should be reflected by conservatism in the applied value of the Doppler coefficient. If no three-dimensional space-time kinetics calculation is performed, the reactivity effect of spatially weighting the core average temperature rise in both the axial and radial directions should be calculated.

9. Control rod reactivity insertion during trip versus time should be obtained by combining the differential rod worth curve with a rod velocity curve based on maximum design limit values for scram insertion times. If the rod worth curve (reactivity vs. depth of insertion) is not obtained from a "true" representation (i.e., an x, y, z, t or an r, z, t calculation); the conservatism of the approximate calculation should be shown. The difference in the depth of insertion at zero power and at full power should be accounted for in calculating the available scram reactivity.

10. The reactor trip delay time, or the amount of time which elapses between the instant the sensed parameter (e.g., pressure or neutron flux) reaches the level for which protective action is required and the onset of negative reactivity insertion, should be based on maximum values of the following: (a) time required for instrument channel to produce a signal, (b) time for the trip breaker to open, (c) time for the coil to release the rods, and (d) time required before scram rods enter the core if the tips lie above the core-reflector interface.

11. The computer code used for calculating the transient should be a coupled thermal, hydrodynamic, and nuclear model with the following capabilities: (a) incorporation of all major reactivity feedback mechanisms, (b) at least six delayed neutron groups, (c) both axial and radial segmentation of the fuel element, (d) coolant flow provision, and (e) control rod scram initiation on either coolant system pressure or neutron flux.

12. The analytical models and computer codes used should be documented and justified and the conservatism of the models and codes should be evaluated both by comparison with experiment, as available, and with more sophisticated spatial kinetics codes. In particular, the importance of two- or three-dimensional flux characteristics and changes in flux shapes should be investigated, and the conservatism of the flux shapes used for reactivity input and feedback, peak energy deposition, total energy, and gross heat transfer to the coolant should be evaluated. Also, sensitivity studies on variations of the Doppler effect, power distribution, fuel element heat transfer parameters, and other relevant parameters should be included.

13. The pressure surge should be calculated on the basis of conventional heat transfer from the fuel, a conservative metal-water reaction threshold, and prompt heat generation in the coolant to determine the variation of heat flux with time and the volume surge. The volume surge should then be used in the calculation of the pressure transient, taking into account fluid transport in the system, heat transfer to the steam generators, and the action of the pressurizer relief and safety valves. No credit should be taken for the possible pressure reduction caused by the assumed failure of the control rod pressure housing.

14. The number of fuel rods experiencing clad failure should be calculated and used to obtain the amount of contained fission product inventory released to the reactor coolant system. It should be assumed that clad failure occurs if the heat flux equals or exceeds the value corresponding to the onset of the transition from nucleate to film boiling (DNB), or for other appropriate causes.

The margin to DNB is expressed in terms of a departure from nucleate boiling ratio (DNBR). The DNBR at any position in the hottest channel is the ratio of the DNB heat flux to the actual heat flux. The DNB heat flux should be evaluated using correlations based on recognized studies and experimental heat transfer DNB data. A minimum DNBR should be determined from the evaluation of the experimental data to ensure a 95% probability with a 95% confidence level that DNB has not occurred for the fuel element being evaluated. One example of a correlation which has been used to date is given by Tong (Ref. 7). The use of this correlation and the above probabilities and confidence level yields a minimum DNBR of 1.30. Other DNB or clad failure correlations may be used if they are adequately justified by analytical methods and supported by sufficient experimental data.

REFERENCES

1. A. F. Henry, "Computation of Parameters Appearing in the Reactor Kinetics Equations," WAPD-142, December 1955.

2. G. E. Moore and K. K. Kelley, "High Temperature Heat Contents of Uranium, Uranium Dioxide and Uranium Trioxide," Journal of the American Chemical Society, Vol. 69, p. 2105, 1949.

3. R. A. Hein and P. N. Flagella, "Enthalpy Measurements of UO_2 and Tungsten to 3260°K," GEMP-578, February 1968.

4. L. Leibowitz, L. W. Mishler, and M. G. Chasanov, "Enthalpy of Solid Uranium Dioxide from 2500°K to its Melting Point," Journal of Nuclear Materials, Vol. 29, pp. 356-358, 1969.

5. M. G. Chasanov, Argonne National Laboratory, Reactor Development Progress Report, ANL-7618, September 1969.

6. E. Hellstrand et al., "The Temperature Coefficient of the Resonance Integral for Uranium Metal and Oxide," Nuclear Science and Engineering, Vol. 8, pp. 497-506, 1960.

