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**LICENSING TOPICAL REPORT:
APPLICABILITY OF DIVISION 1
REGULATORY GUIDES TO HIGH-TEMPERATURE
GAS-COOLED REACTORS**

by
J. H. LEWIS

Prepared under
Contract DE-AT03-76ET35300
for the **San Francisco Operations Office**
Department of Energy

DECEMBER 1980

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**GENERAL ATOMIC PROJECT 6400
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FOREWORD

This Licensing Topical Report has been prepared for submittal to the U.S. Nuclear Regulatory Commission (NRC) as part of a review of issues that may have a bearing on the licensing of commercial high-temperature gas-cooled reactors (HTGRs). The plants addressed herein are graphite moderated and helium cooled with ceramic fuel in prismatic fuel elements.

This report evaluates the applicability of the Division 1 (power reactors) regulatory guides to the HTGR. Although many of the guides are applicable to all types of reactors, some are written for light water reactors. The degree of applicability of the latter to the HTGR varies and may be subject to interpretation. An NRC review of the applicability of these guides prior to application for a construction permit would expedite the licensing process.

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ABSTRACT

The application of Division 1 (power reactors) regulatory guides to high-temperature gas-cooled reactors (HTGRs) is discussed. About eighty of the Division 1 guides can be applied to any type of reactor; the remaining sixty, mostly written for light water reactors (LWRs), are divided between (1) those not applicable to the HTGR because of fundamental differences in design, (2) those applicable in intent but containing positions specific to LWRs, and (3) those written for LWRs but of sufficient generality to be applied to the HTGR without major exception. Emphasis is placed on issues which involve the unique characteristics of the HTGR.

The regulatory guides evaluated are those extant as of early 1980. The positions presented are subject to periodic updating owing to the continuing modification of the guides and because the design options for the HTGR are at various stages of development. Nevertheless, this report is believed to provide a sound basis for evaluating conformance with existing Division 1 guides.

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1. INTRODUCTION

Regulatory guides are issued for several purposes: (1) to describe acceptable methods of implementing specific parts of Nuclear Regulatory Commission (NRC) regulations; (2) to delineate techniques used by the NRC to evaluate specific problems or postulated accidents; and (3) to provide guidance to applicants seeking construction permits or operating licenses for nuclear power plants. Division 1 guides apply to power reactors and for the most part are concerned with compliance with 10CFR50* and 10CFR100**.

Section 1.8 of Regulatory Guide 1.70, Revision 3, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants" (November 1978), requires the inclusion of a table indicating the extent to which an applicant intends to comply with all applicable regulatory guides. The table must identify the sections of the Safety Analysis Report (SAR) to which the guide applies and indicate any proposed exceptions to the regulatory position.

This report addresses the applicability of the guides to the high-temperature gas-cooled reactor (HTGR) and the exceptions to these guides due to the unique characteristics of the HTGR. Although a particular regulatory guide may may not have a position statement, it should not be inferred that there would be no exceptions or alternate approaches at the time the Preliminary Safety Analysis (PSAR) is submitted. Approximately 80 of the current 140 guides are applicable to all types of reactors. Most of the remainder are written for light water reactors (LWRs), and a few

*"Domestic Licensing of Production and Utilization Facilities, " in particular, Appendices A and B, "General Design Criteria for Nuclear Power Plants," and "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," respectively.

**"Reactor Site Criteria."

are specific to boiling water reactors (BWRs), pressurized water reactors (PWRs), and high-temperature reactors. The majority of the guides written for LWRs can either be applied directly to HTGRs or their intent and/or regulatory positions are applicable. In a few instances, the guides are completely inappropriate because of their specificity to an LWR, and a comparable system, method, or test may be required for the HTGR.

Each of the Division 1 regulatory guides has been evaluated to determine its applicability to the HTGR. The guides reviewed were those available in early 1980, and they have been considered without regard to their status, i.e., whether they were issued or issued for comment. A statement of justification is provided for guides which were not considered applicable. Guides applicable in intent are considered if significant exceptions or clarification are needed. Minor technical differences have not been discussed, since they can best be addressed in the context of an application for a specific plant and the guides in effect at that time.

Because the design options for the HTGR are in various stages of development, specific engineering solutions which ensure conformance with guides are not always available, and the positions stated will be subject to modification as the design matures. Likewise, as new guides are issued and existing guides revised, the positions may need to be changed.

The principal objective of this report is to obtain the concurrence of the NRC on the applicability of Division 1 regulatory guides as described herein. This will allow the design effort, and later the preparation of the PSAR, to proceed with greater confidence in the regulatory guidance to be observed and should, to some extent, simplify the review of the PSAR and minimize the potential for impact of the plant design due to differing opinions on the applicability of the regulatory guides.

2. APPLICABILITY OF DIVISION 1 REGULATORY GUIDES

Table 1 lists all the Division 1 regulatory guides and gives their applicability as follows:

1. Not applicable: guides which are so specific to LWRs that they are not appropriate for HTGRs, and there are no comparable requirements which apply to HTGRs.
2. Applicable or applicable with qualification: guides written for power reactors in general (or for high-temperature reactors) and assumed to apply to HTGRs.
3. Applicable intent: guides written specifically for LWRs but whose subject matter is appropriate in whole or in part to HTGRs.

Section 3 describes the objectives of some of the guides and presents position statements justifying or clarifying their applicability.

