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UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

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REACTOR TECHNOLOGY MEMORANDUM No 8 - COMBUSTIBLE GAS CONTROL SYSTEMS EVALUATION GUIDELINES

The enclosed Reactor Technology Memorandum (RTM) sets forth guidelines with respect to minimum requirements for combustible gas control systems (CGCS) to be installed in power reactor facilities and such other facilities as may be appropriate. This information is intended to be used in evaluating the protection provided against failure of containment and other engineered safety features due to uncontrolled combustion of hydrogen gas mixtures following loss-of-coolant accidents.

This RTM addresses methods for predicting hydrogen and oxygen generation rates, limits for gas mixture compositions, radiological constraints to be considered for the use of purging as a combustible gas control system, and guidelines for evaluating the acceptability of a combustible gas control system as an engineered safety feature.

A CGCS has not yet been required for any operating nuclear facility nor have any of the systems proposed by applicants for future plants received final DRL approval. However, the safety concerns for the present generation of power reactors are sufficient to warrant early implementation of the requirements of this RTM upon its approval. An implementation program for operating reactor facilities and for those in the licensing process is being prepared.

Comments on this RTM are requested on or before July 11, 1969, in order that necessary revisions can be made prior to further distribution. The requirements of this RTM will be implemented only after a revision

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that considers comments is approved by the Director, DRL. A copy of any correspondence on this matter should be sent to C. W. Moon, Safety Systems Technology Branch, DRL.

Saul Levine, Assistant Director
for Reactor Technology
Division of Reactor Licensing

RT-457A
DRL:SSTB:WRJ

Enclosure:
RTM 8 - Combustible Gas Control Systems Evaluation Guidelines

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REACTOR TECHNOLOGY MEMORANDUM NO. 8

COMBUSTIBLE GAS CONTROL SYSTEMS

I. INTRODUCTION

It has been demonstrated analytically that flammable mixtures of hydrogen and oxygen gas can be generated within the containment of a large, water moderated power reactor in the period following a loss of coolant accident. Burning or combustion of these gases could result in damage to vital systems within the containment, or to the containment structure itself. This RTM establishes a requirement for installation of an appropriate Combustible Gas Control System (CGCS) in those facilities for which the potential for such an event exists, and provides guidelines for evaluation of such systems.

Hydrogen generation in the post-LOCA period may result from fuel clad metal-water reactions, corrosion, and radiolytic decomposition of aqueous emergency cooling solutions. Oxygen is also a product of radiolysis, and even in the case of an inerted containment, its generation can lead to combustible mixtures. For the present generation of light-water power reactors, the time interval which exists between a LOCA and the onset of flammable gas concentrations is governed primarily by the containment free volume.

The CGCS must operate to preclude flammable mixtures at any time subsequent to the accident, and must be capable of being started under DBA conditions in a time which conforms to predicted onset of combustible concentrations. As an engineered safety feature, a CGCS must meet presently accepted standards of quality and reliability.

The effects of all variables which govern radiolytic H_2 and O_2 generation are not sufficiently well known to allow accurate predictions to be made of their concentrations under conditions which might exist after a LOCA. There are similar uncertainties associated with H_2 generation by metal-water reactions and corrosion. Therefore, the guidelines established by this RTM are intended to be used in current licensing actions, and they shall be amended as more, pertinent information becomes available.

II. REQUIREMENT FOR CGCS

A CGCS shall be required if necessary to prevent damage to the containment or other engineered safety features from the uncontrolled combustion of flammable mixtures of gases that might be accumulated within the containment following a loss of coolant accident. The necessity for such a system in a given reactor facility shall be established by a conservative estimate of potential post-LOCA gas concentrations. This estimate shall be made using appropriate parameter values specified by this RTM.

III. EVALUATION GUIDELINES

A. Combustible Gas Sources

The following values shall be used in the determination of the combustible gas generation rates and accumulation within the reactor containment after a LOCA. Detailed bases for the selected values for each source are enclosed as Appendices.

1. Metal-Water Reaction (Appendix A)

   a. Assume that 5% of Zircaloy fuel clad undergoes metal-water reaction as a result of the design basis LOCA.

   b. Assume that each pound of Zirconium yields 7.9 ft^3 H_2 (STP).

2. Corrosion (Appendix B)

   Values are presented for aluminum only since it is a common structural material which under attack by the basic solutions proposed for PWR emergency cooling and spray systems (0.15 m NaOH - 0.28 m H_3BO_3), undergoes rapid corrosion with the evolution of relatively large quantities of H_2. (Exclusive consideration of aluminum here should not preclude efforts to identify other sources of H_2 and O_2 resulting from corrosion or other chemical reactions).

   a. Assume that aluminum corrodes at the following rates:

Condition	Corrosion Rate
(1) Subject to spray, basic (pH > 7) solution	0.20 in/year
(2) Immersed in basic solution	0.10 in/year
(3) Neutral or slightly acidic (5.5 < pH < 7)	0.02 in/year

   b. Assume that each pound of aluminum which corrodes yields 19.3 ft^3 H_2 (STP).

3. Radiolysis of Aqueous Solutions (Appendix C)

Radiolytic gas generation rates are determined by taking the product of the radiation energy absorption rate in the solution (ev/sec) and a G, or yield, value (molecules/100 ev). In the post-LOCA period, radiolytic decomposition of the emergency coolant solution is expected to result from two radiation sources; the reactor core, as the coolant passes through it; and the radioactive isotopes which become dissolved in the coolant itself.

a. Assume the following:

(1) Reactor Core: 10% of total core gamma-ray energy is absorbed by the coolant in the core region.

(2) Dissolved Isotopes: All the gamma-ray and beta-particle kinetic energy from 50% of the core radio-iodine inventory, and 1% of the total fission product inventory is absorbed in the coolant directly.

b. Assume the following G values:

(1) PWR - in core and solution:

$G(H_2)$ = 0.50 molecules/100 ev absorbed

$G(O_2)$ = 0.25 molecules/100 ev absorbed

(2) BWR - in core and solution *

(a) First 24-hr period after LOCA:

$G(H_2)$ = 0.25 molecules/100 ev absorbed

$G(O_2)$ = 0.13 molecules/100 ev absorbed

(b) Time after first 24-hr period:

$G(H_2)$ = 0.50 molecules/100 ev absorbed

$G(O_2)$ = 0.25 molecules/100 ev absorbed

*Low G values for a BWR are assumed for the first day based on data for pure and low impurity water. Beyond this time the coolant purity cannot be assured, and the higher value applies.

4. Other Gas Sources

Other sources of combustible gases may exist and should be identified and evaluated on an individual case basis. Examples of such sources are:

a. H_2 gas used as fuel for a flame recombiner, which through leakage, or incomplete combustion, may add to the net containment H_2 concentration.

b. For an inerted containment, in which a high H_2 concentration may develop following a LOCA, an air leak into the containment could result in a combustible mixture.

B. Gas Concentration Limits

Containment gas mixture compositions shall be controlled such that local concurrent concentrations of H_2 and O_2 do not exceed 4 v/o (volume percent) and 5 v/o, respectively. The containment volume average concurrent concentrations of H_2 and O_2 shall not exceed 3 v/o and 4 v/o, respectively.

C. Redundancy and Radiological Requirements

A CGCS must operate to maintain combustible gas concentrations inside the containment within specified limits, and must be capable of being started and operated under post-LOCA conditions. Many schemes for gas mixture control, employing different principles, are practicable. Techniques which have already been proposed by applicants are a flame recombiner to reduce the H_2 concentration, and a containment purge system by which the containment gas mixture is exhausted to the atmosphere and is replaced by fresh air.

The CGCS shall consist of one or more completely independent subsystems, each having redundancy of all active components and each being capable of performing the required gas control function. A CGCS operating on the purge principle is simple and reliable, and in the event of failure of the other CGCS subsystems, purge can always be considered as an acceptable alternative to the potential containment damage resulting from gas combustion. However, purge involves the intentional release of the radioactive containment atmosphere. The major consideration to be used in evaluating the number and the operating principle of proposed CGCS subsystems shall be estimated radiation dose to persons outside the exclusion zone as a result of CGCS operation, and possible failure. The basic requirement is that performance of a CGCS should result in offsite doses within the limits of 10 CFR 20, even with the failure of any single active component in the system. The limits of 10 CFR 100 are used in determining the depth of protection to be provided by system redundancy and diversity. The evaluation procedure is presented in the following diagram:

CGCS REQUIREMENT EVALUATION

CASE	D_p	$D_p + D_A$	MINIMUM CGCS REQUIREMENT No. of Subsystems	MINIMUM CGCS REQUIREMENT Acceptable Subsystem Types
1	$\leqslant$ 1.5 Rem Thyroid $\leqslant$ 0.5 Rem Whole Body	$<$ 300 Rem Thyroid $<$ 25 Rem Whole Body	1	1 PSS
2	$>$ 1.5 Rem Thyroid $>$ 0.5 Rem Whole Body	$<$ 300 Rem Thyroid $<$ 25 Rem Thyroid	2	1 NPSS + 1 PSS(BU)
3	$>$ 1.5 Rem Thyroid $>$ 0.5 Rem Whole Body	$>$ 300 Rem Thyroid $>$ 25 Rem Whole Body	3	2 NPSS + 1 PSS(BU)

Key:

D_p = Calculated radiation dose to persons outside the exclusion zone resulting from purging the containment to control combustible gases concentrations.

D_A = Calculated radiation dose to persons outside of the exclusion zone resulting from containment leakage after a Design Basis LOCA.

PSS = CGCS subsystem employing the purge principle.

NPSS = CGCS subsystem employing a physical principle other than purge.

PSS(BU) = CGCS subsystem employing the purge principle, but used only as backup in the event of failure of another subsystem.

A single CGCS subsystem employing the purge principle shall be acceptable if the calculated doses at the exclusion zone as a result of its operation are within the limits specified by 10 CFR 20 (CASE 1). In particular, the one-year-average dose at the exclusion zone boundary for this release shall be less than 0.5 rem, whole body, and 1.5 rem, thyroid.

If the calculated dose as a result of purging is greater than the 10 CFR 20 limits, but the combined accident and purge dose is within the guidelines of 10 CFR 100 (CASE 2), at least one CGCS subsystem using a principle other than purge shall be required. As a minimum, a single back-up purge subsystem shall also be required.

