066

July 19, 1973

Mr. P. Marcus, Research Assistant
Environmental Awareness Center
University of Wisconsin
1201 West Dayton Street
Madison, Wisconsin 53706

Dear Mr. Marcus:

This is in answer to your request for information on the nuclear safety criteria applied by the Advisory Committee on Reactor Safeguards to nuclear generating facility construction. In its review of proposed facilities, there are many standards, codes, and guides which are normally considered by the Committee in formulating its recommendations to the AEC. These include: published, and some proposed, Commission Regulations, which are the basic requirements established for licensing; AEC Regulatory Guides, which describe acceptable methods of meeting the general requirements; and Industry Codes and Standards. It should be noted that these standards are not all inclusive or universally applicable since reactor type, size, location, etc., are variables that must be considered. The Committee frequently recommends additions to or variations of these requirements on a case-by-case basis. These recommendations are included in the Committee's reports to the Atomic Energy Commission on the cases that are reviewed by the ACRS.

The attached list includes many of the relevent examples of each. In addition, I understand that you can find specific examples of the criteria used in specific cases in the library of the Department of Nuclear Engineering at the University. Dr. Max Carbon, Head of the Department of Nuclear Engineering, has indicated that he would be willing to make this information available to you. Additional information regarding specific projects, the Kewaunee Nuclear Plant and the Point Beach Nuclear Plant, is available at the Public Document Rooms located at:

Kewaunee Public Library
Attn: Mrs. O. H. Molle, Librarian
314 Milwaukee Street
Kewaunee, Wisconsin 54216

Manitowoc Public Library
808 Hamilton Street
Manitowoc, Wisconsin 54220

In addition, I have attached an article describing the activities of the ACRS which may be of further use to you.

Sincerely,

Original Signed by
R. F. Fraley

R. F. Fraley
Executive Secretary

Attachments:
1) List of AEC Regulations
2) AEC Regulatory Guides
3) Industry Codes and Standards
4) Article re Activities of ACRS

Atomic Energy Commission Regulations

10 CFR 20* - Standards for Protection Against Radiation

10 CFR 50 - Licensing of Production and Utilization Facilities

Appendix A - General Design Criteria for Nuclear Power Plants

Appendix B - Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants

Appendix E - Emergency Plans for Production and Utilization Facilities

Appendix F - Policy Relating to the Siting of Fuel Reprocessing Plants and Related Waste Management Facilities

Appendix G - Fracture Toughness Requirements (Proposed)

Appendix H - Reactor Vessel Material Surveillance Program Requirements (Proposed)

Appendix I - Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "as Low as Practicable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents (Proposed)

10 CFR 55 - Operators' Licenses

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* Part 20 of the Title 10 of the Code of Federal Regulations

10 CFR 71 - Packaging of Radioactive Material for Transport and Transportation of Radioactive Material Under Certain Conditions

10 CFR 100 - Reactor Site Criteria

Interim Acceptance Criteria for Emergency-core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors

AEC REGULATORY GUIDES

1.1 Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps (formerly Safety Guide 1)
1.2 Thermal Shock to Reactor Pressure Vessels (formerly Safety Guide 2)
1.3 Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors (Revision 1, 6/73, of former Safety Guide 3)
1.4 Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors (Revision 1, 6/72, of former Safety Guide 4)
1.5 Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors (formerly Safety Guide 5)
1.6 Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems (formerly Safety Guide 6)
1.7 Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident (formerly Safety Guide 7)
1.8 Personnel Selection and Training (formerly Safety Guide 8)
1.9 Selection of Diesel Generator Set Capacity for Standby Power Supplies (formerly Safety Guide 9)
1.10 Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures (Revision 1, 1/2/73, of former Safety Guide 10)
1.11 Instrument Lines Penetrating Primary Reactor Containment (formerly Safety Guide 11)
1.12 Instrumentation for Earthquakes (formerly Safety Guide 12)
1.13 Fuel Storage Facility Design Basis (formerly Safety Guide 13)
1.14 Reactor Coolant Pump Flywheel Integrity (formerly Safety Guide 14)
1.15 Testing of Reinforcing Bars for Concrete Structures (Revision 1, 12/28/72, of former Safety Guide 15)
1.16 Reporting of Operating Information (formerly Safety Guide 16)
1.17 Protection of Nuclear Plants Against Industrial Sabotage (Revision 1, 6/73, of former Safety Guide 17)
1.18 Structural Acceptance Test for Concrete Primary Reactor Containments (Revision 1, 12/28/72, of former Safety Guide 18)
1.19 Nondestructive Examination of Primary Containment Liner Welds (Revision 1, 8/11/72, of former Safety Guide 19)
1.20 Vibration Measurements on Reactor Internals (formerly Safety Guide 20)
1.21 Measuring and Reporting of Effluents from Nuclear Power Plants (formerly Safety Guide 21)
1.22 Periodic Testing of Protection System Actuation Functions (formerly Safety Guide 22)
1.23 Onsite Meteorological Programs (formerly Safety Guide 23)
1.24 Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Gas Storage Tank Failure (formerly Safety Guide 24)
1.25 Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (formerly Safety Guide 25)
1.26 Quality Group Classifications and Standards (formerly Safety Guide 26)
1.27 Ultimate Heat Sink (formerly Safety Guide 27)
1.28 Quality Assurance Program Requirements (Design and Construction) (formerly Safety Guide 28)
1.29 Seismic Design Classification (formerly Safety Guide 29)
1.30 Quality Assurance Requirements for the Installation, Inspection and Testing of Instrumentation and Electric Equipment (formerly Safety Guide 30)
1.31 Control of Stainless Steel Welding (Revision 1, 6/73, of former Safety Guide 31)
1.32 Use of IEEE Std 308-1971, "Criteria for Class IE Electric Systems for Nuclear Power Generating Stations" (formerly Safety Guide 32)

1.33 Quality Assurance Program Requirements (Operation) (formerly Safety Guide 33)
1.34 Control of Electroslag Weld Properties (12/28/72)
1.35 Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures (2/5/73)
1.36 Nonmetallic Thermal Insulation for Austenitic Stainless Steel (2/23/73)
1.37 Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants (3/16/73)
1.38 Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants (3/16/73)
1.39 Housekeeping Requirements for Water-Cooled Nuclear Power Plants (3/16/73)
1.40 Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants (3/16/73)
1.41 Preoperational Testing of Redundant On-site Electric Power Systems to Verify Proper Load Group Assignments (3/16/73)

1.43 Control of Stainless Steel Weld Cladding of Low Alloy Steel Components (5/73)
1.44 Control of the Use of Sensitized Stainless Steel (5/73)
1.45 Reactor Coolant Pressure Boundary Leakage Detection Systems (5/73)
1.46 Protection Against Pipe Whip Inside Containment (5/73)
1.47 Bypassed and Inoperable Status Indication for Nuclear Plant Safety Systems (5/73)
1.48 Design Limits and Loading Combinations for Seismic Category I Fluid System Components (5/73)

1.50 Control of Preheat Temperature for Low-Alloy Steel Welding (5/73)
1.51 Inservice Inspection of ASME Code Class 2 and 3 Nuclear Plant Components (5/73)

1.53 Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems (6/73)
1.54 Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants (6/73)
1.55 Concrete Placement in Category I Structures (6/73)
1.56 Maintenance of Water Purity in Boiling Water Reactors (6/73)
1.57 Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components (6/73)

Industry Codes and Standards

1. Sections III and XI of the ASME Boiler and Pressure Vessel Code.

2. USA Standard Code for Pressure Piping (USASB 31.1)

3. USA Standard Code for Pressure Piping (USASB 31.7)

4. Institute of Electrical and Electronic Engineers Criteria for Nuclear Power Plant Protection Systems (IEEE-279)