TABLE 1
APPLICABILITY OF DIVISION 1 REGULATORY GUIDES TO HTGRS

No.	Title	Rev.	Issued (Yr/Mo)	Applicability
1.1	Net Positive Suction Head for Emergency Core Cooling Containment Heat Removal System Pumps (Safety Guide 1)	--	70/11	Not applicable ^(a)
1.2	Thermal Shock to Reactor Pressure Vessels (Safety Guide 2)	--	70/11	Not applicable ^(a)
1.3	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors	2	74/06	Intent applicable ^(a,b)
1.4	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors	2	74/06	Intent applicable ^(a)
1.5	Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors (Safety Guide 5)	--	71/03	Not applicable; specific to BWRs
1.6	Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems (Safety Guide 6)	--	71/03	Applicable to Class 1E power systems as defined in Fig. 1 of IEEE 308
1.7	Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident	2	78/11	Intent applicable (a)
1.8	Personnel Selection and Training	1-R	77/05	Applicable
1.9	Selection, Design, and Qualification of Diesel-Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants (for comment)	2	79/12	Applicable
1.10	Mechanical (Cadmold) Splices in Reinforcing Bars of Category I Concrete Structures	1	73/01	Intent applicable
1.11	Instrument Lines Penetrating Primary Reactor Containment (Safety Guide 11) Supplement to Safety Guide 11, Backfitting Considerations	--	71/03 72/02	Applicable
1.12	Instrumentation for Earthquakes	1	74/04	Applicable
1.13	Spent Fuel Storage Facility Design Basis (for comment)	1	75/12	Applicable
1.14	Reactor Coolant Pump Flywheel Integrity (for comment)	1	75/08	Not applicable; flywheels are not used in HTGR systems
1.15	Testing of Reinforcing Bars for Category 1 Concrete Structures	1	72/12	Intent applicable
1.16	Reporting of Operating Information -- Appendix A Technical Specifications (for comment)	4	75/08	Applicable

TABLE 1 (Continued)

No.	Title	Rev.	Issued (Yr/Mo)	Applicability
1.17	Protection of Nuclear Power Plants Against Industrial Sabotage	1	73/06	Applicable
1.18	Structural Acceptance Test for Concrete Primary Reactor Containments	1	72/12	Intent applicable
1.19	Nondestructive Examination of Primary Containment Liner Welds (Safety Guide 19)	1	72/08	Applicable
1.20	Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing	2	76/05	Intent applicable ^(a)
1.21	Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants	1	74/06	Intent applicable; details of compliance will depend upon operating procedures and expected effluents
1.22	Periodic Testing of Protection System Actuation Functions (Safety Guide 22)	--	72/02	Applicable
1.23	Onsite Meteorological Programs (Safety Guide 23)	--	72/02	Applicable
1.24	Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure (Safety Guide 24)	--	72/03	Intent applicable; release assumptions and inventories of radioactive gases will be appropriate for HTGR
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 (Safety Guide 25)	--	72/03	Intent applicable; release assumptions will be appropriate for HTGR fuel and systems
1.26	Quality Group Classifications and Standards for Water-, and Radioactive Waste-Containing Components of Nuclear Power Plants (for comment)	3	76/02	Intent applicable ^(a)
1.27	Ultimate Heat Sink for Nuclear Power Plants (for comment)	2	76/01	Applicable
1.28	Quality Assurance Program Requirements (Design and Construction)	2	79/02	Applicable. ^(c)
1.29	Seismic Design Classification	3	78/09	Intent applicable
1.30	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment (Safety Guide 30)	--	72/08	Applicable ^(c)
1.31	Control of Ferrite Content in Stainless Steel Weld Metal	3	78/04	Intent applicable

TABLE 1 (Continued)

No.	Title	Rev.	Issued (Yr/Mo)	Applicability
1.32	Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants	2	77/02	Applicable
1.33	Quality Assurance Program Requirements (Operation)	2	78/02	Applicable
1.34	Control of Electroslag Weld Properties	--	72/12	Intent applicable
1.35	Inservice Inspection of UngROUTED Tendons in Prestressed Concrete Containment Structures	2	76/01	Applicable
1.36	Nonmetallic Thermal Insulation for Austenitic Stainless Steel	--	73/02	Intent applicable
1.37	Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants	--	73/03	Intent applicable ^(c)
1.38	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants	2	77/05	Intent applicable
1.39	Housekeeping Requirements for Water-Cooled Nuclear Power Plants	2	77/09	Intent applicable
1.40	Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants	--	73/03	Intent applicable ^(a)
1.41	Preoperational Testing of Redundant On-Site Electric Power Systems to Verify Proper Load Group Assignments	--	73/03	Applicable
1.42	Withdrawn	--	--	--
1.43	Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components	--	73/05	Intent applicable
1.44	Control of the Use of Sensitized Stainless Steel	--	73/05	Intent applicable ^(a)
1.45	Reactor Coolant Pressure Boundary Leakage Detection Systems	--	73/05	Intent applicable ^(a)
1.46	Protection Against Pipe Whip Inside Containment	--	73/05	Intent applicable; conflicts with Branch Technical Position (BTP) MEB 3-1 shall be resolved in favor of the BTP
1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems	--	73/05	Applicable
1.48	Design Limits and Loading Combinations for Seismic Category I Fluid System Components	--	73/05	Intent applicable ^(a)
1.49	Power Levels of Nuclear Power Plants	1	73/12	Applicable

TABLE 1 (Continued)

No.	Title	Rev.	Issued (Yr/Mo)	Applicability
1.50	Control of Preheat Temperature for Welding of Low-Alloy Steel	--	73/05	Intent applicable
1.51	Withdrawn	--	--	--
1.52	Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants	2	78/03	Intent applicable; pressures, temperatures, radiation levels, and iodine concentrations will be appropriate for the HTGR
1.53	Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems	--	73/06	Applicable
1.54	Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants	--	73/06	Intent applicable ^(c)
1.55	Concrete Placement in Category I Structures	--	73/06	Applicable
1.56	Maintenance of Water Purity in Boiling Water Reactors (for comment)	1	78/07	Not applicable; specific to BWRs
1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components	--	73/06	Intent applicable to portions of containment penetrations that resist pressure and are not backed by structural concrete
1.58	Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel	--	73/08	Applicable
1.59	Design Basis Floods for Nuclear Power Plants	2	77/08	Applicable
1.60	Design Response Spectra for Seismic Design of Nuclear Power Plants	1	73/12	Applicable
1.61	Damping Values for Seismic Design of Nuclear Power Plants	--	73/10	Applicable
1.62	Manual Initiation of Protective Actions	--	7/30	Applicable
1.63	Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants	2	78/07	Intent applicable; actual x/r ratios and interrupting times will be used
1.64	Quality Assurance Requirements for the Design of Nuclear Power Plants	2	76/06	Applicable ^(c)
1.65	Materials and Inspections for Reactor Vessel Closure Studs	--	73/10	Intent applicable
1.66	Withdrawn	--	--	--
1.67	Installation of Overpressure Protection Devices	--	73/10	Intent applicable