When the combined accident and purge dose exceeds the 10 CFR 100 guidelines, (CASE 3) at least two CGCS subsystems using principles other than purge shall be required. As a minimum a single backup purge subsystem shall also be required.

D. Determination of Gas and Radioactivity Concentrations

A system shall be available for the determination of the H_2, O_2, and radioactivity concentrations within the containment. The system shall consist of at least two subsystems, each of which meets design redundancy, and reliability standards equivalent to those established for the CGCS, and each of which is capable of independent performance of the required function.

For each subsystem, sufficient collection points within the containment shall be provided to allow a reasonable determination to be made of the average gas and radioactivity concentrations, and collection points shall be located in areas of suspected high local concentrations.

The time required for the determination of gas concentrations shall be commensurate with the conservatively predicted rate of combustible gas generation and accumulation. The accuracy and performance of gas concentration measuring devices, in the range of the gas concentration limits (3-4 v/o H_2; 4-5 v/o O_2) shall be demonstrated to be sufficient to allow concentration predictions to $\pm$ 0.2 v/o, of either gas under post LOCA operating conditions.

Subsystems which remove airborne radioactive material from the containment, shall be designed to accommodate conservatively predicted activity concentrations. Considerations should include radiation exposure to operators, radiation monitoring, the ability to isolate lines in the event of failure, and the effect of radiation on the accuracy of the concentration measurement.

E. Additional Design Requirements

The CGCS and the system for the determination of gas and radioactivity concentrations shall comply with the following additional design requirements:

1. Power

Each subsystem shall be capable of accomplishing its function when operating on normal (off-site) power or when operating on emergency (on-site) power.

2. Isolation

Fluid lines which penetrate the containment shall comply with the isolation valve requirements established in the General Design Criteria.

3. Structural Design

The system shall be designed as a Class I (seismic design) system (see RTM-3). The system shall be designed to withstand natural phenomena other than earthquakes (e.g., tornadoes, see RTM-1) in combination with other applicable forces except that these other natural phenomena need not be considered as occurring simultaneously with the loss-of-coolant accident.

4. Sharing

Systems, subsystems, and/or components may be shared between units for a multiple plant facility. Within a unit, sharing with other systems shall be allowed if it can be rigorously demonstrated: (a) that the quality standards of all systems are equivalent, and (b) that the full design potential of all systems required following an accident can be provided.

5. General

All components shall, as a minimum, be designed and fabricated to the latest applicable codes and quality assurance standards.

The system shall be designed to function in the post-accident environment and the effects considered should include pipe whipping, missiles from components, coolant blowdown effects, and the pressure, temperature, moisture, radioactivity and chemical conditions resulting from a loss-of-coolant accident.

To the extent practical redundant subsystems and/or components shall be placed in different locations to ensure that damage in a specific area will not impair the operability of more than one element.

F. Controls and Instrumentation

Provisions for automatic initiation of the CGCS need not be required, unless the predicted time to the onset of excessive combustible gas mixtures is so short (~1 hour) as to preclude assured manual start-up. System controls and instrumentation shall be designed to protection system standards and shall be adequate to determine the performance of the system, to indicate component failures, and allow for switching to non-failed subsystems. Redundancy requirements shall be similar to those for the CGCS system as a whole.

G. Failure Mode and Reliability Analyses

Comparative reliability analyses shall be used as one method of supporting the choice of a proposed CGCS design and to demonstrate the effect of proposed sharing on reliability. Failure mode analyses shall also be conducted to demonstrate that during operation, the CGCS can accommodate the failure of any single active component, and still fulfill its required functions.

H. Performance Demonstrations and Periodic Testing

Each component of a CGCS shall be tested to demonstrate its ability to meet its functional performance requirements within specified limits under environmental conditions which simulate, as nearly as possible, those which would conservatively be calculated to exist at its' location following a design basis LOCA. These include, as a minimum, conditions of temperature, pressure, relative humidity, containment spray, external radiation, and air borne or gaseous radioactivity.

Performance demonstrations should include operation under simulated accident conditions for periods of time in excess of those for which the component might be needed, and should indicate freedom from systematic failure modes which might result in the loss of redundant components of the system or subsystem.

An installed CGCS shall be periodically tested to demonstrate its continued ability to perform its design function. Such tests shall include operation of the entire system, and to as great a

degree possible shall simulate post-LOCA conditions. This requirement is not intended to specify that hazardous, or near hazardous gas mixtures be created for such tests, but that sufficient combustible gas input be simulated to demonstrate minimum acceptable CGCS performance.

The testing schedule for a CGCS shall, in general, conform to those for other vital engineered safety features, although more frequent tests may be required in the case of a system whose reliability has not been demonstrated.

APPENDIX A

METAL-WATER REACTIONS

I. INTRODUCTION

The three major sources of combustible gas evolution in event of a LOCA are clad-water reactions, corrosion, and radiolysis of the post-accident emergency coolant. This appendix discusses the extent of clad-water reaction that should be assumed in evaluating the acceptability of proposed combustible gas control system designs.

II. RECOMMENDATION

We recommend that 5% of the fuel clad be assumed to undergo a metal-water reaction as a result of a design basis loss of coolant accident, and that the hydrogen gas evolved by the reaction be considered to appear at the time of the LOCA.

III. DISCUSSION

It is generally agreed that the calculational models used by applicants (and reactor vendors) to predict ECCS performance are conservative. Such calculations indicate that even degraded ECCS performance is sufficient to restrict maximum clad temperatures to the range of 2000-2500 °F for a design basis LOCA, and to limit the concomitant clad metal-water reaction to a few tenths of a percent or less. Experimental results related to specific portions of the models in use tend to verify their conservative nature of these portions of the models.

Two major assumptions are made in analytical modeling to predict ECCS performance: (1) that the original core heat transfer geometry is retained during the LOCA, and (2) that the gross and local hydrodynamic behavior of the core during the blowdown and refilling transients can be conservatively predicted. It is obvious that these assumptions interact. It is also apparent that the accuracy of any predictions made by the models is dependent upon the validity of the assumptions.

Fuel clad perforations are predicted to occur extensively in a reactor core undergoing a LOCA. The exact nature of such perforations, by ballooning or splitting, has not been completely resolved experimentally. In either case, the perforation represents a local departure from the original core geometry and a possible flow restriction. A violent perforation may cause damage to adjacent fuel rods and local flow restrictions, as a result of a single perforation, could lead to additional failures of adjacent rods. Both of these events, as well

as other core geometry perturbations, provide mechanisms by which local failures can propagate to affect a larger area of the core. As reactor power densities increase and power distributions become more uniform the potential for failure propagation in high performance fuel regions will be increased. The flow restrictions and impaired cooling capacity associated with such failures are not considered in ECCS models.

Successive flow-starving and cooling of mechanically disturbed core regions could provide the initiation source for "chugging", or coolant flow oscillations. Such flow oscillations have been observed in scale model blowdown tests, but are not accounted for by ECCS performance models. Large scale flow anomolies may be caused by systems-related effects, such as steam-binding, to further cast doubt on predicted core cooling capabilities.

Experimental programs sponsored by the AEC (FLECHT, FHUST, and LOFT) and by the various applicants are underway which may resolve many of the areas of concern on the prediction of ECCS performance. However, the experimental data presently available, while extremely valuable, are not sufficient to verify an analytical model of the blowdown accident incorporating the many potentially significant interactions, which may effect core cooling and subsequent metal-water reaction predictions.

At the other end of the ECCS performance scale, there are estimates which indicate that full core melt, as a result of total ECCS failure, would be accompanied by some 25% to 50% clad metal-water reaction.

IV. CONCLUSIONS

When the performance of one engineered safety feature strongly influences the functional performance requirements of another, we believe that these performance requirements should be based on the assumption that the influencing system performance is degraded. The extent of degradation should be selected on the basis of known and assured performance capabilities. The performance of the ECCS strongly effects the amount of hydrogen that might be evolved from clad metal-water reactions during the LOCA. This hydrogen, in turn, serves as input in establishing the functional performance requirements for the CGCS. In view of the uncertainties concerning ECCS performance predictions, we conclude that, for the purpose of evaluating the acceptability of a CGCS, it is reasonable to assume that the ECCS performance is degraded to the extent that calculations of subsequent events with any additional degradation would be highly questionable. We believe that, for current designs, a core melt of 10 to 15% establishes such a limit. Further, we believe a metal-water reaction of about 5% is not an unreasonable value to associate with a core melt of this magnitude. On this basis we have recommended that a 5% clad-water reaction be assumed for the purpose of evaluating proposed combustible gas control systems.

APPENDIX B

ALUMINUM CORROSION

I. INTRODUCTION

Significant amounts of hydrogen can be generated by the corrosion of aluminum under post-accident conditions. Reasonably conservative aluminum corrosion rates should be assumed in order to predict this contribution to the post-accident hydrogen concentrations within the containment.

II. CORROSION RATES

The corrosion rates of several aluminum alloys have been measured for spray and immersion conditions, using basic solutions of the type proposed for PWR emergency cooling and containment spray systems (Reference 1). The corrosion rates were high (150-200 mil/year for spray, and 25-100 mil/year for immersion conditions). For the purpose of evaluating hydrogen generation due to corrosion of aluminum in post-LOCA situations in which the material is exposed to sprayed, basic solutions, we believe that a corrosion rate of 200 mil/year should be assumed. For aluminum totally immersed in basic solutions, we believe that a rate of 100 mil/year should be assumed.

The corrosion rate of aluminum in neutral and slightly acidic aqueous solutions ($5 < pH < 7$) is generally low, of the order of fractions of a mil/year to several mil/year (References 2 and 3). However, the rates may be increased by increasing the solution temperature or changing the chemical composition of the solutions. We believe that a corrosion rate of 20 mil/year should be assumed for neutral or slightly acidic solutions. This value is conservatively high, but is not restrictive.

Aluminum corrosion results in 19.3 ft^3 (STP) of hydrogen gas per pound of metal consumed. For thin aluminum sections, such as a reflective covering on insulation, corrosion-produced H_2 need only be considered until the section has been completely consumed.

III. REFERENCES

1. J. C. Griess and A. L. Bacarella, "Spray Solutions Corrosion Studies", ORNL-TM-2479 (1969) p. 74.

2. H. H. Uhlig, Corrosion and Corrosion Control, John Wiley and Sons, Inc. (1954) p. 299.

3. Reactor Handbook Vol. I, Materials (2nd Ed.), Interscience Publishers (1960) p. 497.

APPENDIX C

RADIOLYTIC GENERATION OF HYDROGEN AND OXYGEN

I. INTRODUCTION

Hydrogen and oxygen gas will be produced by radiolytic decomposition of emergency core cooling solutions in the post-LOCA period. Gas generation in the core region resulting from solution radiolysis induced by fission product gamma radiation is a certainty. Similarly, radiolytic gas generation will result from fission products which are released from the core and become dissolved, or suspended, in the emergency cooling solutions.

