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Safety Related Criteria and Design Features in the Clinch River Breeder Reactor Plant*

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Abstract

The Clinch River Breeder Reactor Plant (CRBRP) is in the preliminary design phase. A site on the Clinch River in Tennessee has been selected for this demonstration plant.

The preliminary design is based on guidelines specified by Project Management Corporation at the initiation of the project. The guidelines include requirements for reliability, safety and ready licensing capability. This paper describes the overall design approach to assure safe operation and the development of safety related design bases.

The overall design approach includes three levels of design that provide defense in depth. The first level provides a technically sound design that results in high reliability and minimizes the occurrence of accidents. The second level provides protection against ~~all foreseeable~~ failures or misoperations which might occur in spite of precautions taken in the design, construction and operation of the plant, in a manner which minimizes plant damage and ensures safety to the public and the operating staff. The third level provides margin in the plant design as additional assurance that protection to the public is provided even in the event of extremely unlikely and unforeseen circumstances.

make consistent with text in section 2.

The safety related design features that have resulted from applying the three levels of design approach are described and areas in which design flexibility has been retained are indicated.

Safety Related Criteria and Design Features

in the Clinch River Breeder Reactor Plant

1.0 Summary Design Description

The Clinch River Breeder Reactor Plant (CRBRP) is being designed to be located on a TVA site in Tennessee adjacent to the Oak Ridge National Laboratory. The plant will demonstrate an alternate power option which could greatly extend our fuel resources. The plant net electrical output of about 350 megawatts will be utilized in the TVA system.

The principal plant parameters are indicated in Table I and the reactor parameters in Table II. The reactor core plan is shown in Figure 1 and the reactor elevation in Figure II. The core, consisting of hexagonal fuel assemblies in stainless steel cans, is divided into two enrichment zones, each fueled with mixed uranium-plutonium dioxide. The fuel assemblies include upper and lower blankets of depleted uranium dioxide. Radial blanket assemblies surround the core, followed by stainless steel reflector assemblies. The breeding ratio is approximately 1.2.

Sodium enters the reactor vessel at its lower end and flows upward through the orificed assemblies and is discharged from the vessel above the core. The fuel assemblies are hydraulically held down; and a mechanical holddown plate above the core serves as a backup.

Three parallel but well separated sets of loops are used to remove the heat generated by the reactor. The heat in the primary sodium system is transferred to non-radioactive sodium in the intermediate heat exchanger. This heat is then transferred from the intermediate sodium system to the feedwater to generate steam to supply the turbine-generator. The steam conditions are 1450 psig and 900°F.

The primary sodium system components are housed in concrete compartments that are steel lined and inerted. A low leakage containment provides the final barrier between the potential sources of radioactivity and the environment.

Table I

Principal Plant Characteristics

Reactor power, MWt	975
Gross electrical power, MWe	380
Number of primary heat transport loops	3
Leg location for sodium pumps, primary/intermediate	Hot/Cold
Primary plant materials	304 SS
Reactor vessel outlet temperature, °F	995
Total core sodium flow rate, 10^6 lb/hr	41.45
Total intermediate sodium flow rate, 10^6 lb/hr	38.34
Feedwater temperature, °F	452
Steam pressure at turbine throttle, psig	1450
Steam temperature at turbine throttle, °F	900
Total steam flow to turbine, 10^6 lb/hr	3.34
Turbine generator plant gross efficiency, %	39.0

Table II

Principal Reactor Parameters

Core Fuel Assemblies	
Core fuel material	PuO_2/UO_2
Fuel cladding and assembly duct material	316 SS
Fuel rod outer diameter, in	0.23
Rod pitch to diameter ratio	1.25
Core height, in	36
Axial blanket height at both ends, in	14
Fuel rods per assembly	217
Number of core assemblies	198
Peak fuel burnup goal, MWd/t	150,000
Maximum linear power, kw/ft	~~15~~.5
Average linear power, kw/ft	7
Radial Blanket Assemblies	
Blanket fuel material	Depleted UO_2
Rod outer diameter, in	0.52
Cladding thickness, mils	15
Fuel rods per assembly	61
Number of radial blanket assemblies	150
Maximum linear power, kw/ft	17.5
Control Rod Assemblies	
Poison material	B_4C
Number of control rods	19
Refueling	
Frequency, mo	12
Average number of core assemblies replaced	72
Average number of radial blanket assemblies replaced	30
Nuclear Performance	
Initial fissile loading to power ratio, kg/MWe	2.92
Initial breeding ratio	1.23
Simple doubling time, yr	23
Compound doubling time, yr	14

2.0 Design Guidelines

The CRBRP preliminary design is based on the guidelines specified by Project Management Corporation (PMC) in its March 15, 1972 request for proposals from the three participating reactor manufacturers. These guidelines focus attention on the following objectives for the plant, in approximately the order indicated:

- Reliability, safety, and ready licensing capability - achieved through the use of a proven technology base, and design simplicity and practicality

- High plant availability and maintainability, with a high degree of in-place component inspectability

- Minimal environmental effect and impact and good public acceptance, to the extent that these are not already covered under "licensability"

- Component prototypicality and extrapolability to commercial plants

- Demonstration of a reasonable breeding capability, not less than a 1.2 breeding ratio

- Demonstrating the prospect for economic competitiveness of commercial LMFBRs.[1]

The over-riding emphasis on the guideline criteria for reliability, safety, and licensing capability in turn dictates that the demonstration plant design proceed along the following path: (1) maximum use of proven experience as derived from design and operating experience of existing and planned fast reactors, with a substantial incorporation of applicable FFTF technology[2,3,4]; (2) use of the "three levels of design approach" which provides defense in depth through accident prevention, arrest of off-normal conditions and margins to prevent accidental releases to the environment; (3) capability to achieve the desired plant and component design life of 30 years; and (4) operational and design characteristics providing visible and demonstrable margins to assure reliable and safe operation.