TABLE 1 (Continued)

No.	Title	Rev.	Issued (Yr/Mo)	Applicability
1.68	Initial Test Programs for Water-Cooled Nuclear Power Plants	2	78/08	Intent applicable ^(a)
1.68.1	Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants	1	77/01	Not applicable; specific to BWRs
1.68.2.	Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants	1	78/07	Intent applicable ^(a)
1.69	Concrete Radiation Shields for Nuclear Power Plants	--	73/12	Applicable
1.70	Stanford Format and Content of Safety Analysis Reports for Nuclear Power Plants	3	78/11	Intent applicable; LWR version will be revised as necessary to adapt it to HTGR systems
1.71	Welder Qualification for Areas of Limited Accessibility	--	73/12	Applicable
1.72	Spray Pond Piping Made from Fiberglass-Reinforced Thermosetting Resin	2	78/11	Applicable
1.73	Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants	--	74/01	Applicable; source term will correspond to that for the design basis depressurization accident (DBDA)
1.74	Quality Assurance Terms and Definitions	--	74/02	Applicable ^(c)
1.75	Physical Independence of Electric Systems	2	78/09	Applicable
1.76	Design Basis Tornado for Nuclear Power Plants	--	74/04	Applicable
1.77	Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Resources	--	74/05	Not applicable ^(a)
1.78	Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release	--	74/06	Applicable
1.79	Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors	1	75/09	Not applicable; specific to PWRs, but the design of the HTGR core auxiliary cooling system (CACS) will allow appropriate testing.
1.80	Preoperational Testing of Instrument Air Systems	--	74/06	Applicable
1.81	Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants	1	75/01	Applicable
1.82	Sumps for Emergency Core Cooling and Containment Spray Systems	--	74/06	Not applicable; sumps not used for CACS
1.83	Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes	1	75/07	Intent applicable; a comparable program will be developed for Class 1 heat exchangers

TABLE 1 (Continued)

No.	Title	Rev.	Issued (Yr/Mo)	Applicability
1.84	Design and Fabrication Code Case Acceptability -- ASME Section III, Division 1	15	79/05	Applicable; approved code cases will be evaluated on a case-by-case basis, and approval requests for additional required code cases will be submitted by applicant
1.85	Materials Code Case Acceptability -- ASME Section III, Division 1	15	79/05	Applicable; approved code cases will be evaluated on a case-by-case basis, and approval requests for additional required code cases will be submitted by applicant
1.86	Termination of Operating Licenses for Nuclear Reactors	--	74/06	Applicable
1.87	Guidance for Construction of Class 1 Components in Elevated-Temperature Reactors (Supplement to ASME Section III Code Cases 1592, 1593, 1594, 1595, and 1596)	1	75/06	Applicable
1.88	Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records	2	76/10	Applicable ^(c)
1.89	Qualification of Class 1E Equipment for Nuclear Power Plants	--	74/11	Applicable with qualification ^(a)
1.90	Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons	1	77/08	Applicable
1.91	Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants (for comment)	1	78/02	Applicable
1.92	Combining Modal Responses and Spatial Components in Seismic Response Analysis	1	76/02	Applicable
1.93	Availability of Electric Power Sources	--	74/12	Applicable
1.94	Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants	1	76/04	Applicable
1.95	Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release	1	77/01	Applicable
1.96	Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants	1	76/06	Not applicable; specific to BWRs
1.97	Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant Conditions During and Following an Accident	1	77/08	Intent applicable ^(a)

TABLE 1 (Continued)

No.	Title	Rev.	Issued (Yr/Mo)	Applicability
1.98	Assumptions Used for Evaluating the Potential Radiological Consequences of a Radioactive Offgas System Failure in a Boiling Water Reactor (for comment)	--	76/03	Not applicable; specific to BWRs
1.99	Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials	1	77/04	Not applicable ^(a)
1.100	Seismic Qualification of Electric Equipment for Nuclear Power Plants	1	77/08	Applicable
1.101	Emergency Planning for Nuclear Power Plants	1	77/03	Applicable
1.102	Flood Protection for Nuclear Power Plants	1	76/09	Applicable
1.103	Post-Tensioned Prestressing Systems for Concrete Reactor Vessels and Containments	1	76/10	Applicable
1.104	Withdrawn	--	--	--
1.105	Instrument Setpoints	1	76/11	Applicable
1.106	Thermal Overload Protection for Electric Motors on Motor-Operated Valves	1	77/03	Applicable
1.107	Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures	1	77/02	Applicable
1.108	Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants	1	77/08	Applicable
1.109	Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I	1	77/10	Applicable
1.110	Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors (for comment)	--	76/03	Intent applicable
1.111	Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors	1	77/07	Intent applicable ^(a)
1.112	Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Reactors	O-R	77/05	Not applicable; however, a comparable evaluation of environmental impact from gaseous and liquid effluents will be required
1.113	Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I	1	77/04	Applicable

TABLE 1 (Continued)

No.	Title	Rev.	Issued (Yr/Mo)	Applicability
1.114	Guidance on Being Operator at the Controls of a Nuclear Power Plant	1	76/11	Applicable
1.115	Protection Against Low-Trajectory Turbine Missiles	1	77/07	Applicable
1.116	Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems	0-R	77/05	Applicable ^(c)
1.117	Tornado Design Classification	1	78/04	Intent applicable; structures, systems, and components to be protected will be specific to the HTGR
1.118	Periodic Testing of Electric Power and Protection Systems	2	78/06	Applicable
1.119	Withdrawn	--	--	--
1.120	Fire Protection Guidelines for Nuclear Power Plants (for comment)	1	77/11	Applicable with qualification ^(a)
1.121	Bases for Plugging Degraded PWR Steam Generator Tubes (for comment)	--	76/08	Not applicable; however, a comparable basis for plugging tubes of Class 1 heat exchangers will be developed
1.222	Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment for Nuclear Power Plants	1	78/02	Applicable
1.123	Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants	1	77/07	Applicable ^(c)
1.124	Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports	1	78/01	Intent applicable
1.125	Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants	1	78/10	Applicability is site-related; physical hydraulic models may not be necessary
1.126	An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification	1	78/03	Not applicable; HTGR fuel is not of the type addressed in the guide
1.127	Inspection of Water-Control Structures Associated With Nuclear Power Plants	1	78/03	Applicability is site-related
1.128	Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants	1	78/10	Applicable
1.129	Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants	1	78/02	Applicable
1.130	Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports	1	78/10	Intent applicable
1.131	Qualification Tests of Electric Cables, Field Splices, and Connections for Light-Water-Cooled Nuclear Power Plants (for comment)	--	77/08	Intent applicable ^(a)

