HBA
HELIUM BREEDER ASSOCIATES

July 31, 1979

P-456

Mr. Steve Varga
Acting Assistant Director for
Light Water Reactors
Division of Project Management
Office of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
Washington, DC 20555

Subject: Review of Gas-Cooled Fast Breeder Reactor (GCFR) General Design Criteria

References: (1) "Gas-Cooled Fast Breeder Reactor Preliminary Safety Information Document," General Atomic Company Report GA-10298, dated February 15, 1971

(2) Letter from J.J. Scoville to R.S. Boyd, "New Applications Survey," dated December 22, 1978

(3) Letter from R.S. Boyd to J.J. Scoville, "New Applications Survey," dated November 7, 1978

Dear Mr. Varga:

With this letter, we are transmitting thirty-five copies of Revision 1 to Amendment 8 to the GCFR Preliminary Safety Information Document (Ref. 1) on the subject of GCFR General Design Criteria. We are requesting that the United States Nuclear Regulatory Commission review the proposed criteria and discussions submitted in the amendment and to develop formal comments on the content.

To justify such a review, we would like to note the very broad basis of support which the Gas-Cooled Fast Breeder Reactor currently enjoys from the U.S. electrical generating utility community as well as the DOE. Currently the GCFR is supported through Helium Breeder Associates (HBA) by utilities which represent about one-third of the total electrical generating capacity in the United States. DOE funding for the GCFR in fiscal year 1979 is $26 million, with a similar funding level projected in fiscal year 1980.

In developing a commercialization plan for the GCFR, HBA and DOE have divided the program into several sequential phases such that all necessary information from the research and development efforts will be available at certain decision points for review of the program before major capital

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Mr. Steve Varga
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commitments are made to proceed with the next phase. The first and current phase of the program is called the Program Definition and Licensing Phase. Its major objectives are to establish the licensability of the GCFR demonstration plant and to complete appropriately detailed design which will enable a definitive cost estimate to be made. Safety and licensing issues must be addressed early and in parallel with design and development efforts to ensure that utility and DOE funds are not misdirected toward unlicensable features.

Consistent with the current program phase, we have identified a list of five preapplication submittals on topics which are felt to be of key significance to GCFR development activities. These were described in our letter to Mr. Roger Boyd (Ref. 2) responding to his request for the identification of planned submittals through December 31, 1981 (Ref. 3). As a logical first topic for obtaining the needed NRC guidance, we proposed that agreement should be reached upon the general design criteria appropriate for GCFRs.

Toward this end, we are submitting the attached amendment to the GCFR Preliminary Safety Information Document (PSID). In the original issue of the GCFR PSID, the extent to which each of the 1967 AEC General Design Criteria (GDC) for LWR's was met in the GCFR design and was discussed in Appendix A. This was outdated, however, in 1971, when revised LWR GDC were published in 10CFR50, Appendix A. In February 1977, Amendment 8 to the PSID was submitted to the NRC and contained general design criteria specific for the GCFR compatible with the revised LWR GDCs. The 1977 submittal has not as yet been reviewed by the NRC. The attached document revises the GDCs transmitted in the 1977 Amendment 8 submittal consistent with more recent HTGR criteria and NRC positions on the Clinch River Breeder Reactor (CRBR). The NRC is requested to review and to develop comments on the attached GCFR criteria. Subsequent to satisfactory resolution of such comments and incorporation of any required addenda into the document, the staff is requested to provide written approval that the proposed General Design Criteria adequately account for the characteristics of a GCFR reactor plant and, as an appropriate interpretation of 10CFR50 Appendix A, are sufficient to govern design of such a plant.

Should you or your staff have any questions on the attached submittal or related subject matter, please contact Mr. Dave Buttemer of my staff on (714) 294-9500.

POOR ORIGINAL

Sincerely,

J. J. Scoville

J. J. Scoville
Vice President and
General Manager

cc: P. Williams - NRC (w/o encl)
R. Simon - GA (w/o encl)
G. Newby - DOE
K. Highfill - GCRA

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HBA

STATE OF CALIFORNIA
COUNTY OF SAN DIEGO

On July 31, 1979, before the undersigned, a Notary Public for the State of California, personally appeared J. J. Scoville, known to me to be the person whose name is subscribed to the within instrument, and acknowledged that he executed the same.

Parmely T. Ferrie

Parmely T. Ferrie
Notary Public

OFFICIAL SEAL
PARMELY T. FERRIE
NOTARY PUBLIC - CALIFORNIA
SAN DIEGO COUNTY
My comm. expires NOV 18, 1980

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GCFR GENERAL DESIGN CRITERIA

INTRODUCTION

In accordance with 10CFR50.34, an application for a nuclear power plant construction permit must include the principal design criteria for the proposed facility. In the original issue of the GCFR PSID, the extent to which each of the 1967 general design criteria (GDC) for light-water reactors (LWRs) was met in the GCFR plant design was discussed. In February 1977, Amendment 8 to the PSID was submitted to the NRC and contained specific general design criteria for the GCFR compatible with the 1971 revision of the LWR criteria. Amendment 8 was not reviewed by the NRC.

This report updates the GDCs transmitted to the NRC in Amendment 8 to the PSID. The objective of this revision to Amendment 8 to the PSID is to obtain NRC concurrence with recommended interpretations of the general design criteria which are worded specifically for the GCFR. These changes delineate the intent of the GDC for the GCFR which will clarify the licensing requirements and simplify the licensing review of a GCFR nuclear power plant.