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Definition of foreseeable

3.0 Overall Design Approach to Assure Safe Operation

The overall approach to design recognizes that a critically evaluated functional design is the controlling factor in attaining the desired high level of safety. Consequently, the design provides defense in depth through inherent sound reliability and through accident prevention, and thus it provides protection of the public and plant personnel for ~~all foreseeable~~ occurrences, which etc. (use phrase in std text). In addition, capabilities for providing public protection for extremely unlikely and unforeseen circumstances are provided. This approach is expressed in terms of three levels of design.

The <u>first level</u> of design addresses the reliability of operation and the prevention of accidents through the intrinsic features of the design of the plant and the quality, redundancy, testability, inspectability, and fail-safe features of the components of the reactor and plant. The design must be such that the plant will be safe in all phases of operation and will have a maximum tolerance for errors, off-normal operation and components malfunction. Analyses will be made and test programs conducted to determine those types of malfunctions or faults that could affect reliability of operation so that they can be guarded against by design, quality assurance, or inherent fail-safe characteristics, as appropriate. As a basic part of the LMFBR development program, a large number of large scale engineering proof tests are being conducted to verify each design concept. The testing process in the first level is to provide predictability of performance and, hence, safety through assurance of the use of proven materials and technology.

Fabrication and construction of the plant will be performed using applicable codes and standards, and under rigorous, documented quality assurance measures. Where adequate codes and standards do not exist, where appropriate, new ones will be developed and applied to the design

Extensive preoperational test programs will be conducted in the plant to assure conformance of components and systems to the established performance requirements. Key parameters will be monitored continuously or routinely and a well-defined surveillance, in-service inspection and preventive maintenance program will be carried out by a trained operating and maintenance staff to provide assurance that the as-built high quality is retained throughout the life of the plant.

The <u>second level</u> of design provides protection against failures or misoperations (such events as partial loss of flow, reactivity insertions, failure of parts of the control system, or fuel handling errors) which might occur in spite of the care taken in design, construction and operation of the plant. This additional level of defense for the public and the operating staff is provided by reliable protection devices and systems, designed to assure that such events will be prevented, arrested, or accommodated. The requirements for these protection systems are based on a spectrum of occurrences which could lead to off-normal operation which the plant design must safely accommodate. Conservative design practices, including redundant detecting and actuating equipment, will be incorporated in the protection systems to assure both the effectiveness and reliability of this second level of defense. These systems will be designed to be routinely monitored and tested to provide full assurance that when they are required to operate, they will do so reliably.

The <u>third level</u> of design supplements the first two through the demonstration of adequate margin in the plant design such that protection to the public is provided even in the event of the occurrence of extremely unlikely and unforeseen circumstances. It is the intent to design at Levels 1 and 2 to prevent all initiating events and/or control consequences for all foreseeable events. Therefore, the third level design effort evaluates and assures capability in the design to cope with incidents of such low probability so as to be inappropriate as design bases for structures, systems and components. Since the design must be such that no Identified ~~expected~~ sequence of events leads to accidents of sufficient severity to test the third level design features (such as the reactor containment), the accidents evaluated are postulated by assuming low probability events such as independent failures of the redundant protective systems simultaneously with the accident they are intended to control. Although the specific hypothetical accidents evaluated in level three are not used as design bases, the spectrum of evaluations is used to determine appropriate additional capability for unforeseen occurrences. The evaluations are used to demonstrate that the plant has adequate inherent capability or to determine what alternate features should be incorporated into the plant to provide increased capability. The provision of added third level features must be considered in the context of the exceedingly low probability of third level events and the provision of such features must not jeopardize the adequacy of the capabilities provided by design levels 1 and 2.

4.0 Design Bases

To implement the overall design approach described above, design bases are developed for the overall plant and for each system or major component. These design bases consider both natural phenomena (seismic, climatic, tornadic and flood conditions) and operational conditions (normal, off-normal and potential accidents) and assure that the plant objectives noted in Section 2 can be met. The following discussion emphasizes the design bases related to the safety and licensing objectives.

4.1 Natural Phenomena Design Bases

Conditions associated with extremely unlikely natural phenomena, which bound the most severe that have been historically reported for the site and the surroundings, are used as design bases for the plant.

The CRBRP preliminary design is based on a Safe Shutdown Earthquake (SSE) maximum horizontal ground acceleration of 0.18g. This is consistent with the design basis for other TVA nuclear plants in the vicinity and with preliminary evaluations for the Clinch River Site. Additional site investigations are in progress to confirm the seismic characteristics.

The tornado design bases are consistent with those being applied to other nuclear plants east of the Great Divide. The tornado wind force is based on a 360 mph wind, consisting of a combined 300 mph rotational velocity and 60 mph translational velocity, applied over the full height of the structures. The bases also include a tornado associated pressure variation of 3 psi in 3 seconds and typical tornado generated missiles.