Reliable predictions of radiolytic gas production require the knowledge of the amount of radiation energy absorbed in the cooling solution and a G-value relating gas moles generated per unit of absorbed energy. In principle, the amount of energy absorbed can be calculated from basic data now available. G-values, on the other hand, are complex functions of many variables and must be empirically determined. Parameters known, or believed, to affect G-values include the linear energy transfer (LET) of the radiation in question; the chemical composition and pH of the solution; temperature, pressure, flow rate, and steam production rate in the solution; and the relative gas and liquid phase volumes of the irradiated systems (References 1, 2 and 3).

II. IN-CORE ENERGY ABSORPTION

Under the assumption that the reactor core retains its pre-accident geometry, the system in which the absorbed energy calculation must be made is fixed. Studies made by several PWR manufacturers verify that there is essentially no direct contribution to the absorbed energy in cooling solutions due to fission product beta radiation, which is absorbed in the fuel rods (References 4 and 5). The data of Bettet, et al, and Dudziak, can be used to obtain an adequate representation of the post-shutdown gamma-ray intensity as a function of gamma-ray energy, times of operation, and periods of shutdown (References 6 and 7). Selected representations of these results are given in Figures 1 and 2, for a 3000 MWt reactor with a 2-year operating history.

Estimates of the fraction of fission product gamma-ray energy absorbed by coolant in the core, using several calculational models, range from 0.03 to 0.11, (References 4, 5 and 8) with most calculations resulting in values of 5-7%. None of the calculations reviewed to date have provided a precise description of gamma-ray transport and absorption, nor have they included energy absorption in the coolant due to secondary effects such as Compton electron escape from the clad, or bremsstrahlung radiation

from the fuel due to electron energy loss there. These secondary effects would increase the fractional absorption of fission product energy by the coolant.

We believe that a value of 10% of the fission product gamma-ray energy should be assumed to be absorbed by cooling solutions within the core region, with no corrections made for the release of gaseous fission products from the fuel, as a reasonably conservative value for radiolysis evaluation purposes. Figure 1 gives the integrated gamma-ray energy release, from 10^3 to 10^8 seconds after shutdown, for a 3000 Mwt reactor.*

III. IN-SOLUTION ENERGY ABSORPTION

The amount of radiation energy absorbed by the emergency coolant due to fission products dissolved or suspended in it depends directly on the release fractions from the core and the degree of uptake of the released isotopes by the coolant.

For those isotopes which become dissolved or suspended in the coolant, a realistic yet conservative procedure is to assume that all the gamma-ray energy and that fraction of the beta decay energy appearing as kinetic energy of beta-particles (33% E_{max}) is absorbed by the coolant.

We believe that for the purpose of evaluating direct radiolytic gas production in the coolant, 50% of all radioiodines and 1% of all solid fission products initially in the core should be assumed to be dissolved, or suspended, in the coolant. These values are consistent with those used in the past for accident evaluation purposes (Reference 9). The iodine energy absorption may be taken directly from Curve A of Figure 2. Curve B of Figure 2 is the sum of the gamma-ray and beta-particle energy released by 1% of all fission products.

IV. G-VALUES

Values of $G(H_2)$ and $G(O_2)$ have been measured by ORNL, using several typical PWR spray solutions exposed to Co-60 gamma radiation in glass vials, with varying gas-to-liquid phase ratios (References 3 and 10). The latest data indicate $G(H_2)$ values in the range 0.2-0.5 molecules/100 ev, with stoichometric amounts of oxygen produced ($G(O_2)$ = 0.1-0.25 molecules/100 ev),

* The integrated energy release was calculated using an analytical model for the fission product gamma-ray energy release rate for the time period 10^2 sec to 10^7 sec. The predictions of the model and the data of References 6 and 7 for energy release rate are also shown in Figure 1.

except in the case of $Na_2S_2O_3$ solutions in which an O_2 scavenging reaction gave rise to apparent negative $G(O_2)$ values. A most significant result of this and other work is the dependence of gas production on the chemical composition of the irradiated solution (Reference 1 and 2). In general, G-values increase as the solution departs from "pure" water.

Since the ORNL work used only Co-60 gamma radiation, the effect of radiation "quality" was not determined. In a reactor accident situation, a large portion of the radiation energy absorbed by the cooling solution will be delivered by electrons of lower energy than those resulting from Co-60 gamma rays in water (Reference 11). It is expected that G-values would be somewhat higher for the lower energy electrons (higher LET), (Reference 12).

We believe that the following G-values are reasonably conservative and should be assumed for radiolysis evaluation purposes. The use of these values is recommended in view of the lack of complete experimental data, and the fact that the chemistry of emergency core cooling solutions may not be controlled during the post LOCA period (Reference 13).

Normal Values

$G(H_2)$ = 0.5 molecules/100 ev

$G(O_2)$ = 0.25 molecules/100 ev

Special Values (Use when "pure" water can be assumed, say 0-24 hrs after LOCA)

$G(H_2)$ = 0.25 molecules/100 ev

$G(O_2)$ = 0.13 molecules/100 ev

V. REFERENCES

1. A. O. Allen, The Radiation Chemistry of Water and Aqueous Solutions, D. Van Nostrand Co., Inc. (1961).

2. R. G. Sowden, "Radiolytic Problems in Water Reactors", Journal of Nuclear Materials 8 No. 1, 81-101 (1963).

3. H. E. Zittel, "Radiation and Thermal Stability of Spray Solutions", ORNL-TM-2479 (1969).

4. J. Sejvar, "Distribution of Fission Product Decay Energy in PWR Cores", WCAP-7319-L (1969). (Westinghouse Proprietary Report)

5. J. C. Allingham, et al, "An Evaluation of Purging As a Means of Controlling Post-Accident Reactor Building Hydrogen Concentration", BAW-10020 - Draft (1969).

6. M. E. Battat, et al, "Fission Product Energy Release and Inventory from Pu-239 Fast Fission", LA-3954 (1968).

7. D. J. Dudziak - private communication (Computer print-out of Fission Product Inventory Code for U-235 thermal fission, calculation used for LA-3954 U-Pu comparison).

8. M. Chatterton, DRL - Letter to M. Rosen, DRL, April 24, 1969.

9. J. J. DiNunno, et al, "Calculation of Distance Factors for Power and Test Reactors", USAEC-TID-14844 (March 23, 1962).