The NRC staff is requested to review the proposed criteria and discussions submitted in this amendment and to develop formal comments on the content. Subsequent to satisfactory resolution of such comments and incorporation of any required addenda into the document, the staff is requested to provide written approval that the proposed General Design Criteria adequately account for the characteristics of a GCFR reactor plant and, as an appropriate interpretation of 10CFR50 Appendix A, are sufficient to govern design of such a plant.

The scope of the amendment addresses proposed revisions to the Definition and Explanation Section and Criteria Section of Appendix A to 10CFR50 as issued in 1977. The proposed changes are summarized in Table 1. Whenever the wording of a GCFR criterion has been changed from that of a 10CFR50 criterion, the inserted words are underlined and the deleted words are cancelled by dashed lines.

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TABLE 1

SUMMARY OF CHANGES TO APPENDIX A, 10CFR50

Definitions and Explanations

Definition	Summary of Changes
Nuclear Power Unit	No change.
Loss-of-Coolant Accidents	Deleted.
Single Failure	No change.
Anticipated Operational Occurrences	Replaced examples for LWR with examples for GCFR.
Primary Coolant System Boundary	Addition.
Design Basis Depressurization Accident	Addition.

Criteria

Criterion Number	Criterion Title	Summary of Changes
1	Quality Standards and Records	No change.
2	Design Bases for Protection Against Natural Phenomena	No change.
3	Fire Protection	No change.
4	Environmental and Missile Bases	Change reference to "loss of coolant" to "design basis depressurization" accident.
5	Sharing of Structures, Systems and Components	No change.
6-9		Criteria 6-9 do not appear in Appendix A.
10	Reactor Design	No change.
11	Reactor Inherent Protection	No change.
12	Suppression of Reactor Power Oscillations	No change.
13	Instrumentation and Control	Change from "reactor coolant pressure boundary" to "primary coolant system boundary".
14	Primary Coolant System Boundary	Change title from "Reactor Coolant Pressure Boundary".

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Table 1 (Continued)

Criterion Number	Criterion Title	Summary of Changes
15	Primary Coolant System Design	Same as 13.
16	Containment Design	No change.
17	Electric Power Systems	Change terminology to be consistent with other changes.
18	Inspection and Testing of Electric Power Systems	No change.
19	Control Room	Change terminology to be consistent with other changes.
20	Protection System Limits	No change.
21	Protection System Reliability and Testability	No change.
22	Protection System Independence	No change.
23	Protection System Failure Modes	No change.
24	Separation of Protection and Control Systems	No change.
25	Protection System Requirements for Reactivity Control Mal-functions.	No change.
26	Reactivity Control System Redundancy and Capability	No change.
27	Confined Reactivity Control Systems Capability	Delete reference to poison addition by emergency core cooling system.
28	Reactivity Limits	Change of terminology; changes in list of specific accidents mentioned.
29	Protection Against Anticipated Operational Occurrences	No change.
30	Quality of Primary Coolant System Boundary	Similar to 13.
31	Fracture Prevention of Primary Coolant System Boundary	Similar to 13.
32	Inspection of Primary Coolant System Boundary	Change in terminology; relaxed requirement to inspect liner of concrete vessel.
33	Reactor Coolant Makeup	Delete.
34	Residual Heat Removal	Added requirement for two independent and diverse systems.

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Table 1 (Continued)

Criterion Number	Criterion Title	Summary of Changes
35	Core Auxiliary Cooling System	Deleted mention of emergency core cooling system; replaced with auxiliary cooling system requirements.
36	Inspection of Residual Heat Removal Systems	Change terminology to agree with 34, 35 and 37.
37	Testing of Residual Heat Removal Systems	Similar to 36.
38	Containment Heat Removal System	Deleted.
39	Inspection of Containment Heat Removal System	Deleted.
40	Testing of Containment Heat Removal System	Deleted.
41	Containment Atmosphere Cleanup	Change requirements to be compatible with GCFR.
42	Inspection of Containment Atmosphere Cleanup	No change.
43	Testing of Containment Atmosphere Cleanup	No change.
44	Heat Transfer System	No change.
45	Inspection of Heat Transfer System	No change.
46	Testing of Heat Transfer System	Change in terminology.
47-48		Criteria 47-48 do not appear in Appendix A.
49	Prestressed Concrete Reactor Vessel Thermal Control	New.
50	Containment Design Basis	Change requirements to be more flexible and compatible with GCFR.
51	Fracture Prevention in Containment Structure	No change.
52	Capability for Testing Controlled Releases and Leakage from Containment	No change.
53	Provisions for Containment Testing and Inspection	No change.
54	Piping Systems Penetrating Containment	No change.

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Table 1 (Continued)

Criterion Number	Criterion	Summary of Changes
55	Primary Coolant System Boundary Penetrating Containment	Change in terminology.
56	Containment Isolation	No change.
57	Closed System Isolation Valves	Change in terminology.
58-59		Criteria 58-59 do not appear in Appendix A.
60	Control of Releases of Radioactive Materials to the Environment	No change.
61	Fuel Storage and Handling and Radioactivity Control	No change.
62	Prevention of Criticality in Fuel Storage and Handling	No change.
63	Monitoring Fuel and Waste Storage	No change.
64	Monitoring Radioactivity Releases	Deletion of reference to recirculation in "loss of coolant" accident.

DEFINITIONS AND EXPLANATIONS

Nuclear Power Unit

A nuclear power unit means a nuclear power reactor and associated equipment necessary for electric power generation and includes those structures, systems, and components required to provide reasonable assurance the facility can be operated without undue risk to the health and safety of the public.

Discussion

The wording is identical to that in the present Appendix A and is sufficiently general to be applicable to the GCFR.