The CRBRP is designed so that the Probable Maximum Flood (PMF) does not exceed the plant grade elevation. A conservative calculation of the PMF including the effects of dam failure and wave runup results in an elevation of 792 feet, while the plant grade will be at approximately 810 feet.

The plant is designed so that it can be shut down safely and maintained in a safe condition in the event of any of these extremely unlikely natural phenomena. For less severe natural phenomena which would have a significant probability of occurring during the plant life (expected frequency approximately once in one hundred years), the plant is designed to remain operable during the event.

4.2 Operational Design Bases

The full spectrum of plant conditions that can reasonably be postulated is divided into four categories in accordance with the expected frequency of occurrence of each condition. The four categories of design basis events are:

a) Normal Operation

b) Anticipated Faults

c) Unlikely Faults

d) Extremely Unlikely Faults.

Events and conditions associated with each of these categories are evaluated and the design must accommodate the consequences of them. The consideration of this wide range of events in the design bases provides protection for all conditions that can reasonably be postulated. However, to provide an even greater margin of protection, the design bases include dynamic loading conditions that are derived from consideration of a range of hypothetical accidents. Such hypothetical accidents are not design basis accidents since they are precluded by the design, and hence are not included in the four categories described above; however, the inclusion of additional dynamic loading conditions in the design bases provides margins for a spectrum of unforeseen events.

The terminology and definitions selected for the CRBRP are adopted from those in RDT C16-1, "Supplementary Criteria and Requirements for RDT Reactor Plant Protection Systems". An intentional similarity exists between the four frequency classifications and the component operating conditions defined in "Rules for Construction of Nuclear Plant Components", ASME Code, Section III. The classifications were developed jointly by ANS and ASME committees. The ANS sponsored committee developed the ANSI N18.2, which provides guidance for the reactor designer who must select the plant events considered within the different component operating conditions used in ASME Code application. However, a one-to-one correlation between plant events defined here and the ASME component operating conditions is not intended to apply for all components throughout the plant. For example, the fault that would initiate operation of a standby component will be an ASME normal condition for this component. These faults might be in any of the three fault frequency categories described above. For example, a Primary Heat Transport System guard vessel is only required to perform its normal function (containment of primary sodium) following an unlikely or extremely unlikely fault which results in a sodium leak from the enclosed vessel or pipe into the guard vessel.

4.2.1 Normal Operation

Definition

Normal operation includes steady power operation and those departures from steady operation which are expected frequently or regularly in the course of power operation, refueling, maintenance, or maneuvering of the plant.

Examples

a. Steady operation
b. Startup
c. Normal shutdown
d. Stand-by
e. All refueling operations
f. Load following
g. Random fuel rod leakage
h. Operation with specific equipment out of service as permitted by Technical Specifications
i. Routine inspection, testing and maintenance of components and systems during any of the above operations (within Technical Specifications requirements).

4.2.2 Anticipated Faults

Definition

An off-normal condition which individually may be expected to occur one or more times during the plant lifetime.

Examples

a. Uncontrolled withdrawal of a control rod from a subcritical condition
b. Uncontrolled withdrawal of a control rod at power
c. Control rod drop
d. Spurious trip of primary pumps
e. Turbine trip
f. Loss of normal feedwater
g. Loss of off-site power
h. Small steam generator tube leak, not leading to pressurization of the intermediate sodium system
i. Small sodium leak in primary or intermediate system, with leak rate less than normal makeup flow
j. Spurious trip of intermediate pumps
k. Control system malfunction
l. Generator load loss
m. Single error of an operator
n. Spurious reactor trip

4.2.3 Unlikely Faults

Definition

An off-normal condition which individually is not expected to occur during the plant lifetime; however, when integrated over all events in this category, some may be expected to occur during the plant lfietime.

Examples

a. Primary or intermediate sodium pump locked rotor
b. Gaseous waste storage tank rupture
c. Loss of both on-site diesel A-C power units
d. Steam generator tube leak leading to pressurization of the intermediate sodium system

4.2.4 Extremely Unlikely Faults

Definition

An off-normal condition of such extremely low probability that no events in this category are expected to occur during the plant lifetime, but which nevertheless represents extreme or limiting cases of failures which are identified as design bases.

Examples

a. Simultaneous loss of on-site and off-site A-C power sources
b. Major primary sodium pipe rupture
c. A large sodium fire in the containment shell above operating floor
d. Main steam line rupture

4.3 Damage Severity Limits

Damage severity limits are used in combination with the frequency classification to establish design bases for the plant design. Each damage severity limit defines (1) fuel damage; (2) plant equipment damage; and (3) a radioactivity release limit. The following incident definitions provide bases for evaluation of damage severity limits.

No Damage

No damage is defined as (1) no reduction of effective fuel lifetime, or full power capability, below the design value; (2) accommodation of conditions within the fuel and plant operating margins without requiring automatic or manual protective action; and (3) no planned release of radioactivity.

Operational Incident

An operational incident is defined as an occurrence which results in (1) no reduction of effective fuel lifetime, or fuel power capability, below the design value; (2) accommodation with, at most, a reactor trip that assures the plant will be capable of returning to operation after corrective action to clear cause of the trip; and/or (3) plant radioactivity releases that may approach the 10 CFR 20 guidelines.