10. H. E. Zittel, "Radiolysis Studies", ORNL-TM-2230 (1968).

11. R. D. Evans, The Atomic Nucleus, McGraw-Hill Book Co., Inc. (1955) p. 693.

12. A. O. Allen, Loc. Cit. p. 49 et seq.

13. H. E. Zittel, ORNL - Letter to R. C. DeYoung, DRL, May 2, 1969.

FIGURE 1

FISSION PRODUCT GAMMA RAY ENERGY FROM REACTOR OPERATING AT 3000 MWT FOR 2 YEARS

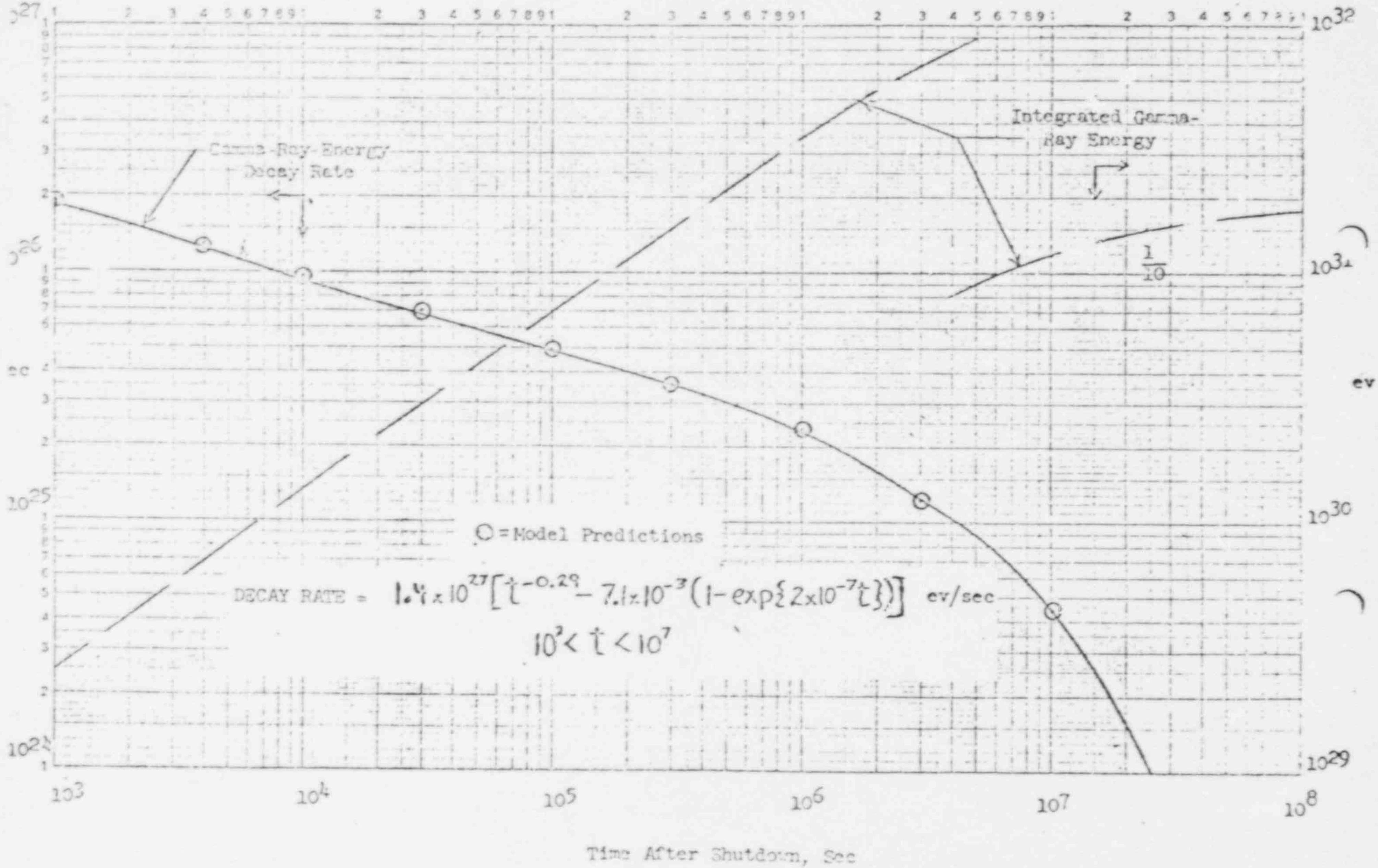
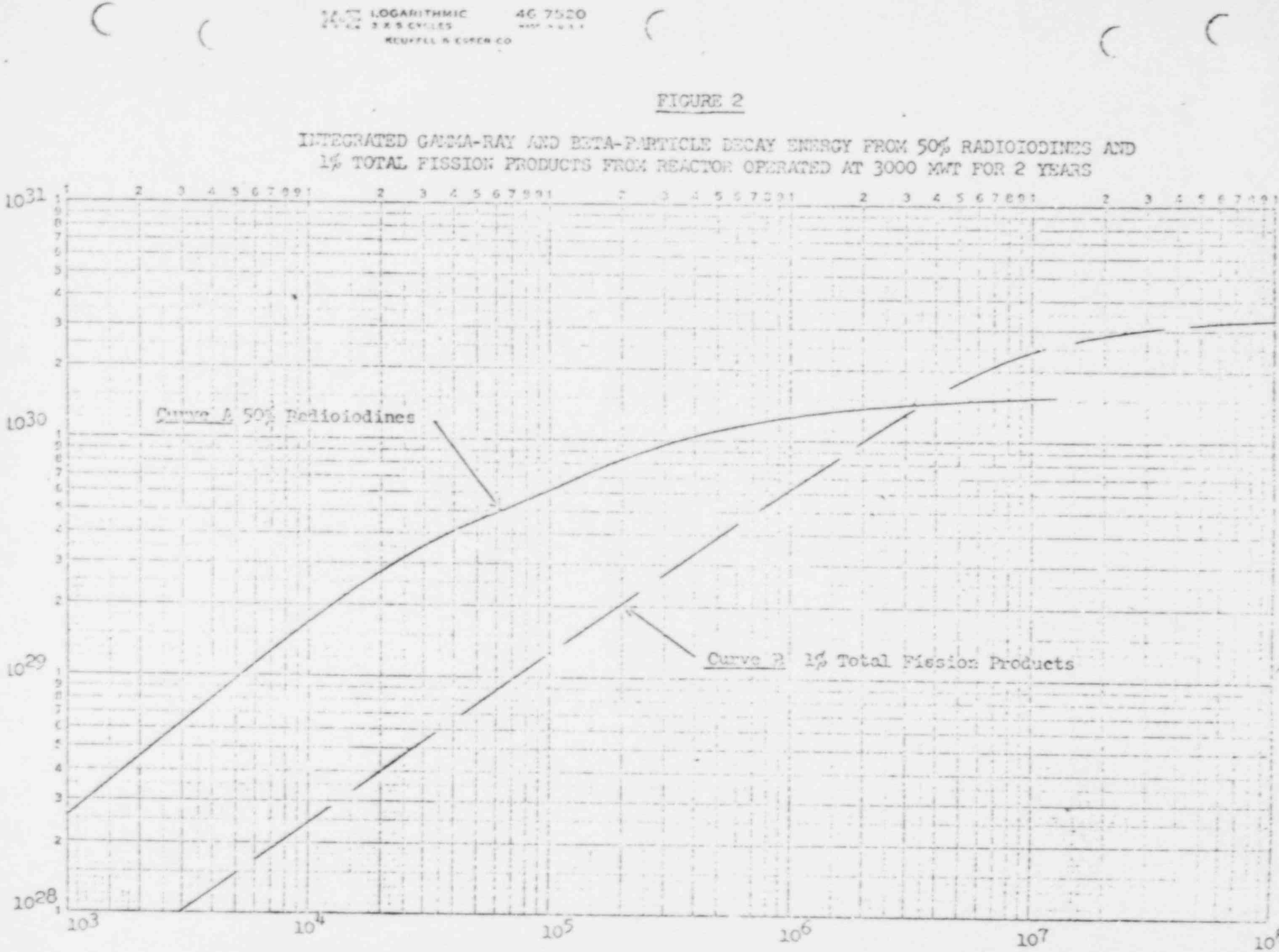


FIGURE 2

INTEGRATED GAMMA-RAY AND BETA-PARTICLE DECAY ENERGY FROM 50% RADIOIODINES AND 1% TOTAL FISSION PRODUCTS FROM REACTOR OPERATED AT 3000 MWT FOR 2 YEARS



UNITED STATES GOVERNMENT

Memorandum

TO : Those Listed Below   DATE: January 29, 1968

FROM : Saul Levine, Assistant Director for Reactor Technology, Division of Reactor Licensing

SUBJECT: REACTOR TECHNOLOGY MEMORANDUM NO. 4 -- EMERGENCY CORE COOLING SYSTEM EVALUATION GUIDELINES

DRL:N&STB:MR   RT-331

The attached RTM sets forth proposed DRL evaluation guidelines for emergency core cooling systems of large power water reactors. Also attached hereto are the bases for these guidelines. We will forward to you in a day or two, another document which contains a discussion of the system changes which should or could occur in PWR's and BWR's, as a result of implementation of the proposed guidelines. It is requested that any comments you care to make, as a result of reviewing these documents, be forwarded to Morris Rosen by February 13, 1968.

Attachment:
RTM-4

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EMERGENCY CORE COOLING SYSTEM

EVALUATION GUIDELINES

January 1968

LIST OF GUIDELINES

Emergency Core Cooling Definitions

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Emergency Core Cooling Definitions

The emergency core cooling system treated below comprises (a) the water storage facilities and the water and additives to be delivered to the core; (b) the piping, fittings, valves, pumps, heat exchangers, injection nozzles, spray headers and associated equipment required for transfer of the water from the storage facilities to the core; and (c) the piping, fittings, valves, pumps, heat exchangers and associated equipment required to transfer core heat to the intermediate heat sinks (e.g., service water system). The guidelines which follow also apply to the functional requirements of the controls and instrumentation required to actuate, monitor, and control emergency cooling of the core.

Systems intermediate to the ECCS and the ultimate heat sink are not covered by these guidelines. However, the plant design should provide at least two methods (intermediate sinks) for transferring heat to the ultimate sink for long-term cooling following a loss-of-coolant accident.

The electrical power systems required to drive the ECCS are not treated by these guidelines.

Active components are those whose mechanical operation is required to control the flow of water (e.g., pumps and all valves including check valves). All other components are defined as passive (e.g., locked valves, heat exchangers, pressure vessels, tanks, and piping).

A loss-of-coolant accident is any decrease in reactor coolant inventory greater than that which should be accommodated by either normal coolant charging systems or normal reactor shutdown, i.e., any coolant loss which requires actuation of the ECCS and reactor shutdown. The normal coolant makeup systems should be designed to cope with a range of small, anticipated leaks, such as failed pump seals and valve packings and broken instrument lines, without the aid of the ECCS. An orderly shutdown of the reactor following these small leaks should then prevent any core damage.

The core, core supports, control rods, control rod drives, and other reactor internals should be designed to accommodate the maximum blowdown forces and related phenomena resulting from a loss-of-coolant accident in combination with other applicable loads without impairing the ability of the ECCS to protect the core in accordance with these guidelines. More specific design requirements for these components will be provided in other documents.

1 - ECCS Functional Requirements

In the event of a loss-of-coolant accident, the functions of the ECCS are to (a) limit the peak clad temperature to well below the clad melting temperature, (b) terminate the temperature transient before the core geometry

necessary for core cooling is lost, (c) limit the fuel clad-water reaction to less than 1 percent of the total fuel clad mass, and (d) reduce the core temperature and remove core heat until the core will remain covered without recirculation and replenishment of coolant. The system shall be designed to perform these functions for all sizes and locations of breaks in the reactor coolant boundary up to and including the instantaneous double-ended rupture of the largest coolant pipe (this break range is herein referred to as the break spectrum).

2 - Margin Requirements

For small breaks the ECCS capability shall include deliberate margin in performance above that which is calculated to be required; this margin may decrease with increasing break size. The margin provided shall be in addition to allowances in the system capability to accommodate such anticipated factors as leakage from components, core bypass flow, spillage from a ruptured pipe, and delay in system initiation. The calculation of required system performance shall identify those parameters which significantly influence the results and shall demonstrate that these parameters have been suitably treated in the calculation to assure that the functional requirements are met.

3 - System Requirements

The ECCS shall supply coolant to the core for two conditions: (a) short-term cooling (until the recirculation mode of ECC is established), and (b) long-term cooling (until ECCS operation is no longer required).

For short-term cooling at least two subsystems shall be provided, each capable of accomplishing the ECC functions. If a short-term subsystem is based on a stored energy concept (one requiring no external controls, signals, or power for its operation), only a single such subsystem need be provided if it has sufficient redundancy. For long-term cooling at least two completely independent subsystems shall be provided, each capable of accomplishing the ECC function.

Short-term subsystems shall provide coolant injection to both the top and the bottom of the core. Long-term subsystems shall provide the following capabilities: (a) coolant injection to the bottom of the core, and (b) either distributed coolant injection to the top of the core or a cavity around the reactor vessel which will quickly be flooded to a height above the top of the core.

In evaluating short-term subsystems, only active component failures need be considered to assure that a single failure cannot disable both subsystems; failures of check valves to operate need not be considered provided the design of the subsystem permits frequent testing of the operability of these valves. In evaluating long-term subsystems, both active and passive component failures shall be considered to assure that a single failure, by any mechanism, cannot disable both subsystems.

Each of the ECC subsystems shall be capable of accomplishing the ECC function when operating on normal (off-site) power and when operating on emergency (on-site) power. An exception to this requirement can be a reactor coolant makeup system (e.g., BWR feedwater or PWR charging) which is required to be running whenever the reactor coolant is at operating temperature and pressure and which is capable of operating on normal power following a loss of coolant accident. Such a system can be considered as one of the short-term ECC subsystems if it complies with the remainder of the guidelines, herein, and if the second subsystem has active component redundancy.