Loss-of-Coolant Accidents

[~~Loss-of-coolant-accidents-means-those-postulated-accidents-that-result-from-the-loss-of-reactor-coolant-at-a-rate-in-excess-of-the-capability-of-the-reactor-coolant-makeup-system-from-breaks-in-the-reactor-coolant-pressure-boundary;-up-to-and-including-a-break-equivalent-in-size-to-the-double-ended-rupture-of-the----largest-pipe-of-the-reactor-coolant-system[1].]~~

[This includes the following footnote.]
~~[[1]Further-details-relating-to-the-type;-size;-and-orientation--of-postulated-breaks-in-specific-components-of-the-reactor-----coolant-pressure-boundary-are-under-development.]-~~

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Discussion

The term "loss of coolant accident" and the associated accident concept are not applicable to gas-cooled reactor nuclear power units. Accidents that involve breach of the primary coolant system boundary for the GCFR are presented below.

Single Failure

Single failure. A single failure means an occurrence which results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure. Fluid and electric systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components function properly) nor (2) a single failure of a passive component (assuming active components function properly), results in a loss of the capability of the system to perform its safety functions.[1]

[The following footnote is also to be included.]

[1] Single failures of passive components in electric systems should be assumed in designing against a single failure. The conditions under which a single failure of a passive component in a fluid system should be considered in designing the system against a single failure are under development.

Discussion

The wording is identical to that in the present Appendix A and is sufficiently general to be applicable to the GCFR.

Anticipated Operational Occurrences

Anticipated operational occurrences mean those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to [~~loss-of-power-to-all-recirculation-pumps~~] tripping of a helium circulator, helium circulator runup, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power.

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Discussion

The deletion shown is appropriate because gas-cooled nuclear power units have no recirculation pumps. Other operational considerations are added involving helium circulators which are components unique to gas-cooled nuclear power units.

Primary Coolant System Boundary

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Primary coolant means the helium gas that flows through and transports heat away from the reactor core. Primary coolant system boundary means the basic physical structure that contains the primary coolant. For gas-cooled reactors, the primary coolant system boundary consists of: a. The liner of the prestressed concrete reactor vessel (PCRV) including cavity and penetration liners which are exposed to primary coolant, in conjunction with the prestressed concrete structure, b. Primary closures that seal penetrations in the liner of the PCRV, c. System piping that contains primary coolant and penetrates the PCRV liner or closures up to and including the second isolation valve, d. System piping within the PCRV cavities that is exposed to primary coolant such as steam generator and other heat exchanger tubes, e. The PCRV overpressure protection system up to and including pressure relief valves, and f. Primary coolant retaining parts of mechanical components such as seals on shafts of helium circulators within the primary coolant system.

Discussion

The insertion of this definition augments the set of definitions in the present Appendix A to define the GCFR "primary coolant system boundary", which is somewhat different from the "reactor coolant pressure boundary" of light-water reactors.

Many portions of the GCFR primary coolant system boundary can be considered equivalent to the reactor coolant pressure boundary of LWRs. These portions, by and large, are designed to the same industry codes as the LWR counterparts. For example, the penetrations and their closures of the PCRV are pressure retaining as well as gas boundaries and are designed to ASME Section III, Division 1 code requirements as are LWR reactor vessels. However, a large portion of the GCFR primary coolant system boundary is the PCRV liner. The liner, by itself, is not considered to be a pressure retaining component although it is a gas boundary. Pressure retention is accomplished by the prestressed concrete structure which backs the liner.

Design Basis Depressurization Accident

Design basis depressurization accident means a postulated accident in which a rapid reduction in primary coolant pressure occurs as a result of egress of a portion of the primary coolant inventory from a breach of the primary coolant system boundary up to a maximum credible flow area such as postulated failure of the largest pipe connected to the primary coolant system boundary.

Discussion

This additional definition replaces its counterpart for water-cooled nuclear power plants -- the loss of coolant accident. It is clear that the intent of the present Appendix A is to address the effects of a possible break in the physical boundary that contains the fluid that passes through and cools the reactor core in a nuclear power unit.

In the case of water-cooled nuclear power units, ruptures can occur in major piping connecting the components in the reactor coolant system. The effect of such ruptures is a loss of a substantial portion of the cooling water, as well as phase changes, changes in liquid level, and changes in the heat transfer characteristics of the coolant. This condition is known as a "loss-of-coolant accident." To accommodate such loss-of-coolant accidents, an Emergency Core Cooling System is required (in water-cooled nuclear power units) that can provide sufficient cooling in a short time frame to prevent unacceptable damage to the reactor core. However, in the case of gas-cooled nuclear power units, hardware failures of the type identified above can lead to a reduction in primary coolant pressure. This depressurization event does not lead to a loss of the ability of the gaseous primary coolant to flow through and adequately cool the reactor core nor to a change in phase of the primary coolant. In such situations, core cooling can be maintained (without the need for supplemental or replacement coolant from a backup or emergency system) as long as the primary coolant can be circulated through the reactor core and as long as heat can be removed from the circulating primary coolant at an adequate rate.

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CRITERIA

CRITERION 1: QUALITY STANDARDS AND RECORDS

Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

DISCUSSION

No change is recommended because the present Appendix A criterion is sufficiently general to apply to GCFRs.

926 327

CRITERION 2: DESIGN BASES FOR PROTECTION AGAINST NATURAL PHENOMENA

Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

DISCUSSION

No change is recommended because the present Appendix A criterion is sufficiently general to apply to GCFRs.

926 328

CRITERION 3: FIRE PROTECTION

Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

DISCUSSION

No change is recommended because the present Appendix A criterion is sufficiently general to apply to GCFRs.