Minor Incident

A minor incident is defined as an occurrence which results in (1) a general reduction in the fuel burnup capability and, at most, a small fraction of fuel rod cladding failures; (2) sufficient plant or fuel rod damage that could preclude resumption of operation for a considerable time and/or (3) plant radioactivity releases that may exceed 10 CFR 20 guidelines, but do not result in interruption or restriction of public use of areas beyond the exclusion boundary.

Major Incident

A major incident is defined as an occurrence which results in (1) substantial fuel cladding failures or distortion in individual fuel rods, but the configuration remains coolable; (2) plant damage that may preclude resumption of plant operation, but no loss of safety function necessary to cope with the occurrence; and/or (3) radioactivity releases that may exceed the 10 CFR 20 guidelines but are within the 10 CFR 100 guidelines.

4.4 Application of the Frequency and Consequence Classifications to the Plant Design and Evaluation

The consequences of plant conditions of a given frequency are restricted by damage severity limits established as design bases. The plant design is required to control the consequences of these plant conditions within the damage severity limits indicated in Table III.

Plant conditions may need to be reclassified as the design is developed. Further detailed failure evaluations or design changes may modify the estimated frequency of occurrence.

Table III

Design Requirements

Frequency Classification	Damage Severity Limit with reactor shutdown systems operable	Damage Severity Limit with complete failure of one of the independent reactor shutdown systems
Normal Operation	No Damage	No Damage
Anticipated Fault	Operational Incident	Minor Incident*
Unlikely Fault	Minor Incident	Major Incident
Extremely Unlikely Fault	Major Incident	No Requirement

*Although failure of one of the shutdown systems concurrent with an anticipated fault for which protection is required is considered extremely unlikely, the design basis establishes that this concurrent failure must be limited to within the Minor Incident category to assure a prudent, conservative design.

5.0 Safety Related Design Provisions

The preliminary design that has resulted from the application of the Design Guidelines and the more detailed design bases provides defense in depth. The principal safety related design features are described below.

5.1 Reactor Features

Reactivity Characteristics

The reactor design provides reactivity coefficients that assure a large stability margin and facilitate the control of the reactor.

The Doppler reactivity coefficient, which is negative in all core regions, is the most important feedback in assuring stability. The core average Doppler constant is -0.006 Tdk/dT. For a totally voided core (a limiting case for certain accident analyses), the average value is approximately -0.004 Tdk/dT.

The overall sodium temperature coefficient including expansion of sodium and structures is negative and small. The coefficient is position dependent, being positive in the central region and negative in the peripheral regions. Sodium voiding can produce a positive reactivity feedback, depending on the assembly location, but the effect is less than 10¢ for any assembly. Voiding the core and blankets (but not the control channels) would result in a reactivity increase of only about 60¢. The additional voiding of the control channels would result in an overall negative reactivity.

Bowing of the fuel assemblies is controlled by the core restraint system design so that a negative power coefficient results at all power levels, under transient and steady-state operation. The fuel and structure reactivity expansion coefficients are small. The core is radially restrained from motion, resulting in a small radial coefficient during a full power transient. The fuel axial expansion reactivity coefficient is also small and negative, but because of uncertainties in fuel-cladding interface forces, no credit for the effect is assumed in the safety analyses. The cladding axial expansion reactivity coefficient is also negative, but is not as prompt as the fuel component alone.

The temperature coefficients combine to give a prompt negative power coefficient for all conditions and a large stability margin exists in the plant.

Fuel Assembly Features

Features are included in the design of the reactor core assemblies to prevent total flow blockage or mislocation of assemblies into positions which could cause overheating or unacceptable reactivity effects.

Each assembly contains multiple openings in the inlet nozzle, so that no foreign object could conceivably result in total flow blockage. Blockage of an opening would not significantly affect the flow rate or the temperatures.

The lower nozzles also provide a mechanical key system to prevent placement of assemblies into adverse unplanned core locations. Typical positioning errors which are precluded include the location of a highly enriched assembly in a higher power zone and the location of a fuel assembly in a control rod location. The keying arrangement guarantees the location of assemblies in positions of acceptable cooling and reactivity.

Holddown Features

The fuel assembly and core support structure designs provide hydraulic holddown during all conditions of flowing sodium. The design provides a low pressure plenum in the core support structure, which is in communication with the core outlet pressure. This low pressure sodium below the fuel assembly largely offsets the normal upward force associated with the pressure drop of the flowing sodium. The assembly is then held down by the net force, which is essentially that due to gravity.

A mechanical holddown grid is included in the upper reactor internals. This grid serves as a backup to the hydraulic holddown and would limit the upward motion of any assembly to less than two inches in the event of failure of the hydraulic holddown. The reactivity effect of such limited motion for one or several assemblies is very small.

5.2 Heat Transport System Features

<u>Provisions for Reliable Cooling</u>

The multiple sodium system loops provide cooling during normal operation, shutdown, and emergency conditions. A malfunction in one loop must not prevent the other loops from functioning. This is accomplished by the following system design features: The primary system operates at a low pressure (less than 250 psia at any point) with the pressurization supplied by the main circulating pumps. Tripping these pumps permits the system to be rapidly depressurized as the flow coasts down to the levels provided by the pump pony motors. Since the pumps are tripped when the reactor is tripped to minimize thermal shock to the components, a relaxation of system dynamic pressure occurs as a standard operating procedure.

