Westinghouse Non-Proprietary Class 3

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IRIS
Plant Description Document

Westinghouse

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WESTINGHOUSE NON-PROPRIETARY CLASS 3

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IRIS Plant Description Document

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FOREWORD

The IRIS project shares a number of design solutions and components, especially outside the reactor vessel, with the Westinghouse passive designs AP600 and AP1000. Due to the limited nature (in terms of time and effort) of this pre-application phase, the approach in this document (as well as other forthcoming documents) has been to address, as exhaustively as possible at the current stage of the IRIS design, the new design and engineering features which are characteristics and unique to IRIS. Those elements and components which are essentially identical to the AP designs have been mentioned only very briefly or omitted altogether.

A complete documentation will be provided later before starting the design certification phase. At this time it was deemed preferable to concentrate our efforts and those of the NRC reviewers on the unique IRIS characteristics. If the staff still feels that other information needs to be provided, we will be ready to comply.

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LIST OF ACRONYMS

ACRS	Advisory Committee on Reactor Safeguards
ADS	Automatic Depressurization System
AE	Architect-Engineer
ALARA	As Low As Reasonably Achievable
ALWR	Advanced Light Water Reactor
AMSAC	Anticipated Transient without Scram Mitigation System Circuit
ANS	American Nuclear Society
ANSI	American National Standards Institute
AP	Westinghouse Advanced Passive Pressurized Water Reactors (AP600, AP1000)
APWR	Advanced Pressurized Water Reactor
ARSAP	Advanced Reactor Safety Analysis Program
ASME	American Society of Mechanical Engineers
ATWS	Anticipated Transient without Reactor Scram
BOP	Balance-Of-Plant
BWR	Boiling Water Reactor
CCW	Component Cooling Water
CCWS	Component Cooling Water System
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
CMP	Canned Motor Pump
CMT	Core Makeup Tank
CPS	Condensate Polishing System
CPSS	Containment Pressure Suppression System
CR	Control Room
CRDM	Control Rod Drive Mechanism
CV	Containment Vessel
CVCS	Chemical and Volume Control System
CWS	Plant Circulating Water System
DBA	Design Basis Accident
DC	Design Certification
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
DOE	Department of Energy
DVI	Direct Vessel Injection
DWS	Demineralized Water Transfer System
EBS	Emergency Boration System
EBT	Emergency Boration Tank
EC	Eddy Current
ECCS	Emergency Core Cooling System
ECR	Emergency Control Room
ECS	Main AC Power System
EDS	Non-Class 1E DC and UPS System
EFPM	Equivalent Full Power Month
EHRS	Emergency Heat Removal System
EMD	Electro-Mechanical Division (Curtiss-Wrigth)
ESF	Engineered Safety Features
ESFAC	Engineered Safety Features Actuation Circuitry
ESP	Early Site Permit
FID	Fixed Incore Detector
FMEA	Failure Modes and Effects Analysis
FMECA	Failure Modes, Effects, and Criticality Analysis
FPS	Fire Protection System
FSAR	Final Safety Analysis Report
FWS	Main Feedwater System
GA	General Arrangement Drawing
GDC	General Design Criteria
GRCA	Gray Rod Cluster Assembly
HCOT	Helical Coil, Once Through
HX	Heat Exchanger
HVAC	Heating, Ventilating and Air-

	Conditioning
I&C	Instrumentation and Control
ID	Inner Diameter
IDS	Class 1E DC and UPS System
IFBA	Integral Fuel Burnable Absorber
IHP	Integrated Head Package
IIS	Incore Instrumentation System
IM/P	Integral Motor/Propeller
INPO	Institute of Nuclear Power Plant Operations
IRC/ORC	Inside Reactor Containment/ Outside Reactor Containment
IRIS	International Reactor Innovative and Secure
ISI	In-Service Inspection
ISIS	Inherently Safe Immersed System
L/D	Length to Diameter
LGMS	Long Term Gravity Makeup System
LMFBR	Liquid Metal Fast Breeder Reactor
LMTD	Log Mean Temperature Difference
LOCA	Loss-Of-Coolant Accident
LTOP	Low Temperature Overpressure Protection
LWR	Light Water Reactor
MCR	Main Control Room
MPS	Makeup Purification System
MFIV	Main Feedwater Isolation Valve
MSIV	Main Steam Isolation Valve
MSR	Moisture Separator Reheater
NPSH	Net Positive Suction Head
NRC	Nuclear Regulatory Commission
NRHRS	Normal Residual Heat Removal System
NSSS	Nuclear Steam Supply System
O&M	Operation and Maintenance
OBE	Operating Base Earthquake
OFA	Optimized Fuel Assemblies
ORE	Occupational Radiation Exposure
OSHA	Occupational Safety and Health Administration
P&ID	Piping and Instrument Drawings
PCCS	Passive Containment Cooling System
PORV	Power Operated Relief Valve
PRA	Probabilistic Risk Assessment
EHRS HX	Emergency Heat Removal System Heat Exchanger
PSS	Primary Sampling System
PWR	Pressurized Water Reactor
QA	Quality Assurance
QAP	Quality Assurance Plan
RC	Reactor Coolant
RCCA	Rod Cluster Control Assembly
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RPI	Rod Position Indicator
RPV	Reactor Pressure Vessel
RSR	Remote Shutdown Room
RV	Reactor Vessel
RVI	Reactor Vessel Internals
RWST	Refueling Water Storage Tank
SAR	Safety Analysis Report
SER	Safety Evaluation Report
SFP	Spent Fuel Pit
SFPCS	Spent Fuel Pit Cooling System
SFW	Startup Feedwater
SFWS	Startup Feedwater System
SG	Steam Generator
SGS	Steam Generator System
SGTR	Steam Generator Tube Rupture
SSE	Safe Shutdown Earthquake
SPX	Superphenix
SV	Safety Valve
SW	Service Water
SWPS	Solid Waste Processing Center
T/G	Turbine Generator
TMI	Three Mile Island
TSC	Technical Support Center
URD	Utility Requirements Document
U.S.	United States

US	Ultrasonic
UT	Ultrasonic Testing
VEHS	Main Control Room Emergency Habitability System
VI	Visual Inspection
VWS	Central Chilled Water System
WGS	Gas Radwaste System
WLS	Liquid Radwaste System
WCAP	Westinghouse Topical Report Prefix
ZOS	Onsite Standby AC Power Systems

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1.0 PLANT DESIGN CRITERIA

1.1 SCOPE AND INTRODUCTION

This section contains current design criteria for the IRIS (International Reactor Innovative and Secure) nuclear power plant, including the NSSS and BOP systems as well as the associated buildings. Some design criteria must be met by the IRIS design for licensing or other compelling technical or economic reasons, while other design criteria are goals for the plant design which will be pursued on a cost effectiveness basis. Where practical the cost effectiveness of achieving design goals will be quantified. In addition to the principal IRIS design criteria included in this section, the IRIS designers will ensure that all applicable regulatory and industry requirements are met. For example, IRIS is being designed to meet the intent of the ALWR utility requirements specified in Volume III of the ALWR Utility Requirements and the criteria provided in this section are consistent with the passive plant requirements contained in Volume 1, ALWR Policy and Summary of Top-Tier Requirements.

IRIS has an integral reactor vessel (RV) that houses not only the nuclear fuel and control rods, but also all the major reactor coolant system (RCS) components. This includes: eight small, canned, reactor coolant pumps (RCPs); eight modular, helical coil bundle, once through steam generators (SGs); and a pressurizer and heaters located in the RV upper head. This integral RV arrangement thus eliminates the individual component pressure vessels and large connecting loop piping between them, which enables the containment diameter to be significantly reduced compared to the containment of a loop-type PWR. Thus, at the same stress level and thickness in the metal shell, the spherical IRIS containment vessel (CV) can take a pressure at least three times higher than today's typical large cylindrical containments.

The IRIS RV is contained in a spherical, steel CV that is 25 meters (82') in diameter (see Figure 1.1-1). The CV is constructed of 44-½ mm (1-¾") steel plate and has a design pressure of 1.3 MPa (188 psig). The containment vessel has a bolted and flanged closure head at the top that provides access to the RV upper head flange and bolting. Refueling of the reactor is accomplished by removing the containment vessel closure head and installing a sealing ring between the CV and RV flanges, and removing the RV head. Note that a permanent seal ring is in place between the refueling cavity and the CV flange. The refueling cavity above the containment and RV is then flooded, and the RV internals are removed and stored in the refueling cavity. Fuel assemblies are vertically lifted from the RV directly into a fuel handling and storage area, using a refueling machine located directly above the CV. Thus, no refueling equipment is required inside containment and the single refueling machine is used for all fuel movement activities.

The IRIS containment includes a pressure suppression system that limits the containment peak pressure to well below the CV design pressure. This suppression pool also provides an elevated gravity driven source of makeup water to the RV following postulated small/medium LOCA events (large LOCAs are made impossible by the integral layout). Additionally, the IRIS containment internal structure includes a flood-up cavity which contains the lower 9 meters (29'-6") of the reactor vessel. This flood-up cavity insures that the lower section of the RV, where the core is located, is surrounded by water following any LOCA event. This water flood-up height is sufficient to provide long-term gravity makeup, so that the RV water inventory is maintained above the core for an indefinite period of time.

The IRIS integral RV, containment, and the RV/CV heat removal system comprises a unique, patent pending, configuration which practically eliminates the need for any safety injection following small-to-medium LOCAs, which are historically the accidents yielding the worst consequences. It is based on thermal-hydraulically coupling the containment and the reactor vessel. The water inventory within the reactor vessel after a LOCA is maintained by rapidly eliminating the pressure differential between the vessel and containment, i.e., the driving force across the break, and therefore the coolant loss. This is accomplished through 1) the small, high design pressure, spherical containment which increases the pressure downstream of the break; 2) the multiple steam generators, which are used to remove the heat directly from inside the reactor vessel, thus reducing the pressure upstream of the break; 3) the Emergency Heat Removal System (EHRS) heat exchangers located outside the containment, which transfer the heat to the environment; and 4) the Automatic Depressurization System (ADS), which depressurizes the RCS for transients that do not result in sufficient reduction in RCS pressure. As shown in Figure 1.1-1, the EHRS heat exchangers are submerged in the Refueling Water Storage Tank (RWST), which is adjacent to the refueling cavity above the CV structure.

Analysis results show that the IRIS safety systems, following the worst combination of LOCA size and axial location, maintain the RV water level and keep the core safely under water for an extended period of time (days), without the need for water makeup. However, water makeup is in any case available, as discussed before. Because no water addition is required even after the worst case postulated LOCAs, IRIS does not require the high capacity, safety grade, high pressure injection emergency core cooling system (ECCS) characteristic of pressurized water loop reactors.

The IRIS containment structures are also designed to minimize the probability of severe accidents that could result in the uncontrolled release of radiation. The flooded reactor vessel cavity, combined with the reactor vessel insulation provides sufficient heat removal from the external RV surface to prevent any vessel failure following beyond design basis scenarios that could result in core damage. In addition, flooding the refueling cavity area and submerging the containment upper closure head and passively cooling the upper containment shell can remove sufficient heat to keep the containment below its design pressure. This containment cooling method, together with the capability to provide gravity makeup to the reactor described above, provides a means of severe accident mitigation that is diverse from the EHRS. This makes the IRIS accident mitigation capability very reliable and will result in reduced core damage frequency.

Figure 1.1-1 provides a conceptual view of the IRIS containment and its main components. Detailed drawings of the IRIS major systems will be presented in Sections 3 and 4.

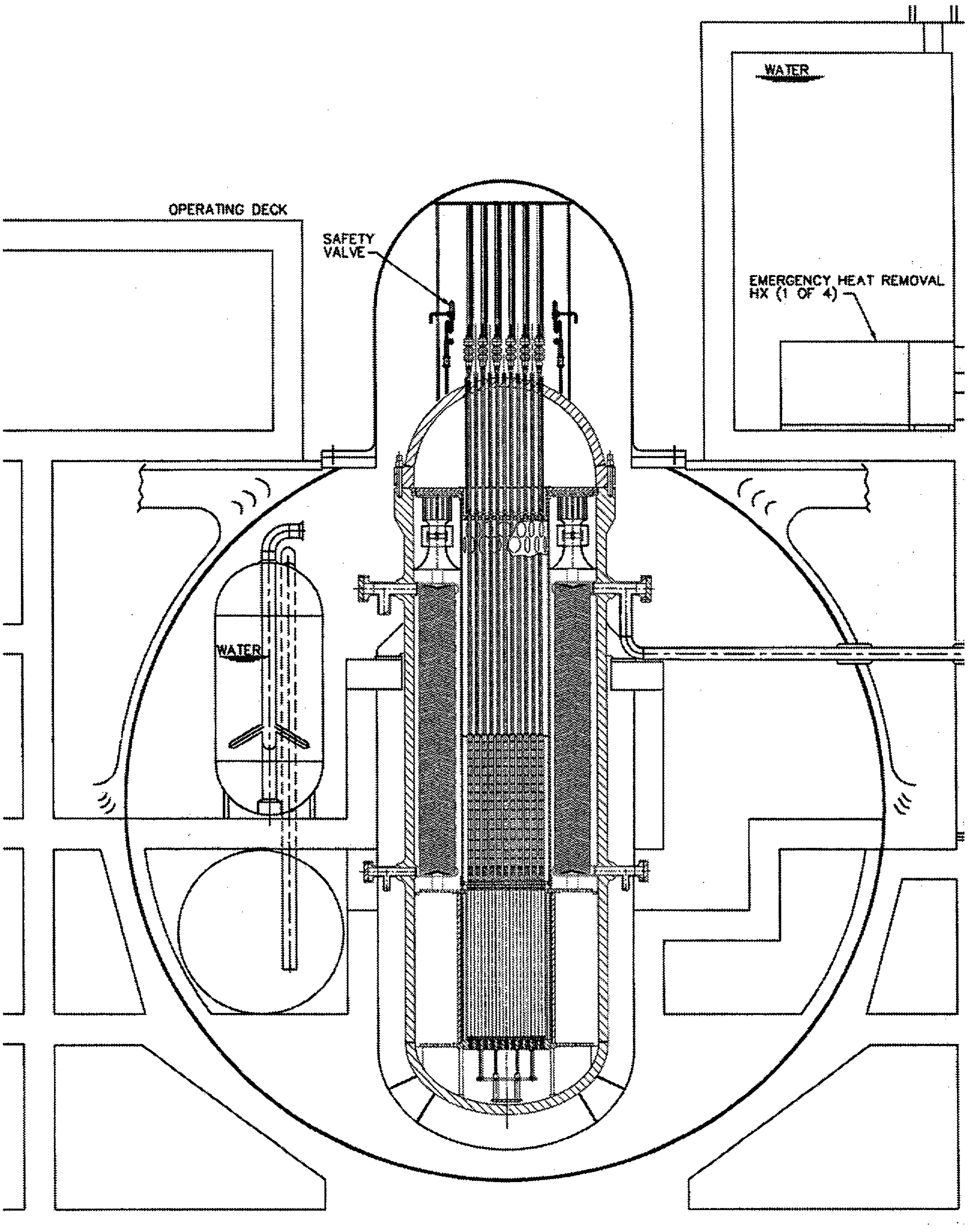

Figure 1.1-1 IRIS Spherical Steel Containment and Major Components

1.2 OVERALL PLANT OBJECTIVES AND DESIGN CRITERIA

1.2.1 Summary

The following is a summary of the major plant design objectives for IRIS.

- Provide a nuclear plant design that can be applied to both developing countries as well as countries with an existing large grid and/or nuclear plant program. This objective is interpreted to mean that the plant should have a small to medium output that is more suitable for addition to small electrical grids, and the plant should be amenable to the series construction of multiple co-located units to meet the demands of a large utility or grid.

- Provide a greatly simplified plant with respect to design, licensing, construction, operation, inspection, and maintenance.

- Reduce the total cost of power so that it is less than or equal to that of other advanced nuclear power plant designs.

- Design the plant so that it can be operated with a five year lead time (owners commitment to commercial operation) or less, and constructed (first concrete to fuel load) in three years or less with a high degree of certainty.

- Ensure that a plant prototype will not be required to obtain design certification, by using proven power generating system arrangements and components, by performing full scale tests of unique components (e.g., spool pumps), or by performing scaled separate effects and/or integral type tests.

- Provide a high degree of licensing certainty by including the following in the design process:

  - Address current licensing issues,

  - Enhance the defense-in-depth safety approach by eliminating, by design, the occurrence of some serious accident sequences, or by decreasing the probability and/or consequences of other serious design basis accidents,

  - Provide passive safety systems that require no operator actions for long periods of time,

  - Reduce core melt frequency to less than $1x10^{-6}$/yr of reactor operation, considering all internal and external events,

  - Reduce significant release frequency to less than $1x10^{-7}$/yr of reactor operation, for all internal and external events,

  - Simplify and reduce the emergency planning zone (EPZ), and as a goal eliminate the need for an EPZ altogether,

- Incorporate the lessons learned from the TMI and Chernobyl related accident issues.

- Design the plant to achieve an availability of 95.2 percent or higher and limit the number of unplanned reactor trips to less than one per year,
- Develop a standard design that is applicable to anticipated US sites.
- Design IRIS to have a 36-month refueling frequency with the potential for a 48-month interval.
- Design the IRIS plant to have a design life of at least 60 years.
- Design the plant in accordance with applicable regulations, codes and standards,
- Ensure that the radiation exposure to plant personnel resulting from normal operation, inspection, and maintenance is less than 65 man-rem/yr.,
- Ensure that the total low level radioactive waste volume produced is less than 870 $ft.^3$/yr. after de-watering. This waste includes spent resins, spent filter elements, tank sludge, chemical wastes, clothing, etc. The total wet radioactive waste volume produced from spent resin and filter elements, tank sludge and chemical waste should not exceed 450 $ft.^3$/yr. (de-watered). Normally non-radioactive, spent, condensate-polishing resins are not included.
- Perform the IRIS plant design, equipment procurement, plant construction, pre-operational testing in accordance with appropriate QA requirements. The QA program shall specify requirements for all the safety related activities, structures, systems, equipment and components. The QA requirements for the non-nuclear systems, components, and structures shall be simplified but will impose requirements for initial purchase, construction, maintenance and spare parts.
- Design the IRIS plant to use an economically optimum number of valves, pumps, heat exchangers, snubbers, and other mechanical components, consistent with the essential functional and safety requirements of the systems, as well as system availability, maintainability, and testing requirements.
- Require a minimum amount of instrumentation, control functions and control loops, consistent with the essential operational and safety requirements of the systems; as well as system availability, maintainability, and testing requirements.
- Design the IRIS plant and control system to simplify operations during all modes of operation, including operator actions to diagnose and manage abnormal and accident conditions.

1.2.2 Power Capability

(1) The net electrical output to the grid shall be at least 335 MWe with a nominal NSSS power (core plus RCP heat) of 1002 MWt.

(2) The plant shall be designed to produce the required net electrical output with up to 5 percent of the steam generator tubes plugged and with a maximum RCS temperature of 623°F.

(3) The design shall be capable of accepting a step load increase or decrease of 10 percent between 25 and 100 percent power without reactor trip or steam dump system actuation, provided the rated power level is not exceeded.

(4) The design shall be capable of accepting a 100 percent load rejection from full power without reactor trip or operation of the pressurizer safety relief valves. The turbine shall be capable of continued stable operation at the minimum house loads.

(5) The design shall be capable of accepting ramp load changes of 5 percent per minute while operating in the range of 25 to 100 percent of full power without reactor trip or steam dump actuation subject to core power distribution limits and provided the rated power level is not exceeded.

(6) The design shall be capable of following the design basis daily load follow cycle for at least 90 percent of the fuel cycle length. The design basis load follow cycle is defined as the daily (24 hour period) cycle of operation at 100 percent power, followed by a 2 hour linear ramp to 50 percent power, operation at 50 percent power, and a 2 hour linear ramp back to 100 percent power. The duration of time at 50 percent power can vary between 2 and 10 hours. This load follow capability must be achievable during 90 percent of each fuel cycle.

- During load following the plant shall be capable of routinely making load changes of ≤10 percent at +/- 2 percent per minute between 50 and 100 percent power without exceeding the core power distribution limits for the purpose of responding to grid frequency changes.

- IRIS shall be designed to minimize changes to the reactor coolant boron concentration during these load follow maneuvers.

1.2.3 Cost and Schedule

(1) The capital cost of an IRIS multiple module power station, on a $/total installed kW basis, shall be equal to or less than other light water nuclear power plant designs with passive safety features built to U.S. requirements.

(2) The cost of electricity produced by IRIS shall be competitive with light water nuclear power plants as well as with fossil power plants of both larger and smaller sizes. In addition, the cost shall be predictable with a high degree of certainty.

As a goal, the cost of electricity produced by this plant should be less than that of other light water nuclear power plants, as well as fossil power plants of both larger and smaller sizes.

The cost of electricity will be the total life cycle costs including capital, interest and escalation, fuel, operating and maintenance, and decommissioning.

(3) As a goal, the fuel cycle cost should be less than that of nuclear power plants with shorter (18 to 24-month) fuel cycles, compared on an equal energy basis.

(4) The total plant lead time for IRIS shall be less than 60 months from time of the project commitment to construct the plant to commercial operation. The owner's commitment to construct is made after certification of the plant and after the owners site has been reviewed and licensing approval has been obtained. The period of time from the first structural concrete pour to completion of the plant warranted demonstration run shall not exceed 36 months, with 6 months for startup testing. In addition, the construction schedule shall be predictable with a high degree of certainty.

1.2.4 Reliability and Availability

(1) The overall plant availability goal of 95.2 percent shall be quantified at the system level. Unavailability allocation goals shall be assigned to individual plant systems with analysis performed on the largest contributing systems to further define their predicted unavailability contributions and critical components.

- Planned Refueling/Maintenance Outage:
  - [ ] (c)
- Minor Outages:
  - [ ] (c)
- Major Outages
  - [ ] (c)

(2) As a goal, the rate of unplanned reactor trips shall be less than one per year.

(3) The plant must have significantly fewer and smaller components, provide significant reductions in building volume, and be easier to construct compared to other passive PWRs. Although these latest design passive safety nuclear power plants have achieved

significant reductions in the number of components, as a goal, the nuclear island should have:

Building Volume	- Containment	75% less/kW
	- Auxiliary Bldg.	10% less/kW
	- Fuel Handling	10% less/kW
	- Turbine Bldg.	10% less/kW
Valves & Pipe	- Safety	50% less
	- Non-safety	10% less
	- Size	20% less/kW
Pumps & Fans	- Safety	none
	- Non-safety	10% less
	- Size	10% less/kW

(4) The plant life shall be 60 years without the planned replacement of the reactor vessel, which shall be conservatively designed to ensure a 60-year life. Note that because of the IRIS integral configuration, the vessel fluence is greatly reduced and it is possible that the life of the vessel can be extended beyond 60 years. Other major components, including the steam generators shall be able to be replaced in a practical manner (no containment modification required), and no cutting and welding of the primary pressure boundary shall be required.

(5) The design of the major components required for power generation such as the steam generators, reactor coolant pumps, fuel, internals, turbine and generator should be based on equipment that has successfully operated in commercial nuclear power plants. Exceptions to this should only be considered where similar equipment has successful operating experience in similar or more severe operating conditions.

If greater changes are incorporated, they shall be conservatively designed and tested to ensure their reliability. A plant prototype shall not be required.

1.2.5 Safety and Licensing

(1) The plant safety systems shall be designed to satisfy both the current deterministic ("single failure," conservative input/methods, etc.) and "defense-in-depth" approach as well as a probabilistic risk assessment approach. This shall include providing diverse means of core cooling and containment cooling in order to reduce the probability of significant damage and/or release of radioactivity.

(2) Licensing certainty of this plant shall be provided as follows:

- Provide greater safety margins and essentially passive safety systems that do not require operator action to maintain core cooling, spent fuel cooling, or containment cooling for at least seven days. After seven days a limited number of operator actions may be required to continue post-accident recovery; e.g., to maintain low containment pressure. Note that in certain unique situations there may be some exceptions that would require limited operator action in less than seven days such as an accident during refueling.

- Provide passive safety systems that do not require operator action to maintain post-accident monitoring, operator habitability, and limited control capability for at least three days.

- Provide on-site capability and equipment to enable the plant personnel to extend the time of all passive system operations to at least one week without off-site assistance; and to at least thirty days with off-site assistance.

- Provide plant safety features that prevent core uncovery following all design basis LOCAs.

- Resolve all current licensing issues; this includes all Unresolved Safety Issues and all high and medium priority, generic, safety issues.

- Assure a core melt frequency of less than 1×10^{-6} per year for internal and external events.

- Reduce the consequences of severe accidents by providing a probability of significant off-site release frequency of less than 1×10^{-7} per year for internal and external events.

- Give special attention to TMI and Chernobyl related accident issues such as operator errors, hydrogen explosions, fires, containment integrity, lessons learned, etc.

- Reduce the potential for serious accident scenarios by designing the plant with a "safety-by-design" approach, which physically eliminates some accident sequences (e.g., large LOCA) and reduces the probability of occurrence or the consequences of other serious design basis accidents. The IRIS safety-by-design approach can eliminate or reduce the probability or consequences of ANS 18.2 design basis accidents.

(3) Severe accidents shall be considered in the design of the containment and its systems:

- IRIS shall be designed such that a postulated core melt will reliably not result in failure of the reactor vessel. With the core retained within the vessel, issues associated with ex-vessel core cooling, core-concrete interaction and non-condensable gas generation, and direct containment heating do not impact containment integrity.

- IRIS shall be designed to provide protection from the reaction of the zircaloy in the cladding of the active fuel by ensuring that the released hydrogen cannot detonate in the containment following postulated core damage events.

- A diverse means of containment heat removal shall be provided so that the probability of containment over-pressure failure, following a LOCA and/or failure to provide core cooling, is essentially eliminated.

- The containment design shall achieve a reduction in the assumed design basis containment leak rate ($\leq$ 0.5 percent/day based on current plant containment volumes), and reduce the difficulty in containment leak testing by facilitating containment leak testing by improving and minimizing penetrations.

(4) As a goal, the need for an emergency planning zone should be eliminated, thereby eliminating the need to develop approved plans for warning, sheltering, and evacuation outside the site boundary. The basis for the elimination is in IRIS safety-by-design approach with increased margins to fuel damage, reduced core damage and severe release frequencies, and reduced off-site doses.

(5) The safety analysis shall be based on nominal full power plus appropriate instrument errors.

(6) The design shall be consistent with applicable NRC regulations and standards except as noted in these design criteria.

1.2.6 Site

(1) As a goal, the plant should be designed for location at most existing nuclear sites in the U.S. and for a reasonable range of new sites including sites in foreign countries. Refer to Table 1.2-1 for a summary of the site interface goals.

(2) As a goal, on site storage capacity for spent fuel corresponding to twelve years of plant operation and for other solid radioactive waste corresponding to six months of plant operation, shall be provided. The spent fuel storage capability should include the removal of the whole core at twelve years. There should also be storage for 1/2 core of new fuel (this storage need not be in the spent fuel pit).

(3) The plant shall be designed to minimize the probability of loss of investment to the plant owner. In addition to serious accidents, this shall include consideration of less severe events that could result in major plant outage, cleanup, and/or repair costs.