CRITERION 4: ENVIRONMENTAL AND MISSILE DESIGN BASES

Structures, systems and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including [~~loss-of-coolant~~] the design basis depressurization accident. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

DISCUSSION

The term "loss-of-coolant accident" is not applicable to a gas-cooled reactor. The more appropriate term "the design basis depressurization accident" is defined in the Definitions and Explanations Section.

926 330

CRITERION 5: SHARING OF STRUCTURES, SYSTEMS, AND COMPONENTS

Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

DISCUSSION

No change is recommended because the present Appendix A criterion is sufficiently general to apply to GCFRs.

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CRITERIA 6, 7, 8, AND 9

Criteria 6 through 9 do not currently appear in the Code of Federal Regulations.

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CRITERION 10: REACTOR DESIGN

The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

DISCUSSION

No change is recommended because the present Appendix A criterion is sufficiently general to apply to GCFRs.

926 333

CRITERION 11: REACTOR INHERENT PROTECTION

The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

DISCUSSION

No change is recommended because the present Appendix A criterion is sufficiently general to apply to GCFRs.

926 334

CRITERION 12: SUPPRESSION OF REACTOR POWER OSCILLATIONS

The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

DISCUSSION

No change is recommended because the present Appendix A criterion is sufficiently general to apply to GCFRs.

926 355

CRITERION 13: INSTRUMENTATION AND CONTROL

Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the ~~{reactor-coolant-pressure}~~ primary coolant system boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

DISCUSSION

The present Appendix A criterion is sufficiently general to apply to GCFRs. The change is made to reflect gas-cooled reactor terminology defined in the Definitions and Explanations Section.

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CRITERION 14: ~~{REACTOR-COOLANT-PRESSURE}~~ PRIMARY COOLANT SYSTEM BOUNDARY

The ~~{reactor-coolant-pressure}~~ primary coolant system boundary shall be designed, fabricated, erected, a..d tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

DISCUSSION

The present Appendix A criterion is sufficiently general to apply to GCFRs. The change is made to reflect gas-cooled reactor terminology defined in the Definitions and Explanations Section.

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CRITERION 15: ~~(REACTOR)~~ PRIMARY COOLANT SYSTEM DESIGN

The [~~reactor~~] primary coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the {~~reactor-coolant pressure~~} primary coolant system boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

DISCUSSION

The present Appendix A criterion is sufficiently general to apply to gas-cooled reactors. Changes are made merely to reflect a difference in terminology that has come into accepted usage for gas-cooled and water-cooled nuclear power plants. The term "primary coolant" is defined in the Definitions and Explanations Section under primary coolant system boundary.

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CRITERION 16: CONTAINMENT DESIGN

Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

DISCUSSION

No change is recommended because the present Appendix A criterion is sufficiently general to apply to GCFRs.

926 339

CRITERION 17: ELECTRIC POWER SYSTEMS

An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the [~~reactor-coolant-pressure~~] primary coolant system boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the [~~reactor coolant-pressure~~] primary coolant system boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a [~~loss-of-coolant-accident~~] design basis depressurization accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.

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Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.

DISCUSSION

Changes are made to account for the difference in terminology that has come into accepted usage for gas-cooled and water-cooled nuclear power plants. The terms are defined in the Definitions and Explanations Section.

26 341

CRITERION 18: INSPECTION AND TESTING OF ELECTRIC POWER SYSTEMS

Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.

DISCUSSION

No change is recommended because the present Appendix A criterion is sufficiently general to apply to GCFRs.

926 342

CRITERION 19: CONTROL ROOM

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including [~~loss-of-coolant~~] the design basis depressurization accident. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

DISCUSSION

The term "loss of coolant accident" is not applicable to a gas-cooled reactor. The more appropriate term "the design basis depressurization accident" is defined in the Definitions and Explanations Section.

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CRITERION 20: PROTECTION SYSTEM FUNCTIONS

The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

DISCUSSION

No change is recommended because the present Appendix A criterion is sufficiently general to apply to GCFRs.

926 344

CRITERION 21: PROTECTION SYSTEM RELIABILITY AND TESTABILITY

The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

DISCUSSION

No change is recommended because the present Appendix A criterion is sufficiently general to apply to GCFRs.

CRITERION 22: PROTECTION SYSTEM INDEPENDENCE

The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

DISCUSSION

No change is recommended because the present Appendix A criterion is sufficiently general to apply to GCFRs.

CRITERION 23: PROTECTION SYSTEM FAILURE MODES

The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

DISCUSSION

No change is recommended because the present Appendix A criterion is sufficiently general to apply to GCFRs.

926 347

CRITERION 24: SEPARATION OF PROTECTION AND CONTROL SYSTEMS

The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

DISCUSSION

No change is recommended because the present Appendix A criterion is sufficiently general to apply to GCFRs.

926 348

CRITERION 25: PROTECTION SYSTEM REQUIREMENTS FOR REACTIVITY CONTROL MALFUNCTIONS

The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

DISCUSSION

No change is recommended because the present Appendix A criterion is sufficiently general to apply to GCFRs.

CRITERION 26: REACTIVITY CONTROL SYSTEM REDUNDANCY AND CAPABILITY

Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

DISCUSSION

No change is recommended because the present Appendix A criterion is sufficiently general to apply to GCFRs.

926 350

CRITERION 27: COMBINED REACTIVITY CONTROL SYSTEMS CAPABILITY

The reactivity control systems shall be designed to have a combined capability[~~, in conjunction with poison addition by the emergency core cooling system,~~] of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

DISCUSSION

Except for the reference to poison injection by the emergency core cooling system which is deleted because it is not relevant to the gas-cooled reactor design, this criterion is sufficiently general to be applicable to GCFRs.