7. L. S. Tong, "Prediction of Departure from Nucleate Boiling for an Axially Non-Uniform Heat Flux Distribution," Journal of Nuclear Energy, Vol. 21, pp. 241-248, 1967.

APPENDIX B

RADIOLOGICAL ASSUMPTIONS

The assumptions given below should be applied in determining a conservative source term and subsequent transport of activity and resulting doses to the public for use in evaluating the radiological consequences of a control rod ejection accident.

1. The assumptions related to the release of radioactive material to the primary containment are as follows:

a. The case resulting in the largest source term should be selected for evaluation.

b. The nuclide inventory in the fuel elements potentially breached should be calculated, and it should be assumed that all gaseous constituents in the fuel-clad gaps are released.

c. The amount of activity accumulated in the fuel-clad gap should be assumed to be 10% of the iodines and 10% of the noble gases accumulated at the end of core life, assuming continuous maximum full power operation.

d. No allowance should be given for activity decay prior to accident initiation, regardless of the reactor status for the selected case.

e. The nuclide inventory of the fraction of the fuel which reaches or exceeds the initiation temperature of fuel melting (typically 2842°C) at any time during the course of the accident should be calculated, and 100% of the noble gases and 25% of the iodine contained in this fraction should be assumed to be available for release from the containment.

f. The effects of radiological decay during holdup in the containment or other buildings should be taken into account.

g. The reduction in the amount of radioactive material available for leakage to the environment by containment sprays, recirculating filter systems, or other engineered safety features may be taken into account, but the amount of reduction in concentration of radioactive materials should be evaluated on a case-by-case basis.

h. The primary reactor containment should be assumed to leak at the leak rate incorporated or to be incorporated as a technical specification requirement at peak accident pressure for the first 24 hours, and at 50% of this leak rate for the remaining duration of the accident.[1] Peak accident pressure is the maximum pressure defined in the technical specifications for containment leak testing.

i. Release of fission products to the secondary system should be computed by assuming that all fission products released from the fuel clad are uniformly mixed in the primary coolant volume.

j. The primary-to-secondary leak rate limitation incorporated or to be incorporated as a technical specification requirement should be assumed to exist until the primary system pressure falls below the secondary system pressure.

k. The release of fission products from the secondary system should be evaluated with the assumption of a coincident loss of offsite power.

2. Acceptable assumptions for atmospheric diffusion and dose conversion are:

a. The 0-to-8-hour ground-level release concentrations may be reduced by a factor ranging from one to a maximum of three (see Figure 1) for additional dispersion produced by the turbulent wake of the reactor building in calculating potential exposures. The volumetric building wake correction, as defined in Section 3-3.5.2 of Meteorology and Atomic Energy 1968 (Ref. 1), should be used only in the 0-to-8-hour period; it is used with a shape factor of 1/2 and the minimum cross-sectional area of the reactor buiding only.

b. No correction should be made for depletion of the effluent plume of radioactive iodine due to deposition on the ground or for the radiological decay of iodine in transit.

c. For the first 8 hours, the breathing rate of a person offsite should be assumed to be 3.47×10^{-4} m^3/sec. From 8 to 24 hours following the accident, the breathing rate should be assumed to be 1.75×10^{-4} m^3/sec. From 24 hours until the end of the accident, the rate should be assumed to be 2.32×10^{-4} m^3/sec. (These values were developed from the average daily breathing rate [2×10^7 cm^3/day] assumed in a report (Ref. 2) of ICRP.[2])

d. The iodine dose conversion factors are also given in Reference 2.

e. External whole body doses should be calculated using "infinite cloud" assumptions, i.e., the dimensions of the cloud are assumed to be large compared to the distance that the gamma rays and beta particles travel. "Such a cloud would be considered an infinite cloud for

[1] The effect on containment leakage under accident conditions of features provided to reduce the leakage of radioactive materials from the containment should be evaluated on a case-by-case basis.