4 - Redundancy Requirements

The ECCS shall have sufficient redundancy to permit surveillance and repair of the principal components (e.g., pumps, turbine drive units, heat exchangers, and stored energy tanks) without decreasing the number of short- and long-term subsystems required by Guideline 3 and without decreasing the performance capability of these subsystems. An exception to this requirement is that one of two subsystems may be incapacitated for a period of up to 48 hours for repair of components provided that during this period the emergency power source is continuously operated and the remaining subsystem is tested for operability.

Redundant subsystems shall be located in different compartments and protected individually to insure that flooding confined to a specific area will not impair the operability of more than one of these subsystems.

exception to 2-loop plts.

A stored energy subsystem shall have sufficient redundancy to accommodate spillage from the reactor coolant system break without decreasing the performance capability of the subsystem.

All emergency core cooling components located within the reactor vessel shall be completely redundant.

5 - Isolation Requirements

Each line which carries emergency core coolant into the containment shall include at least one isolating check valve inside the containment barrier and one automatically actuated isolation valve exterior to the containment. Lines which carry emergency core coolant out of containment (such as those lines from sumps or suppression pools) shall include automatically actuated isolation valves on

each side of the containment barrier. Where isolation valves inside containment are considered to be impracticable, these outflow lines shall have special isolation provisions (e.g., double containment of the lines and the exterior isolation valve, etc.).

Actuation of the isolation provisions shall be completely automatic and shall be initiated by signals of suitable diversity and redundancy (e.g., flow, radioactivity, sump level, etc.). Indication of coolant loss from the ECCS shall be provided in the reactor control room so that the operator can ensure that the failed subsystem was automatically isolated and the second subsystem was automatically started. ECCS components located exterior to the reactor containment shall be housed in a structure which permits venting of releases through iodine filters in the event of ECCS leakage.

Each emergency coolant delivery line which connects with the reactor coolant system shall include a check valve which shall be located as close as practical to that connection. There shall be no branch lines or valves in the reactor coolant pipes between the reactor vessel and the point where an emergency core cooling system connects to the reactor coolant system.

6 - Sharing Restrictions

Except for the ultimate heat sinks which may be shared, no feature or component of the ECCS in one reactor facility on a multiple plant site shall be shared with any system in another facility. Within a facility, no sharing shall be allowed between the ECCS and other systems unless it can be rigorously demonstrated: (a) that the quality standards of the other systems are equivalent to those of the ECCS, (b) that the full design potential of all systems required following an accident can be provided, and (c) by frequent testing that there is a high degree of assurance that the systems and components will perform their ECCS function satisfactorily when required.

7 - Reliability and Failure Mode Analyses

Comparative reliability analyses shall be used as one method of supporting the choice of a proposed ECCS design and to demonstrate the effect of proposed sharing on reliability.

Failure mode analyses shall be conducted to demonstrate that no failure of a single active component can disable both short-term subsystems, and that no single failure of an active or passive component can disable both long-term subsystems.

8 - Design and Quality Assurance

All components of the ECCS as a minimum shall be designed and fabricated to the latest applicable codes and to the same quality assurance standards as the primary coolant system.

The design of the system shall be such that there is a high degree of confidence that it will not significantly increase the potential for leakage from the primary coolant system. Any source of stored energy in the ECCS shall be so designed that leakage from the primary coolant system will not jeopardize the integrity of the stored energy container.

The system shall be designed so that the stored coolant cannot freeze and so that the boron additive cannot precipitate.

9 - Environmental Requirements

The ECCS shall be capable of performing its function considering the effects of the simultaneous combination of (a) the maximum postulated earthquake, (b) the loss-of-coolant accident, and (c) other applicable loads. The systems shall be designed to withstand other natural phenomena (e.g., tornadoes, hurricanes, etc.) in combination with other applicable forces except that these other natural phenomena need not be considered as occurring simultaneously with the loss-of-coolant accident. The ECCS shall be designed to function in the post-accident environment and the effects considered should include possible loss of component function, vessel, and other component displacements, pipe whipping, missiles from components, coolant blowdown forces, and pressure, temperature, moisture, radioactivity and chemical conditions, resulting from such an accident.

10 - Actuation and Control Requirements

The primary mode of actuation for the ECCS shall be automatic, and actuation shall be initiated by signals of suitable diversity and redundancy. Provisions shall also be made for manual actuation, monitoring and control from the reactor control room.

11 - Testing Requirements

All active components used in the ECCS shall be proof-tested before installation under conditions and for time periods as close as practical to the most severe operating conditions to which they may be subjected in their lifetime (component proof tests). Each subsystem shall be designed so that its functional operability can be demonstrated, as installed and with appropriate reactor pressure simulation, prior to reactor operation (preoperational system tests).

The design shall also provide the capability for periodically demonstrating that the system will function properly when an accident signal is received; i.e., it shall be demonstrated that pumps and valves operate on normal and emergency power and water pressure and flow are as designed when the plant is operating (periodic system surveillance). When the plant is shutdown for refueling the system shall be tested for delivery of coolant to the vessel. The design of the system shall include provisions for maximum practical visual inspection when the system is being tested.

BASES FOR ECCS GUIDELINES

January, 1968

1 - ECCS Functional Requirements

The function of the emergency core cooling system (ECCS) is to cool the core in the event of a loss of coolant from the reactor vessel. The break sizes postulated for design of the ECCS range from small leaks, which should be accommodated by normal reactor makeup systems, to the unlikely, double-ended instantaneous rupture of the largest pipe in the primary coolant system.

Peak clad temperatures should be limited to less than the melting temperature to ensure intact, coolable fuel elements. The peak temperature is further restricted to being significantly lower than the melting temperature because of uncertainties in the mechanical and chemical behavior of the cladding near melting (3370°F for zirconium) and uncertainties in the calculation of the temperature transient. As additional heat transfer and clad material data become available it may be possible to enumerate a limit on the peak temperature and this limit may be near the melting temperature. Programs in government and industry are being conducted to determine the behavior of cores at high temperatures with emergency core cooling. However, little data has as yet been published concerning the typical geometries and the typical heating and cooling techniques for zirconium clad temperatures in excess of 1800°F. At the construction permit review an applicant should be allowed to extrapolate from the published data in setting an allowable peak temperature limit. However, the applicants should be required to make commitments to perform tests, prior to applying for an operating license, to confirm the predicted behavior of the core and the ECCS at the calculated peak clad temperature.

The ECCS must limit the clad temperature before geometry changes prohibit cooling of the core. Such geometry changes might be coolant channel blocking by swelling or perforation of fuel elements and severe deformation or failure of vessel internals and core support structures at high temperatures. This limiting of geometry changes by temperature effects is in addition to the requirement that the core and internals withstand the blowdown forces in combination with other applicable loads.

The one percent metal-water reaction limit is intended to restrict the maximum energy content of the total clad mass. The concern is that the energy released from a clad-water reaction will further raise the clad temperature (self-heating) which in turn increases the reaction rate-- an autocatalytic effect which could rapidly lead to clad melting. Since the zirconium-water reaction is temperature dependent and occurs even at the normal operating temperatures of large power reactors (to a negligible and permissible extent), it is impossible to require no clad-water reaction One percent clad-water reaction was chosen as being a negligible mass to undergo oxidation with then only negligible energy additions to the clad. As core cooling data from high temperature tests with zirconium clad elements become available and as analytical methods are verified, it may be possible to raise this clad-water reaction limit.

The reaction of zirconium clad channel boxes (BWR) with water is not mentioned in the guideline. The channel boxes should not become as hot as the fuel elements because the boxes have no internal energy generation (decay heat). Consideration of the channel-water reaction with the clad-water reaction effectively decreases the calculated percent metal-water

reaction for the core. That is, the channel zirconium mass is about the same as the clad zirconium mass and the channel temperature is generally lower than the clad temperature. Clad melting is the phenomenon of concern in setting a metal-water reaction limit, and the amount of clad-water reaction is most descriptive of the clad energy and temperature.

In addition to limiting the clad temperature transient, the ECCS must remove the core stored energy, thereby reducing the core temperature to essentially the temperature of the coolant, and then must maintain this temperature until the core decay heat is negligible (no boil-off hence no need for recirculation and replenishment of coolant) or until the core is dismantled in recovery operations.

2 - Margin Requirements

The ECCS should meet the functional limits with margins and these margins may vary with break size. For example, for the smaller, more easily controlled loss of coolant accidents there should be enough temperature margin to limit clad perforations (peak clad temperature less than 1500°F for zirconium) and other gross reactor damage. This margin can be provided by increases in such design parameters as flow rate and time to initiate flow. Such protection should prevent any radioactivity release form the containment.

As the postulated break size increases, the blowdown and core heatup rates increase to the point that it is difficult to prevent clad perforations. On the other hand, rapid breaks of large pipes are very unlikely. Therefore, the intent is to allow designs that provide little or no margin (with respect to the functional limits) at the upper end of the break spectrum (instantaneous double-ended rupture of largest pipe). The mission of the ECCS to terminate

and control the core temperature transient would be accomplished and radioactivity releases from the containment could be held within acceptable limits.

Conservative analytical techniques should be used to determine the performance capability and margins provided by a proposed ECCS. However, overly conservative analyses have, in the past, often led to gross assumptions regarding the blowdown and core heatup phases of a loss of coolant accident. Therefore, the intent is to encourage realistic or state-of-the-art analyses of the loss of coolant accident. Analytical solutions at the conservative end of any range of uncertainty can then be obtained by identifying the parameters which most significantly influence the calculations and assuring that properly conservative assumptions have been made regarding these parameters.

The ECCS capabiltiy and margin should be evaluated after allowances are made for emergency coolant which, for any anticipated reason, does not reach the core. If, for example, a subsystem injects through reactor coolant lines, allowances must be made for possible spillages of this type and credit for isolation can be taken in determining the margin.

3 - System Requirements

The ECCS function is divided into two categories, short term and long term, to distinguish the kinds of subsystems and the kinds of failures peculiar to these two modes of emergency core cooling.A short term subsystem is required to operate rapidly following a loss of coolant accident, but the total time of operation is short. This time varies with break size but it is generally less than several hours. Failures of passive components are unlikely in this short time span and need not be considered in evaluating

short-term cooling capabilities. On the other hand, failures of active components (e.g., a pump fails to start or a valve fails to open) must be considered in designing and evaluating these subsystems which are required to be quick acting.