TABLE 1 (Continued)

No.	Title	Rev.	Issued (Yr/Mo)	Applicability
1.132	Site Investigations for Foundations of Nuclear Power Plants	1	79/03	Applicable
1.133	Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors (for comment)	--	77/09	Not applicable; detection method not suited to the HTGR; requirement for loose-part detection will be evaluated
1.134	Medical Evaluation of Nuclear Power Plant Personnel Requiring Operator Licenses	1	79/03	Applicable
1.135	Normal Water Level and Discharge at Nuclear Power Plants (for comment)	--	77/09	Applicable
1.136	Material for Concrete Containments	1	78/10	Applicable
1.137	Fuel-Oil Systems for Standby Diesel Generators (for comment)	--	78/01	Applicable
1.138	Laboratory Investigation of Soils for Engineering Analysis and Design of Nuclear Power Plants (for comment)	--	78/04	Applicable
1.139	Guidance for Residual Heat Removal (for comment)	--	78/05	Intent applicable ^(a)
1.140	Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants (for comment)	--	78/03	Intent applicable
1.141	Containment Isolation Provisions for Fluid Systems (for comment)	--	78/04	Intent applicable; specific isolation requirements for the HTGR will be developed to provide comparable isolation integrity
1.142	Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments) (for comment)	--	78/04	Applicable
1.143	Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants (for comment)	--	78/07	Intent applicable
1.144	Auditing of Quality Assurance Programs for Nuclear Power Plants (for comment)	--	79/01	Applicable ^(c)
1.145	Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants (for comment)	--	79/08	Applicable

^(a) See Section 3.

^(b) Intent applicable is used when the guide is specifically for LWRs. Exceptions to the provisions of the guide may or may not be taken when it is applied to the HTGR.

^(c) Nuclear steam supply system (NSSS) and nuclear heat source covered by General Atomic Report GA-A13010A, Amendment 5: "General Atomic Quality Assurance Program," January 1980.

3. OBJECTIVES AND POSITION STATEMENTS

This section describes the objectives of the guides and presents position statements justifying or clarifying their applicability.

Title: Regulatory Guide 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps," November 1970.

Objective: To describe a suitable relationship between increases in containment pressure caused by postulated loss of coolant accidents and net positive suction head of emergency core cooling and containment heat removal system pumps.

Position: This guide is not applicable to the HTGR. A containment heat removal system is not required, and the closed, pressurized, secondary (water) side of the CACS is outside the containment. Owing to the nature of the single-phase coolant, the helium circulators cannot suffer a loss of net positive suction head.

Title: Regulatory Guide 1.2, "Thermal Shock to Reactor Pressure Vessels," November 1970.

Objective: To describe a program which will ensure that the reactor pressure vessel will behave in a nonbrittle manner when a large quantity of cold emergency coolant is injected into the reactor coolant system.

Position: This guide is written for LWRs and is not applicable to the HTGR. Because of its helium coolant and large mass of core graphite and concrete in the prestressed concrete reactor vessel (PCRVR), the HTGR is not subject to rapid changes in temperature. Hence, there is no mechanism by which significant quantities of cold coolant can be injected. However, PCRVR liners and closures are designed to provide fracture integrity for the primary coolant system boundary. The specific requirements imposed on the HTGR are

1. Portions of the PCRVR liner backed by concrete must be designed to provide a leak-tight membrane to contain the primary coolant but are not a structural part of the primary pressure boundary of the PCRVR. These portions of the liner are loaded in a general biaxial compressive state of stress so that brittle fracture is not a serious consideration. To ensure ductile behavior, however, the materials used must satisfy the impact requirements of Section III, Division 2, of the ASME Boiler and Pressure Vessel Code, Article CB-2500.
2. The core cavity liner is the only region of the liner exposed to fast neutron irradiation to an extent that could cause irradiation embrittlement of the material. These irradiated regions are backed by concrete and must satisfy the impact requirements of CB-2500 after allowing for the maximum expected transition temperature shift due to

irradiation. Based on substantial irradiation data on core cavity liner steels and other comparable mild steels, the maximum transition temperature shift due to irradiation is expected to be less than 75°F for base metal and less than 100°F for weld material.

3. The portions of the PCRV penetration liner and their closures that are not backed by concrete for load-carrying purposes must be designed to the ASME Boiler and Pressure Vessel Code, Section III, Division 1. Protection against nonductile fracture of these structures is ensured by satisfying the fracture toughness requirements of Article NB-2300 of the code. Article NB-2300 requires that the material reference nil ductility transition temperature (RT_{NDT}), as determined by drop weight and Charpy V-notch testing, shall be sufficiently below the lowest metal service temperature to ensure ductile behavior under all plant operating conditions. Specific RT_{NDT} requirements for each component are determined by using fracture analysis techniques.

Title: Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences on a Loss of Coolant Accident for Boiling Water Reactors," Revision 2, June 1974.

Regulatory Guide 1.4, "Assumptions for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors," Revision 2, June 1974.

Objective: To provide assumptions to be used to analyze and evaluate the radiological consequences of a postulated loss of coolant accident (LOCA).

Position: These guides are written for LWRs but their intent is applicable to the HTGR. Paragraph 50.34 of 10CFR50 (LOCA evaluation) will be met by using parameters and calculations appropriate for the HTGR to evaluate the consequences of a postulated DBDA. The breathing rates and dispersion methodology of the guide are applicable, but the release of radioactive material will be based on models and data specific to the HTGR.