(4) As a goal, a fuel defect level of 0.025 percent should be used in the design and optimization of the waste processing systems and also in the calculation of normal activity releases. The waste processing systems shall allow operation of the plant with the fuel defect level as high as 0.25 percent. The shielding of equipment not involved in accident mitigation shall be designed for a fuel defect level of 0.25 percent.

Table 1.2-1

Site Interface Goals for a US Site

Normal Heat Sink	
- Source	Wet Cooling Tower
Air Temperature Limits	
- Maximum Safety	115°F dry bulb/80°F coincident wet bulb 81°F wet bulb (non-coincident)
-Minimum Safety	-(40)°F
- Maximum Normal	100°F dry bulb/77°F coincident wet bulb 80°F wet bulb (non-coincident)
- Minimum Normal	-(10)°F
Wind Speed Limits	
- Operating Basis	145 mph; importance factor 1.15 (for safety), and 1.0 (for non-safety)
- Safety Basis	Maximum wind velocity = 300 mph (Tornado) Translational wind velocity = 60 mph maximum = 5 mph minimum
Makeup Water Volume	5200 gpm (1)
Seismic	
- SSE	0.30g peak ground acceleration (2)
- Fault Displacement	No potential
Soil	
- Bearing Strength	≥ 8 KSF allowable static bearing pressure at a depth of 49 feet under all specified conditions.
- Shear Wave Velocity	Greater than or equal to 1000 ft./sec.
- Liquefaction Potential	None

Table 1.2-1 (Cont.)

Site Interface Goals for a US site

Missiles	
- Tornado	4000 - lb. automobile at 105 mph horizontal, and 74 mph vertical 275 lb. 8" shell at 105 mph horizontal, 74 mph vertical 1" diameter steel ball at 91 mph in any direction
Flood Level	Less than or at finished plant grade elev.(max.)
Ground Water Level	Less than 1 meter (3.3 ft.) below grade (max.)
Precipitation	
- Rain	19.4 in./hr. (6.3 in./5 min.)
- Snow/Ice	75 psf static load, with exposure factor 1.0 with importance factor of 1.20 (safety) and 1.0 (non-safety)
Short term (0-2 hour) Dispersion Value - X/Q	$1x10^{-3}$ sec/m^3
Population Distribution	
- Exclusion area (site)	0.5 mi
- Low Population	(3)
- Population Center	(3)

(1) Sufficient water for normal cooling tower, turbine island, and NSSS makeup.

(2) With RG 1.60 spectra, amplified in high frequency range

(3) Population distribution should be consistent with current sites.

1.3 REACTOR COOLANT SYSTEM

1.3.1 Reactor Core

(1) The core shall be designed to provide a 36-month fuel cycle assuming a 95.2 percent capacity factor (34.2 EFPM). The plant design shall also be able to accommodate an advanced core design capable of operating with a 48-month fuel cycle.

(2) The fuel mechanical design shall be capable of achieving an assembly-average burnup of at least 62,000 MWd/MtU.

(3) The moderator temperature coefficient shall be negative over the entire fuel cycle and at any power level with the reactor coolant at normal operating temperature.

(4) A two-region core shall be used in the 36-month base case fuel cycle design.

(5) Adequate margin shall be provided to demonstrate that DNB will not occur on a 95 percent probability, 95 percent confidence basis for all Plant Condition I and II events.

(6) The hot channel bulk exit quality under nominal full power conditions shall be equal to or less than zero.

1.3.2 Reactor Vessel

(1) The reactor vessel shall be designed to contain all the reactor coolant system major components, including the core, the control rods, the control rod supports, the control rod drive-rods, the core support structure, the core reflector, the internals, the pressurizer, the reactor coolant pumps, the steam generators, shielding as required, and instrumentation. (See below for specific criteria for the steam generators, reactor coolant pumps, and pressurizer).

(2) The core shall be located as low as possible in the reactor vessel to maximize the water inventory above the core and the natural circulation capability of the reactor coolant system.

(3) The reactor vessel and internals shall be designed so that the coolant temperatures at the mating flanges between the upper head and vessel cylinder are similar (i.e., at Thot).

(4) The reactor vessel head and penetrations shall be designed to operate in a 15.52 Mpa (2250 psia) saturated steam environment.

(5) The reactor vessel internals shall be designed to include the pressurizer heaters, the pressurizer in and out-surge flow paths, and a means of mixing the pressurizer water with the circulating primary water to ensure uniform soluble boron concentration throughout the reactor vessel (without using flow through the spray nozzle).

(6) The reactor vessel internals, together with the CRDM penetration tube design are to ensure that the CRDM housing contains primary water.

(7) The vessel design life shall be 60 years with less than $2x10^{19}$ nvt fast neutron exposure, (E > 1 Mev). As a goal, the vessel exposure shall be < $1x10^{17}$ nvt, and the vessel life shall be substantially more than 60 years.

(8) The lower internals shall be designed to prevent flow jetting from the reflector into the core.

(9) A radial neutron reflector shall be employed.

(10) The reactor vessel shall not have any penetrations lower than 2 meters above the top of the core.

(11) The reactor vessel shall not have any primary fluid piping penetrations larger than 4-inch, Schedule 160 pipe.

(12) The reactor vessel shall not have any primary fluid piping penetrations larger than 2-inch, Schedule 160 pipe within 8 meters from the top of the core.

(13) The reactor vessel and internals shall be designed to contain and support the steam generators and the reactor coolant pumps.

(14) The reactor vessel internals (with the steam generators) shall be designed to ensure that a single-phase water natural circulation flow path for primary fluid, for the removal of core decay heat via the SGs, is maintained, even at the lowest expected reactor vessel water level.

(15) The reactor vessel shall contain a feed water inlet and steam outlet nozzle for each steam generator. These nozzles shall include a method for attaching and sealing the SG headers to the RV inside wall surface. These nozzles shall be sufficiently large to provide access to the SG headers for tube inspection and normal maintenance (tube plugging and sleeving, and for tube inlet orifice inspection and repair and replacement).

(16) The reactor vessel to steam generator header attachment shall include a method for monitoring and verifying the integrity of the pressure boundary between the primary and secondary fluids, if a bolted attachment is used.

(17) The rector vessel upper head shall be designed to provide the pressurizer function for IRIS.

(18) The reactor vessel head and internals shall be designed for an 89 assembly core, with up to 45 CRDMs, with up to 90 pressurizer heater rods, with up to 40 in-core instrumentation tubes, and with up to 10 excore instrumentation tubes routed to the downcomer region of the reactor vessel.

(19) An integrated head package which contains the CRDMs, instrument columns, cooling

fans, insulation, missile shield and package lift rig shall be employed.

(20) A permanent seal ring adapter shall be provided to allow easy installation of the seal between the vessel flange and the containment flange prior to refueling activities.

(21) The reactor coolant system arrangement shall be such that the maximum expected reactor decay heat can be removed by a single emergency heat removal subsystem and its corresponding two steam generators. This can be accomplished with the RCS operating in natural circulation mode, without exceeding the normal operating temperatures in the primary system.

(22) The reactor vessel and supports shall be designed to be consistent with providing external water cooling of the vessel bottom head sufficient to prevent core melt-through in the event of a postulated core damage event.

(23) The RCS components shall be arranged with consideration of component maintenance, impact on outage schedules, personnel access, and radiation exposure.

1.3.3 Steam Generator

(1) A once through, helical-coil steam generator that is located inside the RV shall be utilized. The steam generator shall employ thermally treated Inconel 690 tubes and shall include the following major features:
- The steam generator shall be designed to have the secondary side feedwater and steam flow inside the tubes, and the primary side reactor coolant flow on the outside of the tubes.
- The steam generator shall be designed so that access to the tubes for inspection and maintenance is provided without the need to remove the reactor vessel head.
- The steam generator shall be designed to produce superheated steam with a pressure and temperature of 5.8 MPa (841 psia) and 317°C (602°F).
- The steam generator tubes, feedwater supply header, and steam discharge header shall be designed for the pressure and temperature of the primary reactor system, namely, 17.24 MPa (2500 psia) external pressure at 343.3°C (650°F), with no internal pressure.
- The steam generator design shall include means to provide a large pressure drop at the feed water inlet of each tube, to minimize parallel flow instability between tubes and between parallel steam generators, throughout the range of expected operating conditions.

(2) The top of the steam generator shroud and center column support shall be designed to accept the direct attachment of a reactor coolant pump.

(3) The steam generator shall be capable of accepting cold feedwater supply after dryout, without damage.

(4) The steam generator feedwater and steam headers shall be designed to attach to the inside of the reactor vessel and if bolted attachments are used, the attachment shall

include a method for monitoring and verifying the integrity of the pressure boundary between the primary and secondary fluids.

(5) For ease of replacement of the steam generator, bolted connections between the steam generators and reactor vessel shall be considered, so that no cutting or welding of connections/supports is required for removal of the steam generator.

(6) The steam generators are to be designed so that at least four (4) independent steam generators are provided.

(7) The steam generators are to be designed to provide, with the reactor vessel internals, a natural circulation flow path for removing heat from the core with no reactor coolant pumps operating and the water level in the reactor vessel reduced below the pump suction elevation.

1.3.4 Reactor Coolant Pump

(1) Hermetically sealed canned spool pumps that require no seal support systems (e.g., seal injection) shall be employed. These pumps shall be designed to be completely within the reactor vessel eliminating the need for large reactor vessel penetrations for mounting the pumps, and the need for the pump motor to have a high-pressure casing.

(2) The pumpsshall be designed to operate submerged in the reactor coolant at Thot temperatures and full reactor pressure. As a goal, the pumps shall not require an external water support system for bearing and/or motor cooling.

(3) The RCP motor shall be conservatively designed and sized so that one pump can operate alone (with back-flow through the idle RCP/SG's) and so that the pump(s) can operate continuously with cold primary coolant.

(4) The RCP discharge shall be attached directly to the steam generator, to direct all pump flow inside the SG outer shroud and over the outside surface of the steam generator tube bundle.

(5) The RCP shall include sufficient internal rotating inertia to mitigate a complete loss of reactor coolant flow accident.

(6) The RCP impeller and diffuser vanes shall be ground and polished to minimize radioactive crud deposition and to maximize pump efficiency.

(7) The RCP and its attachment to, and location within the reactor vessel shall be designed to facilitate maintenance, if needed. Appropriate lifting and handling devices shall be considered.

(8) If the pump should require cooling water, the RCP shall be designed such that it can operate without damage for a sufficient time such that the loss of cooling can be sensed by instrumentation, and the reactor tripped before pump operation must be terminated. Also, the pump flow coast-down shall not be significantly impaired following this short

operating period with no cooling water.

(9) Consideration shall be given in the bearing design to minimize the deposition of crud, for example by providing sufficient purge flow through the bearings.

1.3.5 Pressurizer

(1) The IRIS pressurizer is located within the reactor vessel upper head (See Section 1.3.2) and shall be designed to provide a barrier to prevent mixing between the normally circulating, subcooled, reactor coolant and the saturated fluid in the pressurizer region. The pressurizer design shall include the following features:

- The pressurizer heaters shall provide 2700 kW of heater capacity
- The pressurizer structures shall be designed to support the heaters, the control rod drive rod sleeves, in core instrumentation tubes, and the pressurizer instrumentation.
- The barrier (inverted top hat) shall have in- and out-surge flow paths. The barrier and flow paths shall be designed such that the pressure differential across the barrier due to the largest in- or out-surges does not result in loss of important functions (e.g., control rod operation).
- The pressurizer/barrier shall provide a low flow rate natural circulation flow path that mixes the pressurizer water with the circulating primary water (to provide mixing of the soluble boron in the primary water).

(2) The pressurizer region of the reactor vessel shall be designed to limit the heat loss between the saturated water in the pressurizer and the subcooled water circulating in the reactor vessel. This heat loss should be less than ½ the power input provided by one group of pressurizer heaters.

(3) The pressurizer shall be designed to provide access such that the reactor head penetrations can be inspected and maintained, and that the heater rods can be replaced.

(4) The pressurizer shall have sufficient steam volume such that with no spray flow and no PORV function, a loss of load transient (or any other Plant Condition II transient) shall not cause the pressurizer safety valves to open, assuming realistic conditions.

(5) The pressurizer water level instrumentation shall provide level measurement accuracy consistent with the small level water height available in the upper head region. This accuracy will consider the need for a normal operating level band as well as off-normal high and low level alarms.

(6) The size of the pressurizer and safety valves shall consider ATWS.

(7) The pressurizer barrier (inverted top hat) shall provide support for the top end of the reactor coolant pumps.

1.4 STEAM AND POWER CONVERSION SYSTEMS

1.4.1 Main Steam System

(1) The steam and feed water piping and appurtenances up to and including the steam and feed water isolation valves shall be designed for full reactor primary side pressure and temperature, thus no steam generator safety valves shall be required.

(2) The steam piping, isolation valves, and their supports shall be designed to be water-filled.

(3) At least four separate steam lines (with isolation capability) shall exit the containment.

(4) At least four separate feedwater lines (with isolation capability) shall enter the containment.

(5) The speed of closure of the MSIV and MFIV shall be $\leq$ 5 seconds.

(6) The steam piping and appurtenances downstream of the steam isolation valves to the turbine stop valves shall be designed for at least 8.28 Mpa (1200 psia).

(7) The steam dump valves shall provide sufficient capacity to allow a complete load rejection without requiring a reactor trip.

1.4.2 Main Feedwater and Condensate Systems

(1) The main feedwater and condensate system shall be designed to limit the iron content in the feed water to $\leq$1 ppb and to provide dissolved oxygen control, in order to minimize deposits in the once through steam generator tubes.

(2) The main feedwater and condensate system shall provide full flow condensate polishing capability.

(3) Regenerative feed water heating, moisture separator heating, and a deaerating heater shall be used.

(4) The main feed water and condensate system shall be designed to have adequate capacity and redundancy for key components to meet the plant availability goals. For example, use of 2 x 100% main feed water pumps, booster pumps, and condensate pumps shall be considered.

(5) The main feedwater and startup feedwater isolation valves and upstream piping and appurtenances shall be designed for full primary side design pressure and temperature.

(6) The main feedwater piping shall be designed to minimize the occurrence of damaging water hammer, such that an empty steam generator can be refilled.

(7) The steam generator chemical addition and mixing, and the steam generator cooling

and draining functions will be integrated into the feed water system.

1.4.3 Turbine Generator

(1) The reference design of the turbine and generator will be for a 60 cycle electrical grid. The design impacts of a 50 cycle grid application for deployment of IRIS outside the US shall be considered.

(2) The reference 60 cycle turbine and generator will be limited to 1800 rpm to maximize reliability. The design impacts of a high speed (e.g. 3600) rpm T/G shall be considered.

1.5 AUXILIARY FLUID SYSTEMS

1.5.1 Engineered Safeguards Systems

(1) The safety systems must be designed to mitigate design basis events with a single failure as well as to provide sufficient reliability to support the core melt frequency and core melt consequence goals.

(2) The safety systems shall be greatly simplified and maximize the use of passive devices which use only gravity and natural circulation flow; they shall not use active components such as pumps, fans or diesel generators. These passive systems can use a minimum number of valves for the purpose of initially aligning the safety systems. Where valves are required for re-alignment, they should be "fail safe" wherever possible.

(3) The safety systems shall not require safety grade support systems such as AC power, CCW/SW, HVAC, etc.

(4) The passive safety system mitigation capability must consider all DBA's including those from shutdown conditions and all NRC criteria must be satisfied with appropriate assumptions with respect to initial conditions, use of safety systems, and single failures.

(5) Operator action must not be required to provide core cooling, containment integrity, and spent fuel pool cooling for 7 days following DBA's from at power conditions. For events from shutdown conditions and for PRA type events (multiple failures, etc.), operator actions within 7 days, required to maintain these functions, must be minimized.

(6) Operator action must not be required to provide post-accident control area habitability and to maintain post-accident monitoring for 3 days following DBA's from at power conditions. For events from shutdown conditions and for PRA type events (multiple failures, etc.), operator actions within 3 days, required to maintain these functions, must be minimized.

(7) For functions that are provided for 3 days without operator action, on-site equipment (e.g., self-powered air compressors or generators) shall be provided to enable these functions to be extended to at least 7 days.

(8) As a goal, core cooling and containment integrity should be provided for an indefinite time without offsite assistance.

(9) Offsite assistance may be utilized after 7 days provided:

- Safety grade connections are provided in the plant.
- The required offsite equipment is commonly available and only reasonable quantities are required.

(10) An automatic RCS depressurization feature (ADS) shall meet the following criteria:

- The reliability (redundancy and diversity) of the ADS valves and controls must be sufficiently high to satisfy single failure requirements as well as provide the failure tolerance required to achieve the plant core damage frequency goal.

- The probability of use of the ADS should be low and its impact on plant restart and operation should be minimized. Consideration should be given to both real demands (such as RCS leaks and failure of the CVCS makeup pumps) as well as spurious instrumentation signals. The probability that the use of ADS results in a significant flooding of the containment should be less than once in 600 plant years.

- Spurious use of the ADS should not prevent the plant from restarting in 14 days assuming the availability of normal plant systems to minimize flooding and to assist recovery. Operation of the ADS should not result in anticipated fuel or steam generator tube damage, and it should not be necessary to inspect the fuel, the vessel, the reactor internals, or the steam generator tubes. The operation of installed non-safety equipment should limit the flood up of the containment and the increase in the containment pressure and temperature.

- The RCS and the auxiliary equipment should be designed for several ADS operations in the life of the plant.

(11) For all postulated pipe breaks resulting in the worst LOCA accident ($\leq$ 4" in diameter), the core must remain covered. As a goal, for all postulated pipe breaks resulting in the worst LOCA accident ($\leq$ 4" in diameter), the core will remain covered with no credit for the addition of water from the safety grade or other makeup sources to the reactor vessel.

(12) Actuation of the passive safety systems following anticipated transients (including Automatic Depressurization System use as a PORV) should not result in steaming to containment.

(13) The passive safety systems must be capable of cooling the RCS from normal temperatures down to safe Shutdown Conditions ($\leq$ 400°F/204°C) following shutdown from full power.

(14) The passive safety systems must be capable of borating the RCS from the normal operating boron concentration to the boron concentration required to maintain $k_{eff} \leq 0.99$ at 200°F (93°C) with the rods inserted, and the worst stuck rod assumed.

(15) Containment Heat Removal

- Containment pressure shall be reduced to less than 1/2 of the peak pressure in one day following the worst design basis LOCA, to limit design basis off-site doses.

- With no operator action after seven days, the containment pressure should not exceed its expected failure pressure.

(16) Containment spray additive shall not be required. A passive means shall be provided to adjust the containment sump pH to greater than 7.0 within 8 hours following a design basis event that floods the containment.

(17) Boric acid solution used to provide the means of diverse shutdown (GDC 26/27) shall be enriched boron with $\geq$ 80 percent B^{10}, in order to minimize the volume of solution needed to ensure core nuclear shutdown.

1.5.2 Non-Safety Systems

(1) The non-safety systems should be greatly simplified; the number of systems and components and the complexity of operation and maintenance shall be reduced from current conventional operating plants, and as a goal, from current passive plant designs too.

(2) Licensing design basis accident analysis will be performed using only safety-grade equipment; however, in order to increase investment protection and accident protection, the non-safety systems shall be designed to provide a highly reliable system back-up.

(3) Important non-safety systems that are required for normal plant operation must be sufficiently reliable and capable of providing high plant availability and minimizing economic risk. As a result, these systems must have appropriate redundancy, be powered by an on-site power supply, and have sufficient capacity to prevent automatic passive safety system actuation following anticipated transients (events more frequent than once every ten years).

- The RCS makeup capability must be at least 7.6 10^{-3} m^3/s (120 gpm) which is sufficient for RCS leaks up to 9.52 mm (3/8 inch) in diameter.

- Steam generator feedwater capability (separate from the main feedwater system) must provide sufficient flow for normal plant start-up and be able to remove core decay heat following a reactor trip due to the loss of main feedwater, without actuating the passive safety systems.

- Containment fan coolers used for normal containment cooling must be capable of operating in containment pressures and temperatures that are somewhat elevated above normal operating conditions.

- As a goal, the normal containment sump pumps should be designed such that they can be used to assist in recovery should the containment be flooded.

(4) As a goal, for normal operation, startups and shutdowns and anticipated transients, the steam generator should be fed with heated condensate quality (including O_2 concentration) water. Also, the steam generators should not dryout following an anticipated transient with a single failure and realistic assumptions.

(5) Boric acid solutions must be stored at concentrations that do not require heat tracing or

room temperatures above normal values; e.g., 4 percent ± 0.2 percent requires that the ambient temperature be ≥60°F to keep the boric acid in solution. Consider the use of 2.5 percent boric acid (4400 ppm B) such that special ambient temperature requirements other than freeze protection are eliminated.

(6) The boric acid used in the plant shall be enriched boron, with a B^{10} concentration of ≥80 percent.

(7) As a goal, the use of complex, troublesome, hard to maintain and high exposure equipment shall be eliminated. In particular, eliminated equipment will include the boron recycle and waste evaporators, waste gas compressors, waste gas hydrogen recombiners and fast start (10 second) diesel generators.

(8) As a goal, the HVAC systems and equipment shall be simplified from currently operating PWRs, considering reductions in large volume ducting, large filters and fans. Where possible, the consequences of the limiting accidents, such as fuel drop should be eliminated or reduced. The elimination of all charcoal filters should be considered because the new source terms indicate that almost all released iodine becomes a particulate.

1.6 ELECTRICAL AND CONTROL SYSTEMS

1.6.1 Control and Protection Systems

(1) During normal operation, a single failure in the I&C system shall not result in a reactor trip or ESF actuation even with a single channel out for maintenance or test.

(2) The potential for reactor trip and for safeguards actuation due to failures in the reactor control or protection systems shall be reduced. Special attention shall be given to limiting actuations that could lead to extended plant outage such as inadvertent ADS actuation.

(3) The number of variables used for reactor trip and for safeguards actuation shall be minimized consistent with the safety goals for current operating plants.

(4) The margin between the normal operating conditions and the protection system setpoints shall be maximized in order to reduce the likelihood of ESF actuations.

(5) The potential interaction between control and protection systems shall be reduced relative to current operating plants.

(6) A distributed logic system utilizing multiplexing techniques shall be used to significantly reduce the amount of wiring required in the plant. As a minimum, the Remote Shutdown control feature, the Engineered Safeguards System, and the Integrated Logic Control Systems shall utilize the multiplexing techniques.

(7) An AMSAC (ATWS Mitigation System) system shall be provided, which is diverse (including hardware) from the reactor protection system, and provides a limited number of actuations and/or trips for functions to ensure adequate ATWS protection. Additional actuations that provide significant PRA and licensing benefits shall be considered.

1.6.2 AC and DC Power

(1) There shall not be a requirement for safety grade emergency AC power in the plant. The use of passive type safety systems which do not require AC power and non-safety blackout diesels to power the non-safety systems in the case of loss of off-site power should be sufficient to meet the safety goals as well as provide for operability and utility financial risk reduction. Consideration should be given to use of a gas turbine powered generator as a diverse non-safety blackout power supply.

(2) Emergency DC power must be provided to support very reliable reactor trip and engineered safeguards actuation. Batteries shall be sized for providing the necessary DC power for the protection system actuation, the emergency control room functions including habitability, a few DC powered valves in the passive safety systems, containment isolation, etc. The batteries should provide power for required emergency functions for three days.

1.6.3 Control Room

(1) A main control room (MCR) shall be provided that is able to control the plant during any event, including design basis accidents. Hence, the MCR includes indications and controls capable of monitoring and controlling the plant safety systems as well as the normal control systems.

(2) A remote shutdown room (RSR) shall be provided. The RSR shall contain the safety grade instrumentation and controls that allow an operator to achieve and maintain safe shutdown of the plant following an event when the MCR is unavailable.

(3) The MCR and the RSR shall be serviced by reliable and redundant non-safety power supply and HVAC systems during normal operation. These normal systems shall be reliable enough to limit the use of passive HVAC features to once in 500 years.

(4) In the unlikely event that the normal power supply or HVAC system become unavailable, there will be passive systems (batteries, compressed air) to support one of these control rooms. It is preferred that the MCR be supported by passive safety systems; however, if that is impractical because of its size, number of doors, etc., then it is acceptable to support the RSR. The passive systems shall include sufficient power for emergency lighting in areas of the plant that may require access during shutdown operations with safety-grade equipment.

(5) If the RSR is the control area supported by the passive safety grade power supplies and HVAC, then it shall contain the safety grade instrumentation and controls to allow the operator to achieve and maintain safe shutdown following any design basis event. This I&C is not expected to be significantly different from that required for MCR evacuation.

(6) The passive safety power supplies and HVAC system shall be designed to provide a habitable environment for the operating staff assuming that no AC power is available. Installed equipment shall provide for at least three days of operation. Afterwards it shall be possible to continue operation to 7 days with operator action and on-site equipment.

(7) The RSR shall contain the minimum number of indications and controls consistent with its intended use; that is, the RSR is to be used only in the unlikely event that the MCR is not available.

(8) The RSR capability shall include the mitigation (block) of spurious signals due to fire in the MCR or other parts of the plant. The MCR shall also be capable of mitigating spurious signals from a fire in the RSR.

(9) The safety grade indications and controls in the RSR shall be electrically isolated from those in the MCR.

(10) A mechanism shall be provided to allow the operating staff to transfer controls or instrumentation from the MCR to the RSR.

(11) The transfer of the control of components to the RSR shall be alarmed in the MCR, clearly audible to the operating staff.

(12) Access to the RSR and the transfer mechanism shall be under strict administrative control.

(13) Both the MCR and the RSR shall be designed in accordance with human engineering principles and practices.

(14) Human factors considerations shall be utilized to ensure that the indications and controls in the RSR are similar to those provided in the MCR.

(15) The safety grade instrumentation (equipment racks) shall be maintained at acceptable ambient conditions for three days using only passive systems, following a loss of all AC power. However, it shall be possible to continue operation to 7 days with operator action and on-site equipment.

(16) A technical support center (TSC) shall be provided. The TSC need not be designed to safety or seismic standards.

(17) Because IRIS is likely to be deployed as a multiple unit power station, consideration is being given to using a common control area for all the IRIS reactors on a site. This control concept is being developed as part of the design for other small reactor concepts. The IRIS control concept could include this modification should an acceptable multi-unit concept be demonstrated.

1.7 PLANT ARRANGEMENT AND CONSTRUCTION

1.7.1 Plant Arrangement

(1) The overall plant arrangement shall utilize optimum building configurations and structural designs to minimize the building volumes and bulk quantities (concrete, structural, steel, rebar, etc.) consistent with safety, operational, maintenance, and structural needs.

(2) The total plant schedule shall not exceed 60 months from the project commitment to initial commercial operation. The total plant construction period for the IRIS shall not exceed 42 months. The construction period is defined as the period of time from the first structural concrete to the completion of the plant warranty demonstration run. Included in this time is 6 months for startup testing.

The total schedule consists of the following key events:

Project commitment/site preparation	18 months
Construction (first structural concrete to fuel loading)	36 months
Start up test/Demonstration Run	6 months
Total	60 months

The integral reactor configuration, because of the small size of the containment structure, and the construction of multi-unit sites shall reduce, as a goal, the construction time (first structural concrete to fuel load) for IRIS from 36 months to 30 months or less.