Because a loss-of-coolant accident in a sodium system is essentially a draindown or pump-out condition, elevation and/or containment of the system by guard tanks can be employed, in combination with tripping the pumps, to limit the loss of coolant inventory (See Figure III). By restricting the loss of sodium inventory such that the submergence of the reactor vessel main coolant nozzles is maintained, the other unaffected primary loops and associated primary sodium system components remain operable as redundant emergency, and long-term decay heat removal systems. The flow provided by the coastdown of the pumps, following the reactor and pump trips, cools the core in the period immediately following a system rupture.

Since the reactor vessel outlet nozzles must remain covered after an accident, the elevation or containment of the remainder of the system is considered in relationship to those nozzles. As shown in Figure III, all major components and directly attached piping are contained in guard vessels which provide double containment to an elevation above the reactor vessel outlet nozzles so that the pumps, when operating at pony motor speed, cannot pump sodium above the guard vessels. The primary sodium piping, except for the connections to the components within the guard vessels, is installed above this minimum elevation without additional containment. The guard vessels are sized to prevent the sodium level in the vessel from dropping below the outlet nozzle elevation.

In the event of a pipe leak, the remaining intact primary sodium loops and their associated heat transport systems are available to provide a reliable method of long-term core cooling. Even without electrical power to drive the primary and intermediate sodium system pumps, the system is capable of removing decay heat by natural circulation as long as water is supplied to the evaporators.

Prevention of Gas Entrainment

The primary sodium system is designed to avoid gas entrainment since large coherent bubbles could result in undesirable heat transfer and reactivity effects. The design includes a number of provisions to assure that large gas bubbles will not enter the core:

1. A vortex suppressor plate in the upper vessel plenum provides a quiescent pool and inhibits gas entrainment. The vessel outlet piping and the pump impeller are sufficiently submerged to avoid entrainment.

2. The pump lubricant is isolated from hot sodium by a multiplicity of seals, gas barriers, and leak sumps.

3. No part of the primary sodium system is below atmospheric pressure during normal operation, thereby avoiding in-leakage of external gases.

4. Gas accumulation points are limited to locations which are vented.

These design measures should prevent gas bubbles in the primary system. However, even if a bubble entered the primary system, turbulence in the inlet plenum and the design of the core support structure and fuel assembly inlet nozzles will break up the bubble before it enters the core (as has been shown in tests for FFTF). Consequently, coherent gas bubbles cannot enter the reactor core.

5.3 Plant Protection System

Reactor Shutdown System

A highly reliable reactor shutdown system is an essential feature for the plant. A design requirement for the shutdown system is that the reliability is sufficiently high so that successful shutdown is assured in the event of off-normal conditions. Based on guidance from the Regulatory assessment of Anticipated Transients Without Scram in water reactors[5], the need to assume scram failure in accident analyses is negated if the failure rate is less than about 10^{-7} per reactor year. The design of the shutdown system is aimed at achieving this high reliability.

The design includes two independent fast acting shutdown systems, each capable of terminating anticipated transients without action of the other system. The systems are actuated by functionally diverse signals wherever possible. Equipment diversity is being included in the design to minimize the possibility of degradation of the shutdown system due to postulated common mode failures. Both shutdown systems use mechanical poison rods inserted from above the reactor. Differences in the absorber assembly design, in the control rod drive mechanism design and in the disconnect and insertion features are planned. Extensive testing and evaluation programs are planned to assure reliable performance of all components required for shutdown.

Although no driving force which could eject a control rod has been identified, the drive mechanisms contain an anti-ejection device that positively prevents any rapid upward motion of the rod which would result in more than a few cents of reactivity.

Control rod withdrawal speeds are limited so that any malfunction resulting in inadvertent withdrawal is readily accommodated by the protection system.

Sodium-Water Reaction Protection

The protection system includes provisions to limit the consequences of potential sodium-water reactions that could occur as a result of a failure in the steam generator. The protection system detects leaks through hydrogen or oxygen detection or by an overpressure in the intermediate sodium system. Upon detection of a large sodium-water reaction, the protection system vents the sodium system of the affected loop to a sodium-water reaction products tank and shuts off and drains the water to minimize the extent of the reaction. The plant is designed so that any pressure loadings resulting from sodium-water reactions will not fail the intermediate heat exchanger (so that no radioactive sodium is involved) and will not result in failures in any of the other loops (so that heat removal capability is retained).

Other Plant Protection System Functions

The plant protection system also provides for containment isolation in the event of excessive radiation levels in the plant, startup of emergency diesel generators if offsite power is lost, and actuation of any devices required to remove decay heat upon reactor shutdown.

5.4 Power Supply Features

In addition to ties with offsite electrical transmission systems, an emergency power system is provided to supply important auxiliaries including engineered safety equipment in the event of loss of the preferred power supplies. The emergency power system supplies the loads necessary to shut down the reactor, remove residual heat for extended periods and indicate the status of plant parameters. Two diesel generators, each capable of carrying the emergency load, are automatically started upon loss of offsite power. The diesel generators are hardened and are designed to be up to speed and capable of accepting load in ten seconds. Batteries are provided to maintain continuity of supply to vital instruments. In the event of a delayed startup of the diesels, the batteries can supply vital loads. This requirement is minimized by the natural circulation capability provided in the design. A system of inverters brings these batteries into operation automatically after failure of incoming electrical and diesel supplies.