POOR ORIGINAL

926 351

CRITERION 28: REACTIVITY LIMITS

The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the ~~{reactor-coolant-pressure}~~ primary coolant system boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout (unless prevented by positive means), steam line rupture, changes in ~~{reactor}~~ primary coolant temperature and pressure, ~~{and-cold-water-addition}~~ and ingress of secondary or other fluids.

DISCUSSION

Changes are made to reflect differences in terminology as previously discussed. The addition of the parenthetical statement reflects the impossibility of a rod "dropout" from gas-cooled reactors. The rods are actuated from above, and they move in holes in the core that are open at the top and closed at the bottom, thereby providing a positive mechanical stop that prevents rod dropping out from the core. However, postulated reactivity accidents may include "rod drops" within the confinement of the core. Reference is deleted to reactivity insertion by the thermal effect of adding cold water to the water coolant moderator. It is replaced by the possibility of reactivity being added to the gas-cooled reactor by the ingress of fluids which are good neutron moderators.

POOR ORIGINAL

926 352

CRITERION 29: PROTECTION AGAINST ANTICIPATED OPERATIONAL OCCURRENCES

The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

DISCUSSION

No change is recommended because the present Appendix A criterion is sufficiently general to apply to GCFRs.

926 353

CRITERION 30: QUALITY OF [~~REACTOR-COOLANT-PRESSURE~~] PRIMARY COOLANT SYSTEM BOUNDARY

Components which are part of the [**reactor-coolant-system**] primary coolant system boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of [**reactor**] primary coolant leakage.

DISCUSSION

The present Appendix A version is sufficiently general to apply to gas-cooled reactors. Changes are made to reflect the difference in terminology between gas-cooled and water-cooled reactors as defined in the Definitions and Explanations Section.

POOR ORIGINAL

026 354

CRITERION 31: FRACTURE PREVENTION OF [~~REACTOR-COOLANT-PRESSURE-BOUNDARY~~] PRIMARY COOLANT SYSTEM BOUNDARY

The [~~reactor-coolant-pressure~~] primary coolant system boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady-state and transient stresses, and (4) size of flaws.

DISCUSSION

The present Appendix A criterion is sufficiently general to apply to gas-cooled reactors. Changes are made to reflect the difference in terminology as discussed previously and defined in the Definitions and Explanations Section.

POOR ORIGINAL

926 355

CRITERION 32: INSPECTION OF [~~REACTOR-COOLANT-PRESSURE-BOUNDARY~~] PRIMARY COOLANT SYSTEM BOUNDARY

Components which are part of the {~~reactor-coolant-system~~} primary coolant system boundary shall be designed to permit, to the extent practical, (1) periodic inspection and testing of important areas and features {~~to-assess-their-structural-and~~} as appropriate to assess structural integrity of pressure bearing components or to assess leak-tight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

DISCUSSION

As discussed previously, changes are made to reflect the difference in terminology between gas-cooled and water-cooled reactors. Adding the phrase "as appropriate to assess structural integrity of pressure bearing components" is recommended to differentiate between requirements for pressure boundaries and coolant boundaries. Some components of the primary coolant system may be both a pressure boundary as well as a coolant boundary, as is the case for all components of light-water reactor coolant systems. For such components where a potential for a rapidly propagating failure can exist, inservice inspection methods and frequencies compatible with ASME codes, are invoked to reduce the likelihood. By contrast, however, the liner of the prestressed concrete reactor vessel employed with the GCFR is not a pressure boundary but a leak-tight membrane. The pressure is borne by the prestressed concrete backing the liner. The liner is maintained in compression by the PCRV, except perhaps for a few local areas, and is designed to remain ductile throughout the plant life. Hence, the ASME Section XI, Division 2 code requires that the shift in liner nil-ductility transition temperature be surveyed throughout the reactor life and visual examination be performed of only exposed and accessible areas.

POOR ORIGINAL

926 356

CRITERION 33: REACTOR COOLANT MAKEUP

[~~A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.~~]

DISCUSSION

This criterion is not applicable to gas-cooled reactors since there is no need for supplemental or replacement coolant in the event that primary coolant leaks out of the reactor vessel as a result of small breaks in the primary coolant system boundary. The reason for this is that, in spite of reduction in primary coolant pressure as a result of such leaks, adequate core cooling can be maintained as long as the gaseous primary coolant can be circulated through the reactor core and the heat can be removed from the primary coolant.

POOR ORIGINAL

CRITERION 34: RESIDUAL HEAT REMOVAL

~~{A}~~ Two independent systems to remove residual heat shall be provided. The ~~{system}~~ safety function of each system shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the ~~{reactor}~~ primary coolant ~~{pressure}~~ system boundary are not exceeded. Design techniques, such as diversity in component design and principles of operation shall be used to the extent practical to prevent loss of the safety function.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

DISCUSSION

The change to two independent residual heat removal systems reflects the present design criteria for the GCFR. This is in conformance with requirements placed on the CRBR plant (NRC letter, Denise to CRBR Project, dated May 6, 1976). Changes are also made to reflect the use of proper terminology for gas-cooled reactors.

POOR ORIGINAL

926 358

CRITERION 35: [~~EMERGENCY~~] CORE AUXILIARY COOLING SYSTEM

[~~A-system-to-provide-abundant-emergency-core-cooling-shall-be provided.--The-system-safety-function-shall-be-to-transfer-heat-from-the-reactor-core-following-any-loss-of-reactor-coolant at-a-rate-such-that-(1)-fuel-and-clad-damage-that-could-interfere with-continued-effective-core-cooling-is-prevented-and-(2)-clad metal-water-reaction-is-limited-to-negligible-amounts.~~] A core auxiliary cooling system shall be provided which has the capability of heat removal at a rate sufficient to prevent any damage which could interfere with continued effective core cooling assuming a depressurization accident together with a loss of main loop cooling. Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

DISCUSSION

The deletion is made on the basis that the wording is not appropriate for gas-cooled reactors. The addition is a statement of what has evolved to be the design basis for the core auxiliary cooling system (CACS) in gas-cooled reactors.