An appropriate source term for the reference HTGR fuel will be developed and provided for NRC review. The transport mechanism for fission products will consider (1) circulating activity; (2) transport of fission products through graphite; (3) plateout of the products within the PCRV; (4) lift-off of plated-out fission products; (5) escape of fission products from the PCRV to the containment during depressurization; (6) plateout of fission products within the containment; and (7) leakage of the products from the containment. The assumptions for atmospheric diffusion and dose conversion will be the same as those in Regulatory Guide Position C.2 for scoping calculations, but the atmospheric dispersion models of Regulatory Guide 1.145 will be used for site-specific calculations.

Title: Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," Revision 2, November 1978.

Objective: To describe a method which enables certain LWR plants to meet the requirements for controlling hydrogen, oxygen, and other substances that may be released to the reactor containment following postulated accidents.

Position: This guide is specific to LWRs fueled with uranium oxide pellets within zircaloy cladding, but its intent is applicable to the HTGR. Analyses will be performed for the HTGR to ensure that combustible concentrations of potentially flammable gaseous mixtures will not be accumulated in the containment.

Title: Regulatory Guide 1.20, "Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing," Revision 2, May 1976.

Objective: To present a comprehensive assessment program for flow-induced vibrations to verify the structural integrity of the reactor internals prior to commercial operation.

Position. This guide is written for LWRs, but its intent is applicable to the HTGR. The configuration of the reactor internals for the first large HTGR will be a prototype. The intent of this guide is to qualify the configuration as a valid prototype. To do this, the following must be accomplished:

1. Full vibration assessment of components that could interfere with core cooling (by blocking coolant flow or changing core geometry) or prevent insertion of negative reactivity if a loss of their structural integrity occurred.
2. Components of the primary coolant system that are not covered by item 1 but are potential generators of fluid vibrations that could cause the structural failures described in item 1 should be in place in the primary coolant system for experimental measurements. It is not necessary to measure the response of these components during flow excitation.

If it is not practical to perform preoperational tests with full flow in all reactor cooling loops, tests will be performed with simultaneous partial flow in all loops and with full flow in a single loop. Analytical methods for extrapolating to full-flow conditions will be used.

Title: Regulatory Guide 1.26, "Quality Group Classifications and Standards for Water-, and Radioactive Waste-Containing Components of Nuclear Power Plants," Revision 3, February 1976.

Objective: To describe a quality classification system related to specified national standards that may be used to determine acceptable quality standards for safety-related components, other than Class 1 components, containing water, steam, or radioactive material in light-water-cooled nuclear power plants.

Position: This guide is written for LWRs, but its intent is applicable to HTGRs subject to the following:

1. The reactor coolant pressure boundary defined in 10CFR50(v) is written for light-water-cooled reactors. The analogous definition for the HTGR is that of the primary coolant system boundary; i.e., the PCRV liner, including all cavity and penetration liners exposed to primary coolant, in conjunction with the prestressed concrete structure, forms the primary coolant boundary and includes
 - a. All primary closures that seal penetrations in the PCRV liner.
 - b. All system piping that contains primary coolant and penetrates the PCRV liner or PCRV closures, up to and including the second isolation valve in the piping with free-flow greater than one square inch. If the free-flow area is less than or equal to one square inch, single valve isolation is sufficient.
 - c. All system piping within the PCRV liner cavity that is exposed to primary coolant and is not covered in item b.

- d. The PCRV overpressure protection systems, up to and including the pressure relief valve.
 - e. Parts of mechanical components which retain primary coolant.
- 2. Table 2 delineates the minimum requirements for terminating the primary coolant boundary in fluid system components.
 - 3. Components of the HTGR will be classified by ANS Safety Class 1, 2, and 3 designations in lieu of the quality group designations of Regulatory Guide 1.26.

TABLE 2
MINIMUM REQUIREMENTS FOR FLUID SYSTEM COMPONENTS WHICH
TERMINATE THE PRIMARY COOLANT SYSTEM BOUNDARY^(a)

Free-Flow Area (in. ²)	Number of Valves	Safety Class ^(b)	Type of Isolation Valve	Basis
>10	2 ^(c)	1	Normally closed or automatic ^(d)	Definition of reactor coolant pressure boundary from 10CFR50.2(v) and 10CFR50.55a
10 to 1	2	2	Normally closed or automatic	Footnote 2 of 10CFR50.55a and DP&L SER, page 3-4
1 or less	1	2	Normally closed or automatic ^(e)	Footnote 2 of 10CFR50.55a; the leak rate is low enough to allow an orderly shutdown per footnote 1 of 10CFR50.55a, and dose consequences with pipe break and valve failure are acceptable
Instrument line	1	2	Normally closed, automatic, or remote manual ^(f,g)	Regulatory Guide 1.11

(a) Termination of the primary coolant system boundary means termination of the requirements for Safety Class 1 or 2 components. Components beyond the termination (away from the primary coolant system boundary) may be non-nuclear unless a safety grade is required to comply with safety functions, e.g., shutdown cooling, containment of radioactive waste, or passage through the containment building. The boundary is also terminated if the system is closed and entirely of the indicated safety class. (Instruments such as pressure transducers are outside the jurisdiction of ASME Section III and therefore cannot be nuclear grade.)

(b) The safety class up to and including the isolation valve(s). All SC-1 and SC-2 fluid system components are Seismic Category I.

(c) The pressure relief line is covered by 10CFR50.2(v). Small lines attached to the relief line have a 10-in.² or less free-flow area.

(d) Automatic valves (except check valves) or remote manual valves require instrumentation for detection of abnormal conditions which necessitate valve closure to terminate a leak of primary coolant.

(e) Remote manual operation may be acceptable on a defined basis for free-flow areas which are less than 0.1 in.².

(f) Footnote d applies for valves outside the containment

(g) For the specific case of instrument lines which do not penetrate the containment, the position is as follows. The primary coolant system boundary may be terminated by an SC-2 line which ends at a high-quality instrument which has been certified (by testing, etc.) to withstand all design transients without gross failure of its pressure-containing components. This requirement shall normally be imposed on all safety-related systems (e.g., plant protection system) which receive signals from the instrument. An alternatively acceptable approach to ensuring the integrity of the primary coolant system boundary at instruments is to provide an SC-2 enclosure around the instrument casing. As a minimum, SC-2 piping connected to appropriately protected and service-qualified instruments with a locally operated manual SC-2 valve is required for instruments which form a part of the primary coolant system and do not penetrate the containment.