Accumulator subsystems provide a highly reliable short term supply of emergency coolant because they store the energy needed for operation and they do not depend upon the starting of power supplies and pumping units. Because of the high reliability of this method of short-term coolant makeup only one such subsystem is required. The failures of check valves, herein defined as active components, need not be considered in evaluating the capability of short-term subsystems. However, the check valves should be tested frequently to assure their reliability.

Long term subsystems may be required to operate over extended time periods (e.g., dismantling of a reactor following a major loss of coolant accident might not be possible until several months after the accident). In this time span failures of active and passive components by wear and fatigue, or failures caused by a severe natural phenomenon, (earthquake, etc.) are possible. Therefore, for long term cooling, two completely independent subsystems are required. The redundancy of subsystems permits continuation of long term cooling in the event of failure of any component, no matter what the failure mechanism. The physical independence of the two subsystems (e.g., no shared pipe headers unless double valving is provided) ensures that a single failure would disable only one of the subsystems.

The assurance gained by requiring the ECCS to employ several cooling principles (e.g., top and bottom injection) can be illustrated by consideration of several postulated loss of coolant accidents. First, in the unlikely event

of a break in the reactor coolant boundary below the bottom level of the core, recovering of the core by flooding within the vessel might not be possible. Then, for example, either distributed coolant injection from above the core, which can accomplish the ECC function without core flooding, or a system with the capability of rapidly recovering the core by flooding a special cavity around the vessel, should be provided. Second, an emergency coolant injection system which floods the core from below may be unable to cope with a large pressure buildup above the core. In a pressurized water reactor such a pressure hangup could be caused by steam generation within the core following a break in a cold leg of the primary loop or by release of high energy steam into the primary system from the secondary side of a failed (during blowdown) steam generator. Coolant injection from above the core to enhance mixing of low enthalpy coolant and high enthalpy steam, thereby increasing the pressure decay rate, should be provided.

All of the subsystems required to provided emergency core cooling over the break spectrum should be capable of operating on normal plant power and on emergency power. Emergency power can be supplied by steam turbines for short term cooling and by quick-starting diesel generators for either short or long term cooling. Accomplishment of the ECC function is thus insured in the event a loss of coolant accident coincides with a loss of normal (off-site) power.

It may be necessary to require that the emergency power source be capable of supplying several ECC subsystems simultaneously. For example, in the case of a Westinghouse PWR both the high head safety injection subsystem (short term) and one of the residual heat removal subsystems (long term) should be started simultaneously upon indication of a loss of coolant accident. Then if the break

is small the HHSIS will be running to provide makeup and to aid the depressurization of the vessel, or if the break is large the RHRS will be running to begin coolant recirculation after the accumulators have delivered and the vessel pressure has decayed.

A normally running coolant makeup system can be credited as one of the two short term subsystems if it complies with the remainder of these guidelines (including design for seismic loads), even if it has no emergency power source. Its operability is continuously demonstrated for long periods of time and in the event of a loss of coolant accident is would be unnecessary to start pumps and open valves. However, the second short-term subsystem must have an emergency power source and active component redundancy. Thus the loss of off-site power and the failure of an active component cannot prevent short-term cooling.

4 - Redundancy Requirements

Redundancy of certain components in the ECCS is required to enable the reactor plant to stay in service, in compliance with these guidelines, in the event one of these components must be isolated for repair. With no redundancy the isolation of a component is some subsystem may violate the two subsystem requirements or it may decrease the performance capability of that subsystem.

The exception to this requirement recognizes that there is a compromise to forcing the reactor to be subjected to a shutdown transient because of an inoperable ECCS component. That is, depending upon one subsystem rather than two for a specified short period of time is preferable to forcing a plant shutdown. For increased assurance that the remaining subsystem can accomplish its ECC functions, the subsystem will be tested and the emergency power source will be run continuously during the period that one subsystem is incapacitated.

The time limit is a matter for technical specifications on the individual plants, but a time on the order of 48 hours appears sufficiently long to accomplish repairs and it is not unreasonably long from the standpoint of accident probability over the life of the plant.

Component redundancy for single failures is inherent in the two subsystem ECCS concept. That is, either subsystem, by itself, can accomplish its appropriate emergency core cooling functions. The two short term subsystems provide sufficient redundancy to accomodate a single active failure and still accomplish the short term ECC functions. The two completely independent long term subsystems provide sufficient redundancy to accomodate a single failure, active or passive, and still accomplish the long term ECC functions.

Component redundancy within the reactor vessel is required for increased assurance that the component function (e.g., spray distribution by a core spray sparger) will be provided. This extra assurance is necessary because components within the vessel could be subjected to more severe blowdown loads than components located outside the vessel.

5 - Isolation Requirements

Pipes which penetrate containment are potential avenues for the release of radioactivity following a loss of coolant accident. Such lines should be designed to standards comparable to the containment design standards (see Guidelines 8 and 9). Nevertheless, failures of any of the components in these lines, including pumps and heat exchangers exterior to the containment and pipes which penetrate a sump or a suppression pool, are to be considered in the failure mode analyses required by these guidelines.

An automatic-actuation isolation capability for all ECCS penetrations of containment is required. For those lines which carry coolant from outside to inside containment, a check valve on the inside, not necessarily close to the containment liner, and an automatically actuated isolation valve on the outside are required. Lines which carry coolant from a sump or suppression pool to the outside of containment present a more difficult problem. Because motor operated valves on the inside of containment could be subjected to a severe environment (radioactivity and flooding following a loss of coolant accident), it may be difficult to provide valves in this position. Another solution is to require special designs (e.g., sealed pipe sleeves or bellows sealing to provide a double containment, etc.) for the sump or suppression pool lines up to and including remote actuation isolation valves located just outside containment. In addition to automatically isolating one subsystem, it is required that the second subsystem be automatically started to insure continued core cooling.

The detection of ECCS leakage external to containment is important if the resultant doses to the public are to be kept within 10 CFR Part 100. Indirect leak detectors, in the form, perhaps of flow meters or flow pressure indicators, would allow rapid indication of a gross failure thus enabling the rapid, automatic isolation of the failed subsystem. Development of leak detection schemes such as sump level, particulate monitors, etc., should be required in order to provide similar protection in the event of smaller ECCS leaks. The ECCS components located external to containment are to be housed in a structure which permits filtering of releases from an ECCS leak before venting to the atmosphere. Guidance as to the calculation of doses from a postulated ECCS failure will be provided in a separate RTM. These doses when summed with those attributable to containment system leakage should not exceed 10 CFR Part 100.

Isolation valves on ECC delivery pipes which connect to the reactor coolant system are required in order to protect low pressure subsystems from high pressure primary coolant and to ensure that failures in the ECCS cannot cause a loss of coolant accident. Check valves are an acceptable means of providing this protection. Branch lines or valves between the ECCS injection points and the reactor vessel are not allowed in order to eliminate potential routes for bypass flow or leakage during emergency coolant delivery.

6 - Sharing Restrictions

Sharing of systems or components between the ECCS in one facility and any other system in any other facility is not allowed. This permits continued

operation, in compliance with these guidelines, of facilities on a multi-plant site regardless of the conditions (accident, normal operating, or maintenance) of any other facility on the site. This removes the problem of deciding whether to shutdown two or more plants simultaneously (an action utilities would protest regardless of adminstrative rules) because of off-normal conditions in just one plant causing the reduction of ECC capability in other plants. A possible exception to this restriction may be the sharing of two cooling water storage facilities on a three plant site.

Sharing of components between the ECCS and other systems in one facility is allowed only if the functional performance of the ECCS is guaranteed. With such sharing the reactor operator may have increased control of all engineered safety features thus enabling him to choose that combination of operating systems which best ensures the safety of the plant, based on his firsthand knowledge of post-accident conditions.

Sharing of redundant components between ECC subsystems is implicitly allowed by these guidelines. For example, two low head safety injection subsystems each employing one full capacity pump may perhaps share a third full capacity redundant pump for repair purposes. Final judgment of such sharing will have to be based upon consideration of the individually proposed ECCS designs and upon evaluation of the applicant's failure mode analysis for such a system.

7 - Reliability and Failure Mode Analyses

Comparative reliability (availability) analyses are to be performed for various system configurations to demonstrate reasonable optimization of the

proposed ECCS. An ultimate goal should be to attach absolute reliability numbers to the proposed systems, but at present there are insufficient data on these systems and components operating under the proper conditions. Therefore, comparative reliability analyses, which do not require absolute reliability numbers, of the most promising configurations are to be required in the interim until sufficient reliability data can be collected from operating nuclear power plants.

Where components are shared with systems that perform other functions, analyses should be performed to demonstrate that there is an insignificant difference in the reliability of the ECCS with and without the proposed sharing of components. These analyses will in large part be the basis for the judgment of compliance with Guideline 6.

The required failure mode analyses will serve to demonstrate compliance of a proposed system with the physical independence requirements of Guideline 3 and the isolation requirements of Guideline 5.

8 - Design and Fabrication Requirements

The intent of this guideline is to ensure the quality of design and fabrication required to enable the ECCS to accomplish its mission. For example, components that contain pressurized water might be designed according to the ASME code, controls and instrumentation might be designed to applicable IEEE codes, and components which contain water with spray additives, because of sharing between an ECCS and the containment spray system, should be designed to standards which consider corrosion. The design and fabrication standards of

the individual vendors and applicants should be reviewed for each ECC component to determine equivalency with the intent of this guideline and with the quality assurance standards of the reactor coolant system.

ECC temperatures are required to be above the freezing temperature of water to insure the delivery of emergency coolant during the winter, and above the boron precipitation temperature to insure boron solubility. As more is learned about the thermal shock associated with emergency coolant injection to the reactor vessel, this temperature limit may be increased (e.g., if coolant temperature is a sensitive parameter in such considerations).

9 - Protection Requirements

Although it is difficult (and even misleading) to define mechanisms for the instantaneous, double-ended pipe rupture, it is conceivable that if such a highly improbable event were to occur, it would be during a time of maximum postulated earthquake. Thus, it is required that the earthquake and loss of coolant loads and other applicable loads (e.g., structural loads on components) be appropriately summed, and that components be designed to withstand the combined loads within a limit that ensures operation of the ECCS following this accident. For assessing the ability of the system to perform adequately under these circumstances, consideration should be given to, at least, the following: degrading of the system by loss of components, pipe whipping and missiles from any part of the plant, coolant blowdown forces, displacement of all reactor components including the vessel, and the anticipated pressure, temperature, moisture, radioactivity, and chemical conditions.