POOR ORIGINAL

926 359

CRITERION 36: INSPECTION OF [~~EMERGENCY-CORE-COOLING~~] RESIDUAL HEAT REMOVAL SYSTEMS

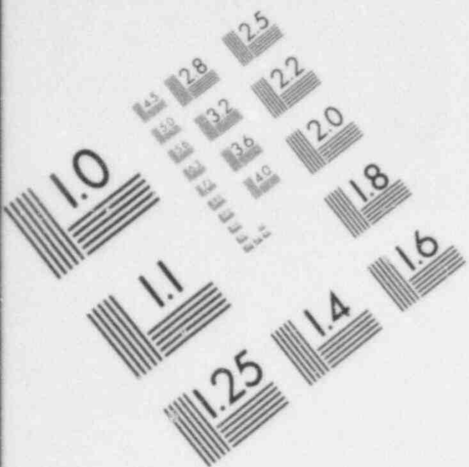
The [~~emergency-core-cooling~~] residual heat removal systems (including the CACS) shall be designed to permit appropriate periodic inspection of important components [~~such-as-spray rings-in-the-reactor-pressure-vessel,-water-injection-nozzles, and-piping~~] to assure the integrity and capability of the systems.

DISCUSSION

The change is consistent with that made for Criteria 34, 35 and 37. The deletion eliminates a reference to equipment unique to LWRs that is not used for gas-cooled reactors.

POOR ORIGINAL

926 360



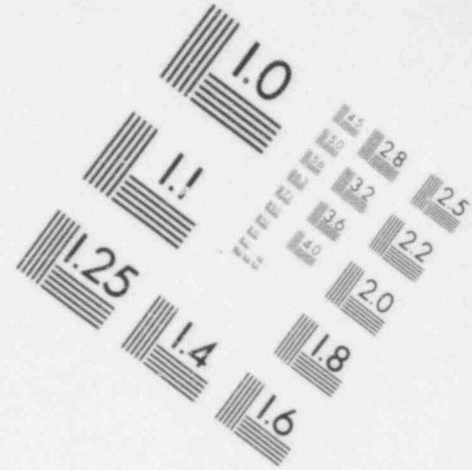
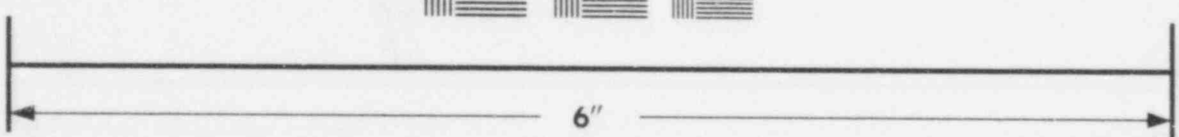
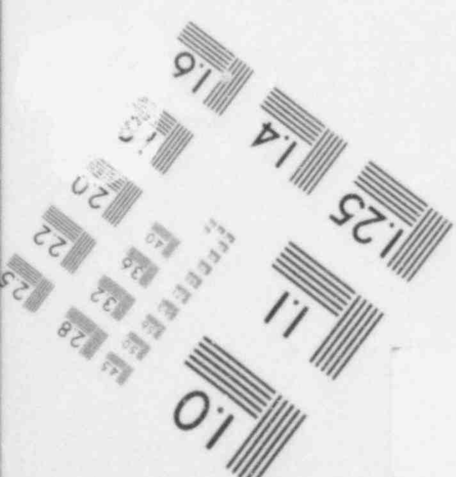


IMAGE EVALUATION TEST TARGET (MT-3)





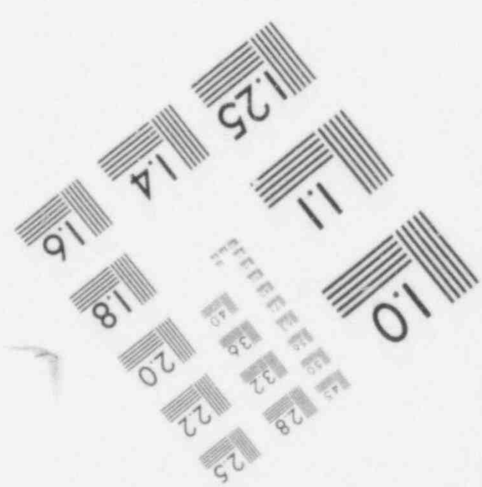


IMAGE EVALUATION TEST TARGET (MT-3)

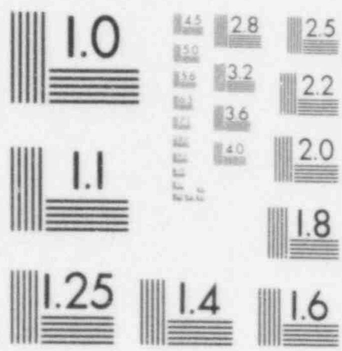


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IMAGE EVALUATION
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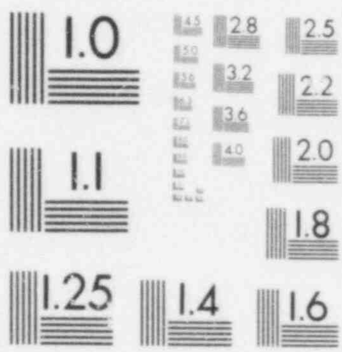

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CRITERION 37: TESTING OF [~~EMERGENCY-CORE-COOLING~~] RESIDUAL HEAT REMOVAL SYSTEMS

The [~~emergency-core-cooling~~] residual heat removal systems (including the CACS) shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

DISCUSSION

The change is consistent with that made for Criteria 34, 35 and 36.