The redundancy of supplies, transformer and buses and the division of critical loads among the buses results in a system of high reliability. The physical separation of buses, switchgear, and components is intended to limit or localize the consequences of electrical faults or mechanical failures occurring anywhere in the system.

5.5 Containment Features

The fuel rod cladding and the primary sodium system boundary provide the first two barriers to prevent the escape of radioactive fuel and fission products to the environment. The plant design provides additional barriers. These include the compartments which house the reactor vessel and the other primary sodium system components. These compartments are heavy concrete structures lined with steel plate. An inert atmosphere (nitrogen) is used in those compartments of the containment where there are large volumes of sodium such as the primary sodium system. The inert atmosphere prevents sodium fires and minimizes the consequences of any sodium spills or leaks.

The outer barrier is a low leakage containment designed for a pressure of 10 psig and a leak rate of 0.1%/day at the design pressure. This pressure capability provides a large margin since no accidents are calculated to pressurize the containment to more than a few psig. The containment is designed to withstand the extremely unlikely natural phenomena including tornado generated missiles.

Plutonium handling and storage will be limited to specifically designated and controlled areas within the containment. Strict security measures and accountability procedures will be enforced to safeguard all plutonium at the plant.

5.6 Capabilities in Design Level Three

The design features described above provide capability to accommodate the four categories of plant conditions identified in Section 4; namely, Normal Operation, Anticipated Faults, Unlikely Faults, and Extremely Unlikely Faults. Many of these same design features provide capability to accommodate even more severe occurrences than are considered in the four categories of plant conditions. For example, the reactor containment capability is not challenged by any of these events and thus provides a large margin for unforeseen events.

Other design areas in which substantial capability is being provided in design level three include:

1. The reactor vessel head assembly is made as thick as can be adequately welded and tested to code requirements.

2. An array of head holddown bolts is designed to restrain the head and absorb energy in the event of a large impact loading on the head.

3. The primary system is designed to retain its integrity and its ability to remove decay heat following dynamic loadings associated with a core energy release of about 300 MW-sec available work energy. The clearance between the reactor vessel and guard vessel is designed to avoid interaction in the event of these dynamic loadings and the vessel support is designed to accommodate the transmitted loads.

4. The fuel assembly and core support structure designs provide substantial barriers to any molten fuel. Heat removal paths are available to cool accident products in the core or in the core support structure.

5. The intermediate sodium system and the steam system are designed so that a major sodium-water reaction will not release primary sodium system products to the environment or damage heat removal equipment in the other loops.

6. The reactor cavity is designed with a pressure capability of 35 psig although predicted pressures from accident conditions are substantially less.

These capabilities provide margin for a spectrum of unforeseen occurrences even more severe than those in the four categories of plant conditions. Analyses of a range of hypothetical occurrences are in progress to assess their consequences and to provide a further measure of the plant capabilities. These hypothetical occurrences include flow transients and reactivity insertion transients with an arbitrarily assumed failure to scram; also, limiting pipe ruptures with protective action are being evaluated. Although these analyses are not yet complete, it is anticipated that the results will show that the plant design can accommodate the best estimate of the consequences of these hypothetical occurrences.

The plant design has retained the flexibility for a number of potential fallback options in the event that the continued assessment of plant capabilities and the evaluation of level three events indicates that additional margin is desirable. This flexibility includes the potential addition of a sealed head compartment to provide an additional barrier to the release of radioactive materials through the reactor vessel closure head, and a space below the reactor vessel cavity in which additional cooling equipment could be located. The use of these fallbacks could entail penalties in areas such as maintainability and inspectability and a detailed review is required to determine whether such features would be effective and would contribute to increased overall safety.

The consideration of design level three events and capabilities must not overshadow the areas in which the greatest safety assurance can be developed; i.e., in design level one, which provides reliable operation to prevent accidents; and in design level two which assures that all foreseeable accidents can be accommodated safely.