Title: Regulatory Guide 1.40, "Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants," March 1973.

Objective: To describe a method of complying with NRC regulations on qualification testing of prototype continuous-duty Class I motors installed inside the containment of water-cooled nuclear power plants.

Position: This guide is written for LWRs, but its intent is applicable to the HTGR subject to the following clarification. The CACS motor is encased within the structural concrete of the PCRV and is therefore not subject to the global environmental conditions associated with the DBDA and steam water or water line rupture. Associated equipment for the CACS motor is qualified to the global conditions expected during the DBDA and steam or water line rupture. In addition, the CACS motor is qualified to the helium and radiation environment anticipated during normal operation and to the dynamic depressurization effects of the DBDA. All rotating mechanical motor components (e.g., bearings) are qualified to the anticipated service life of the motor.

Title: Regulatory Guide 1.44, "Control of the Use of Sensitized Stainless Steel," May 1973.

Objective: To describe methods of implementing requirements for controlling the application and processing of stainless steel and avoiding severe sensitization that could lead to stress corrosion cracking.

Position: This guide is written for LWRs and is not directly applicable to the primary coolant system of the HTGR in that the type of intergranular or stress corrosion cracking described in the guide occurs in aqueous environments. Stainless steel is not known to be susceptible to stress corrosion cracking in the low-impurity helium atmosphere of the HTGR primary coolant system. The secondary coolant of the HTGR steam cycle does provide a steam-water environment; however, 304 and 316 stainless steel are not in contact with the secondary coolant.

With regard to the piping of the CACS, the intent of the guide will be applied to unstabilized, austenitic steels, if used, except that intergranular corrosion testing is not considered necessary for welding of materials which have a carbon content greater than 0.03%. This position is based on experience with and the effectiveness of imposed welding controls for minimizing heat input.

Title: Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

Objective: To describe methods of implementing 10CFR50, Appendix A, General Design Criterion 30 requirements with regard to selection of leakage detection systems for the reactor coolant pressure boundary.

Position: This guide is written for LWRs, but its intent is applicable to the HTGR. To meet this intent and thereby satisfy the requirements of General Design Criterion 30, the HTGR provides instruments capable of indicating helium leakage from the primary coolant system boundary by measuring

1. Low reactor coolant pressure.
2. High containment pressure, temperature, or radioactivity.
3. PCR/V penetration seal system flow, which indicates purified helium inleakage.

Low-rate helium leakage can be monitored by daily comparison of the total loss of helium to the helium purification system and the total flow of helium from the helium makeup system. These daily records can be used to monitor the long-term performance of the primary coolant system boundary.

Title: Regulatory Guide 1.48, "Design Limits and Loading Combinations for Seismic Category I Fluid System Components," May 1973.

Objective: To delineate design limits and appropriate load combinations associated with normal operation, postulated accidents, and specified seismic events for Seismic Category I fluid system components.

Position: This guide is written for LWRs, but its intent is applicable to the HTGR. Load combinations and allowable stresses for Seismic Category I components are given in Tables 3 through 5. The following exceptions apply to the guide:

1. Regulatory Positions C.4, C.10, and C.12. Footnote 7 of Regulatory Guide 1.48 requires analysis of secondary stress for emergency and faulted plant conditions when an active Class 1 valve is designed by analysis. This requirement is supplemental to the code and is not industry practice. If such a requirement is sustained, the proposed allowable conservative stresses (plant conditions 1, 2, and 3) should give a sufficient guarantee of valve operability within the pressure boundary.
2. The operating basis earthquake (OBE) will not be superimposed on plant condition 1, 2, and 3 stresses for times when the components are subject to routine testing or special operational conditions.

TABLE 3
LOAD COMBINATIONS FOR INITIATING EVENTS AND SEISMIC EVENTS

Plant Condition(a)	ASME III, Division 1 And Other Safety-Related Components(b,c)	ASME III, Division 2 Components(d)
1	NOL	NOL
2	NOL + L ₂	NOL + L ₂
3	NOL + L ₃ ^(e) NOL + L ₂ + OBE ^(f,g)	NOL + L ₃ ^(e) NOL + L ₂ + OBE ^(f,g) NOL + L ₃ + OBE ^(f,g)
4	NOL + L ₄	NOL + L ₄ NOL + L ₄ + SSE ^(g)
5	NOL + L ₅ NOL + L ₅ + SSE ^(g)	NOL + L ₅ NOL + L ₅ + SSE ^(g)

(a) Plant conditions (PCs) are defined in terms of expected best-estimate frequency of occurrence (f) per reactor year, as given below. The approximate equivalent in Regulatory Guide 1.48 is also shown. When using this table in conjunction with Division 2 of the ASME Code, the following applies: PC-1 = normal; PC-2 = abnormal; PC-3 = severe environmental; PC-4 = extreme environmental; PC-5 = failure.

PC	f	Regulatory Guide 1.48
1	Planned operations	Normal
2	$f \geq 10^{-1}$	Upset
3	$10^{-1} > f \geq 10^{-2}$	Upset
4	$10^{-2} > f \geq 10^{-4}$	Emergency
5	$10^{-4} > f \geq 10^{-6}$	Faulted

(b) NOL = normal operating loads; L₂, L₃, L₄, and L₅ are the transient loads associated with PCs 2, 3, 4, and 5, respectively.

(c) The basis for stress and primary pressure limits are given in the position statement.

(d) Closures and penetrations not backed by concrete are analyzed by Division 1 rules.

(e) When the OBE is taken as the initiating event, the dynamic loads should include those due to the vibratory motion of the earthquake and those due to the overall system response to the earthquake.

(f) The OBE as an initiating event is PC-3.

(g) SSE = safe shutdown earthquake. When combining earthquake loads with other loads, the initiating events and earthquake should be assumed to occur simultaneously. It is not necessary to arbitrarily assume that the earthquake occurs at the time other stresses are maximum. Component stresses shall be evaluated using the loading combinations given in Tables 3 and 4. The design shall be based on the most limiting combination.