POOR ORIGINAL

CRITERION 38: CONTAINMENT HEAT REMOVAL

[~~A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss of coolant accident and maintain them at acceptably low levels.~~

~~Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.~~]

DISCUSSION

It is recommended that this criterion be deleted because a containment heat removal system is not needed in gas-cooled reactors. Such heat removal systems are employed with light water cooled reactors because a loss of coolant accident (LOCA) releases a considerable amount of energy to the containment due to the large heat capacity of the reactor cooling water. By contrast, the helium coolant of GCFRs has low heat capacity with the result that, in the event of a design basis depressurization accident passive heat sinks afforded by structures and plant components within the containment remove heat at a fast enough rate to limit the containment atmosphere pressure and temperature to acceptably safe levels.

POOR ORIGINAL

CRITERION 39: INSPECTION OF CONTAINMENT HEAT REMOVAL SYSTEM

[~~The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system.~~]

DISCUSSION

As discussed under Criterion 38, no active containment heat removal system is required for the GCFR.

POOR ORIGINAL

927 003

CRITERION 40: TESTING OF CONTAINMENT HEAT REMOVAL SYSTEM

[~~The-containment-heat-removal-system-shall-be-designed-to permit-appropriate-periodic-pressure-and-functional-testing to-assure-(1)-the-structural-and-leaktight-integrity-of-its components;-(2)-the-operability-and-performance-of-the-active components-of-the-system;-and-(3)-the-operability-of-the-system as-a-whole-and;-under-conditions-as-close-to-the-design-as practical;-the-performance-of-the-full-operational-sequence--that-brings-the-system-into-operation;-including-operation-of-applicable-portions-of-the-protection-system;-the-transfer between-normal-and-emergency-power-sources;-and-the-operation of-the-associated-cooling-water-system.~~]

DISCUSSION

As discussed under Criterion 38, no active containment heat removal system is required for the GCFR.

POOR ORIGINAL

CRITERION 41: CONTAINMENT ATMOSPHERE CLEANUP

Systems to control [~~fission-products;~~] radioactivity, [~~hydrogen; oxygen;~~] and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of [~~fission-products~~] radioactivity released to the environment following postulated accidents. [~~and-to-control-the-concentration-of-hydrogen-or-oxygen-and other-substances-in-the-containment-atmosphere-following postulated-accidents-to-assure-that-containment-integrity-is maintained;~~]

Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.

DISCUSSION

The present Appendix A criterion is sufficiently general to apply to gas-cooled reactors. The change to radioactivity is made to address all potential forms (fuel aerosols and fission products) being in the containment atmosphere. At the present time, it has not been shown that the potential for release of hydrogen to the containment is sufficient to make it necessary to provide a system for hydrogen control for a gas-cooled reactor.

The requirement to control hydrogen concentration stems from concern about the potential for generating hydrogen by metal-water reactions during the course of a loss-of-coolant accident in a light-water reactor plant. The intent of the criterion is maintained by requiring control of "other substances", "as necessary".

POOR ORIGINAL

927 005

CRITERION 42: INSPECTION OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems.

DISCUSSION

No change is recommended because the present Appendix A criterion is sufficiently general to apply to GCFRs.

CRITERION 43: TESTING OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

DISCUSSION

No change is recommended because the present Appendix A criterion is sufficiently general to apply to GCFRs.

CRITERION 44: COOLING WATER

A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

DISCUSSION

No change is recommended because the present Appendix A criterion is sufficiently general to apply to GCFRs.

927 008

CRITERION 45: INSPECTION OF COOLING WATER SYSTEM

The cooling water system shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.

DISCUSSION

No change is recommended because the present Appendix A criterion is sufficiently general to apply to GCFRs.

CRITERION 46: TESTING OF COOLING WATER SYSTEM

The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for [~~loss-of-coolant~~] design basis depressurization accident, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

DISCUSSION

The present Appendix A criterion is sufficiently general to apply to gas-cooled reactors. The change made is to reflect the use of proper terminology for gas-cooled reactors.

POOR ORIGINAL

CRITERION 47 AND 48

Criteria 47 and 48 do not currently appear in the Code of Federal Regulations.

CRITERION 49: PRESTRESSED CONCRETE REACTOR VESSEL THERMAL CONTROL

To help maintain the integrity of the primary coolant system boundary, thermal control shall be provided to limit the temperatures of the reactor vessel elements in order to protect against temperature effects which could cause degradation of structural material and to limit thermal stresses in the prestressed concrete reactor vessels (PCRV). The cooling system portion of PCRV thermal control shall include sufficient redundancy such that the probability of the loss of its capability is minimized for all anticipated operational occurrences and postulated accidents. Capability shall be provided for tolerating the consequences of or detection of inoperable or leaking portions of the system. In addition, capability shall be provided to assure that adequate thermal control of critical portions of the reactor vessel is maintained for any postulated accident such that cooling the core to a safe shutdown condition will not be impaired. Means for inspection and testing of the thermal control components shall be provided as appropriate.