Reactor components located inside the containment would be protected from other natural phenomena such as tornadoes or hurricanes. This guideline requires that all ECCS components, including those outside containment, must be similarly protected. Because it is less likely that other natural phenomena will cause a loss of coolant accident, and because degradation of the ECCS by such a combination of events would be less severe than the combination considered above, the simultaneity of other natural phenomena and a loss of coolant accident need not be considered.

The requirement to individually protect the independent subsystems from flooding is necessary to insure double coverage for long term cooling. That is, part of the reason for two subsystems is to provide a backup system in the event a single subsystem is lost. Thus, the subsystems must be separated and independently protected to insure that they cannot be simultaneously flooded.

10 - Actuation and Control Requirements

Automatic actuation should be provided because the loss of coolant accident can proceed rapidly once a break has occurred. For example, in the event of an instantaneous double-ended pipe rupture, pumping systems and diesel generators may be required to operate within about 30 seconds after the break. This is not sufficient time for a reactor operator to assess the accident conditions and take the proper actions. The small breaks place less stringent timing requirements on the ECCS, but because of severe pressure upon the operator to make the right decision in a time span of several minutes, it is preferable to remove the element of human error and require automatic actuation. The

manual actuation, monitoring and control provisions in the reactor control room enable the operator to control long-term cooling based on his firsthand knowledge of the accident conditions.

11 - Testing Requirements

The best way to prove a system design is to test it. The testing program should thoroughly investigate the behavior of the ECCS under conditions as close as practical to the most severe conditions envisioned, and the program should provide for continuing periodic testing of the system.

The required test program is three-fold:

(1) Component proof tests - A representative sample (one may be enough) of each type of active component proposed for use in the ECCS should be tested under the most severe operating conditions (e.g., time, temperature, humidity, pressure, corrosion, and radioactivity) to which that component may be subjected.

(2) Preoperation system tests - The ECCS should be designed so that it can be tested as installed prior to reactor operations. Although radioactivity and temperature are difficult, if not dangerous, to simulate, pressure simulation may, perhaps, be more easily and safely provided for these tests.

(3) Periodic system tests - The design of the ECCS should provide for functional tests of all active components during normal plant operations. During plant shutdown for refueling each subsystem should be tested for delivery of coolant to the vessel.

In addition, it is required that the capability be provided for visual inspection of ECC components when the system is being tested to check those components for leakage.

Decker

OPTIONAL FORM NO. 10
MAY 1962 EDITION
GSA GEN. REG. NO. 27

5010-107

UNITED STATES GOVERNMENT

Memorandum

TO : All Technical Personnel
Division of Reactor Licensing

DATE: April 29, 1968

FROM : Peter A. Morris, Director P. A. Morris
Division of Reactor Licensing

SUBJECT: REACTOR TECHNOLOGY MEMORANDA

DRL:ADRT:SL RT-631

The attached Reactor Technology Memorandum #1 on Tornado Considerations is forwarded for your use in the review of power reactor facilities. Additional RTM's will be issued on a continuing basis to provide guidance for the review of other areas. Their development will require all technical personnel to contribute to the review of draft material to assure that the evolving standardization of this new technology is done expeditiously and well.

This document has established design bases and review requirements for a difficult and complex area in engineering design. Definitions of additional areas affecting tornado design will be provided later as indicated in the RTM. Consistent use of this RTM, and others to be issued later, by DRL personnel will do much to improve the quality and efficiency of our reviews.

Attachment:
RTM-1

cc: H. L. Price, REG, w/att.
C. K. Beck, REG, w/att.
M. M. Mann, REG, w/att.
R. L. Doan, REG, w/att.
J. A. McBride, DML, w/att.
E. G. Case, DRS, w/att. (10)
L. D. Low, CO, w/att. (10)

A/1

8105290362 1p.

OPTIONAL FORM NO. 10
MAY 1962 EDITION
GSA GEN. REG. NO. 27
5010-107

UNITED STATES GOVERNMENT

Memorandum

TO : All Technical Personnel
Division of Reactor Licensing

DATE: April 10, 1968

FROM : Saul Levine, Assistant Director for Reactor Technology
Division of Reactor Licensing

SUBJECT: REACTOR TECHNOLOGY MEMORANDUM NO. 1 -- TORNADO CONSIDERATIONS

DRL:C&CTB:RCDeY RT-603

The attached RTM sets forth the DRL technical position with respect to minimum requirements for the consideration of tornadoes and their effects that are to be used in safety evaluations of power reactor facilities and such other facilities as may be appropriate.

Attachment:
RTM-1

cc: H. L. Price, REG, w/att.
C. K. Beck, REG, w/att.
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APPROVED: Peter A. Morris
Peter A. Morris, Director
Division of Reactor Licensing

8105290434 8pp.

April 10, 1968

REACTOR TECHNOLOGY MEMORANDUM NO. 1

TORNADO CONSIDERATIONS

I. INTRODUCTION

Minimum requirements concerning tornado considerations to be taken into account in the design of nuclear facilities are provided by this memorandum. The requirements were developed using the basic approach that:

(1) The facility design should be such that the design tornado or missiles associated with it should not affect vital structures, systems, and components so as to cause an accident releasing radioactivity to the environs in excess of 10 CFR 20 limits.

(2) Except for those items provided to contain and process airborne radioactivity under post-accident conditions, the facility should also be designed to prevent the design tornado or missiles associated with it from affecting vital structures, systems and components so as to interfere with necessary consequence limiting functions following occurrence of the design basis loss-of-coolant accident. These exceptions are considered warranted in view of (a) the low probability* of a severe tornado strike on the facility simultaneous with or during a reasonably short period of time following a design basis loss-of-coolant accident, (b) the high probability for extensive dilution and dispersion of airborne radioactivity released into a tornadic atmosphere, and (c) the reduction in dose rates due to radioactive decay during the time interval between the occurrence of the accident and the tornado.

II. MINIMUM REQUIREMENTS

(1) **Location**

The effects of tornadoes should be taken into consideration in the design of vital structures, systems, and components for a nuclear facility located:

(a) On any site within the continental United States east of the 109° West Longitude Line, or

(b) On any other site where the probability for experiencing a tornadic event is greater than about 0.01 in 40 years.

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* The probability of a tornado hitting the facility within a time period of a few days is 10^{-4} or less.

(2) Design Tornado - Characteristics

Unless it can be rigorously demonstrated that lesser values pertain to a given site the minimum values of characteristics for the tornado considered for design purposes should be:

(a) Tangential velocity - 300 mph

(b) Transverse velocity - 60 mph

(c) Pressure drop - 3 psi in 3 seconds*

(3) Design Tornado - Occurrence**

The design tornado should be assumed capable of occurring at any time except that the simultaneous occurrence with the design basis accident or with any other limiting site related event such as an earthquake or flood need not be considered for design purposes.

(4) Load Combinations

Vital structures, systems, and components capable of being subjected to tornado forces should be designed for the loads induced by such forces, including missile loads, in combination with applicable functional design loads.

(5) Stress Limits

Stresses due to the selected load combinations should not exceed 90% of yield stress in steel nor 75% of ultimate stress in concrete.

(6) Missiles - Types

The types of missiles to be taken into account should be selected on the basis of an analysis of representative potential missiles capable of being generated from ground level and applicable elevated locations by the design tornado.

(7) Design Margins

The margins of safety provided in the design should be determined by defining the upper limit tornado wind speeds that the facility could be subjected to without a resultant accident releasing activity in excess of 10 CFR 20 to the environment, or without damage to vital structures, systems, or components such that essential post-accident safety functions would be negated.

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* These values are related to the wind velocities by tornado theory.

** Tornado considerations should not be factored into dose calculations associated with design basis accidents.

III. ITEMS BE FURNISHED AT A LATER DATE

The following will be provided at a later date:

(1) Recommended design basis missiles.

(2) Definitive guidance on how tornado loads are to be applied.

(3) A specific list of necessary consequence limiting functions as discussed in I (2) above.

(4) Technical bases for the RTM

IV. APPENDIX

Appendix A provides a tabulation of tornado data provided in applications representing 34 sites.

TORNADO DATA FROM SAFETY ANALYSIS REPORTS

Plant	Tornado Interval years	Transverse Velocity mph	Tangential Velocity mph	P, psi	Missiles	Stress Limits Steel, % Yield	Stress Limits Conc., % Ult.	Remarks
San Onofre								No mention of tornadoes
Conn Yankee								"Tornado probability very small"
Malibu								No mention of tornadoes
Oyster Creek								Data given in Amendment No. 9
Nine Mile Pt.								No mention of tornadoes
Dresden II			300					
Brookwood								No mention of tornadoes
Millstone Pt.	1250		300					"Safe shutdown assured"
Indian Pt.								No mention of tornadoes
Dresden III			300					
Turkey Pt. 3&4	5000		337	2.25	Plank, 2 ton car		100	"Safe shutdown before, during, or after a Florida tornado"
Quad-Cities 1&2	1250		300	1.18	Pole(r=7"),1 ton car	100	100	Blowout panels at 0.25 psi.
Palisades	"Rare"	40	300	3.0	Pipe(r=1.5"),4"x12" Planks, 2 ton car, 20 ton flatcar, 120 ton locomotive	90	85	
Browns Ferry 1, 2, & 3	5880		300					
H. B. Robinson			300	3.0	2 x 4 Timber	120		"Accept damage and shutdown for repairs"

TORNADO DATA FROM SAFETY ANALYSIS REPORTS

Plant	Tornado Interval years	Transverse Velocity mph	Tangential Velocity mph	P, psi	Missiles	Stress Limits Steel, % Yield	Conc., % Ult.	Remarks
Monticello	2000		.00	2.0	Pole(r=7"),1 ton car	90	85	Shutdown and make necessary repairs to equipment.
Pt. Beach 1&2	3030	60	300	3.0	Plank, 2 ton car	90	85	
Ft. St. Vrain		50	300	0.377/sec	In progress			
Oconee 1,2,&3		40	300	3.0	Pole, 1 ton car	100		
Vermont Yankee	1040		300		2 x 4 timber	90	85	Blowdown panels 0.25 psi.
Diablo Canyon								No mention of tornadoes
Peach Bottom 2&3	2600		300	1.18	Pole(r=7"),1 ton car	100	100	
Surry 1&2					1500 lbs at 550 mph			"...orderly shutdown should tornado strike."
Ft. Calhoun			350	3.0				Expert total revision later.
Indian Pt. 3								No mention of tornadoes
Three Mile Island	0.7 in 25 mile radius		300	3.0	Utility Pole	100		
Pilgrim	1700		300	3.0	Plank,2 ton car	100		Steel framework to 300 mph on upper reactor bldg. Blowout panels.
Zion 1&2	800 to 1600	60	300	3.0	Pole (r=4")	100	85	"Safe shutdown for winds in excess of 300 mph"
Cooper			300					"Safe shutdown assured in 300 mph winds"
Easton								No mention of tornadoes