(3) The plant arrangement shall be compatible with advanced construction techniques including fabrication of on-site and off-site (factory assembled) modules of building structures, components, and plant systems.

(4) As a goal, the standard design of the plant arrangement, which is based on the envelope of site parameters given in Table 1.2-3, should be applicable to the majority of potential sites available in the continental U.S. In addition, the plant arrangement shall be adaptable to modifications that may be required by specific site characteristics that are outside the bounds of the design parameters, including sites in foreign countries.

(5) The plant arrangement shall provide separation between safety and non-safety systems to preclude adverse interaction between safety grade and non-safety grade equipment. Separation between redundant safety grade equipment and systems shall be provided where required to assure that the safety functions can be performed. In general, this separation will be provided by substantial barriers (concrete walls) or by separated building areas, and will provide protection from internal fire and flooding events.

(6) The plant arrangement shall provide separation for radioactive (dirty) and non-radioactive (clean) equipment and shall provide separate pathways to these clean and dirty areas for personnel access.

(7) All pathways through the plant shall be designed to accommodate equipment maintenance and equipment removal. The size of the pathways shall be dictated by the largest piece of equipment that may have to be removed or installed after initial installation of the associated module. If required, laydown space shall be provided for disassembling large pieces of equipment to accommodate the removal or installation process.

(8) Adequate space shall be provided in the layout for equipment maintenance, laydown, removal and inspection. Hatches, monorails, hoists, and removable shield walls shall be provided where required to facilitate maintenance.

(9) The containment shall be of a spherical shape with a top mounted closure head through which access to the reactor vessel head, integral reactor vessel equipment including the reactor coolant pumps and steam generators, and reactor vessel internals is provided. All refueling activities are to be conducted through this top mounted closure head.

(10) The containment closure flange shall be provided with a permanently installed seal ring that attaches to the refueling cavity in the fuel handling area located directly above the containment closure head. This seal ring, combined with the reactor vessel to containment flange seal ring (see below) will allow the refueling cavity to be flooded above the reactor vessel.

(11) The containment lower closure flange and the reactor vessel lower closure flange shall have provisions for installing a seal ring for refueling operations. This seal, together with the containment flange to refueling cavity seal (see above), will allow the refueling cavity to be flooded above the reactor vessel.

(12) The fuel handling area will contain the refueling machine, the refueling cavity, the spent fuel pit, the cask loading and wash-down pits, and associated gates and equipment; thus, all refueling operations are performed from/within this area.

(13) The fuel handling area shall have adequate space for new fuel storage, reactor vessel head storage, containment closure head storage, spent fuel cask storage, new fuel container storage, and an overhead crane for lifting all anticipated loads. The fuel handling area overhead crane shall have direct access to/from a rail car.

(14) The plant arrangement shall be comprised of four principal building structures; a Nuclear Island, a Turbine Island, a mechanical/electrical Annex Building, and an Access Control Building.

(15) The Nuclear Island shall be structurally designed and constructed to meet Seismic Category I requirements, as defined in RG 1.29. The Nuclear Island shall consist of a free-standing steel containment structure, a concrete shield building, a main steam and feedwater penetration area, mechanical/electrical penetration areas, a fuel handling area, all safety related mechanical/electrical components and systems. The preferred

foundation for the Nuclear Island is an integral basemat, which supports the above structures. This Nuclear Island therefore contains the traditional containment building, the auxiliary building, the fuel building, and the refueling water storage tank in an integrated structure

(16) The Turbine Island shall be structurally designed and constructed to the Uniform Building Code. The Turbine Island shall be supported on a single basemat foundation.

(17) The radwaste systems shall be located in a building separate from the Seismic I Nuclear Island. The building shall be designed and constructed to the seismic requirements given in RG 1.143. Specifically the foundation and structural walls of this building, which are located below grade elevation, shall be designed to the applicable seismic criteria of RG 1.143 to a height sufficient to contain the maximum liquid inventory expected to be in this building.

(18) Access control features such as plant security, the health physics area, the technical support center, and the showers and locker rooms shall be located in a non-seismic building.

(19) The Nuclear Island shall be designed to withstand the effects of natural phenomena such as hurricanes, floods, tornadoes, tsunamis, and earthquakes without loss of capability to perform the safety functions. Design for natural phenomena shall be based on the industry standards and applicable regulatory codes.

(20) The plant arrangement/structures shall be designed to minimize the potential for man-made hazards external to the plant from affecting the plant safety related functions. This shall include consideration of toxic or hazardous gas releases, gas explosions, fires, sabotage, airplane crash, surface vehicle accidents, etc.

(21) The plant arrangement/structures shall be designed to withstand the effects of any postulated internal event such as fires and flooding without loss of capability to perform safety functions.

(22) Radioactive equipment and piping shall be arranged and shielded to minimize to radiation exposure to help meet the man-rem/yr goal.

1.7.2 Construction

(1) The plant shall apply advanced construction techniques and practices to the maximum extent practical. Both on-site and off-site prefabricated modules shall be considered for all building structures and plant systems.

(2) The standard off-site modules shall be sized for commercial rail shipment because this provides the most convenient form of transportation for many potential sites. However, the standard modules shall also be adaptable for pre-assembly into a much larger barge shippable modules for sites that are accessible by water. Note that the reactor vessel will require barge shipment to the plant location, with special handling required for its final placement.

(3) The standard off-site modules shall be pre-assembled, pre-tested, and pre-inspected in the fabricator's shop prior to shipment. Consideration shall be given to a construction plan that would assemble several rail shippable modules into a larger module after they have arrived at the site, but prior to being integrated into the plant, in order to accomplish additional pre-testing and pre-inspection and to allow parallel construction.

(3) The application of modularization shall be considered for all aspects of the plant design and construction. The applicability for modularization shall be based on the overall cost/schedule benefit.

(4) The use of site fabricated modules shall be considered for plant structures that are too large for rail shipment. Modularization shall be considered for basemat reinforcement, in-containment structural steel, in-containment water storage tanks, support structures for the reactor vessel, major sections of the containment structure, containment penetrations, pre-cast beams, condenser, etc.

(5) IRIS, like other small to medium power reactors, is intended for multiple-unit installations for applications with a large electrical grid. The construction of many identical, small units is being actively studied to determine the most cost-effective ways to combine functions between these multiple units. IRIS intends to incorporate the results of these on-going studies into its overall design and construction plan, while maintaining the high degree of overall plant safety and plant operability.

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2.0 DESIGN PARAMETERS LIST

This section of the report summarizes the pertinent parameters that define the major features and operating characteristics of the plant. In particular, parameters are provided for the Integral Reactor Coolant System, Reactor Pressure Vessel, the Reactor Coolant Pumps, the Pressurizer, the Steam Generators, the Turbine and the Core. For the latter component, the fuel rods, fuel assembly and control rod design are described in detail and coolant temperature and flow rates are presented.

Design Flows

The flow rates are listed for four sets of conditions, based on accepted definitions as discussed below:

- Best Estimate Flow (BEF)
- Minimum Measured Flow (MMF)
- Thermal Design Flow (TDF)
- Mechanical Design Flow (MDF)

Best Estimate Flow

The Best Estimate Flow is considered to be the most likely value for the normal full power operating condition. This flow is based on the best estimate of the reactor vessel, steam generator and piping flow resistances, and on the best estimate of the reactor coolant pump head-flow capability, with no uncertainties assigned to either the system flow resistance or the pump head. The Best Estimate Flow provides the basis for establishing the other design flows required for the system and component design. The Best Estimate Flow and head also define the performance requirement for the reactor coolant pump.

Minimum Measured Flow

The Minimum Measured Flow is specified in the Technical Specifications as the flow that must be confirmed or exceeded by the flow measurements obtained during plant startup, and is the flow used in reactor core DNB analyses for plants applying the Improved Thermal Design Procedure. When no excess flow margin is provided, Minimum Measured Flow also equals the lowest measured flow predicted by the Expected Flow Measurement Range.

Thermal Design Flow

The Thermal Design Flow is specified as the conservatively low design flow for thermal-hydraulic analyses where the design uncertainties are not combined statistically and, therefore, additional flow margin must be included to cover the flow measurement uncertainties. The Thermal Design Flow is defined by subtracting the plant flow measurement uncertainty from the Minimum Measured Flow.

Mechanical Design Flow

Mechanical Design Flow is the conservatively high flow used as the basis for the mechanical design of the reactor vessel internals, fuel assemblies and other system components. Mechanical Design Flow is established at 104 percent of Best Estimate Flow.

2.1 PLANT PARAMETERS

Parameter	
General	
Plant design life, years	60
NSSS power, MWt (BTU/hr)	1002 (3.421×10^9)
Reactor coolant pressure, operating, MPa (psia)	15.513 (2,250)
Reactor coolant pressure, design, MPa (psia)	17.235 (2,500)
Reactor coolant system liquid volume m^3 (ft^3)	~ 400 (~14128)
Average Core inlet temperature, °C (°F)	292 (557.6)
Average core outlet temperature, °C (°F)	330 (626)
Steam Pressure, MPa (psia)	5.8 (841)
Steam Temperature, °C (°F)	317(602.6)
Loop Piping	**N.A. for IRIS**
Reactor vessel and Internals Parameters	
Reactor vessel I.D., m (in.)	6.223 (244.96)
Reactor vessel O.D., m (in.)	6.783 (267.04)
Reactor vessel length, m (ft.)[a]	22.21 (72.86)
Cold leg nozzle I.D., m (in.)	N.A.
Hot leg nozzle I.D., m (in.)	N.A.
Downcomer annulus width, m (in.)	1.68 (66.1)
Core barrel I.D., m (in.)	2.75 (108.27)
Core barrel O.D., m (in.)	2.85 (112.2)
Reactor vessel design temperature, °C (°F)	343.33 (650)
Reactor vessel neutron fluence, n/cm^2	$\ll 10^{18}$

(a) From bottom of the lower head to the top of the upper head, without CRDM housing.

Parameter	
Reactor Coolant Pumps	
Type of RCP	Immersed Spool Pumps
Number of RCPs	8
Model	TBD
Variable Speed Controller	No
Motor Speed (rpm)	1800
Motor Input Power per pump, hp	[ ] (a,c)
Total power to coolant per pump, hp	[ ] (a,c)
Total power to coolant per pump, MWt	[ ] (a,c)
Total power to coolant (per 8 pumps), MWt	[ ] (a,c)
Pump efficiency	[ ] (a,c)
Motor efficiency	[ ] (a,c)
Overall Efficiency	[ ] (a,c)
RCP rated flow per pump, m^3/s (GPM)	[ ] (a,c)
RCP rated head, m (range)	[ ] (a,c)
RCP rated head, ft (range)	[ ] (a,c)
Estimated coastdown beta	[ ] (a,c)
Pump Weight, kg (lbm)	[ ] (a,c)
Pressurizer	
Number of Units	1
Total Volume, m^3 (ft^3)	71.41 (2,522)
Water Volume, m^3 (ft^3)	22.45 (793)
Spray capacity	N.A.
Inside Diameter	N.A.
Surge line volume	N.A.
Rated pressurizer heater capacity, kW	2,430

Parameter	
Steam Generator [1]	
Type	Inside vessel, Once through, Helical coil
SG Power, MWt/unit	125
Number of units	8
Number of tubes per unit	655
Surface area, m^2/unit (ft^2/unit)_primary side	1149.7 (12375)
Tube outer diameter, mm (inches)	17.46 (0.688)
Tube wall thickness, mm (inches)	2.11 (0.083)
Tube inner diameter, mm (inches)	13.24 (0.521)
Tube Average Length, m (ft)	32 (104.99)
Radial tube pitch[2], mm (inches)	[ ] (a,c)
Axial tube pitch[2], mm (inches)	[ ] (a,c)
Total bundle height[3], m (inches)	7.9 (311.0)
Overall height[3], m (inches)	8.5 (334.6)
Upper shell I.D./O.D.	N.A.
Lower shell I.D./O.D.	N.A.
Tube sheet thickness, mm (inches)	TBD
Shell design pressure	N.A.
Zero load temperature, °C (°F)	TBD
Feedwater temperature, °C (°F)	223.9 (435.02)
Bundle pressure drops[4], KPa (psia)	296 (42.93)
Primary Side pressure drops, KPa (psia)	72 (10.44)
Exit Steam Pressure, MPa (psia)	5.8 (841)
Exit Steam Temperature, °C (°F)	317(602.6)
Steam flow, kg/s (lb/hr) per S/G	62.85 (0.5×10^6)
Total steam flow, kg/s (lb/hr)	502.8(3.99×10^6)

(1) All data assume no tube plugging.

(2) Radial tube pitch is defined as the pitch between rows, while the axial tube pitch is defined as the pitch between tubes.

(3) Bundle Height is between headers centerline, while Overall Height is the total height of each SG module including headers.

(4) Only for the tube bundle: to avoid parallel channel instabilities an equivalent pressure drop is assumed at the tube inlet orificing (thus the overall SG secondary side pressure drops would be twice the value indicated).

Parameter	
SG Blowdown rates (Nominal)	N.A.
SG Blowdown rates (Maximum Intermittent)	N.A.
Primary water volume, m^3 (ft^3)	TBD
Secondary water volume, m^3 (ft^3)	TBD
Secondary steam volume, m^3 (ft^3)	TBD
Secondary water mass, kg (lbm)	TBD
Fouling factor, external $m^2 \circ C/W$ ($hr\text{-}\circ F\text{-}ft^2/BTU$)	TBD
Fouling factor, internal $m^2 \circ C/W$ ($hr\text{-}\circ F\text{-}ft^2/BTU$)	[ ] (a,c)
Steam outlet quality, %	Super-heated
Tube plugging allowance, %	5%

Parameter	
Turbine / Electric	
Turbine Configuration	1 – double flow HP/LP
No. of reheat stages	1
No. of feedwater heaters	7
Feedwater temperature, °C (°F)	223.9 (435.02)
Feedwater line nominal pipe size, mm (in.)	TBD
Main Steam line nominal pipe size, mm (in.)	TBD
Gross turbine electric output, MWe	365.17
Gross turbine heat rate, BTU/KWh	9,344
Auxiliary load-BOP, MWe	9.86
Auxiliary load-rest of Plant, MWe	15 est.
Steam generator outlet pressure, MPa (psia)	5.8 (841)
Maximum plant net electrical output, MWe	340.31
Turbine delivery pressure, MPa (psia)	TBD
Plant net heat rate, BTU/kW-hr	10,028
Plant net efficiency	34.3%
Turbine Inlet Temperature, °C (°F)	317 (602.6)
Moisture Separator Parameters	TBD
Reheater Pressure, psia	TBD
Outlet Temperature, °C (°F)	TBD
Condenser Vacuum, MPa (in. HgA)	0.005 (1.48)
Feedwater Heater Outlet Temperature, °C (°F)	
Heater 1	TBD
Heater 2	TBD
Heater 3	TBD
Heater 4	TBD
De-Aerating Heater (Heater 5)	TBD
Heater 6	TBD
Heater 7	TBD

2.2 THERMAL HYDRAULIC PARAMETERS

Parameter	
NSSS power, MWt	1,002
Reactor power, MWt	1,000
Best Estimate vessel flow, kg/s (lb/hr)	4698 ($37.25x10^6$)
Best Estimate core flow, kg/s (lb/hr)	4498 ($35.70x10^6$)
Reactor coolant pressure, MPa (psia)	15.513 (2,250)
Vessel/core inlet temperature, °C (°F)	292 (557.6)
Core average temperature, °C (°F) [(5)]	312.1 (593.72)
Vessel average temperature, °C (°F) [(5)]	310.2 (590.36)
Riser outlet temperature, °C (°F)	328.4 (623.1)
Average core outlet temperature, °C (°F)	330 (626)
Average velocity in core, m/s (ft/s)	2.45 (8.04)
Total core bypass flow, (pct. of total flow)	4.25 est.
Steam Generator leakage paths	TBD
Thimble flow	TBD
Reflector cooling flow	TBD
Core or R/V Pressure Loss, KPa (psi)	[ ] (a,c)
Piping Pressure Loss, KPa (psi)	N.A.
S.G. Pressure Loss, KPa (psi)	72 (10.44)
Total Pressure Loss, KPa (psi)	TBD
Minimum Measured Flow (M.M.F.) kg/s (lb/hr)	4563.1 ($36.22 x10^6$)
Thermal Design Flow (T.D.F.) kg/s (lb/hr)	4282.1 ($33.99 x10^6$)
Mechanical Design Flow (M.D.F.) kg/s (lb/hr)	4677.9 ($37.13 x10^6$)

[5] [

] (a,c)

Parameter	
Notes:	
M.M.F. = 0.970 x B.E.F	
T.D.F. = 0.952 x B.E.F	
M.D.F. = 1.04 x B.E.F	

2.3 CORE PARAMETERS

Parameter	
Core power, MWt	1000
No. of fuel assemblies	89
No. of fuel rods	23,496
Fuel assembly pitch, mm (in.)	[ ] (a,c)
Inter-assembly gap, mm (in.) (at Inconel top and bottom grids)	[ ] (a,c)
Active core length, m (in.)	4.2672 (168)
Overall[6] Fuel Assembly length, m (in.)	[ ] (a,c)
Core loading, MTU	48.5
Geometric fuel density, percent of theoretical	96.0
Dishing volume fraction, percent	[ ] (a,c)
Average linear power, kW/m (kW/ft)	9.974 (3.04)
Average power density, kW/liter	51.26
Average specific power, kW/kgHM	20.6
Fuel rod heat transfer area, m^2 (ft^2)	2,992.2 (32,208)
Average heat flux, W/m^2 ($BTU/hr\text{-}ft^2$)	334,201 (105,941)
Water volume fraction (in core)	61.98 %
UO_2 volume fraction (in core)	26.81 %
Zirconium Alloy volume fraction (in core)	9.83 %
Water-to-fuel volume ratio (in core)	2.311
Fraction of heat generated in the fuel	0.974
Design value of F_Q	2.6
Design value of $F_{\Delta H}$	1.65

[6] From top of lower support plate to bottom of upper support plate

2.4 FUEL ASSEMBLY PARAMETERS

Parameter	
Assembly	
Lattice	Square array, 17X17 RFA EnM XL
No. of fuel rods	264
No. of guide thimbles	24
No. of instrument tubes	1
No. of Non Mixing Vane Grids	2
No. of Mixing Vane Grids	8
No. of Intermediate Fluid Mixers (IFMs) Grids	4
Fuel assembly pitch, mm (in.)	[ ] (a,c)
Fuel assembly length, m (in.)	[ ] (a,c)
Fuel assembly weight, kg (lbs.)	TBD
Fuel rod pitch, mm (in.)	13.2842 (0.523)
Pitch over Diameter (P/D) ratio	1.3984
Fuel assembly loading, kg U (lb U)	545 (1202)
Fuel Rods	
Fuel rod outer diameter, mm (in.)	9.4996 (0.374)
Clad inner diameter, mm (in.)	8.3566 (0.329)
Pellet-clad diametral gap, mm (in.)	0.1651 (0.0065)
Pellet diameter, mm (in.)	8.1915 (0.3225)
Cladding thickness, mm (in.)	0.5715 (0.0225)
Guide Thimbles	
Guide thimble outer diameter, mm (in.)	[ ] (a,c)
Guide thimble inner diameter, mm (in.)	[ ] (a,c)
Guide thimble thickness, mm (in.)	[ ] (a,c)

Parameter	
Instrument tube outer diameter, mm (in.)	[ ] (a,c)
Instrument tube inner diameter, mm (in.)	[ ] (a,c)
Instrument tube thickness, mm (in.) (Top entry instrumentation)	[ ] (a,c)

2.5 CORE COMPONENT PARAMETERS

Parameter	
Total Number of Control and Gray Rod Cluster Assemblies	37 est.
Rod Cluster Control Assembly (RCCA)	
No. of RCCA's	25-29 est.
No. of rodlets per RCCA	24
Control rod clad material	SS-304
Control rod outer diameter, mm (in.)	[ ] (a,c)
Control rod clad inner diameter, mm (in.)	[ ] (a,c)
Cladding thickness, mm (in.)	[ ] (a,c)
Absorber material	Ag-In-Cd or B4C
Pellet diameter, mm (in.)	[ ] (a,c)
Gray Rod Cluster Assembly (GRCA)	
No. of GRCA's	8-12 est.
No. of rodlets per GRCA (total)	24
No. of rodlets with SS-304 pellets	20
No. of rodlets with Ag-In-Cd pellets	4
Gray rod clad material	SS-304
Gray rod outer diameter, mm (in.)	[ ] (a,c)
Gray rod clad inner diameter, mm (in.)	[ ] (a,c)
Cladding thickness, mm (in.)	[ ] (a,c)
Gray rod pellet diameter, mm (in.)	[ ] (a,c)
Reflector	
Type	Stacked rings
Number of Rings	10
Reflector ring material	SS-304

Parameter	
No. of Reflector cooling holes and diameter, mm (in.)	[ ] (a,c)
Total weight of reflector assemblies, kg (lb)	~50,000 (~110,000)
Miscellaneous	
Unrodded guide thimble locations	1248 est.

2.6 BUILDING DESIGN PARAMETERS

Parameter	
Nuclear Island, Containment Structure	
Type	Spherical with top mounted closure flange and closure head
Design Pressure, MPa (psi)	1.2 (174)
Design Code	ASME Section III, Div. 1
Seismic Design Category	Seismic I
Containment Atmosphere	N_2
Dimensions	
Containment Internal Diameter, m (ft)	25 (82.02)
Containment Shell Thickness, mm (in.)	44.45 (1.75)
Containment Material	Carbon Steel
Containment Volume, m^3 (ft^3)	8181.23[7] (288,917)
Containment Free Air Volume, m^3 (ft^3)	4700[8] (166,000)
Volume occupied by Reactor Vessel, m^3 (ft^3)	711 (25,109)
Internal Structures/Equipment	
Vessel Cavity Height, m (ft)	10 (32.81)
Suppression Pool overall Volume, m^3 (ft^3)	1200 (42377.6)
Suppression Pool Water Fraction	0.25
Nuclear Island, Containment Shield	
Description	Concrete Cylinder
Diameter ID, m (ft)	27 (88.58)
Wall Thickness, m (ft)	1.5 (4.92)
Seismic design category	Seismic I

[7] Estimated as the volume of a sphere of 25 m in diameter.

[8] Containment Free Volume estimated as 75% of the Volume obtained by subtracting the Vessel Volume and Pool volume from the Containment Volume.

Parameter	
Nuclear Island, Fuel Handling Area	
Seismic design category	Seismic I
Crane capacity (bridge), tons	200
Crane capacity (cask handling), tons	Same as above
Spent fuel pit volume, gross, m^3 (ft^3)	250 (8830)
Nuclear Island, Auxiliary Bldg. Area	
Seismic design category	Seismic I
RWST volume, m^3 (gallons)	1344 (355,000)
No. Of Divisions	2
Radwaste Building	
Seismic design category	Seismic II/Non-seismic
Turbine Building	
Seismic design category	Seismic II
Crane capacity (main), tons	100
Annex Building(s)	
Seismic design category	Seismic II
Diesel Generator Building	
Seismic design category	Non-safety, uniform building code

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3.0 INTEGRAL REACTOR COOLANT SYSTEM

The IRIS Reactor and Reactor Coolant System (RCS) are designed to generate heat in the reactor core and to remove or to enable removal of this heat during all modes of plant operation, including shutdown and accident conditions. In conventional pressurized water reactors these functions are performed by components that are connected by large piping circuits; these components include the reactor vessel, steam generators, pressurizer, and pumps. The corresponding IRIS RCS components are all contained within a single pressure vessel with no external connecting piping circuits. This single integral vessel therefore contains the IRIS Integral Reactor Coolant System.

3.1 SYSTEM CONFIGURATION

IRIS employs an integral RCS configuration, in which the reactor vessel (RV) houses not only the nuclear fuel, internals, and control rods, but also all the major RCS components (see Figure 3.1-1a). This includes: eight (8) small, spool type, reactor coolant pumps (RCPs); eight (8) modular, helical coil, once through steam generators (SGs); a steel reflector which surrounds the core to improve neutron economy and reduce neutron fluence on the core barrel and RV; and a pressurizer located in the RV upper head. This integral RV arrangement eliminates the individual component pressure vessels and large connecting loop piping between them, resulting in a more compact configuration and in the elimination of the large loss-of-coolant accident as a design basis event. Because the IRIS integral vessel contains all the RCS components, it is larger than a traditional RV, and has an ID of 6.21 meters (20'-4") and an overall height of 22.214 meters (72'-10").

The primary coolant main flow path is illustrated in Figure 3.1-1(b). Water flows upwards through the core and through the riser region (defined by the extended core barrel). At the top of the riser, the coolant is directed into the upper part of the annular plenum between the extended core barrel and the RV inside wall, where the suction of the reactor coolant pumps is located. Eight coolant pumps are employed, and the flow from each pump is directed downward through its associated helical coil steam generator module. The flow path continues down through the annular downcomer region outside the core to the lower plenum and then back to the core completing the primary coolant flow path.

The IRIS Integral reactor coolant system consist of the following components:

Reactor Core and Associated Components (Section 3.2) - This consists of the reactor core and its supporting structure inside the reactor vessel. The associated control rods, rod drivelines, and core instrumentation are connected via penetrations through the top of the reactor vessel head.

Integral Reactor Vessel and Internals (Section 3.3) – The IRIS vessel contains the reactor core and its associated components, and the remaining RCS main components which are designed to remove and transfer heat from the core during all plant operating modes. In the IRIS configuration the vessel constitutes the RCS pressure boundary, and the vessel internals define the RCS coolant path.

Reactor Coolant Pumps (Section 3.4) - A total of eight, spool-type, canned motor pumps pump hot, pressurized fluid from the core through the steam generators, where the water is cooled and returned to the core. Each of the eight pumps is coupled with one of the eight steam generators.

Steam Generators (Section 3.5) - The steam generators provide the heat exchange surface through which heat is transferred from the reactor coolant that exits the core during normal operation, resulting in the heatup and vaporization of water supplied to the secondary side of the steam generator heat transfer surface. During normal operation, the steam produced is used to power the turbines, which turn the electric generator. The steam generator provides the boundary between the reactor coolant system and the steam and feed water supply systems.

Pressurizer (Section 3.6) - The pressurizer is located within the reactor vessel upper head and contains saturated water and steam. The saturated steam volume maintains the reactor coolant system at a constant pressure, and provides a surge volume for minimizing the effects of plant transients.

The Reactor Coolant System is also served by a number of auxiliary systems, including the Chemical and Volume Control System (CVCS), the Normal Residual Heat Removal System (NRHRS), the Steam Generator System (SGS), the Primary Sampling System (PSS), the Liquid Radwaste System (WLS), and the Component Cooling Water System (CCWS).