[2] International Commission on Radiological Protection.

a receptor at the center because any additional [gamma and] beta emitting material beyond the cloud dimensions would not alter the flux of [gamma rays and] beta particles to the receptor." (Ref. 3) Editorial additions were made to the quotation so that gamma as well as beta emitting material could be considered. Under these conditions, the rate of energy absorption per unit volume is equal to the rate of energy released per unit volume. For an infinite uniform cloud containing χ curies of beta radioactivity per cubic meter, the beta dose in air at the cloud center is:

$$_{\beta}D'_{\infty} = 0.457\,\overline{E}_{\beta}\chi$$

The surface body dose rate from beta emitters in the infinite cloud can be approximated as being one-half this amount (i.e., $_{\beta}D'_{\infty} = 0.23\,\overline{E}_{\beta}\chi$). For gamma emitting material, the dose rate in air at the cloud center is:

$$_{\gamma}D'_{\infty} = 0.507\,\overline{E}_{\gamma}\chi$$

From a semi-infinite cloud, the gamma dose rate in air is:

$$_{\gamma}D'_{\infty} = 0.25\,\overline{E}_{\gamma}\chi$$

where

- $_{\beta}D'_{\infty}$ = beta dose rate from an infinite cloud (rad/sec)
- $_{\gamma}D'_{\infty}$ = gamma dose rate from an infinite cloud (rad/sec)
- $\overline{E}_{\beta}$ = average beta energy per disintegration (Mev/dis)
- $\overline{E}_{\gamma}$ = average gamma energy per disintegration (Mev/dis)
- χ = concentration of beta or gamma emitting isotope in the cloud (Ci/m^3)

f. The following specific assumptions are acceptable with respect to the radioactive cloud dose calculations:

(1) The dose at any distance from the reactor should be calculated based on the maximum concentration in the plume at that distance, taking into account special meteorological, topographical, and other characteristics which may affect the maximum plume concentration. These site-related characteristics must be evaluated on a case-by-case basis. In the case of beta radiation, the receptor is assumed to be exposed to an infinite cloud at the maximum ground-level concentration at that distance from the reactor. In the case of gamma radiation, the receptor is assumed to be exposed to only one-half the cloud owing to the presence of the ground. The maximum cloud concentration always should be assumed to be at ground level.

(2) The appropriate average beta and gamma energies emitted per disintegration, as given in the Table of Isotopes (Ref. 4), should be used.

g. The atmospheric diffusion model should be as follows:

(1) The basic equation for atmospheric diffusion from a ground-level point source is:

$$\chi/Q = \frac{1}{\pi u \sigma_y \sigma_z}$$

where

- χ = the short-term average centerline value of the ground-level concentration (Ci/m^3)
- Q = amount of material released (Ci/sec)
- u = windspeed (m/sec)
- σ_y = the horizontal standard deviation of the plume (meters) [see Figure V-1, Ref. 5].
- σ_z = the vertical standard deviation of the plume (meters) [see Figure V-2, Ref. 5].

(2) For time periods greater than 8 hours, the plume should be assumed to meander and spread uniformly over a 22.5° sector. The resultant equation is:

$$\chi/Q = \frac{2.032}{\sigma_z u x}$$

where

- x = distance from the point of release to the receptor; other variables are as given in paragraph g. (1), above.

(3) The atmospheric diffusion model[3] for ground-level releases is based on the information in the table below.

Time Following Accident	Atmospheric Conditions
0-8 hours	Pasquill Type F, wind speed 1 m/sec, uniform direction
8-24 hours	Pasquill Type F, wind speed 1 m/sec, variable direction within a 22.5° sector
1-4 days	(a) 40% Pasquill Type D, wind speed 3 m/sec (b) 60% Pasquill Type F, wind speed 2 m/sec (c) wind direction – variable within a 22.5° sector.

[3]This model should be used until adequate site meteorological data are obtained. In some cases, available information, such as meteorology, topography, and geographical location, may dictate the use of a more restrictive model to insure a conservative estimate of potential offsite exposures.

Time Following Accident	Atmospheric Conditions
4-30 days	(a) 33.3% Pasquill Type C, wind speed 3 m/sec (b) 33.3% Pasquill Type D, wind speed 3 m/sec (c) 33.3% Pasquill Type F, wind speed 2 m/sec (d) Wind direction – 33.3% frequency in a 22.5° sector.

(4) Figures 2(A) and 2(B) give the ground-level release atmospheric diffusion factors based on the parameters given in paragraph g.(3), above.

REFERENCES

1. D. H. Slade, ed., "Meteorology and Atomic Energy – 1968," TID-24190, Division of Technical Information, U.S. Atomic Energy Commission, July 1968.

2. Report of Committee II, "Permissible Dose for Internal Radiation," ICRP Publication 2, 1959.

3. D. H. Slade, *op. cit.*, Section 7.4.1.1.

4. C. M. Lederer, J. M. Hollander, and I. Perlman, "Table of Isotopes," Sixth Edition, University of California, Berkeley, Lawrence Radiation Laboratory.

5. F. A. Gifford, Jr., "Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion," Nuclear Safety, Vol. 2, No. 4, June 1961.