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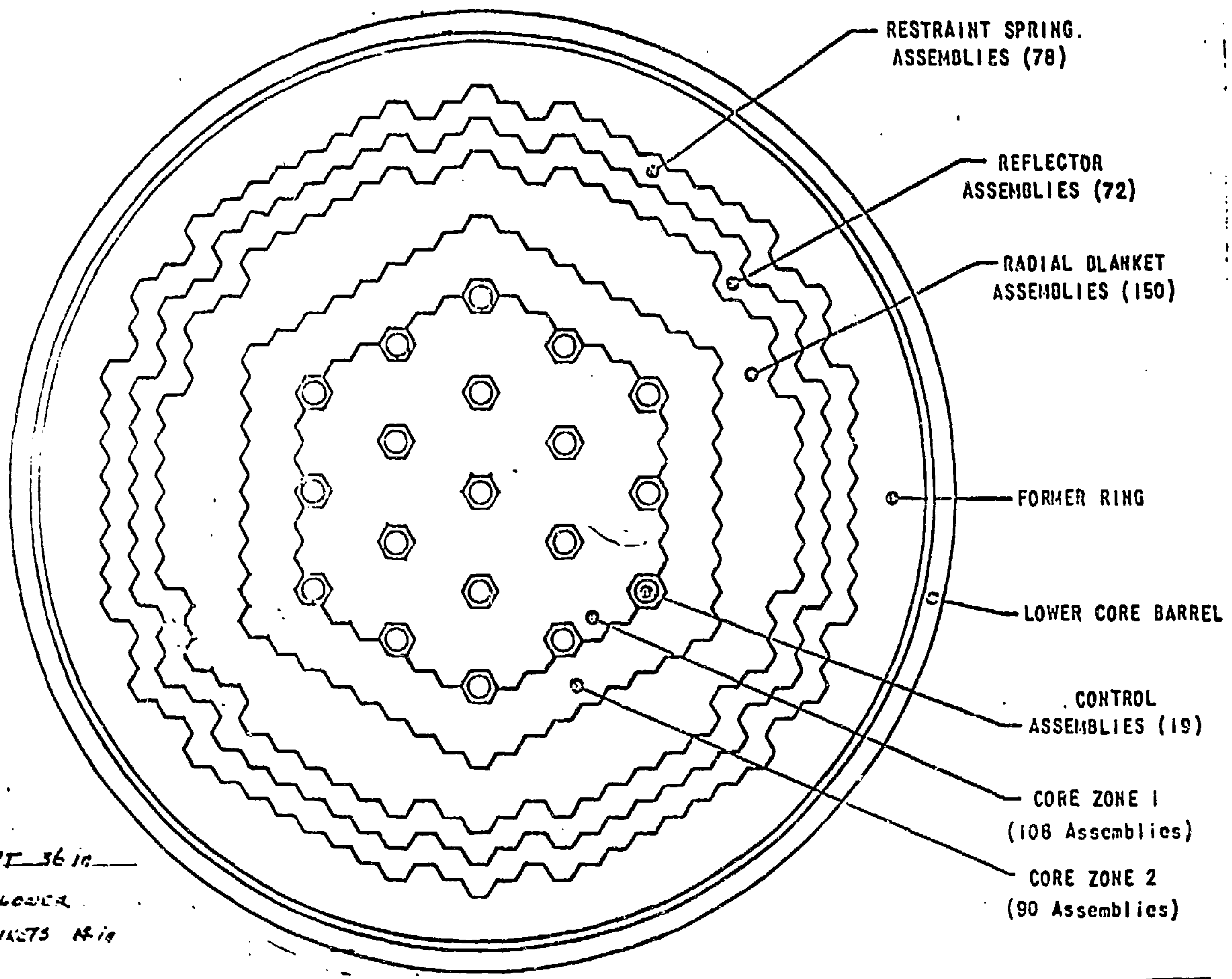