TABLE 4
LOAD COMBINATIONS WITH AN ASSUMED SINGLE FAILURE

Assumed Condition ^(a)	Response Within Limits for	All Safety-Related Components ^(b,c)
PC-2 + SF	PC-3	NOL + L ₂ + effects of SF
PC-3 + SF	PC-4	NOL + L ₃ ^(d) + effects of SF
PC-4 + SF	PC-5	NOL + L ₄ + effects of SF
PC-5 + SF	PC-5	NOL + L ₅ + SSE + effects of SF

(a) PC = plant condition; SF = single failure. Plant conditions are defined in terms of expected best-estimate frequency of occurrence (f) per reactor year, as given below. The approximate equivalent in Regulatory Guide 1.48 is also shown. When using this table in conjunction with Division 2 of the ASME Code, the following applies: PC-1 = normal; PC-2 = abnormal; PC-3 = severe environmental; PC-4 = extreme environmental; PC-5 = failure.

PC	f	Regulatory Guide 1.48
1	Planned operations	Normal
2	$f \geq 10^{-1}$	Upset
3	$10^{-1} \geq f > 10^{-2}$	Upset
4	$10^{-2} \geq f > 10^{-4}$	Emergency
5	$10^{-4} \geq f > 10^{-6}$	Faulted

(b) NOL = normal operating loads; L₂, L₃, L₄, and L₅ are the transient loads associated with PCs 2, 3, 4, and 5, respectively.

(c) Effects of SF include dynamic loads which may result from the SF and the consequential changes in system conditions (e.g., temperature) which may alter the allowable stresses.

(d) When the OBE is taken as the initiating event, the dynamic loads should include those due to the vibratory motion of the earthquake and those due to the overall system response to the earthquake.

TABLE 5
ASME SECTION III, DIVISION 1 AND 2 SERVICE LIMITS

Plant Condition	Service Limit for Safety Function		
	Pressure-Retaining Integrity(a)	Active Component Operability(b)	Support and Dimensional Stability(c)
1	A	A	A
2	B	B	B
3	B(d)	B	B(d)
4	C	B(e)	B(f)
5	D	B(g)	B(h)

(a) Pressure-retaining integrity refers to the capability of the component to withstand the pressure associated with the designated service load combination.

(b) Active component operability pertains to components that require mechanical motion to perform their safety function, e.g., pumps and valves.

(c) Dimensional and support stability refers to the capability of the component to perform its intended safety function. Support stability is the capability of the support to maintain the safety function of the supported component. Dimensional stability refers to the capability of components such as pipes (including elbows and tees), vessels, nozzles, and valves to perform safety functions such as maintenance of the fluid flow path or prevention of unacceptable valve seat leakage.

(d) For Division 2 only, Service Limit C shall be permitted if functional capability is demonstrated under the service load combination.

(e) Service Limit C shall be permitted if the functional capability of the active component and its supports is demonstrated under the service load combination.

(f) Deformation or distortion of components may be considered in the evaluation of the component's ability to perform its safety function. Service Limit C shall be permitted if functional capability is demonstrated under the service load combination.

(g) Service Limit D shall be permitted if the qualification in footnote e is met.

(h) Service Limit D shall be permitted if the qualification in footnote f is met.

Title: Regulatory Guide 1.68, "Initial Test Programs for Water-Cooled Nuclear Power Plants," Revision 2, August 1978.

Objective: To describe the general scope and depth of initial test programs for light-water-cooled reactor power plants, to provide a representative listing of plant structures, systems, components, and design features, and to suggest tests to demonstrate performance capability.

Position: This guide is written for LWRs, but its intent is applicable to the HTGR. Certain tests in Appendix A of this guide do not apply to the HTGR owing to the inherent fluid dynamic differences between the water coolant of the LWR and the helium coolant of the HTGR. The design of the HTGR will, however, facilitate system tests which are the same as or similar to those in Appendix A.

Title: Regulatory Guide 1.68.2, "Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants," Revision 1, July 1978.

Objective: To describe an initial startup test program for demonstrating hot standby capability and the potential for cold shutdown from outside the control room. This capability is required by 10CFR50, Appendix A, General Design Criterion 19. Hot shutdown corresponds to hot standby as defined in the standard technical specifications for light-water-cooled reactors. Hot standby is used in this guide. Cold shutdown is also defined in the standard technical specifications.

Position: This guide is written for LWRs, but its intent is applicable to HTGRs to the extent that provisions for testing are included in the plant design. The following exceptions apply:

1. This guide draws a definite distinction between hot and cold shutdown capabilities. Although this may be significant in an LWR because of a need to depressurize to use a core decay heat removal system, it is not required in the HTGR, since the main and the auxiliary loops are capable of bringing the reactor and primary system down from operating temperature to a cold shutdown condition after the reactor is either automatically tripped by the plant protection system or manually tripped.
2. Regulatory Position C.2.d states that it is necessary to verify that control is not possible from the main control room after control from the remote shutdown stations has been established. Regulatory Position 2.C(1) prescribes that provision should be made during the demonstration test for assumption of immediate control of the plant by the

control room observers if an emergency or unsafe condition develops during testing. This is difficult to implement without special switchover provisions installed for the test or some carefully designated procedures for rapid transfer of control.

Title: Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," May 1974.

Objective: To identify analytical methods and assumptions that may be used in evaluating the consequences of a rod ejection accident in uranium-oxide-fueled PWRs.

Position: This guide is not applicable to the HTGR. The design features of the HTGR preclude control rod ejection as a credible accident, and it is therefore not assumed to be a design basis accident.

Refueling penetrations for the HTGR, within which the control rod drives are installed, are designed and fabricated to the requirements of the ASME Code for Section III, Division I, Class 1 vessels. All pressure boundary welds are full penetration and 100% radiographed during fabrication. They are subject to volumetric examination during in-service inspection in accordance with ASME Section XI, Division 2 code rules. Primary and Secondary shear anchors in the PCRV concrete are provided. For these reasons, the penetrations are not postulated to fail.