DISCUSSION

Concrete tends to lose strength and creep more when it is exposed to elevated temperatures for an extended period of time. In addition, stresses induced by thermal gradients in the reactor vessel must be maintained within acceptable limits. Therefore, to preserve the integrity of the reactor vessel and the primary coolant system boundary, a cooling system is employed to remove heat from the concrete near the interface with the metallic liner, an insulating thermal barrier is employed to limit the heat load to the cooling system. Together, the thermal barrier and cooling system form the thermal control. Appropriate inspection and testing includes provisions for visual examination of exposed and accessible areas of the thermal control components and provisions for monitoring of thermal control components exposed to primary coolant. This proposed Criterion 49 provides requirements for gas-cooled reactor systems that supplement the more general Criterion 15 which governs primary coolant system design.

CRITERION 50: CONTAINMENT DESIGN BASIS

The reactor containment structure, including access opening and penetrations, [~~and-the-containment-heat-removal-system~~] shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rates and with sufficient margin, the calculated pressure and temperature conditions resulting from [~~any-loss-of-coolant~~] the limiting design basis accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators, {~~and-energy-from-metal-water-and other-chemical-reactions-that-may-result-from-degraded-emergency core-cooling-system-functioning~~} (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

DISCUSSION

Reference to the containment heat removal system is removed to be consistent with the deletion of Criterion 38.

Reference to the loss-of-coolant accident is deleted because it is not applicable to a GCFR. Flexibility is provided in defining the design basis event for the containment by stipulating that the "limiting design basis accident" be established and incorporated into the design. It is recognized that the "limiting" containment design basis accident for the GCFR has been taken to be the "design basis depressurization accident." The change, however, allows for determination of the appropriate limiting event on a case-by-case basis. Flexibility is also provided by deleting reference to energy sources which are explicit to light-water reactors. By deleting these items, application of Criterion 50 in licensing proceedings would then require that appropriate other "energy sources" be identified, if not already included in the "limiting design basis accident."

POOR ORIGINAL

CRITERION 51: FRACTUPE PREVENTION OF CONTAINMENT PRESSURE BOUNDARY

The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady-state, and transient stresses, and (3) size of flaws.

DISCUSSION

No change is recommended because the present Appendix A criterion is sufficiently general to apply to GCFRs.

927 014

CRITERION 52: CAPABILITY FOR CONTAINMENT LEAKAGE RATE TESTING

The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

DISCUSSION

No change is recommended because the present Appendix A criterion is sufficiently general to apply to GCFRs.

927 015

CRITERION 53: PROVISIONS FOR CONTAINMENT TESTING AND INSPECTION

The reactor containment shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows.

DISCUSSION

No change is recommended because the present Appendix A criterion is sufficiently general to apply to GCFRs.

CRITERION 54: PIPING SYSTEMS PENETRATING CONTAINMENT

Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

DISCUSSION

No change is recommended because the present Appendix A criterion is sufficiently general to apply to GCFRs.

927 017

CRITERION 55: [~~REACTOR~~] PRIMARY COOLANT SYSTEM BOUNDARY PENETRATING CONTAINMENT

Each line that is part of the [~~reactor~~] primary coolant [~~pressure~~] system boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis: (1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or (2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or (3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or (4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

POOR ORIGINAL

DISCUSSION

The presenc Appendix A criterion is sufficiently general for gas-cooled reactors. The change made is to reflect the difference in terminology between gas-cooled and water-cooled reactors.

CRITERION 56: PRIMARY CONTAINMENT ISOLATION

Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis: (1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or (2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or (3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or (4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

DISCUSSION

No change is recommended because the present Appendix A criterion is sufficiently general to apply to GCFRs.

CRITERION 57: CLOSED SYSTEM ISOLATION VALVES

Each line that penetrates primary reactor containment and is neither part of the [reactor] primary coolant [pressure] system boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

DISCUSSION

The present Appendix A criterion is sufficiently general for gas-cooled reactors. The change made is to reflect the difference in terminology between gas-cooled and water-cooled reactors.

POOR ORIGINAL

CRITERIA 58 AND 59:

Criteria 58 and 59 do not currently appear in the Code of Federal Regulations.

CRITERION 60: CONTROL OF RELEASES OF RADIOACTIVE MATERIALS TO THE ENVIRONMENT

The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

DISCUSSION

No change is recommended because the present Appendix A criterion is sufficiently general to apply to GCFRs.

CRITERION 61: FUEL STORAGE AND HANDLING AND RADIOACTIVITY CONTROL

The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

DISCUSSION

No change is recommended because the present Appendix A criterion is sufficiently general to apply to GCFRs.

927 024

CRITERION 62: PREVENTION OF CRITICALITY IN FUEL STORAGE AND HANDLING

Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

DISCUSSION

No change is recommended because the present Appendix A criterion is sufficiently general to apply to GCFRs.

CRITERION 63: MONITORING FUEL AND WASTE STORAGE

Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

DISCUSSION

No change is recommended because the present Appendix A criterion is sufficiently general to apply to GCFRs.

CRITERION 64: MONITORING RADIOACTIVITY RELEASE

Means shall be provided for monitoring the reactor containment atmosphere, [~~spaces-containing-components-for-recirculation-of loss-of-coolant-accident-fluids;~~] effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

DISCUSSION

The change deletes criteria not applicable to gas-cooled reactors. Whereas a sump and a recirculation pumping system are provided in a water cooled nuclear power plant to collect the fluids released as a result of loss of coolant accidents and recirculate these through the core, these are not required for a depressurization accident with gas-cooled reactors. First, the gas discharged into containment tends to mix throughout the containment atmosphere. Second, core cooling is maintained by circulating the gas remaining in the primary system by use of circulators which are part of the main or auxiliary cooling systems. No additional system(s) for the recirculation of this air-gas mixture is required. A system is provided for containment atmosphere cleanup (see Criterion 41).

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