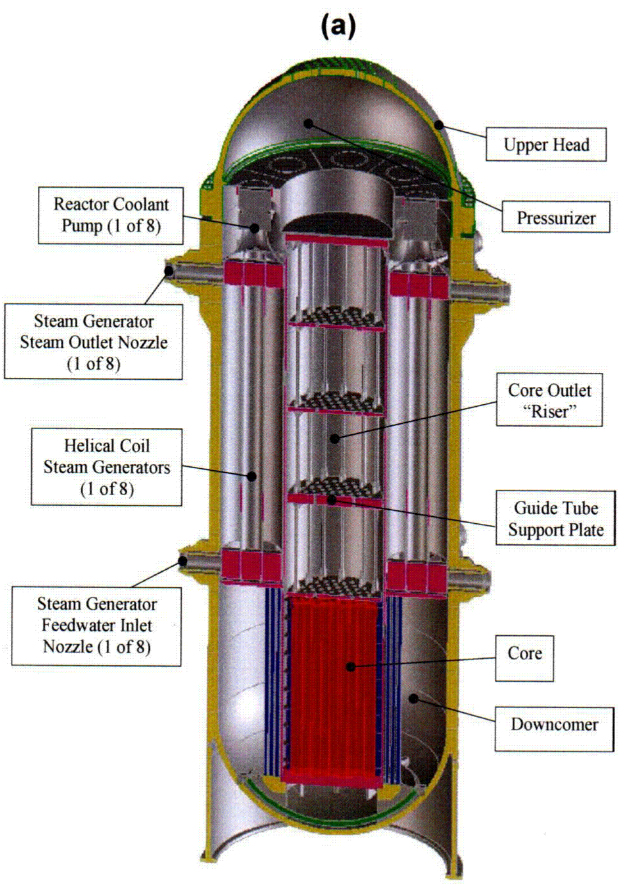

Figure 3.1-1: IRIS integral layout: (a) main components; (b) main flow path

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3.2 CORE DESIGN

The IRIS core and fuel characteristics are similar to those of a conventional Westinghouse PWR design. However, several features have been modified to enhance performance as compared to conventional plants, while retaining existing technology. A low power density core reduces the average linear power by about 25 percent as compared to AP600. The improved thermal margin provides increased operational flexibility, while enabling longer fuel cycles and increased overall plant capacity factors. The IRIS fuel employs a lattice design with optimized (i.e., slightly enhanced) moderation that results in increased discharge burnup or reduced enrichment requirements. Another feature that contributes to lowering fuel cycle cost and extending reactor life is the use of a stainless steel radial neutron reflector. This reflector reduces neutron leakage thereby improving core neutron utilization. An additional fuel design feature, longer fission gas plenum, will enable achieving higher fuel burnups in the future.

Components directly related to the core neutronic performance are described in this section, including the core and fuel, RCCA (Rod Cluster Control Assembly) and Control Rod Drive Mechanism (CRDM), incore instrumentation, and the radial reflector.

3.2.1 Core and Fuel Design

An IRIS fuel assembly consists of 264 fuel rods in a 17x17 square array. The fuel rods consist of enriched uranium, in the form of cylindrical pellets of uranium dioxide, contained in ZIRLO™ tubing. The fuel rod O.D. is 9.5 mm (0.374 in). The central position is reserved for incore instrumentation, while the remaining 24 positions have guide thimbles (Fig. 3.2-1). The IRIS fuel assembly design is similar to the Westinghouse 17x17 XL Robust Fuel Assembly design and AP1000 fuel assembly design. However, IRIS fuel employs a lattice with somewhat enhanced moderation (similar to that of the Westinghouse OFA fuel) that results in increased discharge burnup or reduced enrichment requirements. The fission gas plenum volume is increased as compared to the current PWR fuel to allow achieving higher fuel burnup in the future. Due to the integral vessel layout, where the vessel height is mostly determined by the steam generators, this plenum increase is possible with no corresponding increase in vessel height.

Low power density is achieved by employing a core configuration consisting of 89 fuel assemblies with a 14-foot (4.267 mm) active fuel height, and a nominal thermal power of 1,000 MWt. This results in an average linear power heat generation rate of 9.84 kW/m (3 kW/ft), i.e., about 25 percent lower than in AP600. The improved thermal margin provides increased operational flexibility, while enabling longer fuel cycles and increased overall plant capacity factors.

The core fuel loading of IRIS consists of 48.5 metric tons of uranium. Several reloading strategies are available depending on the utility/customer requirements and priorities. When the cycle length is the primary objective, straight-burn core design utilizing enrichment close to 5% can provide a four-year cycle lifetime with a burnup of ~40,000 MWd/tU. A more traditional multi-batch reloading enables achieving average batch discharge burnup of ~55,000 MWd/tU (for a two-batch reload scheme), or up to ~65,000 MWd/tU (for a three-batch reload scheme). The two-batch reload is compatible with the currently licensed maximum allowed burnup (62,000 MWd/tU lead rod average) and is therefore the current reference core design. The three-batch reloading scheme may be implemented in the future to improve fuel economy, when fuel with higher allowed discharge burnup (e.g., 62/75,000 MWd/tU batch and lead rod average) becomes licensed, or, it may be currently used to reduce enrichment requirements.

Reactivity control is achieved in the traditional PWR manner by a combined use of soluble boron, burnable absorbers, and control rods. However, soluble boron concentration is reduced, as compared to conventional PWR cycles, to improve core response in transients (more negative reactivity coefficients) and to reduce the amount of water to be processed. Burnable absorbers of the same type as those used in existing Westinghouse fuel are foreseen for use in IRIS. These include ^{10}B as IFBA (thin zirconium diboride fuel pellet coating) and erbia as integral fuel absorber (intermixed with the fuel), or some combination of the two. It should be noted that the increased plenum volume will accommodate a larger He release, and thus enables using higher ^{10}B loading than is the current practice, if deemed desirable for the reactivity/peaking control purposes.

A complete set of pertinent core parameters is given in Sections 2.3, 2.4 and 2.5.

3.2.2 Control Rods and Control Rod Drive Mechanisms

The control rod drive mechanism (CRDM) is a hermetically sealed electro-mechanical device used to move the rod control cluster assemblies (RCCA) and gray rod cluster assemblies (GRCA) in and out of the core. The CRDMs are coupled to these two types of assemblies by a drive rod, which extends through the upper vessel head. The assemblies are inserted and withdrawn in discrete steps or may be held stationary at any position. If power to the CRDMs fails or is removed, the CRDMs release the drive rods and the assemblies fall into the core by gravity. The design of the IRIS CRDMs is similar to the Westinghouse standard design used in present plants.

The entire assembly consists of the control rod drive mechanism and its pressure housing, the control rod drive rod, which is moved up and down by the mechanism, and the RCCA or GRCA which is attached to the drive rod and moves up and down within the fuel assembly. This RCC/GRC assembly holds the absorber rods, which are used to control the nuclear reaction within the fuel assembly. The RCCA/GRCA is guided by a guide tube which is part of the upper internals. The drive rod assembly consists of a hollow tube with external latching grooves, which are engaged by latches for holding and moving the rod. A coupling attached to the lower end of the drive rod provides the means for connecting or disconnecting the RCCA and GRCA clusters. The disconnection is accomplished by pulling a central rod upwards, pulling a button from an inside diameter taper and permitting the two halves of the coupling to collapse. The coupling can then be withdrawn from the RCCA hub.

The CRDM is an air cooled vertically oriented magnetic jack mechanism. Each mechanism is constructed so the drive rod is completely enclosed within a pressure housing that forms part of the reactor coolant pressure boundary. The latch assembly moves the drive rod using two latches; this is accomplished by energizing the coil stacks in sequence. The coil stack assembly is a separate assembly that is installed outside of the pressure housing and provides the magnetic force necessary to operate the latch assembly. The pressure housing encloses the drive rod and latch assemblies.

The function of the latch assembly is to provide the motion for drive rod stepping and holding, and to release the rod during scram. The latch assembly utilizes the grooves in the drive rod to lift and hold the RCCA in position. The latch assembly is composed of three operating sections, each having a separate function. Each section is made up of two flat faced poles which provide axial motion in response to magnetic flux. Two sections, the moveable gripper and the stationary gripper, have a set of linkages which translates the axial pole motion into radial motion, causing latches to engage the grooves in the drive rod. The lift section is attached to the moveable gripper section, and lifts the moveable gripper section a distance equal to the step length. By operating the three sections in the proper sequence, the drive rod can be stepped out, stepped in, or held stationary.

The function of the coil stack assembly is to provide the magnetic flux, which actuates the plunger magnets of the latch assembly. Each section of the latch assembly (lift, moveable gripper and stationary gripper) has a corresponding electrical coil in the coil stack. When energized, the coil produces a level of magnetic flux, which passes through the non-magnetic pressure housing and closes the corresponding plunger section.

The CRDM is designed so that all surfaces in contact with the reactor coolant are manufactured from corrosion resistant materials. The pressure housing of the CRDM is constructed of 304 LN stainless steel to ASME Section III Class 1 guidelines. Drive rods are manufactured from 410 stainless steel. Latches, latch arms, linkages, etc. are manufactured from corrosion resistant materials either 304 or 410 stainless steel or Inconel depending on the function of the part. Standard manufacturing techniques would be employed. High wear locations within the latch assembly use Stellite hard facing or are made from Haynes-25 alloy.

The rod control cluster assemblies consist of 24 absorber rods fastened at the top end to a common hub, or spider assembly. Each absorber rod consists of an alloy of silver-indium-cadmium, which is clad in stainless steel. The rod cluster control assemblies are used to control relatively rapid changes in reactivity and to control the axial power distribution.

The gray rod cluster assemblies consist of 24 rodlets fastened at the top end to a common hub, or spider assembly. Geometrically, a gray rod cluster assembly is the same as a rod cluster control assembly except that 20 out of 24 rodlets are fabricated of stainless steel, while the remaining 4 are silver-indium-cadmium clad in stainless steel. This core design feature is common with the AP600 and AP1000 design, and the primary use of these reduced-worth ("gray") rods is to achieve daily load follow while minimizing the required change in the soluble boron concentration of the reactor coolant.

The upper head has 45, four inch outer diameter penetrations for the CRDM housings. Preliminary analysis assumes that the core will utilize a total of 37 RCCAs and GRCAs (thus leaving 8 spare penetrations). One possible control rod pattern is shown in Fig. 3.2-2; while the core positions occupied by the control rods are final, the bank assignment will be defined as the core physics and safety analyses are being refined.

3.2.3 Incore Instrumentation

The Incore Instrumentation System (IIS) provides the capability of measuring the core operating parameters. The IIS consists of incore instrument thimble assemblies which house fixed incore detectors and core exit thermocouples within an inner and outer sheath assembly. The thimble assemblies are designed to be inserted directly through the RV head and upper internals package into the active fuel assembly prior to plant operation. The thimbles are extracted from the RV prior to the removal of the upper head for refueling. Each incore instrument thimble assembly is composed of multiple fixed incore detectors and one thermocouple. The neutron detectors provide signals that are used to generate an on-line three dimensional measurement of the core power distribution. The thimble assembly is positioned in the fuel assembly during plant operation and exits through the top of the reactor vessel to the containment where the fixed detector and thermocouple cables are routed to different data conditioning and processing stations. The data are processed and the results are displayed in the main plant control room and other areas.

The thimble assembly is inserted through the top of the integrated head package (IHP) seismic support plate through the reactor internals into the fuel. The upper head penetrations for the incore instrumentation guide tubes are 1.5 inch in diameter. The thimble assembly is supported and guided through the internals into the fuel in a manner which is designed to prevent flow induced vibrations and misalignment. In the IHP a guide tube projects from the reactor head and forms the pressure boundary around the thimble. This tube extends through the IHP seismic support plate to the thimble pressure seal. The guide tube penetrates the reactor head and a guide extension protects the thimble assembly to the upper internals support column. The support column guides the thimble assembly down to the top of the fuel. At the bottom of the fuel assembly instrument tube there is a leakage path to allow coolant flow around the thimble and prevent boiling on the thimble.

A sufficient number of incore instrument thimbles is located across the reactor core in a pattern, which provides a radial distribution that adequately monitors each fuel assembly. This pattern is designed to allow up to 50 percent loss of instrument signals before the loss of accuracy requires reduction of the reactor power. It is estimated that incore instrumentation will be located at 26 to 30 core positions to provide the required functionality.

The signals from the incore detectors are processed by power distribution algorithms in the BEACON System to produce a detailed 3-D power distribution at time intervals of at least once per minute. The BEACON System provides the reactor operator with detailed knowledge of the current reactor operating conditions and the available margin, including a display of the current axial power distribution relative to the applicable operating limits, as well as a display which allows the operator to anticipate changes in the axial power distribution. The BEACON System also provides displays capable of assisting the operator to quickly identify and correct the cause of radial and axial power perturbations.

In addition to the described, already proven incore detector technology, Westinghouse is developing a novel concept based on advanced SiC detectors, for excore and/or incore neutron/power monitoring. If the pace of its development, performance validation and qualification permits, it will be considered for use in IRIS.

3.2.4 Radial Reflector

Another feature that contributes to lowering the fuel cycle cost and extending reactor life is the use of a stainless steel radial neutron reflector (Fig. 3.2-3). This reflector reduces radial neutron leakage thereby improving neutron economy. As a result, fuel utilization is improved, fuel cycle may be extended, and the average discharge burnup may be increased. The radial reflector has the added benefit of reducing the fast neutron fluence on the core barrel and reactor vessel, as well as reducing the dose and material activation outside the vessel

The reflector is made from rings, which surround the core and serve as formers. The rings are stacked inside the core barrel and are aligned and connected by tie rods which pass through all the rings. The tie rods are bolted to the lower core support and prevent lift off during operation. Holes of varying size are drilled in the rings to provide cooling necessitated by gamma heating within the metal. Approximately one percent core bypass flow is allocated to cooling the reflector. The reflector rings are forged 304 LN material. Other parts are fabricated from 304 or 316 material, such as the fasteners. The cobalt content is controlled in all materials so that the integrated average content is less than 0.02 percent.

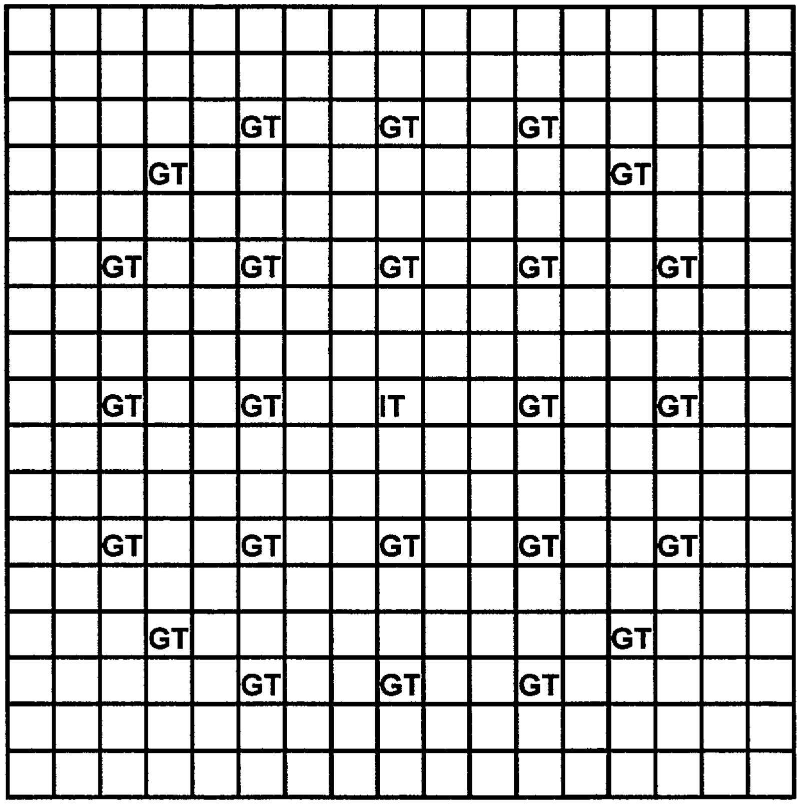
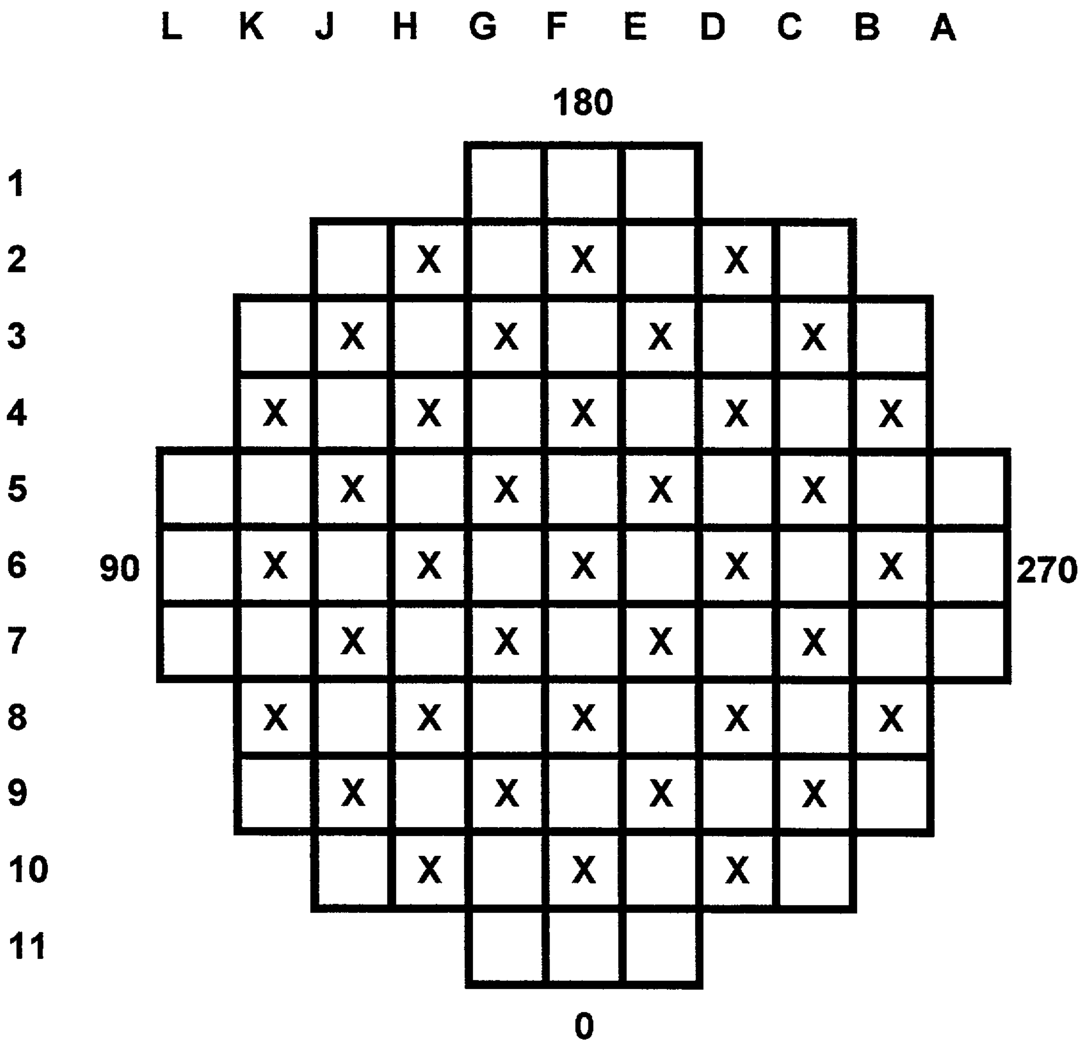


Figure 3.2-1 IRIS 17x17 fuel assembly layout
(GT = guide thimble; IT = instrumentation tube)



Bank	Number of Clusters
MA (Gray Bank A)	8
MB (Black Bank B)	4
MC (Black Bank C)	4
AO (A.O. Control Bank)	5
SD1 (Shutdown Bank 1)	4
SD2 (Shutdown Bank 2)	4
SD3 (Shutdown Bank 3)	8
TOTAL	37

Figure 3.2-2 IRIS Control Rod Locations

(Individual bank positions not shown, not finalized)

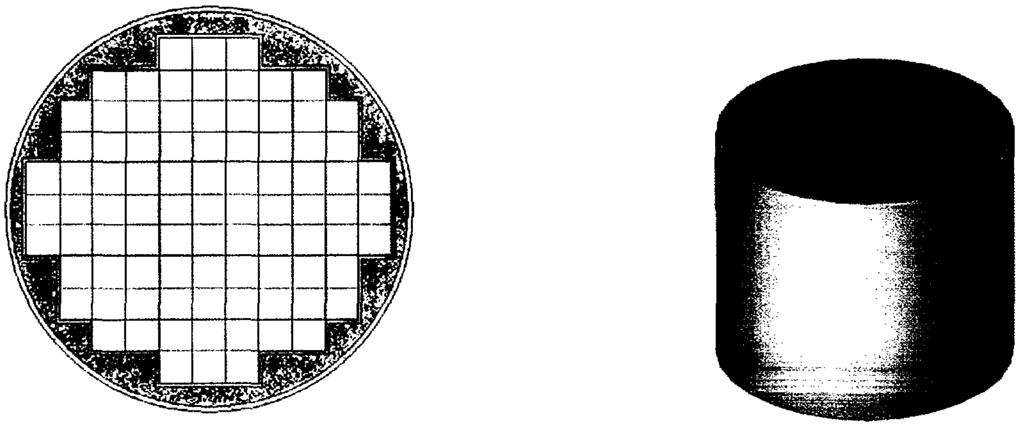


Figure 3.2-3 IRIS radial neutron reflector assembly (conceptual)

3.3 REACTOR VESSEL AND INTERNALS

The design of the IRIS reactor pressure vessel and internals is conceptually similar to the existing PWRs; it contains the pressurized primary coolant water and houses the reactor core and internals, the reactor control rods and drive lines, and the core monitoring instrumentation. However, the IRIS vessel is longer and has a larger diameter than a typical PWR vessel in order to also house the eight (8) steam generators, eight (8) primary coolant pumps and pump diffusers, the pressurizer and its heaters, and the radial reflector. Figures 3.3-1 to 3.3-9 depict the IRIS reactor vessel and internals preliminary design.

3.3.1 REACTOR VESSEL

As shown in Figure 3.3-1, the reactor vessel consists of a cylindrical shell made of several courses, a semispherical dished bottom head, and a flanged and gasketed removable upper head. Stainless-steel cladding of 6 mm. minimum thickness covers all internal surfaces of the vessel. The reactor vessel size and configuration is dictated largely by the space required by the steam generators and internally mounted reactor coolant pumps. The reactor vessel's main design data are summarized in Table 3.3-1.

Table 3.3-1 IRIS Reactor Vessel Parameters

Overall length of assembled vessel	22214 mm (874.57 in)
Inside diameter (to base metal)	6223 mm (245 in)
Nominal base metal thickness	280 mm. (11.02 in)
Minimum cladding thickness	6 mm. (0.236 in)
Design pressure	17.24 MPa (2500 psia)
Design temperature	343.3°C (650°F)
Vessel material	Carbon steel, SA 508, Gr.3, Cl.2
Cladding material	Stainless steel

(a,c)

Figure 3.3-1 IRIS Reactor Vessel, Internals and Integral Components Cross Section.

(a,c)

Figure 3.3-2 IRIS Reactor Vessel

(a,c)

Figure 3.3-3 Reactor Vessel Upper Head

(a,c)

Figure 3.3-4 Inverted Top Hat

(a,c)

Figure 3.3-5 Barrel

(a,c)

Figure 3.3-6 Upper Reactor Internals

(a,c)

Figure 3.3-7 Core Neutron Reflector

(a,c)

Figure 3.3-8 Lower Core Support Plate

(a,c)

Figure 3.3-9 Core Support and Thermal Shields

The removable upper head of the vessel contains a bolting flange with 72, 8 inch diameter studs and nuts. Hydraulic tensioning of the studs will permit uniform nut loading on the clusters. Two hollow, metallic O-rings form a pressure-tight seal in concentric grooves in the head flange. The closure head is penetrated by the control rod drive mechanisms (CRDMs), the pressurizer heater rods, the core monitoring instrumentation tubes, and the nozzles for piping connections to the safety valves and for the auxiliary pressurizer spray line.

The reactor vessel cylindrical wall has 8 feedwater inlet nozzles located just above the core level and 8 steam outlet nozzles located below the vessel flange. The cylindrical wall is also penetrated by smaller piping connections for auxiliary systems including connections to and from the chemical and volume control system (CVCS), the normal residual heat removal system (NRHRS), and the long-term gravity makeup system (LGMS). These piping connections are small in size, the largest being the normal residual heat removal system suction lines, which are 4 inch, Sch. 160 lines.

During normal operation, the reactor coolant is totally contained within the reactor vessel and is pumped in a closed circuit within the vessel (Fig. 3.1-1) with the exception of the flow to and from the CVCS purification equipment. The coolant passes up through the core, turns radially outward at the top of the upper internals, flows up to the eight primary pumps, is pumped downward by the pumps and through the steam generators, passing over the tube bundle, down the annulus between the core barrel and inside reactor vessel wall, then upward through the core support assembly. Steam generator feedwater passes through the feedwater nozzle into a feedwater header, enters the steam generator tubes and flows upward inside the tubes, first being heated to saturation then boiled, and then heated to dry superheated steam, which then flows into the upper steam discharge header and flows out through the steam outlet nozzles to the turbines.

The IRIS pressurizer is located in the reactor vessel upper head (shown in Figure 3.3-3 and 3.3-10), and is discussed in Section 3.6.

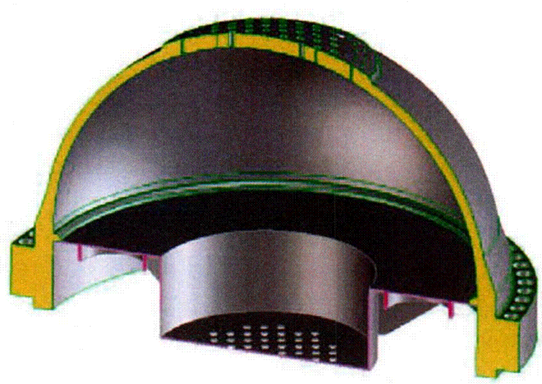


Figure 3.3-10 IRIS Reactor Vessel Upper Head and Pressurizer volume

The reactor vessel and closure head are designed as a Class 1 vessel in accordance with the ASME Code, Section III. The design life for the reactor vessel is at least 60 years. In general, all attachments and pressure containing parts have full penetration welds.

C02

3.3.2 Reactor Vessel Support

The reactor vessel shown in Figure 3.1-1 is supported vertically by means of a conical skirt welded to the cylindrical shell between the steam generator inlet and outlet nozzles. The support is designed to restrain lateral, vertical, and rotational movement of the reactor vessel and still allow for thermal growth. This type support and location provides a vessel structure with high natural vibration frequency, which can reduce the seismic response of the reactor vessel and the dynamic interaction of the different components. The possibility of using a cylindrical skirt welded to a "Y" forging between the lower cylindrical shell and the semispherical lower head is also being evaluated.

The dynamic evaluation of seismic effects is one of the most important considerations in the design of the reactor vessel external support and of the internal structures. The natural frequency of the reactor vessel has been determined for the lower and middle vessel skirt designs. This will be used together with seismic and other forcing frequencies to determine possible resonant conditions. This dynamic analysis will determine the influence of the overall supporting scheme on the reactor and internals.

Another consideration in the design of the reactor vessel skirt is the thermal stress due to the temperature gradient of the skirt at the attachment to the reactor vessel. Detailed thermal stress analysis of this area using finite-element techniques will be performed to determine primary plus secondary stresses of heatup and cooldown thermal transients. In order to provide good heat flow from the reactor vessel to the skirt, a forged skirt attachment with full penetration welds and the selective use of insulation in the crotch area will be used.