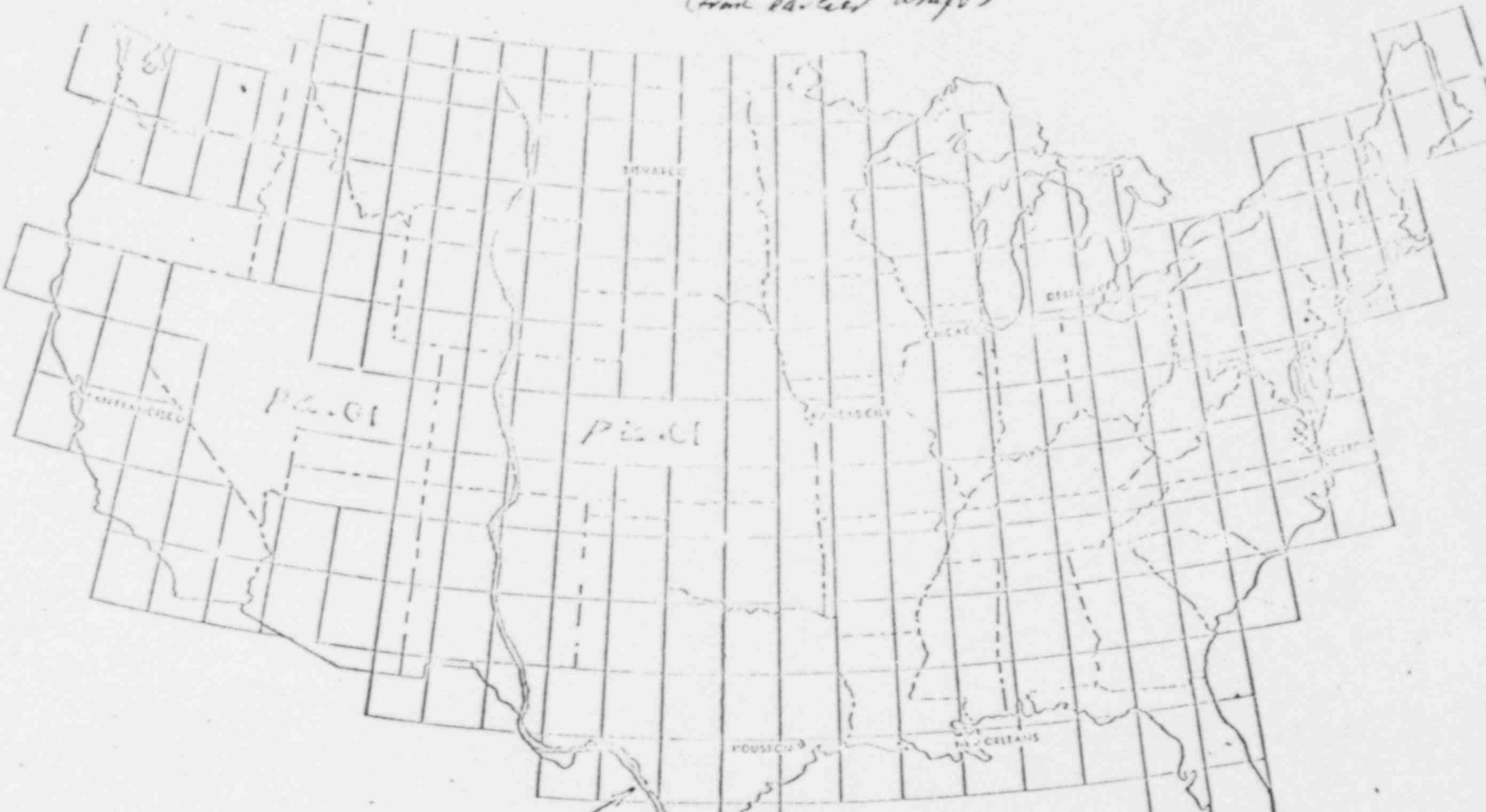
APPENDIX A (Cont'd.)

TORNADO DATA FROM SAFETY ANALYSIS REPORTS

Plant	Tornado Interval years	Transverse Velocity mph	Tangential Velocity mph	P, psi	Missiles	Stress Limits Steel, % Yield	Stress Limits Conc., % Ult.	Remarks
Crystal River 3	1210 to 2350		300	3.0	Utility Pole,1 ton car			
Kewaunee	1750 to 3030	60	300	3.0	Plank, 2 ton car	1.33 allowables		
Prairie Island	1250	60	300	3.0	Plank, 2 ton car	90	85	
Bolsa Island Two Units								No mention of tornadoes in "site" volume.

Probability of Tornado Occu[illegible]

Not part of Tornado Rtr
(from earlier draft)



Klecker

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

May 21, 1968

Roger S. Boyd, Assistant Director
for Reactor Projects, DRL
THRU: Saul Levine, Assistant Director
for Reactor Technology, DRL

STRESS LIMIT IN CONCRETE FOR TORNADO LOADS

Reactor Technology Memorandum No. 1 on Tornado Considerations, dated April 10, 1968, has been approved by the Director, Division of Reactor Licensing. The stress limit for concrete as specified in the approved RTM differs from that specified in previous drafts. The limit had been specified as 85% of ultimate; it is now specified as 75% of ultimate.

The 75% limit is strongly supported by our seismic design consultants. We believe it is not inconsistent with the use of the 85% limit generally accepted for combined normal plus extreme environmental plus accident loads. We believe that where, for example, the large earthquake and design basis accident loads are combined, the low probabilities associated with such a combination provide a measure of conservatism that warrants use of the higher limit for concrete. In addition, the 75% limit for concrete is more consistent with the 90% of yield stress limit specified for steel in RTM No. 1.

We believe the 75% limit should be factored into our evaluations as expeditiously as possible and suggest that the initial implementation be made for the Donald C. Cook facility. The use of a 75% limit was discussed with this applicant at a recent technical meeting without a strong reaction against its use.

We also intend to use the lower limit on all future CP evaluations.

for Richard C. DeYoung, Chief
Containment & Component Technology Branch
Division of Reactor Licensing

DRL:C&CTB:RCDeY
RT-670

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UNITED STATES
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WASHINGTON, D.C. 20545

June 11, 1969

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THRU: P. A. Morris, Director P. A. Morris
Division of Reactor Licensing

REACTOR TECHNOLOGY MEMORANDUM NO. 6 -- CONTROL ROOM DESIGN CONSIDERATIONS

The enclosed RTM sets forth proposed guidelines with respect to minimum requirements for control room design considerations.

This information is intended to be used in safety evaluations of power reactor facilities and such other facilities as may be appropriate. Comments on this RTM are requested on or before July 11, 1969, in order that necessary revisions can be made prior to further distribution. A copy of any correspondence pertaining to this RTM should be sent to C. W. Moon, Safety Systems Technology Branch, Division of Reactor Licensing.

R. C. DeYoung, for
Saul Levine, Assistant Director
for Reactor Technology
Division of Reactor Licensing

RT-391A
DRL:I&PTB:ODP

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RTM-6

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Issued: June 11, 1969

REACTOR TECHNOLOGY MEMORANDUM NO. 6

CONTROL ROOM DESIGN CONSIDERATIONS

I. INTRODUCTION

The purpose of this RTM is to provide minimum requirements which should be taken into account in the evaluation of the control room design for a nuclear facility against Criterion 11, Part 50, General Design Criterion for Nuclear Power Plant Construction Permits. The requirements were developed using the basic approach that:

1. The control room should be designed to allow occupancy during all accidents which have been analyzed for the facility up to and including the design basis accident.

2. If access to the control room is lost, it shall be possible to shut the reactor down and maintain it in a safe condition from a location(s) outside the control room.

II. MINIMUM REQUIREMENTS

1. Radiation Protection

The control room shall be so designed as to provide adequate radiation protection for personnel within the limits defined by 10 CFR Part 20. As established by 10 CFR 20, the present exposure limit is 3 rem whole body dose in any calendar quarter for individuals in a restricted area. This protection shall be designed so as to permit access, even under accident conditions, to equipment in the control room or other areas as necessary to shut down and maintain safe control of the facility. The exposure limit for the operating personnel during a nuclear incident should not exceed 5 rem whole body dose in any calendar year.

2. Fire Protection

The control room design shall be such as to minimize the possibility of fire. Continuing occupancy where possible should be provided for in the case of control room fire or smoke. The control room building components, finish materials and furnishings shall be noncombustible. Combustible supplies

such as logs, records, procedures and manuals should be limited to the amounts required for plant operation. Fire fighting equipment including fire extinguishers and breathing apparatus should be available to the control room.

3. Evacuation of Control Room

In the event that it becomes necessary to evacuate the control room, it shall be possible to shut the reactor down and maintain it in a safe condition from a location(s) outside the control room.

(a) It should be assumed that, during normal plant operation with all plant equipment operable, access to the control room is lost for a relatively long time.

(b) The facility should be examined to assure that hot shutdown from full power can be accomplished from outside the control room in a relatively short time of the order of an hour or so. The applicant should provide a general plan and show that adequate instrumentation and control are available to allow the plant to be safely placed in hot shutdown from outside the control room.

(c) The facility should be further examined to assure that without necessarily adding any equipment, existing equipment, instrumentation, panels, etc., can be manipulated (including opening panels, jumpering wires, etc.) in order to achieve cold shutdown in a period of time not to exceed several days The applicant should provide a general plan and show that it is feasible to safely bring the plant to cold shutdown from outside the control room.

(d) The facility should be examined to establish the length of time that it can be easily maintained in a hot standby condition from outside the control room. This period of time should exceed that in (c) above.

III. ITEMS TO BE FURNISHED AT A LATER DATE

The following will be provided at a later date:

(1) Control room ventilation system radiation protection requirement.

(2) Control room lighting.

(3) Control room communications.

(4) Technical bases for the RTM.