CORE HEIGHT 36 in
UPPER AND LOWER
AXIAL BLANKETS 14 in

Figure I. Plan View of Initial Core Arrangement

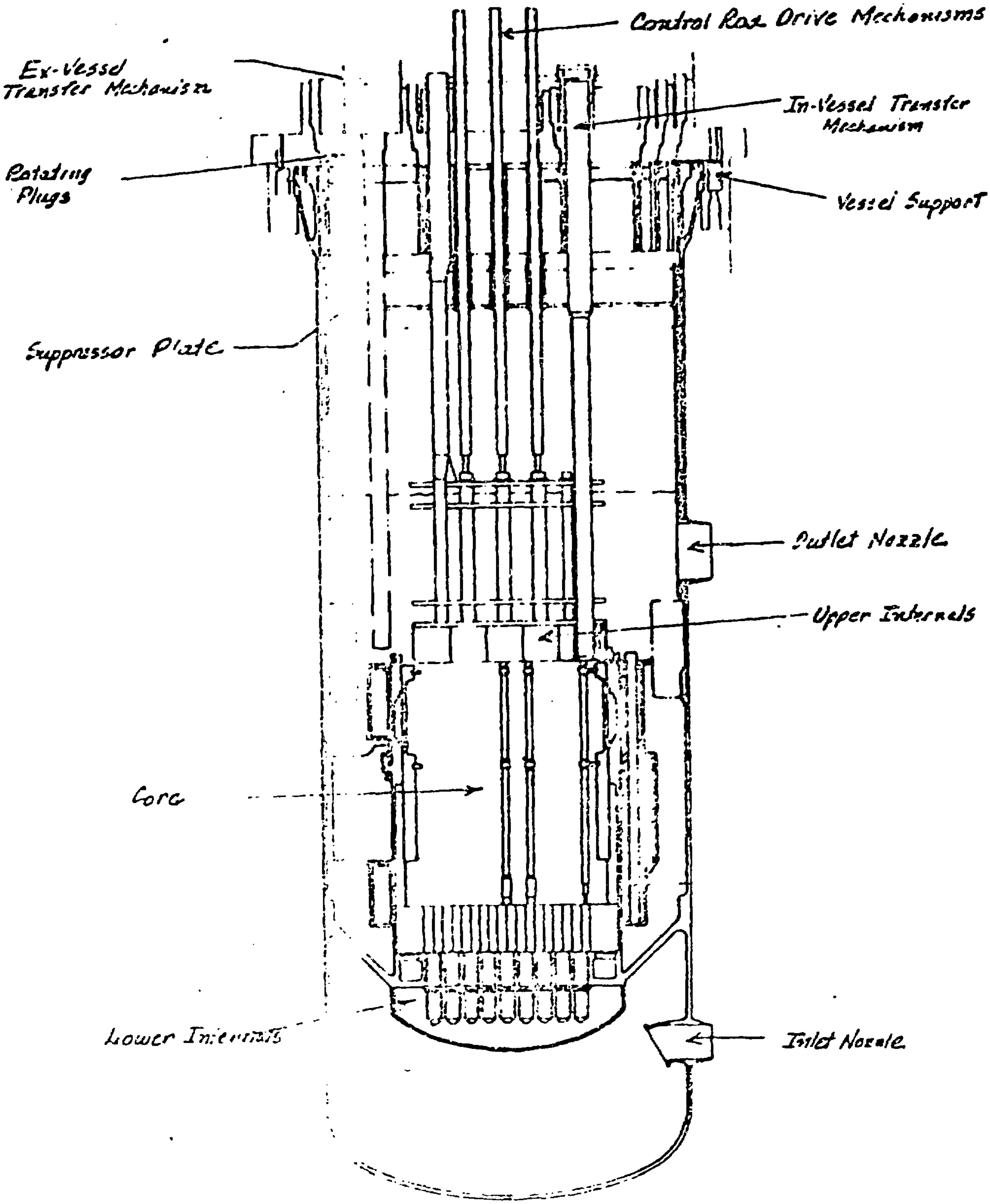

Figure II. Reactor and Closure Head Schematic

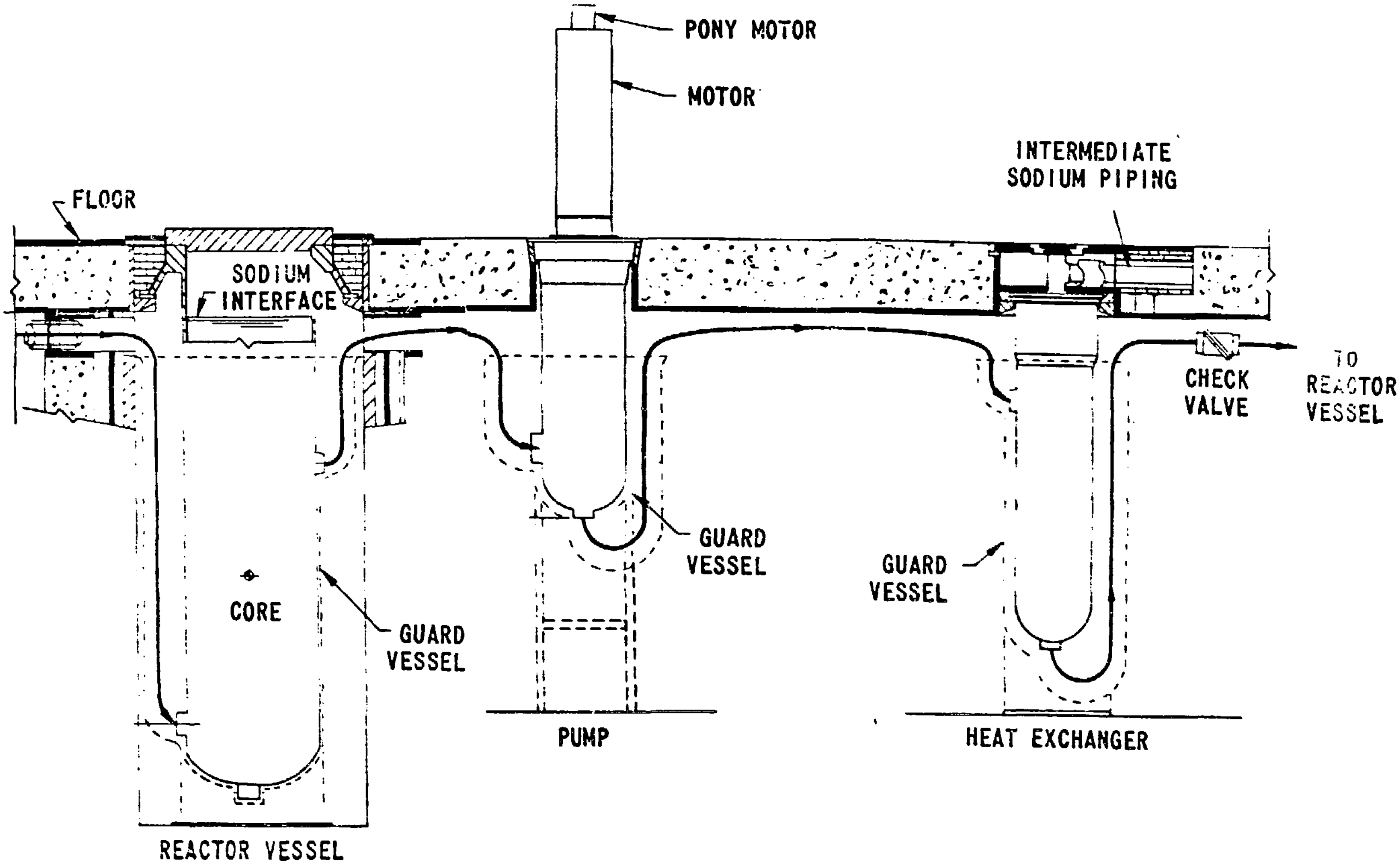

Figure III. Primary Sodium Loop Schematic