3.3.3 Reactor Vessel Internals

The IRIS reactor vessel internals (RVI) are similar to those of current PWRs in that they support the core, core barrel, control rods, control rod drive lines and they also form the circulation path for the flow of coolant through the core. In IRIS, however, the RVIs provide the additional functions of supporting the internally mounted steam generators, reactor coolant pumps and radial reflector. In addition, the IRIS RVIs must provide support for the pressurizer heater rods, additional lateral support for the longer than normal control rod drivelines and provide an extended length upper core barrel to form the core flow path. The internals are designed to withstand the forces due to weight, preload of fuel assemblies, control rod dynamic loading, vibration and earthquake acceleration.

The IRIS reactor vessel and internals are designed to permit the refueling operations to be conducted in the same manner as in current PWRs, providing access to the fuel assemblies after removal of the closure head and upper internals. Refueling operations do not require the removal of the coolant pumps or steam generators; however, their support structures are designed to permit removal of these components for out-of-vessel inspection and replacement. The components of the reactor internals are divided into two parts:

1. The lower core support structure (including the entire core barrel and reflector)

2. The upper core support assembly.

3.3.3.1 Lower Core Support Structure

The major restraining and support member of the reactor internals is the lower core support structure, shown in Figure 3.3-11. This support structure assembly consists mainly of the core barrel, the core reflector, the lower core plate, the shield plates, the triangular shaped core support members which are welded to the bottom head, and the core support ring which also functions as neutron shielding for a portion of the lower head. Another intermediate shield plate will be placed below the lower core plate to complete the shielding of the vessel bottom head; ledges welded to the bottom head support it. All the major components of this structure are supported at the bottom head, as shown in the lower portion of Figure 3.3-11. The core barrel is restrained in its transverse movement by a bolted connection to the support ring, which rests on the (triangular) support members. Within the core barrel is the reflector, which is attached to the core barrel wall and forms the enclosure periphery of the assembled core. The lower core plate is positioned at the bottom level of the core below the reflector and provides support and orientation for the fuel assemblies. The lower core plate is perforated and contains the locating pins for the fuel assemblies. The lower core support structure (principally the core barrel) also serves to define the passage-ways for the primary coolant flow through the core.

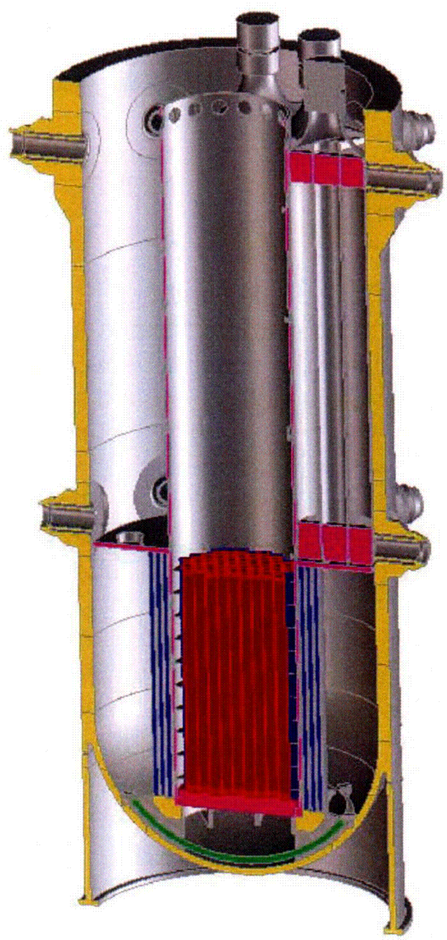


Figure 3.3-11 Lower Core Support Structure

C03

At this stage of the design, the project has not yet decided if there will be cylindrical shield plates between the reflector and the vessel. The final decision will be based on a cost-benefit analysis. In any case, the IRIS design will be able to accommodate cylindrical thermal shield plates by supporting them on horizontal ledges, which are welded to the internal surface of the vessel and supported on the vertical members welded to the bottom head. This bottom support allows for differential axial growth of the shields with respect to the core barrel but restricts radial or horizontal movement of the bottom of the shields.

The vertically downward loads from weight, fuel assembly preload, control rod dynamic loading, and earthquake acceleration are carried by the lower core plate partially through the support ring to the lower (triangular) support members and to the bottom head. Transverse loads from earthquake acceleration, coolant crossflow and vibration are carried by the core barrel shell to be shared by the horizontal ledges, the support ring, and the vessel shell. Transverse acceleration of the fuel assemblies is transmitted to the core barrel shell by direct connection of the lower core support plate to the barrel wall and by a radial support-type connection of the upper core plate to slab-sided pins pressed into the core barrel.

With this design, the internals are provided with a support at the furthest extremity, with the core barrel bolted to the column supports, and may be viewed as a beam simply supported at the bottom. Radial and axial expansions of the core barrel are accommodated, but transverse movement of the core barrel is restricted by this design, keeping cyclic stresses in the internal structures within the ASME Section III limits, which essentially eliminates any possibility of failure of the core support.

3.3.3.2 Upper Core Support Assembly

The upper core support assembly, Figure 3.3-6 and 3.3-12, consists of the upper support plate, upper core plate, support columns, middle support plates and guide tube assemblies (not shown). The support columns establish the spacing between the upper support plate, middle support plates and the upper core plate and are fastened at top and bottom to these plates. These support columns transmit mechanical loadings between the plates and serve the supplementary function of supporting in-core and ex-core instrumentation conduits. The guide tube assemblies sheath and guide the RCCA rodlets and drive shafts, but provide no other mechanical functions; they are fastened to the lower middle support plate and are guided by pins in the upper core plate for proper orientation and support.

The main radial support system between the core barrel and the upper internals is accomplished by key and keyway joints. At equally spaced points around the circumference and coinciding with the level of each support plate, Inconel blocks are welded to the inside diameter of the core barrel. Each of these blocks has a keyway geometry; and opposite each of these is a key which is attached to the upper internals support plates. At assembly, as the internals are lowered into the vessel, the keys engage the keyways in the axial direction.

The upper core support assembly, which is removed as a unit during refueling operations, is positioned in its proper orientation with respect to the lower support structure by flatsided pins pressed into the core barrel which in turn engage in slots in the upper core plate. Slots are milled into the core plate at the same positions. As the upper support structure is lowered into the main internals, the slots in the plate engage the flat-sided pins in the axial direction. Lateral displacement of the plate and of the upper support assembly is restricted by this design. Fuel assembly locating pins protrude from the bottom of the upper core plate and engage the fuel assemblies as the upper assembly is lowered into place. Proper alignment of the lower core support structure, the upper core support assembly, the fuel assemblies, and control rods is thereby assured by this system of locating pins and guidance arrangement.

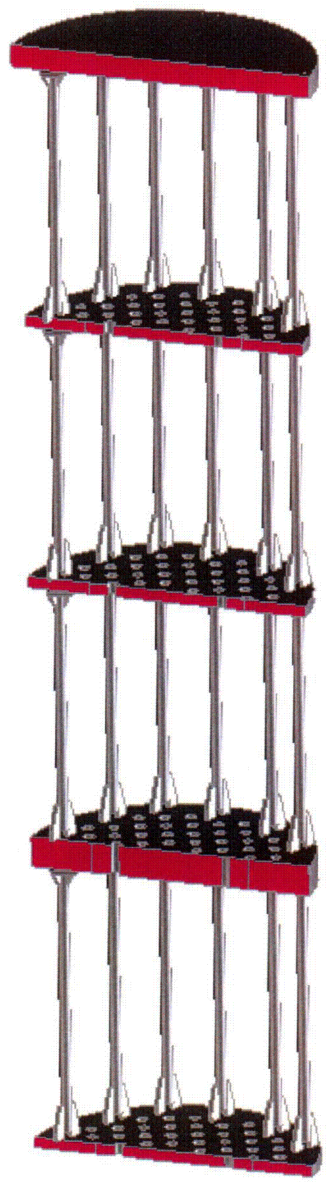


Figure 3.3-12 Upper Core Support Structure

C04

3.3.4 Materials and Construction

Large integral type forgings have been used for the construction of large primary reactor components for nuclear power plants in order to reduce the length of welds, which reduces both the manufacturing time and the time required for in-service inspections. Figure 3.3-13 shows a possible course layout for the IRIS reactor vessel design.

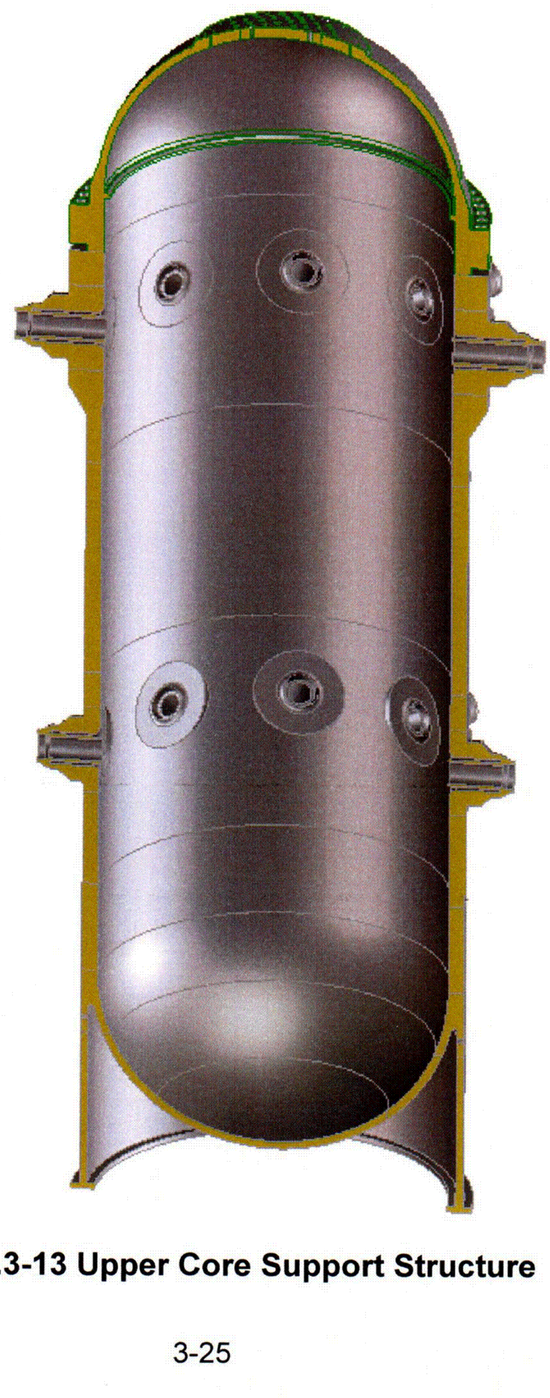


Figure 3.3-13 Upper Core Support Structure

C05

In order to decrease the overall weight, high strength SA 508, Gr.3, Cl.2 carbon steel has been selected for the reactor pressure vessel shell, flanges, and upper and lower heads. The resulting decrease in weight is beneficial from the standpoint of reducing loads resulting from earthquake. This material is used because of its strength properties, availability in the required sizes and thickness, satisfactory service in a neutron and gamma field, and the capability of producing high quality welds. The material is also compatible with the weld overlay cladding of stainless steel. All surfaces of the reactor vessel in contact with reactor coolant are either clad with, or made from 300 series stainless steel and/or Inconel 690. Based on tensile and impact properties, Type SA 540, Class 3 carbon steel has been selected for closure studs, nuts, and washers.

The vessel shell material can be further protected from fast neutron flux and gamma heating effects by the assembly of shield plates made of stainless steel and located between the reflector and the vessel wall. Studies are under way to identify the extent of shielding required and its costs/benefits will be assessed.

The current design has verified the adequacy of materials and suitable construction techniques for the large ring forgings that make up the IRIS vessel. The challenges rising from the manipulation required for the assembly/construction and inspection of the vessel with its very large internal diameter and wall thickness are being carefully studied and will be implemented prior to the manufacturing stage. Also, the transportation of such a large and heavy piece of equipment to the plant site and its installation are additional areas of current study. However, the use of the integral reactor vessel configuration allows the plant construction strategies to incorporate changes that directly meet the challenge of shortening construction time. The first benefit of this integral reactor system configuration, and probably the most obvious, is the large reduction of bulk quantities and components used. Because there are no external cooling loops, steam generators, pressurizer, or pumps, the containment building housing the reactor coolant system can be significantly reduced in diameter and volume. Fewer quantities and components mean less installation time with the obvious corollary that if there is less to install, there will be fewer installation mistakes and associated rework.

3.3.5 Integrated Head Package

The Reactor Vessel also includes the integrated head package (IHP), which houses the control rod drive mechanisms that drive the control rods up and down for reactor control; the incore instrumentation system extensions; and the seismic support plate and cooling shroud (which is provided to direct cooling air flow around the control rod drive mechanisms).

The integrated head package (IHP) combines several separate components in one structure to simplify refueling of the reactor. Refueling outages require considerable time and effort to prepare the reactor vessel head for movement between the normal operation location and the storage location. Some of these tasks are critical path for refueling outage. In addition, radiation levels are high during this operation and significant radiation exposure may occur. Thus, the purpose of the integrated head package is to reduce the outage time and personnel radiation exposure by combining operations associated with movement of the reactor vessel head into a single structure.

The design of the IRIS IHP is similar to the Westinghouse standard design used in present plants. The IHP consists of the shroud assembly and cooling system, the lifting rig, missile shield, CRDM seismic supports, shielding, guide structure for incore instrumentation thimble assemblies, cables, messenger tray and cable bridge. With the IHP concept, the CRDMs and rod position indicators (RPI) remain with the reactor vessel head within the cooling shroud assembly at all times. The shroud assembly is a carbon steel structure, which encloses the CRDMs above the reactor head. During normal operation it directs the flow of cooling air for the CRDMs coil stacks and RPIs. Structurally the shroud is integrated with the head lifting system, the missile shield and the CRDM seismic supports. The lifting rig is provided to lift the vessel head and IHP as a unit. It consists of three vertical lift rods attached to the shroud, extending up through the IHP and the incore instrumentation housing assemblies and connecting at the top to a spreader and three lift legs. The seismic support plate consists of a single horizontal plate of steel located in the region directly above the rod travel housings; its purpose is to support the upper end of the CRDM and act as a spreader bar for the lifting legs.

The upper head has 45 four (4) inch outer diameter penetrations for the CRDM housings, 38 one and a half (1.5) inch penetrations for the incore instrumentation guide tubes, 90 penetrations for pressurizer heater rods, 8 penetrations for excore neutron flux detectors, 2 penetrations for the 4 inch safety valve piping, and 1 penetration for the auxiliary spray (and head vent) piping connection. Twelve IHP lugs are welded to the top of the closure head.

Incore Instrumentation was discussed in detail in Section 3.2.3.

CRDMs were discussed in Section 3.2.2.

The messenger cable tray encircles the outside of the IHP above the control rod travel housings. It provides support and routing for the CRDM power cables, the RPI power and instrumentation cables, and the pressurizer heater rod power cables. These cables terminate at the connector plate, which constitutes the interface with the mating cables. Cable disconnects are made at this connector plate. Note that because IRIS has additional cables for the pressurizer heater rods, two levels of cable trays may be used in the final IHP design. Also, the interfaces between the connector plate(s), and the cable routing to the containment penetrations are not yet finalized. The intent is to have cable connections that are easily disconnected and which allow unimpeded removal of the IHP in preparation for refueling.

3.4 REACTOR COOLANT PUMP

An advanced, spool-type reactor coolant pump (RCP) that can be totally contained within the reactor vessel has been adopted as the reference for the IRIS reactor, with the backup design being the canned pump similar to the type used in AP600. The "spool type" pump that is envisioned for IRIS, is an extension of marine applications requiring high flow rates and low developed head.

In its simplest form, the spool type RCP motor and pump impeller consist of two concentric cylinders, where the outer ring is the stator and the inner ring is the rotor that carries high specific speed pump impellers. This pump has several advantages over typical RCPs that have the pump/impeller extending through a large opening in the pressure boundary with the motor extending outside. In the case of canned motor pumps, the motor casing becomes part of the pressure boundary and is typically flanged and seal welded to the mating pressure boundary surface. The spool type pump would be located entirely within the reactor vessel eliminating the need for the high pressure casing, large vessel openings and closure flanges; only small penetrations for the electrical power cables and for water cooling supply and return piping, if necessary, are required. Furthermore, the use of high temperature motor windings and bearing materials are being investigated in order to eliminate even the need for cooling water and the associated small piping connections.

In addition to the above advantages deriving from its integral location, other advantages of the spool pump result from its geometric configuration, which is amenable to provide sufficient inertia for flow coastdown, and flow run-out capability. Both these characteristics contribute to mitigate the consequences of loss-of-flow-accidents (LOFAs). Because of the low developed head, spool pumps have never been candidates for nuclear applications. The integral configuration, low-pressure drop IRIS can accommodate these pumps and take advantage of their characteristics.

3.4.1 Design Background

The Curtiss-Wright Electro-Mechanical Corporation has been designing, manufacturing, and testing canned motor pumps (one of its primary products) for both fossil and nuclear power generation service for over 50 years. The critical nature of these applications demands high reliability, and quality. To date, Curtiss-Wright Electro-Mechanical Corporation has designed, manufactured, and delivered over 1700 canned motor pumps. Almost all of these units have operated without repair or maintenance over their life. These units were originally developed to meet the rigorous demands of circulating high-temperature, high-pressure, radioactive fluids in pressurized water nuclear reactor systems with zero leakage. Besides nuclear applications, these canned motor pumps (CMP) were found to be ideal for non-nuclear applications and were installed as fossil boiler circulating pumps for very high pressure and temperature applications with supercritical water. This canned motor technology was applied to meet needs for deep sea submersible vehicles and other marine applications and resulted in a fluid moving device called the Integral Motor/Propeller (IM/P)™, (see Figure 3.4-1), which is the fore-runner of the spool type pump for the IRIS. Figure 3.4-2 identifies the IM/P™ basic components, and illustrates how typical canned motor pump components are reconfigured.

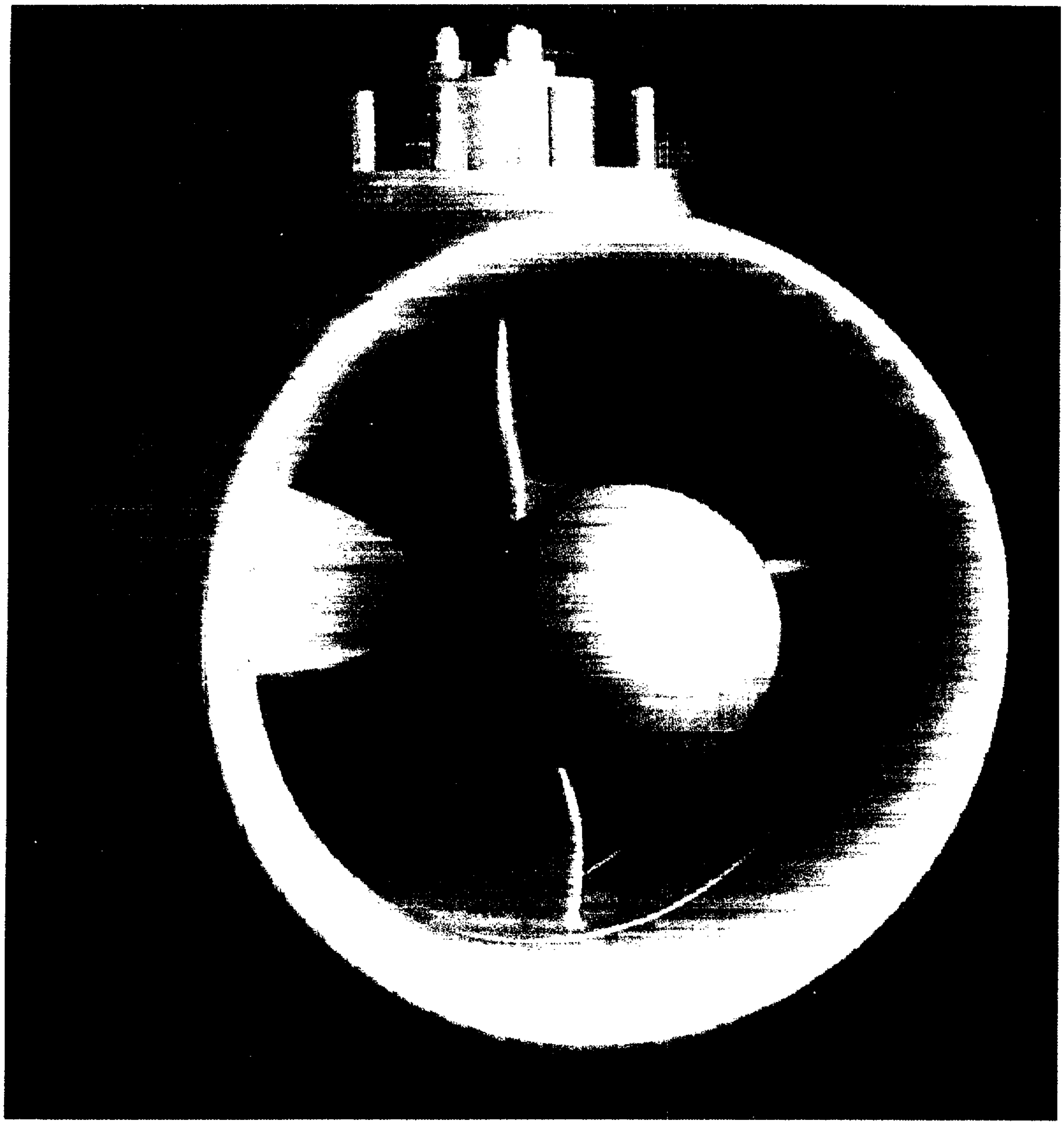


Figure 3.4-1 Curtiss-Wright Electro-Mechanical Corporation IM/P^{TM}

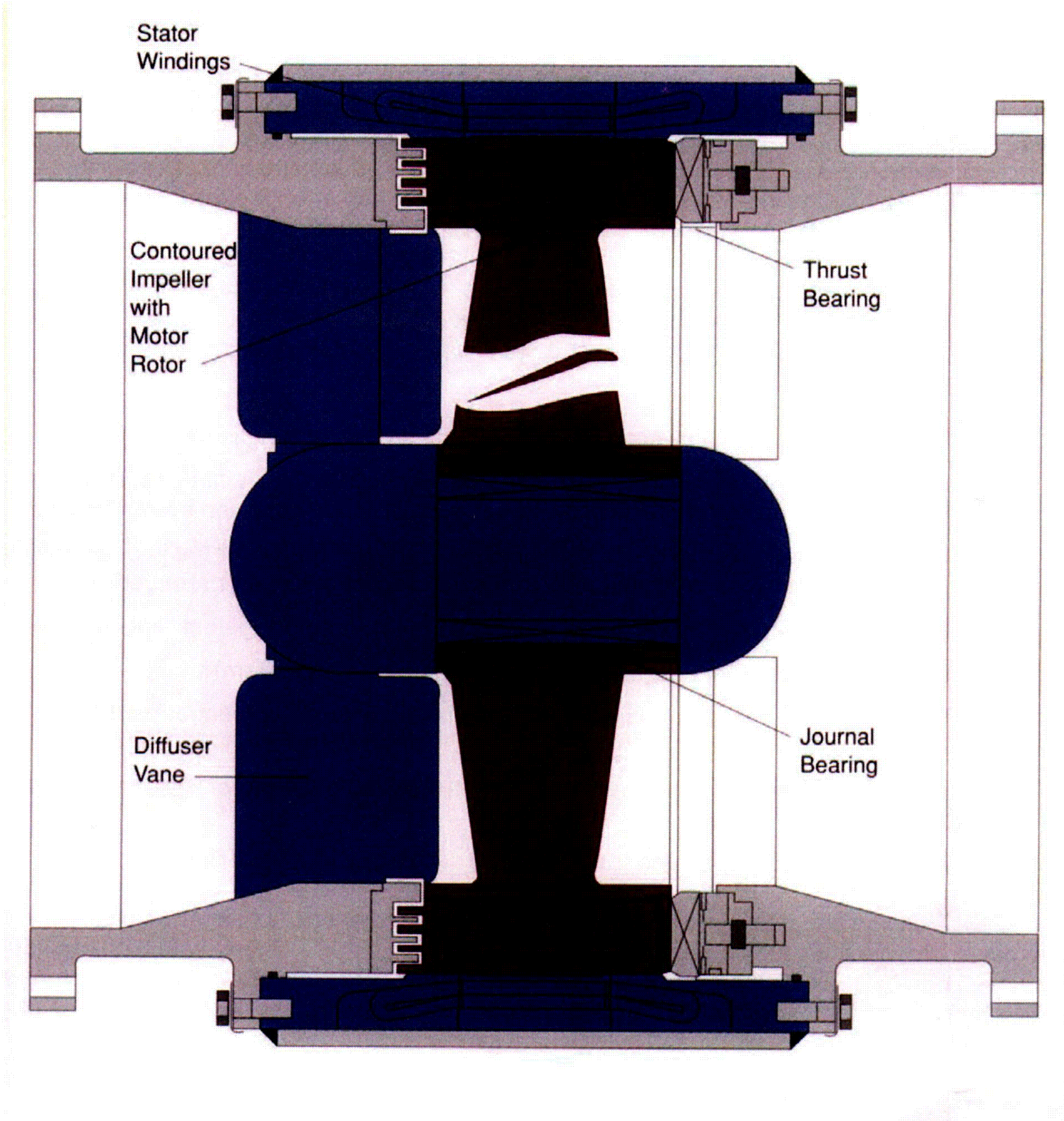


Figure 3.4-2 Curtiss-Wright IM/P Basic Components

C06

The first of two key items of the spool pump technology is the canned and hermetically sealed rotor and stator. As in the canned motor pump, they permit the motor to operate in literally any fluid without sealing problems. The other key item is the use of process fluid lubricated bearings, which eliminate the need for rotating seals and a lubrication system. The complete IM/P™ consists of a canned motor, comprised of the rotor and stator with the rotor winding surrounding the propeller, the bearings, and the frame and support structure. The bearings, radial and thrust, can be located either on a stationary rotor hub or on the rotor rim in a hub-less arrangement, depending on the loads from the potential applications.

By integrating the electric motor with the impeller, the IM/P™ achieves a number of advantages over conventional pumping devices that used a traditional propeller/shaft/motor arrangement. These same advantages apply to the use of a spool type pump in the IRIS reactor vessel, namely:

- Powerful, compact, lightweight (no high-pressure pump casing is needed)
- No dynamic mechanical pressure vessel penetrations (no shaft penetration through the pressure boundary with attendant seal injection and lubrication systems required)
- No large pressure vessel flanged connections (no penetration for the impeller/diffuser bowel, only an electrical penetration through the reactor vessel is required)
- Direct water cooled and lubricated bearings (no seal support system required)
- Direct water cooled motor (no water or air cooled cooling system)
- Rugged, reliable, easy to maintain (no seal inspection/replacement or periodic maintenance required)
- Reduced hydraulic and mechanical vibration (fully supported rotor, no cantilevered shaft/impeller/diffuser)
- Canned hermetically sealed rotor and stator (no periodic maintenance)

The IM/P™ design and deployment have provided a large experience base on which to base the development of spool-type pumps for an application like IRIS. Designs from 32 hp to 350 hp have been built, and qualified in tests exceeding 10,000 hours, and have operated for years with almost no maintenance issues. Conceptual designs are being developed for units as large as 50,000 HP.