Title: Regulatory Guide 1.89, "Qualification of Class 1E Equipment for Nuclear Power Plants," November 1974.

Objective: To provide a method for complying with NRC regulations on design verification of safety-related electric equipment.

Position: This guide is applicable to the HTGR. Class 1E equipment will meet its requirements with the following clarification. The appropriate source term given in Regulatory Position C.2 for qualification of Class 1E equipment in the containment is that corresponding to the DBDA release, i.e., the design primary coolant activity plus a conservative estimate of the plateout activity released during the event. Credit will be taken for locations and shielding. The equipment will be qualified so that it meets the intended performance requirements for the environmental and operating conditions in which it will be required to function and for the length of time during which it must function.

Title: Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," Revision 1, August 1977.

Objective: To describe an acceptable method of complying with NRC requirements for providing instrumentation to monitor plant variables and systems during and following an accident.

Position: This guide is written for LWRs, but its intent is applicable to HTGRs subject to the following qualification. The instruments and their ranges given in Regulatory Position C.3 are not necessarily appropriate for the HTGR. A limited number of high-range monitoring instruments shall be selected. These instruments shall have ranges sufficient to measure the selected parameters beyond DBDA, thereby meeting the intent of this guide.

Title: Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials," Revision 1, April 1977.

Objective: To describe general procedures which are acceptable on an interim basis for predicting the neutron radiation damage effects of the residual elements copper and phosphorus on the low-alloy steels currently used for light-water-cooled reactor vessels.

Position: This guide is specifically written for light-water-cooled reactors in steel pressure vessels. It is not applicable to the HTGR for the following reasons:

1. The pressure vessel of the HTGR is prestressed concrete, and the membrane liner of the PCRV is not subject to significant tensile stresses.
2. The guide is limited to irradiation of steel at $500^{\circ} \pm 25^{\circ}\text{F}$, a temperature far above that of the PCRV liner (normally 125°F).
3. The guide specifies the use of a neutron fluence with an energy greater than 1 MeV. The HTGR liner is predominantly exposed to thermal neutrons, and there is no experimental evidence that the presence of copper or phosphorus is detrimental when most of the radiation exposure is to thermal neutrons.

Title: Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977.

Objective: To describe the basic features of calculational models and the assumptions for the estimation of atmospheric transport and dispersion of gaseous effluents in routine releases from land-based LWRs.

Position: This guide is written for LWRs, but its intent is applicable to the HTGR subject to the following exceptions:

1. Regulatory Position C.1.c(3). If sufficient site-specific data are not available for determining site-specific corrections, terrain adjustment factors for Eq. 3 (Sagendorf model) will be as given in Revision 0.
2. Regulatory Position C.2.a. For elevated releases, the terrain adjustment methodology will be determined on a site-specific basis and will not necessarily be the maximum terrain height between the release point and receptor location.
3. Regulatory Position C.4. The wind direction in calm conditions for constant mean wind (hourly data) and variable trajectory models will be assigned the value of the last noncalm data point.
4. Regulatory Position C.4. If a variable trajectory model is used, the wind speed value assigned to calm conditions and reported by regional meteorological stations not operated by or for the applicant (e.g., National Weather Service or Federal Aviation Administration Stations) will not be less

than the minimum wind speed determined for the on-site instrumentation, which is one-half the starting threshold of the vane or anemometer, whichever is higher.

Title: Regulatory Guide 1.120, "Fire Protection Guidelines for Nuclear Power Plants," Revision 1, November 1977.

Objective: To present acceptable guidelines for implementing General Design Criterion 3 of 10CFR50, Appendix A, which requires the development of a fire protection program for nuclear power plants. The purpose of the fire protection program is to ensure the capability of shutting down the reactor and maintaining it in a safe shutdown condition and to minimize radioactive releases to the environment in the event of a fire. This guide addresses fire protection programs for safety-related systems and equipment and other plant areas containing fire hazards that could adversely affect safety-related systems.

Position: This guide is applicable to the HTGR. The plant design will generally be in compliance with the provisions of the guide; however, if the plant design and arrangement suggest that deviations are in order, these deviations will be identified on a case-by-case basis for staff review and acceptance.

Title: Regulatory Guide 1.131, "Qualification Tests of Electric Cables, Field Splices, and Connections for Light-Water-Cooled Nuclear Power Plants," August 1977.

Objective: To describe a method for complying with NRC regulations on qualification testing of electric cables, field splices, and connections to ensure that they can perform their safety-related functions. The fire test provisions of this guide do not apply to qualification for an installed configuration.

Position: This guide is written for LWRs, but its intent is applicable to the HTGR subject to the following clarifications and exceptions:

1. Regulatory Position C.1. The environmental conditions used for qualification tests will be appropriate for the HTGR and may be different from those specified in IEEE Std. 323-1974.
2. Regulatory Position C.4. Aging data will be developed consistent with the methods and procedures in common use at the time of cable qualification. Relative to the "ongoing qualification procedure," other methods which adequately demonstrate projected cable life may be used in lieu of those recommended in the guide.
3. Regulatory Position C.5. The radiation environment for qualification of safety-related cables, splices, and/or connectors will be based upon normally expected radiation exposure over the expected life of the plant plus the radiation exposure resulting from an accident, with credit taken for locations, shielding, and the length of time during which the equipment is required to perform its intended functions.

4. Regulatory Position C.6. An "aged" cable is defined as a sample that has been brought to the end-of-life condition by the accelerated aging technique or an otherwise technically justifiable method to simulate the effect of environmental conditions on equipment performance. This aging may not include the radiation dose received during the design basis event.

Title: Regulatory Guide 1.139, "Guidance for Residual Heat Removal,"
May 1978.

Objective: To describe an acceptable method for removing decay heat and
sensible heat after reactor shutdown.

Position: This guide is written for LWRs. The intent of Positions C.1 and
C.5 is met for the HTGR by means of the CACS. This redundant,
safety-grade, single-failure-proof system is capable of taking
the reactor from normal operating conditions to cold shutdown
using either on-or off-site power. Provisions are made for pre-
operational testing which will meet the requirements of
Regulatory Guide 1.68 as applied to the HTGR.