3.4.2 IRIS Reactor Coolant Pumps Design

The preliminary requirements for IRIS RCPs are summarized in Table 3.4-1:

Table 3.4-1 IRIS RCPs design requirements

(a,c)

These parameters indicate that some mixed flow needs to be provided. Thus the spool pump concept is modified from a strict axial flow pump, by adding a bulbous section to provide radial flow and to generate the required developed head, as shown in Figure 3.4-3. One radial bearing will be below the motor portion, and the other bearings would be typically located in the hub.

Main thermal-hydraulic characteristics for IRIS RCPs, including head, NPSH required and efficiency versus flow curves are provided in Figures 3.4-4 to 3.4-6.

3.4.3 IRIS Spool-type Pump Development Efforts

In order to adapt the spool-type pump technology to IRIS conditions, several design development activities are required. This development effort includes the use of high temperature motor stator and rotor windings that can operate inside the IRIS reactor vessel without the need for external cooling water and cooling coils within the motor. Also, the material used for the hydrostatic, water lubricated bearings must be verified by test, since the IRIS water temperatures are somewhat higher than current bearing materials are designed for.

Because the pump is located in the hot fluid portion of the IRIS reactor vessel, the winding insulation system and the critical bearing design must be able to operate in a ~329.4°C (625°F) and 15.5 MPa (~2235 psig) water environment. Curtiss-Wright has developed the technologies to meet these conditions, including a winding insulation system designed to operate in a 500°C (932°F) environment, and a number of bearing configurations that have been shown by test to be acceptable. Canned motor pump tests utilizing a 500°C (932°F) insulation and bearing system have been performed. The test report concluded: "the motor has been successfully demonstrated; thereby, proving feasibility of eliminating plant auxiliary cooling for the main coolant pump and adding to plant reliability. The design utilized commercially available materials." This report also noted that, "Additional tests should be performed in order to verify long term durability and reliability prior to acceptance of this motor for nuclear applications. Thermal cycling, starts and stops, and extended periods of run at full operating temperature are required." A detailed plan, which includes vibration testing, thermal cycling, radiation aging, and material availability confirmation will be included in the IRIS development program test plan.

The bearing materials used in the testing are fairly common hard/soft material couples. The bearing design included additional features to address the relative growth of the bearing surfaces to maintain the critical bearing clearance. Since these tests, design advances on thermal bearing issues and hard on hard bearing designs were performed. A study is being conducted to establish the best design and material selection fit for a spool pump application such as IRIS.

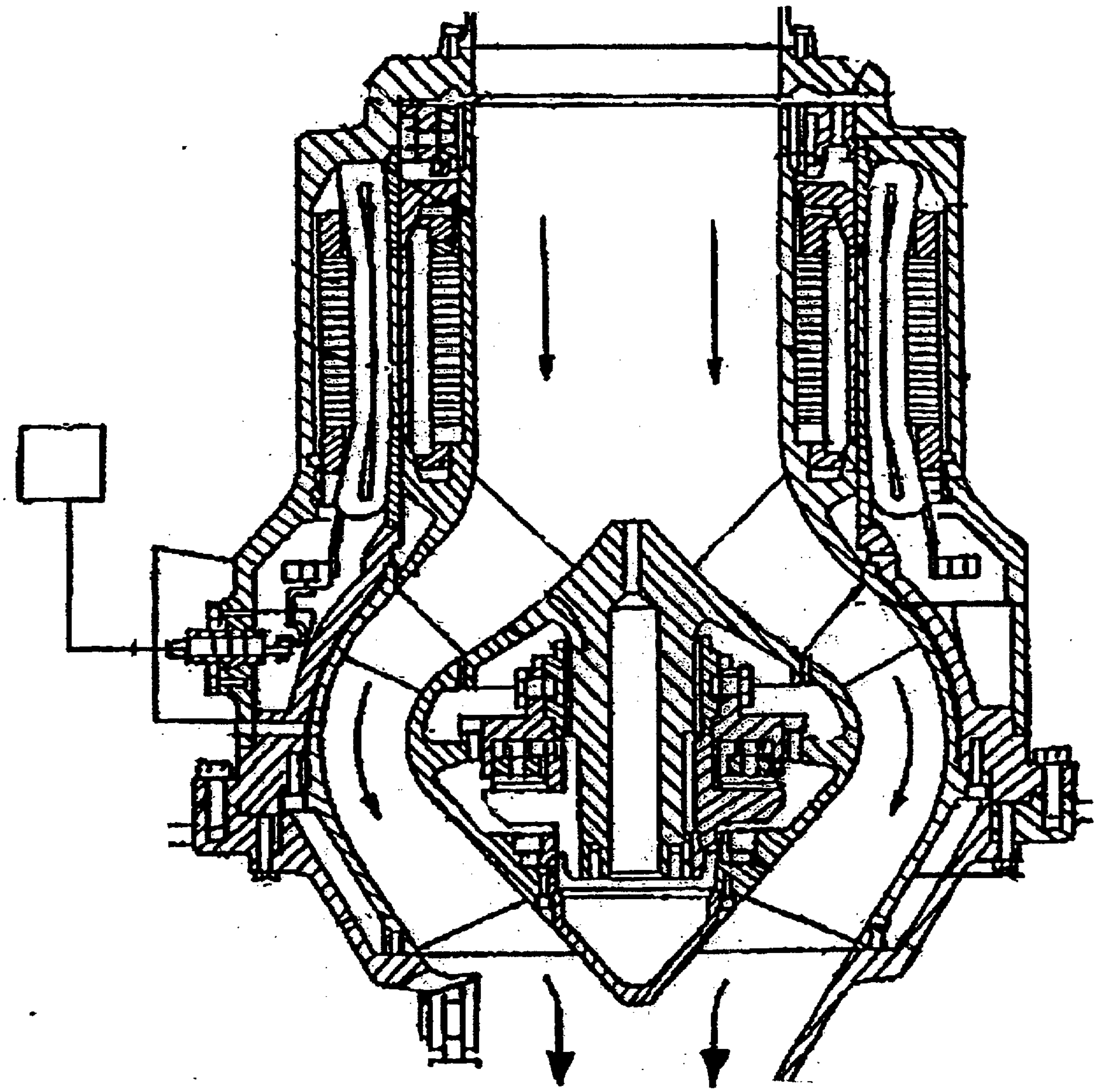


Figure 3.4-3 Example of Spool-type Pump with Mixed Flow Hydraulics

(a,c)

Figure 3.4-4 IRIS Spool pumps Head versus Flow characteristic curve.

(a,c)

Figure 3.4-5 NPSH required versus Flow curve

(a,c)

Figure 3.4-6 Hydraulic Efficiency versus Flow Curve

3.5 STEAM GENERATOR DESIGN

The basic function of the IRIS steam generator (SG) is to transfer heat from the single-phase reactor coolant water through the helical coil steam generator tubes to the boiling, two-phase steam mixture in the secondary side of the steam generator. Figures 3.5-1 to 3.5-3 show an overall view of the IRIS steam generators and their location in the integral reactor coolant system.

The IRIS SGs are a once-through, helical-coil tube bundle design with the primary fluid outside the tubes. Eight steam generator modules are located in the annular space between the core barrel and the reactor vessel (Figure 3.5-1). Each IRIS SG module consists of a central inner column that supports the tubes the tube support system, and an outer wrapper. The central column is attached to the RV wall via cantilevered arms, so that the weight of the SG is supported from the RV wall. The feed water and steam headers are bolted separately to the vessel from the inside of the feed inlet and steam outlet pipes. A double gasket, with a monitor leak-off, provides the pressure boundary between the primary coolant and the secondary side feed water inlet and steam outlet penetrations in the reactor vessel. The enveloping outer diameter of the tube bundle is 1.64 m (5.38 ft) (Figure 3.5-2).

Each SG has 656, 17.46 mm (0.688 inch) outside diameter, 2.11 mm (0.083 inch) wall thickness, Inconel-690 thermally treated tubes. The tubes and headers are designed for the full RCS pressure applied to the outside surfaces. The vertical sides of the lower feedwater header and the upper steam header are the tubesheet surfaces and the tubes are connected to these vertical sides of the headers. A 3D rendering of an IRIS SG is provided in Figure 3.5-3.

In operation, feed water enters the SG through a nozzle in the reactor vessel wall and enters the lower feed water header. The feedwater enters the SG tubing, and is heated to saturation temperature, boiled to steam, and superheated as it flows upward to the upper steam header. Steam then exits the SG through the nozzle in the reactor vessel wall.

The helical SG tube bundle is contained within an outer wrapper (flow shroud) that directs the primary water flow from the top of the SG, downward through the bundle (outside the tubes), and out the bottom of the bundle into the reactor vessel downcomer region. Each of the eight reactor coolant pumps is attached directly to the top of its corresponding SG flow shroud, so that its flow is entirely directed through the SG bundle region.

3.5.1 Design Background

Several configurations have been examined for the IRIS steam generator: straight-tube, U-tube, helical tube, C-tube, bayonet tube. Based on overall lifecycle costs, design and manufacturing experience, and high reliability, a helical-coil tube bundle steam generator was selected. The helical-coil tube bundle is a proven design that has operated in various reactors, including the French LMFBR Superphénix, and the PWR powered German nuclear ship Otto Hahn with its 38 MW SG. The helical-coil tube bundle design is capable of accommodating thermal expansion without excessive mechanical stress, has high resistance to flow-induced vibrations, and is designed to have thermal performance second only to a straight-tube design (which was discarded because of the high loads due to thermal expansion caused by temperature

transients, resulting in large compressive forces between the feed and steam headers).
As background information, the IRIS helical-coil, once-through (HCOT) SG design was developed in the 90's by Ansaldo SA as part of an effort to develop an integral PWR called ISIS (Inherently Safe Immersed System). The innovative aspects of the ISIS SG were successfully tested in an extensive test campaign as discussed in the following section.

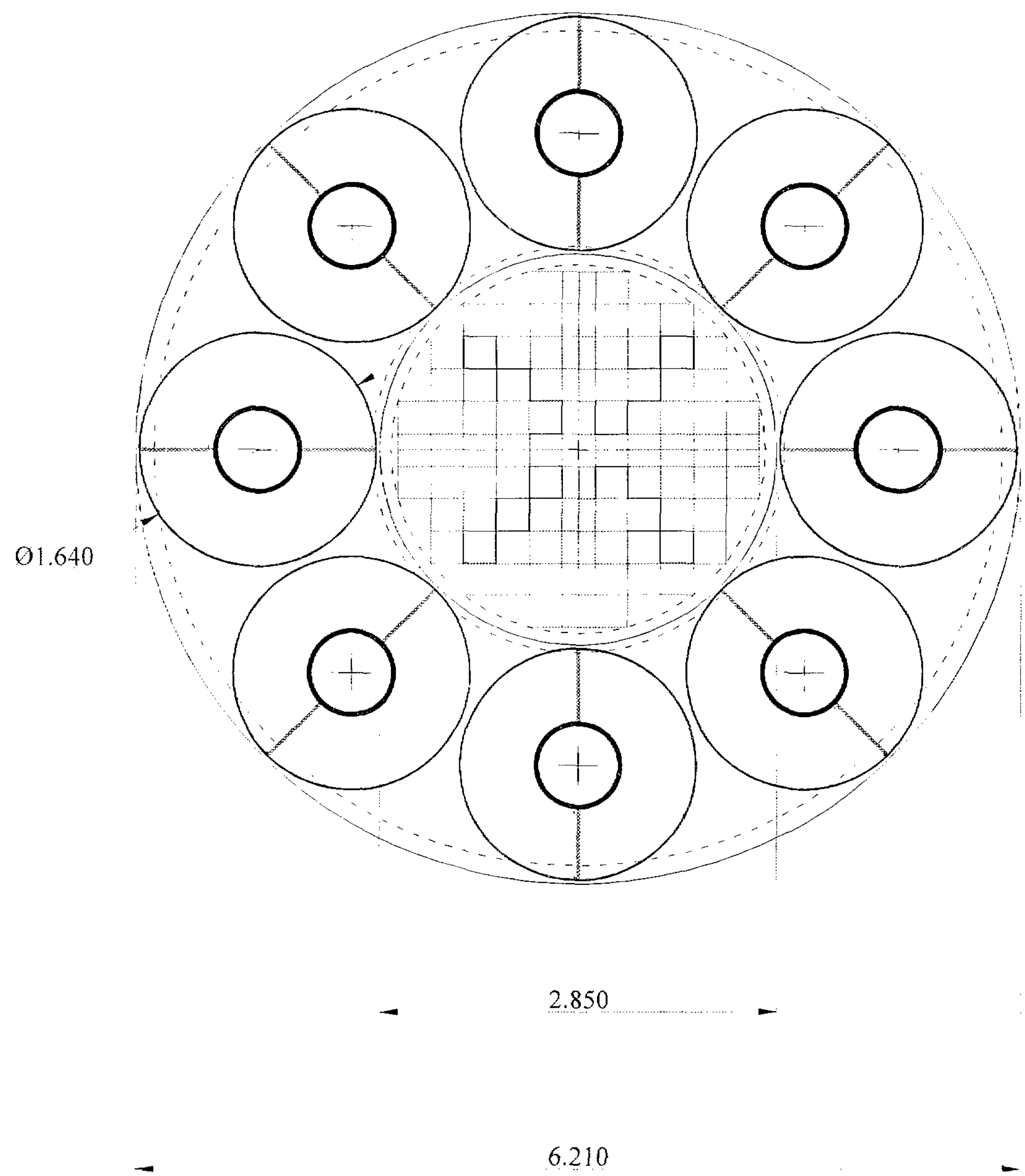


Figure 3.5-1 IRIS SGs layout in the Integral Reactor Coolant System

(a,c)

Figure 3.5-2 SG module layout and main data

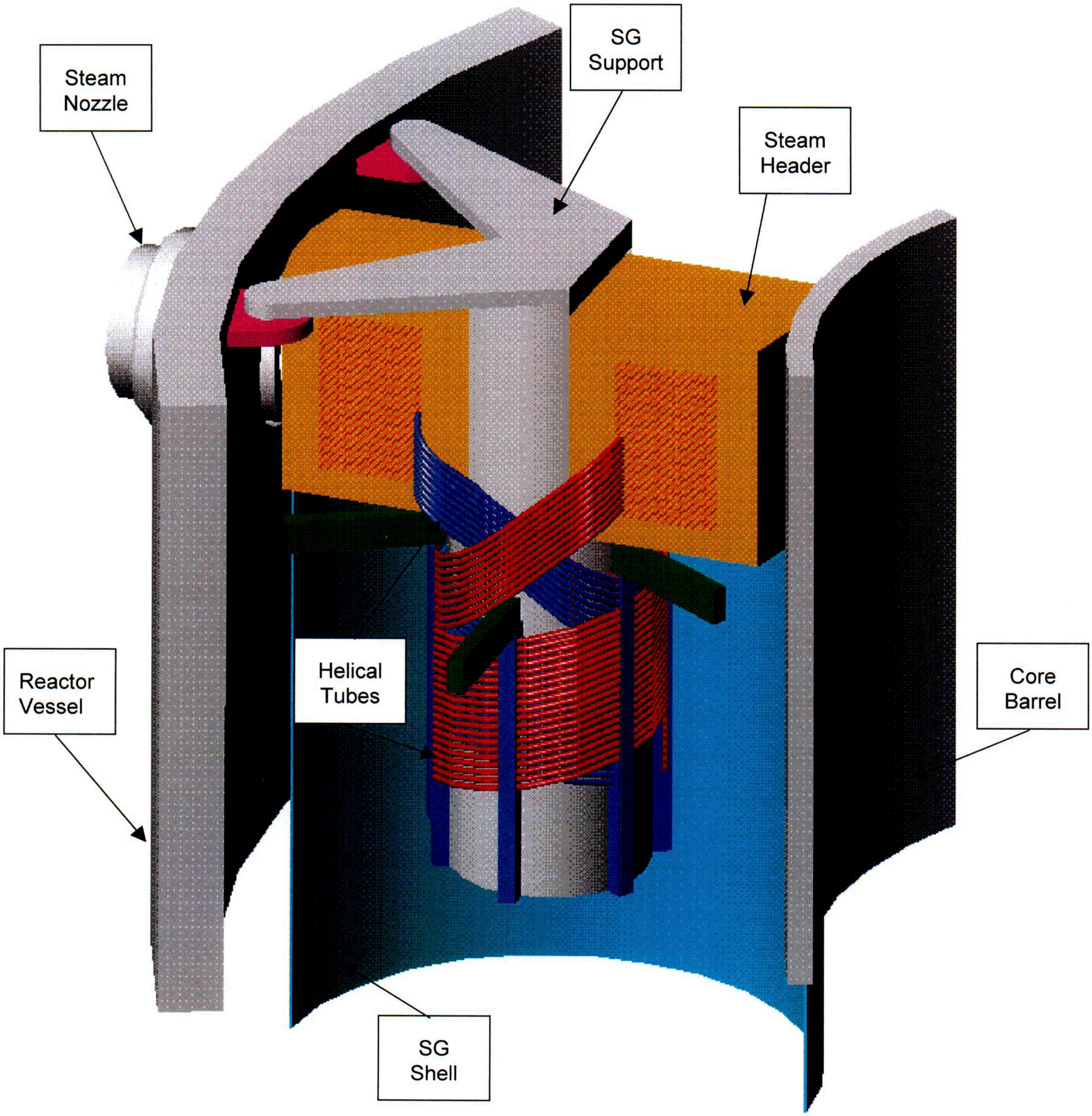


Figure 3.5-3 3D Representation of an IRIS SG module

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3.5.2 Experimental Design Basis

The steam generator proposed for IRIS will profit from design experience and test results of the ISIS (Inherently Safe Immersed System), a modular light water reactor with innovative, full-passive safety characteristics developed by Ansaldo in the 90's. A mock-up of the ISIS steam generator was constructed in 1997 and tested by Ansaldo.
The mock-up consisted of a 20 MW_t full diameter, reduced height, test article with 50 tubes arranged in 5 rows of 10 tubes, each row forming - alternately - 3 clockwise and 2 counterclockwise coils. The construction of the mock-up demonstrated the feasibility of building this compact SG. A test campaign was carried out on the SG mock-up of ISIS, to confirm the thermal hydraulic and mechanical performance (thermal, vibration, pressure losses) of the SG. Tests were also aimed at identifying instability thresholds to identify any flow regimes that could cause secondary-side parallel channel flow instability between tubes. The concern was that because steam generation and superheat takes place inside the tubes, a condition with potential parallel-channel flow instability exists. Furthermore the tube bundle support system had to be tested in order to ensure minimal flow-induced vibrations.

The test campaign was carried out at the SIET test facility in Piacenza, Italy. The main test results were as follows:

- Confirmation of the predicted thermal performance.
- Identification of about 25% margin on the calculated primary side pressure losses.
- Identification of the domain of stable operation as a function of:
  - Primary coolant inlet temperature;
  - Secondary coolant flow rate;
  - Secondary coolant inlet temperature; and
  - Secondary pressure.

The results of the test campaign of the ISIS SG mock-up are applicable to, and have been used as the basis for the IRIS SG module design. Both designs employ similar tube lengths and bundle diameters and the thermal-hydraulic performance and mechanical behavior at the primary and secondary-side temperatures and flow rates are representative for assessing the IRIS SG module.

3.5.3 Considerations on IRIS SG's In-Service Inspection

The IRIS steam generators tubes and headers can be inspected and normal maintenance can be performed by simply removing the blind flange bolted on the steam nozzle, without removing the steam lines or having to operate from inside the reactor, and without requiring the SGs to be removed from the reactor vessel. The inspection and maintenance will be performed using tooling similar to that developed for the Super Phenix intermediate loop steam generators which were also a once-through, helical coil, design.

Framatome and Ansaldo jointly developed and fabricated a system for the ultrasonic (US) and visual inspection (VI) of the SuperPhenix (SPX) steam generator tubing, that was used for a tube inspection campaign conducted in 1997/98. This inspection included mechanically cleaning the tubes, visual inspection, and an ultra-sonic inspection of the tubes. Since the SPX SG tubes were comprised of several welded pieces, the inspection required the use of ultrasonic, volumetric inspections of the welds. Ultrasonic and visual inspection of the SPX SG tubing was a complex operation, owing to the strict tolerance about flaw size detection, the high inspection speed required, and the fact that the tubes were very long (~100 m) and had a tortuous geometry (helical tubes with orthogonal bends) and high associated friction. The main characteristics of the US probe developed for the SPX inspection were as follows:

- 80 multiple adjacent piezo-elements;
- High speed inspection with 360° electronic scanning;
- High inspection flexibility with electronic focusing and beam steering;
- Inspection of bent tubes with small radius of curvature;
- Center frequency from 8 MHz to 15 MHz.

This SPX inspection verified the feasibility of performing the tube inspection through the steam/feed nozzles using the space-saving circular array tooling developed and verified that high resolution detection was achievable at high (10 MHz) frequency.

The SG Inspection System, including its Signal Acquisition System, the tube Cleaning and Gauging System, and the Hydraulic Probe Driving System, is shown schematically in Figure 3.5-4. The US probe is shown in Figure 3.5-5. An eddy current module (not shown in the figure) was also developed and used to detect the tube support locations and to screen for flaws that were then examined with the US probe. An umbilical cord provides the electrical and mechanical links between probe and the Data Acquisition System.

The *Visual System* was equipped with an axial and a radial micro-camera. The *Hydraulic Driving System*, shown in Figures 3.5-4, uses a reversible water stream to push the probe with attached umbilical cord inside the tube.

The *Cleaning/Gauging System* was designed to remove deposits from the internal surface of the SG tube and to verify the absence of inner geometric deformations of the SG tube . Another *Hydraulic System* feeds a water stream into the SG tube, to push the brushes or the gauging device. The *Brushing Device*, shown in Figure 3.5-6, is a component of the *Cleaning and Gauging System.* In the *Gauging Module Configuration*, the Brushing Device is replaced by a calibrated ball.

The IRIS SG modules have helical-tube bundles, akin to the SPX SG. However, IRIS SGs should be easier to inspect because:

- The IRIS tube length is about 32 meters vs ~100 meter length of the SPX tubes;
- IRIS has a smaller number of bends in the tube adjacent to the tube sheets (4 to 5 bends vs 10 bends in SPX);

- The IRIS tube bends have a higher radii of curvature (6 to 8D) vs 5D in SPX;
- The IRIS tubes are one piece with no intermediate welds vs. the 6-piece welded SPX tubes.

On the other hand, the IRIS SG arrangement presents the following disadvantages compared to the SPX:

- The IRIS tube ID of 14.5 mm is smaller than the 19.8 mm ID tubes of the SPX (reduced, however, to 17.8 mm at the welds and bends);
- The minimum tube coil diameter for the inner-most IRIS tubes is 640 mm vs 1170 mm in SPX.

Based on the experience with the SPX helical bundle SG tube inspection, similar US and EC probes can be used in IRIS. US and EC probes are available for tubes with ID as small as 12 mm. Also, the shorter tube length, the reduced number of bends, and the fact that there are no reduced area welds in the tubes offset the effect of the smaller tubing size in IRIS. Thus, the SPX SG In-Service Inspection System is intended to be used for inspecting the IRIS SG tubes, with only the modifications to the probe assembly required by the IRIS specific header geometry, which features vertical tube sheets on the sides of the headers.
The scheme of a possible mechanism for introducing probes into the tubes is shown in Figure 3.5-7 and details A and B.

The probe assembly consists of three concentric cylinders, complete with stepping motors and gears, which function to place the probe-carrying tube end/tube sheet coupling device exactly on any specific tube hole as shown in Fig 3.5-7 and detail drawing A. The outer cylinder is bolted on the steam nozzle, whereas the intermediate cylinder is rotated and locked in position manually. The inner cylinder, driven by a motor, carries the coupling device one tube hole further, while a second motor coupled to a pinion gear allows the coupling device to reach each row, as shown in detail drawing B. The probe and attached umbilical cord, provided with uniformly spaced beads which match the tube ID, travel along the SG tube and are pushed by water flow provided through the insertion mechanism. The water is returned to the inspection system, via a connection made to the lower header nozzle.

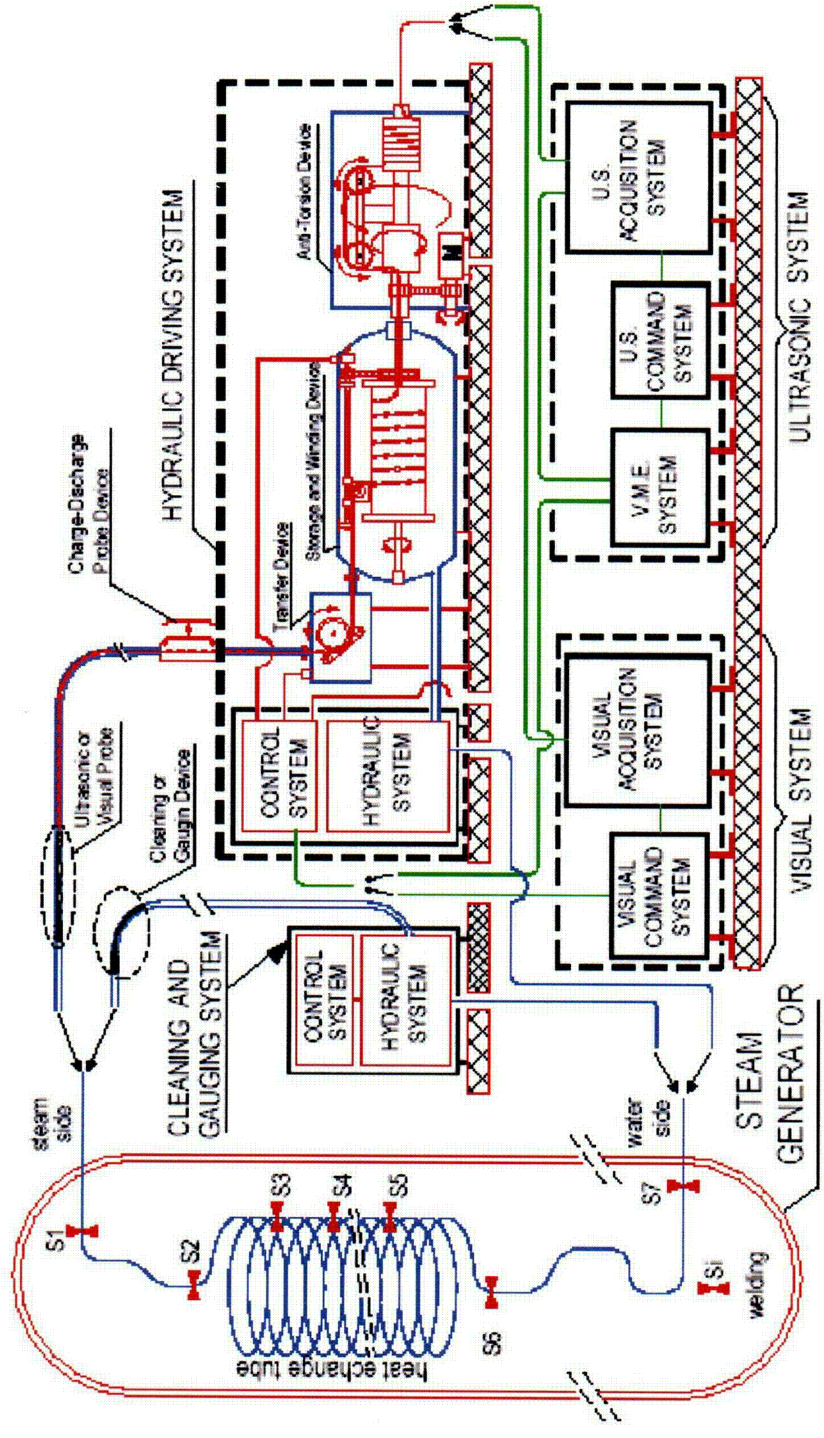

Figure 3.5-4 Helical Coil Steam Generator Inspection System

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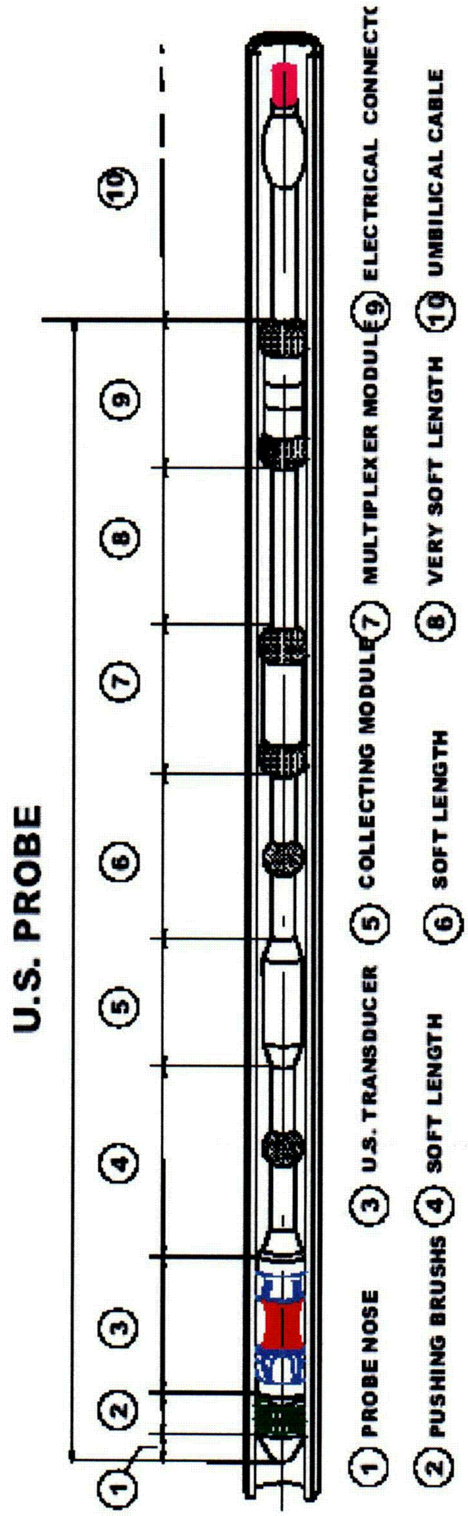

Figure 3.5-5 Schematic of the Ultrasonic Probe

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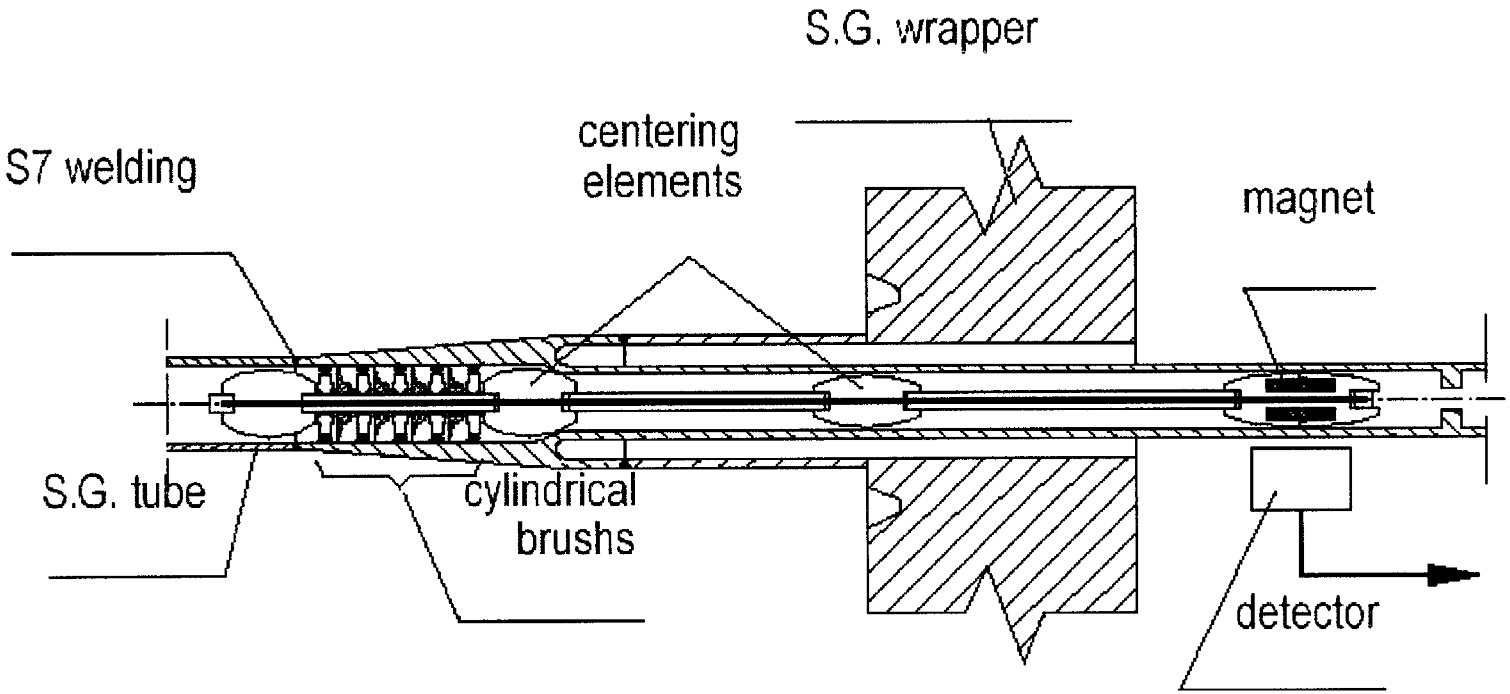
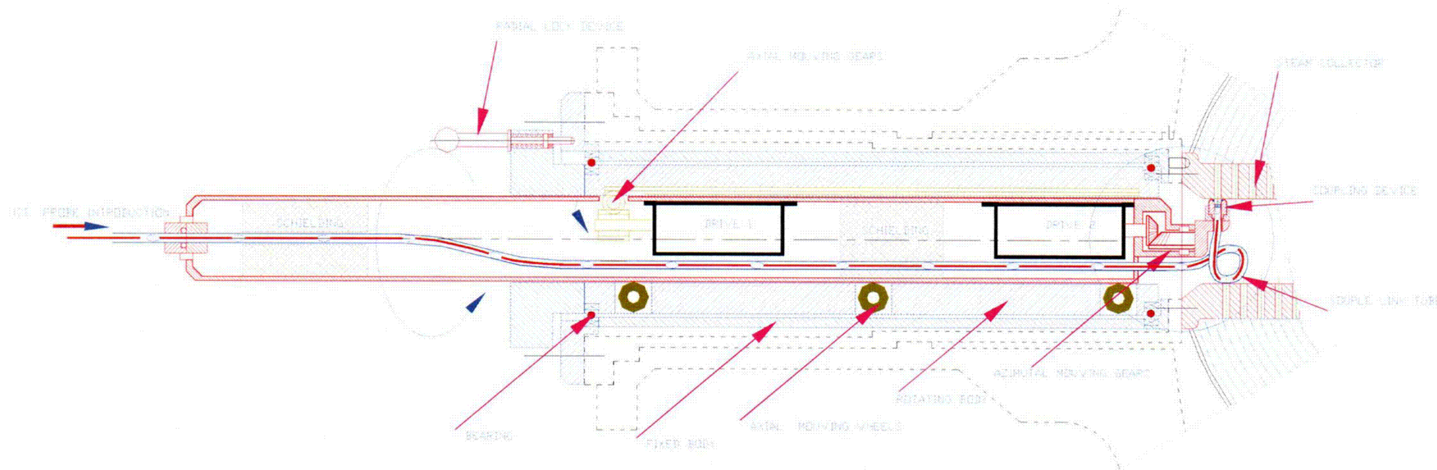
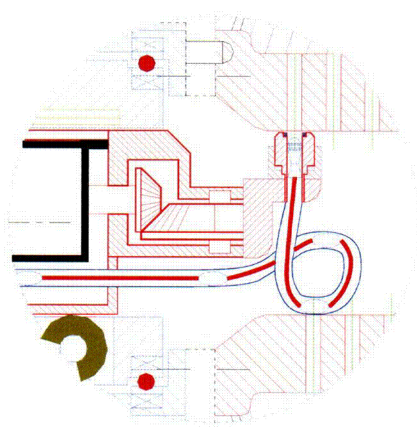
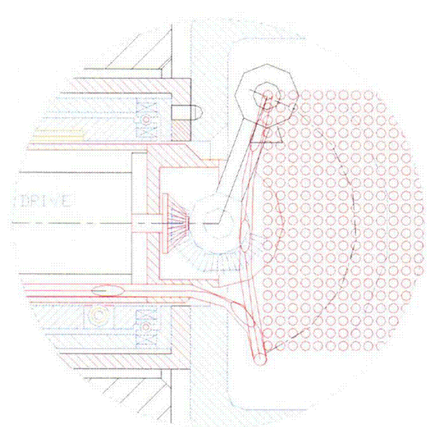


Figure 3.5-6 Brushing Device





Detail "A"



Detail "B"

Figure 3.5-7: Scheme of the ISI probe assembly bolted on the steam nozzle

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3.6 PRESSURIZER

The IRIS pressurizer, a principal component of the RCS pressure control system, is integrated into the upper head of the reactor vessel, where liquid and vapor are maintained in equilibrium saturated conditions to control system pressure. The pressurizer region is defined by an insulated, inverted top-hat structure (see Figure 3.3-4) that divides the circulating reactor coolant flow path from the saturated pressurizer water, as shown in Figure 3.3-10. Figure 3.6-2 shows the basic configuration of the pressurizer and summarizes design and operating parameters.

This structure includes a closed cell insulation to minimize the heat transfer between the hotter pressurizer fluid and the subcooled water in the primary water circulating flow path. Heater rods are located in the bottom portion of the inverted top-hat. The bottom portion of this inverted top-hat contains holes to allow water insurge and outsurge to/from the pressurizer region. These surge holes are located just below the heater rods so that insurge fluid flows up along the heater elements. An auxiliary spray nozzle and two nozzles for connecting the safety and depressurization valve inlet headers are located in the top head.

By utilizing the upper head region of the reactor vessel, the IRIS pressurizer provides a very large water and steam volume, as compared to plants with a traditional, separate, pressurizer vessel. The IRIS pressurizer has a total volume of 71.41 m^3 (2522 ft^3), which at hot full power conditions includes a steam volume of ~49 m^3 (1730 ft^3). This steam volume is about 1.6 times bigger than the AP1000 pressurizer steam space, while IRIS has ~1/3 the core power. This large steam volume to power ratio contributes to the fact that IRIS does not require the use of a pressurizer spray function to prevent the pressurizer safety valves from lifting for any design basis heatup transients. However, the implementation of a spray system is still considered as a design option, depending on the evolution of the design.

During normal transient events, pressure increases caused by insurges are controlled/ mitigated by the large pressurizer volume, such that the high pressure reactor trip setpoint is not reached. During pressure decreases (outsurges), water-to-steam flashing and automatic heater operation keep the pressure above the low pressure reactor trip setpoint. The heaters are also energized on the high water level during insurges to heat the subcooled surge water entering the pressurizer from the reactor coolant loop. A screen at the surge line nozzle, as well as baffles (heater support plates) in the lower section of the pressurizer, prevent cold insurge water from flowing directly to the steam/water interface. The baffles and screen also assist in mixing.

The two pressurizer safety valves are of the totally enclosed pop type. They are spring loaded, self actuated with backpressure compensation, and designed to meet the requirements of the ASME Boiler and Pressure Vessel Code, Section III. Valve set pressure is 2485 psig. Their combined capacity is based on not exceeding the RCS maximum pressure limit during the Condition II loss of load transient.

Operation of the IRIS Passive LOCA Mitigation Safety System includes automatic depressurization of the RCS (see Section 4.3, Automatic Depressurization System). This is accomplished in part by the automatically controlled depressurization valves mounted on the pressurizer, consisting of a parallel set of two valves in series.

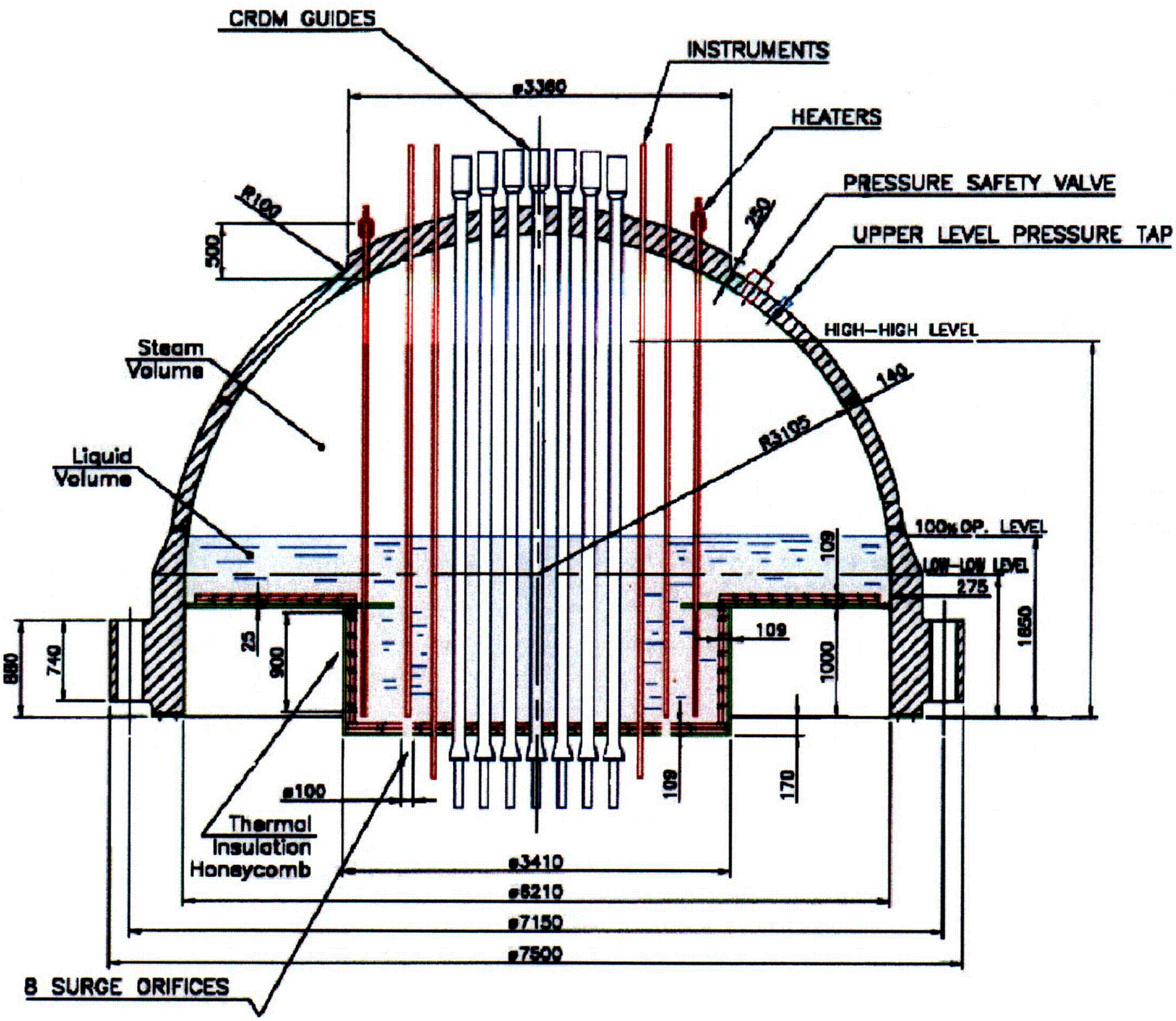

Figure 3.6-2 IRIS Pressurizer simplified drawing

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4.0 NUCLEAR FLUID SYSTEMS

4.1 STEAM GENERATOR SYSTEM

The Steam Generator System (SGS) comprises the safety related portions of the main steam and feed water systems and associated piping connections such as the connections to the Emergency Heat Removal System (EHRS) and connections from the start-up feed water system (SFWS).

4.1.1 System Functions

The SGS performs the following main functions:

- Transports feed water and steam to and from the steam generators during normal operation;
- Isolates the main steam and feed water lines;
- Transports steam and feed water between the steam generators and the Emergency Heat Removal System (EHRS) following EHRS actuation;
- Transports start-up feed water to and steam from the steam generators during normal shutdown operation decay heat removal;
- Provides access to the steam generators for tube inspection, cleaning, and repair;
- Provides steam generator tube rupture (SGTR) mitigation;

4.1.2 System Description and Operation

The SGS consists of the safety grade portion of the feed water and steam lines that connect to the steam generators, the valves that are used to isolate these lines, and the instrumentation and appurtenances related to the steam generator function. IRIS utilizes four feed water and four steam line penetrations through the containment, with each of these lines branching inside containment and connects to two (of eight) steam generator modules. The connections to each of the SG's are made to the individual steam and feed water header nozzles that penetrate the integral reactor vessel. A simplified sketch is provided in Figure 4.1-1.

- <u>Main Steam Line</u> - Each of the four main steam lines (each line collects steam from two integral SG modules) has two series main steam isolation valves (MSIVs). The MSIVs have a bypass line and valve to permit warming of the main steam lines prior to startup, when the MSIVs are closed. The MSIVs are located in the main steam and feed line penetration area adjacent to the containment structure. A main steam drain line is provided to collect and discharge condensate from each of the four steam lines.

- Main Feedwater Line - Each of the four main feedwater lines has two series main feed water isolation valves (MFIVs). The MFIV's are located in the main steam and feed line penetration area adjacent to the containment structure. Each feed line also contains a check-valve inside containment. The main feed water flow and control is provided by the non-safety related main feed water system which is located in the turbine building. The main feedwater control provides flow control over the range of 3 percent to 100 percent power.

- Startup Feedwater Line - Four startup feedwater (SFW) lines are provided which connect to each of the four main feed water lines. Each of the startup feed water lines contains a control valve and one isolation valve located in the penetration area. During normal startups and shutdowns each of the four control valves automatically control the start-up feed water flow to the SG modules, over the range of 0 percent to 5 percent power. This automatic flow control also functions in post reactor trip situations where the startup feedwater is actuated to remove heat from the reactor coolant system.

Operation

The operation of the SGS for the various phases of the reactor plant operation is described in the following paragraphs.

Cold Startup

The primary reactor water is heated using the reactor coolant pump heat and heatup equipment. When the reactor coolant water reach a temperature of approximately 400 F (250 psia steam pressure), startup feed water is added to the eight SG modules in order to produce steam for plant warm-up and to begin turbine roll. [

]$^{(a,c)}$ When the primary water temperature has increased to the No-Load Temperature, the auxiliary heat source is isolated and the steam bypass valve(s) are opened. [

]$^{(a,c)}$ Saturated steam is produced in this mode of operation. When the steam generators approach five percent of design feedwater flow, the capacity of the startup feed pumps will be exceeded and one of the main feedwater pumps is placed in service. At this time each of the four main feed water control valves regulates flow to its SG pair to match core power while maintaining the proper steam superheat, while the condenser steam dump valve(s) maintains proper steam pressure.

Hot Standby

Hot standby conditions are defined as having the reactor subcritical (at zero power) and no-load primary temperature and steam pressure and temperature. A small amount of steam is relieved automatically to maintain the steam pressure by operation of the condenser steam dump system in its pressure control mode. A small amount of feedwater flow is delivered by the Startup Feedwater System with flow controlled by the SG level sensed in each SG pair.

Normal Operation

During normal operation, the SGS transfers main feedwater to the steam generators and steam to the Main Steam System, which in turn transfers steam to the turbine. Normal operation is fully automatic with the feed water controlled by maintaining the desired steam superheat, the proper turbine inlet steam pressure, while matching the reactor power.

Cold Shutdown

Feed water delivery to the SG's is required when bringing the plant to a cold shutdown in order to maintain steam generator water inventory while decreasing steam pressure as heat is being removed. The reactor core decay heat and the primary system sensible heat is removed by steam release via the steam dump system to the condenser. The rate of feed water flow during cooldown is limited to 5 percent of full flow, and the start-up feedwater subsystem is used to supply the feed water makeup to the steam generators while the main feedwater pumps are shutdown. After approximately four hours, the steam pressure will be approximately 125 psia and steam dump may be discontinued after the Normal Residual Heat Removal System is aligned to provide direct RCS cooling.

Abnormal Operations

If the condenser is unavailable, the reactor is tripped and decay heat is removed from the steam generators by condensing steam using the Emergency Heat Removal System (See Section 4.2). The EHRS is comprised of four subsystems and the system is designed such that two of the four subsystems are actuated automatically (note that only one of four subsystems is required to match the core decay heat at shutdown). [

] (a,c) Additional EHRS subsystems can be initiated, as required, to continue the cooldown of the primary system in a controlled manner. Note that during operation of the EHRS, the steam and feed water isolation valves are closed and the steam pressure in the idle SGs will increase to correspond to the primary system water temperature. This is acceptable since the SG modules and their related piping and valves are designed for full primary system pressure and temperature. [

] (a,c)

If the main feedwater is lost and the reactor trips, the startup feedwater pumps will start automatically to provide SG feedwater. [

] (a,c) The condenser steam dump valves are used to remove heat from the SG's in their temperature control mode. If however, both main and startup feedwater pumps fail, the core decay heat will be automatically removed by the EHRS.

Accident Operations

The SG main steam isolation valves and feed water isolation valves automatically close on low steam generator steam pressure, thus assuring that the SGs are isolated following a steam line or feed line break accident. This action, combined with the small water inventory in the once-through steam generators, limits the potential mass/energy input to the containment, and limits the RCS cooldown rate following a steam line break.

Following a postulated steam generator tube rupture, the steam and feed line isolation valves are closed to terminate the loss of mass from the reactor vessel. Because the IRIS steam and feed water piping and valves are designed for full reactor coolant pressure, the affected steam generator and piping fill and pressurize to RCS pressure thus stopping the loss of coolant before core cooling is affected.

The main feedwater isolation valves in combination with the main feedwater control valves and the trip of the main feedwater pumps ensure that main feedwater flow can be terminated to prevent excessive RCS cooldown or SG overfill.

Note that the SFW pumps and the opening of the steam dump valves to the condenser are not safety grade and are not required to mitigate any accidents. The passive EHRS in conjunction with the RWST water provides decay heat removal following accidents. The EHRS also provides containment depressurization following postulated LOCA's.

4.1.3 SGS System Parameters

Steam flow, kg/s (lb/hr)per SG	62.85 (0.5 x10^6)
Total Steam flow, kg/s (lb/hr)	502.8(3.99x10^6)
Feedwater temperature, °C (°F)	223.9 (435.02)
Bundle Steam Pressure, MPa (psia)	5.8 (841)
Bundle Steam Temperature, °C (°F)	317(602.6)
Design Pressure, MPa (psia)	17.235 (2,500)

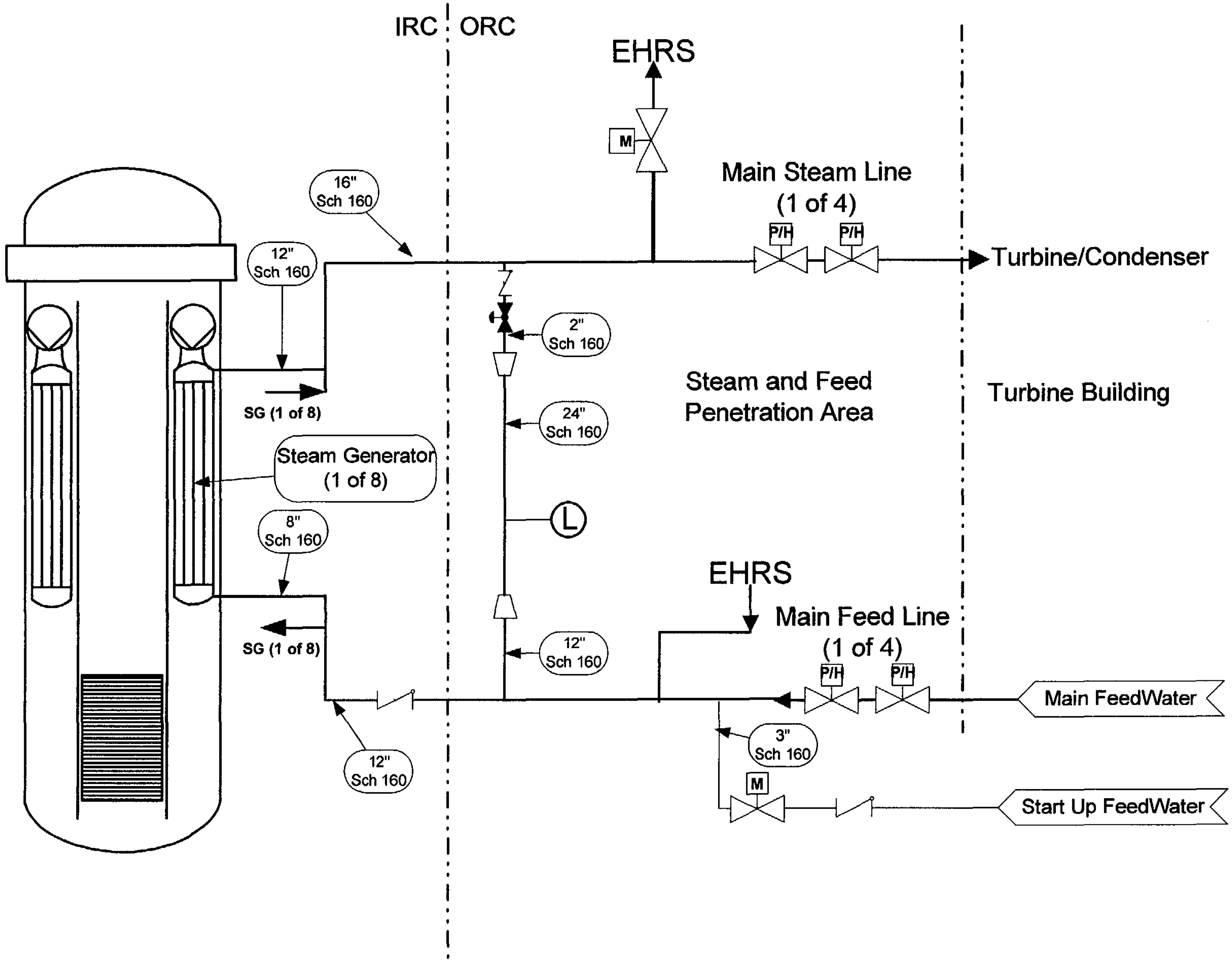

Figure 4.1-1 IRIS Steam Generator System Simplified Sketch

4.2 EMERGENCY HEAT REMOVAL SYSTEM

4.2.1 System Functions

The Emergency Heat Removal System (EHRS) is designed to perform the following major functions:

- Emergency Core Decay Heat Removal - Transfers core decay heat and sensible heat from the reactor coolant to the environment during transients, accidents, or whenever the normal heat removal paths are lost. This heat removal function is available during all plant operating conditions except refueling operations when the vessel head is removed. See Section 4.2.2 for additional information about heat removal during plant shutdown and refueling.

- Emergency Reactor Coolant System Water Inventory Control (LOCA Mitigation) - The EHRS condenses steam within the reactor vessel, following events that result in reduced RCS water inventory. This function reduces the coolant loss following a LOCA by returning the condensed steam to the core and by reducing the reactor vessel pressure. Thus, the EHRS ensures that sufficient water is retained within the vessel to provide adequate core cooling for the complete range of LOCAs, up to and including the double-ended rupture of the largest auxiliary primary pipe connection to the vessel.

- Emergency Containment Pressure Reduction.-.The EHRS reduces the containment pressure following loss of coolant accidents by minimizing the mass and energy release from the reactor vessel into the containment during blowdown and long-term cooling following a LOCA. The EHRS also reduces containment pressure by condensing steam within the reactor vessel (see above), causing the LOCA break flow to reverse.

These protective functions provide safety-related plant protection, and therefore the EHRS is designed as a safety-related system. This includes the use of safety class equipment, redundancy to deal with single failures, environmental qualification, and protection from external events. The EHRS requires a one-time alignment of valves to actuate and perform its functions.

The reliability of the EHRS must be sufficient to meet plant design requirements for protection of the public and to reduce the financial risk to the plant owner by minimizing or preventing damage to major plant equipment.

The RWST, which provides the EHRS heat sink, is also used to store the water required for filling the refueling cavity during plant refueling operations.

4.2.2 System Design Description and Operation

The EHRS consists of four subsystems each having an EHRS heat exchanger, a steam generator water addition tank, and associated valves, piping, and instrumentation. Each subsystem is connected to one of the four steam generator steam and feed water lines in the penetration area outside the containment, and the heat exchangers are submerged within the Refueling Water Storage Tank, which is located within the auxiliary building. A simplified sketch of the EHRS is shown in Figure 4.2-1.

Each EHRS subsystem forms a natural circulation flow path from a pair of steam generators to its associated EHRS heat exchanger, that allows steam to flow from the steam generator to the heat exchanger where it is condensed, with the condensate returned to the steam generator through the normal feed water piping. Actuation of the EHRS is accomplished by opening one of two normally closed, fail-open isolation valves in the EHRS heat exchanger return line. Of course, isolation of the main steam and feed isolation valves is included as part of the EHRS actuation. A steam generator water addition tank is provided for each SG pair/EHRS subsystem, which contains sufficient water to refill a dry SG and to provide makeup during EHRS operation to compensate for leakage. The EHRS heat exchangers transfer heat to the RWST water, which is heated and eventually boils, with the steam produced vented to atmosphere. The RWST contains sufficient water to maintain core decay heat removal for seven days, and the tank is provided with connections for both on-site makeup water addition and for addition of water from off-site sources.

Emergency Core Decay Heat Removal - During a plant transient that results in a reactor trip, core heat removal will normally be performed by the steam generators with feed water supplied by the Startup Feed Water System, and the steam is directed to the condenser via the steam dump valves. If this heat removal path is not available, then the EHRS provides emergency core decay heat removal. The EHRS consists of four independent subsystems, each one of which is designed to provide sufficient heat removal to match core decay heat. Each subsystem is connected to the steam discharge and feed water supply lines from a pair of SGs. The EHRS heat exchangers are located above the SGs and the piping provides a natural circulation flow path for the flow of steam to the HX and for the gravity drain of the condensed steam back to the SG, with the RWST water providing the heat sink. The RWST contains sufficient water to provide core decay heat removal for seven days with no need for operator action to replenish the tank water. This water is heated and then steaming occurs directly to the environment. Note that alignment of the normal residual heat removal system to cool the RWST will prevent steaming and extend the time that the RWST water is available.

Following postulated secondary side accidents, the EHRS heat removal function is employed as the safety grade means of decay heat removal. Following a steam or feed water line break, or a steam generator tube rupture, the EHRS subsystems attached to the unaffected steam and feed lines are available for decay heat removal, and function in a manner similar to that previously described. Following a loss of coolant accident where the water inventory in the reactor vessel is decreased, the EHRS heat removal function condenses primary side steam on any exposed steam generator tube area.

Emergency Reactor Coolant Inventory Control - Normally the Chemical and Volume Control System (CVCS) provides makeup water to the RCS. However, following a design basis LOCA, the CVCS makeup capacity is insufficient and the system is typically considered unavailable for maintaining inventory for core cooling. For these events the ability of the EHRS to remove heat directly from the reactor vessel by condensing steam on the exposed steam generator tube surfaces provides emergency reactor coolant inventory control. By condensing steam within the vessel, the EHRS reduces the in-vessel pressure thus decreasing the break flow rate.

Additionally, the condensed steam (water) remains in the vessel and remains available for core cooling. This EHRS function has been shown to maintain sufficient water inventory to keep the core covered following breaks of the largest piping connections (up to 4 inches in diameter) to the reactor vessel. In the longer term of accident recovery, the EHRS heat removal acts to greatly minimize the mass loss from the vessel through the break, such that the core will remain covered for several days, with no credit for makeup from any other sources, like the long term gravity makeup system (See LGMS, Section 4.5).

Emergency Containment Pressure Reduction - The EHRS heat removal function following a LOCA event also acts to provide emergency containment pressure reduction. Normally heat is removed from containment by an active HVAC system that cools the containment atmosphere and transfers heat to the plant cooling water systems. Following a LOCA event the active HVAC capacity is insufficient and the system is typically considered unavailable for maintaining cooling to effectively reduce containment pressure. The EHRS, by condensing steam inside the reactor vessel, reduces the vessel pressure below the containment pressure. This stops the loss of coolant, and actually results in drawing containment atmosphere into the reactor vessel through the break. This results in a decrease in the containment pressure to a low, quasi-steady state pressure, that corresponds to a balance between the EHRS heat removal capability, which decreases with decreasing pressure and temperature, and the core decay heat, which decreases with time.

System Operation

The operation of the EHRS for the various phases of the reactor plant operation is described in the following paragraphs.

Normal Operation

During normal operation, the four EHRS heat exchangers and their associated discharge piping and steam generator makeup tanks are maintained full of cold water at the secondary side steam pressure. The EHRS heat exchanger discharge lines connected to the feed water line to the steam generators are isolated by two normally closed, fail open, parallel isolation valves. The EHRS heat exchangers are submerged in the RWST water, which acts to condense steam to keep both the heat exchangers and their discharge flow paths full. The EHRS heat exchanger inlet lines connected to their corresponding main steam lines are normally open and filled with steam.

The RWST is maintained full of water during normal operation (by the CVCS, Section 4.9), to ensure that the EHRS heat sink is maintained.

Shutdown Operation

Following a plant shutdown or trip, EHRS actuation will not occur as long as the normal RCS heat removal method (SGs-SFW-steam dump) is maintained. If heat removal from all four pairs of steam generators is lost, due to loss of main feed and startup feed water capability or loss of all AC power, or failure to actuate the steam dump system, the EHRS heat removal function is automatically actuated by the initiation of two of the four EHRS subsystems. The operator can sequentially start the remaining EHRS subsystems to cooldown the RCS to safe shutdown conditions.

LOCA Mitigation

Following a LOCA event, all four EHRS subsystems receive an actuation signal. The EHRS actuation includes closing the feed water isolation valves, followed by the closing of the main steam isolation valves, followed by the opening of the EHRS isolation valves; for each of the four independent EHRS subsystems. The steam generator makeup tanks add water to their associated steam generators, as needed, until the tank water level and steam generator effective level in the tubes are equal. The large surface area of the steam generators is then available to remove heat from the primary coolant. As the reactor vessel level decreases, more of the steam generator surface is available to condense steam, and the EHRS acts to reduce the vessel pressure while retaining the condensed steam for core cooling.

The EHRS heat exchangers transfer heat to the RWST water and the steam condensed in the heat exchanger tubes flows by gravity back to the steam generators via the associated steam line. The RWST water is heated by the EHRS heat removal function and increases in temperature until it boils after approximately 5 hours of operation. The RWST then steams to the environment and its inventory slowly decreases. If AC electrical power is available, the normal residual heat removal system (NRHRS) can be aligned to cool the RWST water, which will prevent boiling and extend the time before replenishment is required. Alternatively, water can be added to the tank to replenish the water inventory using the normally available plant water systems. If no AC electrical power is available, the RWST water inventory is sufficient for seven days of operation, after which the water inventory will need to be replenished. Piping connections to facilitate water replenishment from off-site sources (e.g., fire trucks) are provided.

A range of break sizes and locations are analyzed to verify the adequacy of the EHRS design. The two limiting LOCA cases for IRIS are:

- The rupture of the largest vessel connection (4-in., Sch 160) near the steam generator steam discharge nozzle (high on the reactor vessel),

- The rupture of a direct vessel injection line connection (a 2-in., Sch 160 pipe of the LGMS described below) located 2 meters above the top of the active fuel (closest to the core region).

For these two scenarios, the EHRS successfully removes heat and depressurizes the containment. However, for lower elevation breaks (e.g. at the DVI line elevation) a means of assisting the pressure equalization between the vessel and containment, other than the break itself, is required to avoid a short-term partial core uncovery. This means is the ADS system discussed in the following Section 4.3.

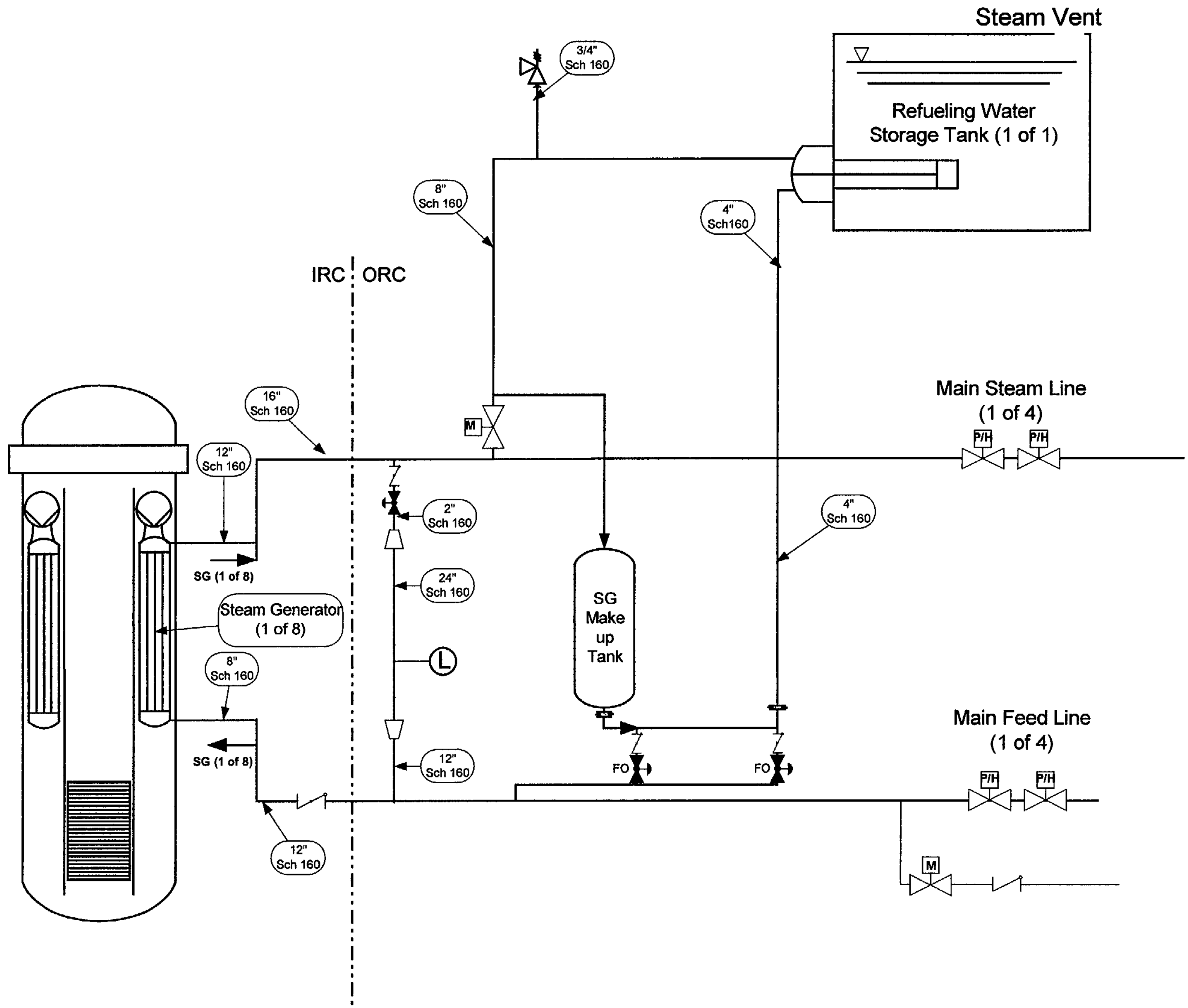

Figure 4.2-1 IRIS Emergency Heat Removal System (EHRS) Simplified Sketch

4.3 AUTOMATIC DEPRESSURIZATION SYSTEM

4.3.1 System Functions

The Automatic Depressurization System (ADS) is designed to perform the following major functions:

- Emergency Depressurization – The ADS assist the EHRS in depressurizing the reactor coolant system following certain postulated LOCAs, such that the reactor vessel pressure is equalized with the containment pressure, before any core uncovery occurs.
- Normal Operation Depressurization – The ADS is available for the operator to, manually and in a controlled manner, reduce or limit the reactor vessel pressure.

These protective functions provide safety-related plant protection, and therefore the ADS is designed as a safety-related system. This includes the use of safety class equipment, redundancy to deal with single failures, environmental qualification, and protection from external events.

4.3.2 System Description and Operation

The Automatic Depressurization System consists of a single 4-inch piping connection from the reactor vessel head (pressurizer region) that is normally closed. This piping branches into two parallel lines each of which contains a pair of 4-inch, normally closed, motor-operated valves. The piping downstream of the isolation valves provides a single flow path to one of the Containment Pressure Suppression System (CPSS) water tanks. The piping terminates in a sparger which assist in condensing any released steam in a controlled manner. The isolation valves are powered by DC electrical motors and are capable of being partially opened in their manual operating mode. A simplified sketch of the EHRS is shown in Figure 4.3-1.

System Operation

The operation of the ADS for the various phases of the reactor plant operation is described in the following paragraphs.

Normal Operation

The ADS is normally isolated and does not have any normal operation functions.

Transient Overpressurization

The ADS is available for manual use by the operator to reduce the reactor vessel pressure in a controlled manner.

LOCA Mitigation

Following an event that results in a LOCA Mitigation signal, combined with a confirmatory Hi-Hi containment pressure signal, the ADS is automatically actuated to assist the EHRS in reducing the reactor vessel pressure. When the vessel pressure matches the containment pressure, the break flow is reduced such that sufficient water inventory remains to keep the core covered.

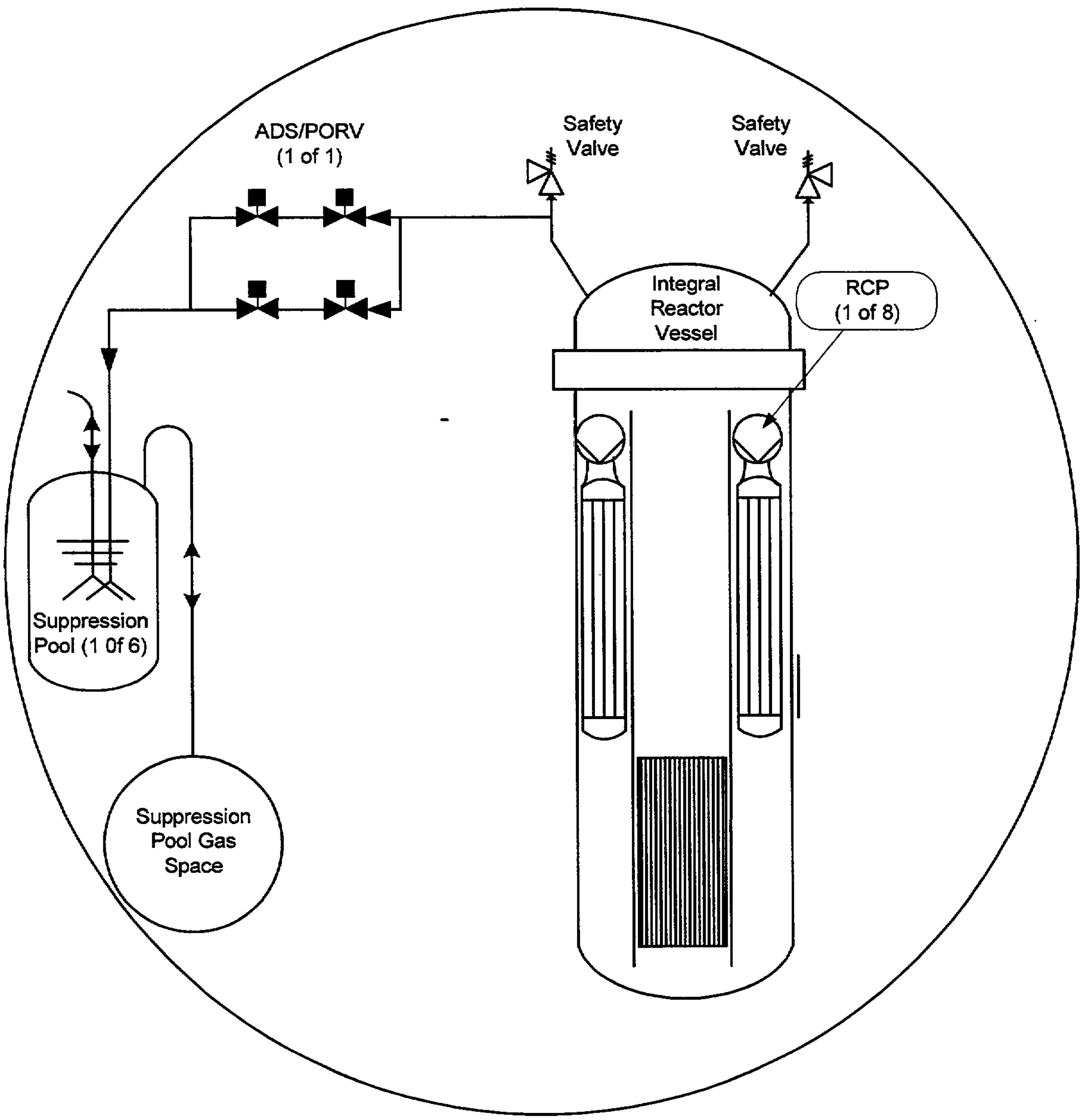

Figure 4.3-1 IRIS ADS Simplified Sketch

4.4 EMERGENCY BORATION SYSTEM

4.4.1 System Functions

The EBS provides limited RCS makeup and provides sufficient borated water for core reactivity control during transients or accidents when the normal RCS makeup supply from the Chemical and Volume Control System (CVCS) is not available or is insufficient.

The EBS protective functions provide safety-related plant protection, and therefore the EBS is designed as a safety-related system. This includes the use of safety class equipment, redundancy to deal with single failures, environmental qualification, and protection from external events.

4.4.2 System Description and Operation

There are two Emergency Boration Tanks (EBTs) located inside the containment at an elevation above the gravity makeup addition piping (Direct Vessel Injection, DVI, line) connections to the reactor vessel. Each tank, which is filled with borated water, has an injection line connected to one of the two direct vessel injection lines into the reactor vessel downcomer. Each tank also has a pressure balance line from the upper portion of the reactor vessel to the top of the EBT. The isolation valve in the pressure balance line is normally open to keep the EBS full of water at RCS pressure. These tanks are sized to provide sufficient borated water to enable the reactor to go to cold shutdown conditions following all postulated non-LOCA events. A simplified sketch of the EBS is shown in Figure 4.4-1.

Normal Operation
During normal operation, the two EBTs are maintained full of cold, borated water at RCS pressure. The EBS discharge line from the bottom of each EBT to the reactor vessel is isolated by two normally closed, fail open, parallel isolation valves. A normally open line from the reactor vessel is connected to the top of each EBT. These tanks are monitored to ensure that their boron concentration is within technical specifications.

Shutdown Operation
Following a plant shutdown or trip, EBS actuation does not occur as long as normal RCS inventory control (CVCS) is maintained. Should normal RCS makeup be unavailable following a plant trip, due to failure of the makeup pumps or loss of AC power, the EBT discharge line isolation valves are automatically opened on low reactor vessel level, and the cold borated water within the tank naturally circulates and mixes with the reactor coolant. This also provides a limited amount of makeup to maintain RCS inventory. If the normal CVCS makeup cannot be restored, the EBS continues to add water, as needed, to the RCS. The EBTs contain sufficient borated water to provide boration of the RCS water for a cold shutdown from hot conditions. The large IRIS reactor vessel contains sufficient water inventory to compensate for shrinkage of the reactor coolant during cooldown without impeding single-phase natural circulation heat removal. Thus, the EBTs together with the large reactor vessel inventory provides ample time for restoration of normal makeup and heat removal functions.

Non-LOCA Events

Following non-LOCA events, where the primary system is cooled down as part of the transient response or eventual recovery, the cooldown reduces the core shutdown margin due to the negative moderator temperature coefficient, and a potential return to power is possible. EBS actuation, following these events, provides sufficient borated water to prevent any reactivity transient and maintain the core in a shut down condition, assuming that the most reactive RCCA is stuck in its fully withdrawn position.

During a steam generator tube rupture, the affected SG steam and feed water lines are isolated to terminate and limit the loss of RCS inventory. Should the CVCS and normal means of core decay heat removal not be available, the EBS injection maintains the RCS inventory and provides RCS boration, while the EHRS provides heat removal.

LOCA Mitigation

Following a LOCA the EBS will be actuated and will perform the RCS makeup and boration function. If/when the RCS water inventory decreases below the EBS pressure balance line RV connection, the tanks will drain into the RV. Thus the EBS provide a limited source of water injection and their borated water also helps to ensure that the core remains in a shutdown condition following LOCA events and until long-term gravity makeup to the reactor vessel occurs.

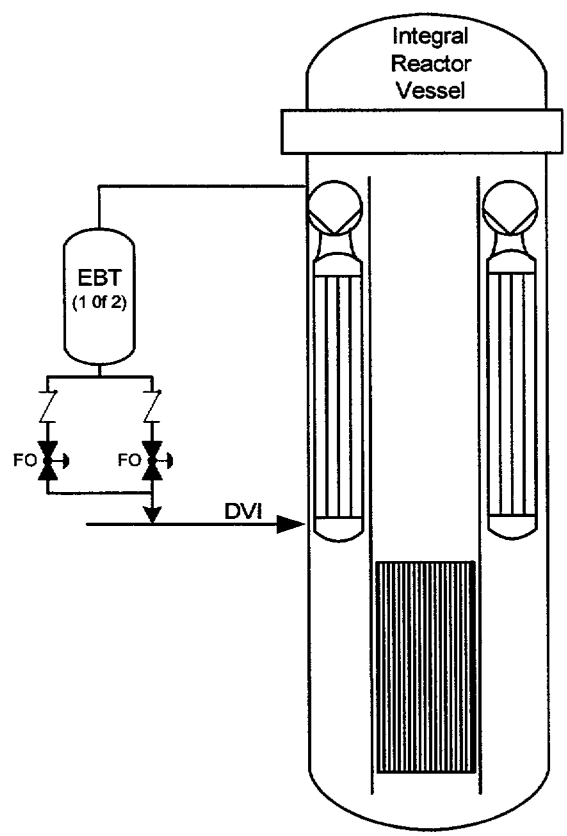

Figure 4.4-1 IRIS EBS Simplified Sketch

4.5 LONG-TERM GRAVITY MAKEUP SYSTEM

4.5.1 System Functions

The Long-term Gravity Makeup System (LGMS) is designed to perform the following major functions:

- Gravity Makeup to the Reactor Coolant System – Provides the means for the addition of water to the reactor vessel from the Containment Pressure Suppression System (CPSS) in-containment stored water (See Section 4.6) or from the containment flood-up water volume using only gravity. This ensures that the reactor core can remain covered for an extended time following reactor depressurization and/or post-accident floodup conditions resulting from postulated LOCA events.

  The LGMS provides screens, etc. to prevent debris in the in-containment flood-up water from hindering or blocking the gravity makeup flow from providing adequate core cooling.

- Containment Sump pH Control - Provides chemical addition to the containment post-LOCA floodup water to increase the retention of radioactive nuclides and to prevent equipment corrosion.

The LGMS provides safety-related plant protection, and therefore it is designed as a safety-related system. This includes the use of safety class equipment, redundancy to deal with single failures, environmental qualification, and protection from external events.

4.5.2 System Description and Operation

The Long-term gravity Makeup System is comprised of two gravity injection flow paths from the CPSS water tanks, two flow paths from the containment flood-up area, two flow paths to ensure that the reactor vessel cavity can be flooded, and associated valves, piping, and instrumentation. A simplified sketch is shown in Figure 4.5-1.

The LGMS can establish and maintain the core in a safe shutdown condition following a LOCA by delivering to the reactor coolant system, the borated water in the CPSS water tanks.

<u>Normal Operation</u>

The LGMS injection lines from both the CPSS water tanks and from the containment flood-up valve rooms are isolated during normal plant operations. The lines from the cavity are provided with diverse valve types to assure that they function reliably to open, providing gravity makeup flow paths to the reactor vessel, and for flooding the reactor vessel cavity.

<u>LOCA Mitigation</u>

Once the heat removal via the EHRS using the steam generator heat transfer surface area to condense steam is initiated, the resulting loss of reactor vessel inventory also results in the initiation of the vessel depressurization. The LGMS isolation valves will automatically open providing multiple, diverse paths to initiate gravity makeup to maintain a long-term injection and core cooling capability, should it be required (Note that this function will not be required for a long time period (days) after the accident due to the cooling provided by the EHRS). The containment is designed such that most of the mass discharged from the RCS will drain into the containment flood-up areas and then into the reactor vessel cavity.

Containment Sump pH Control

Following a severe accident condition with a loss of coolant and high radiation levels in containment, the containment flood-up water pH is automatically adjusted by the presence of pH control dry chemicals that are stored in the two gravity makeup valve rooms. Each of these rooms also contain a containment recirculation screen areas. The pH adjustment chemicals are sufficient to maintain the pH of the reactor and containment flood-up water inventory between 7.0 and 9.5.

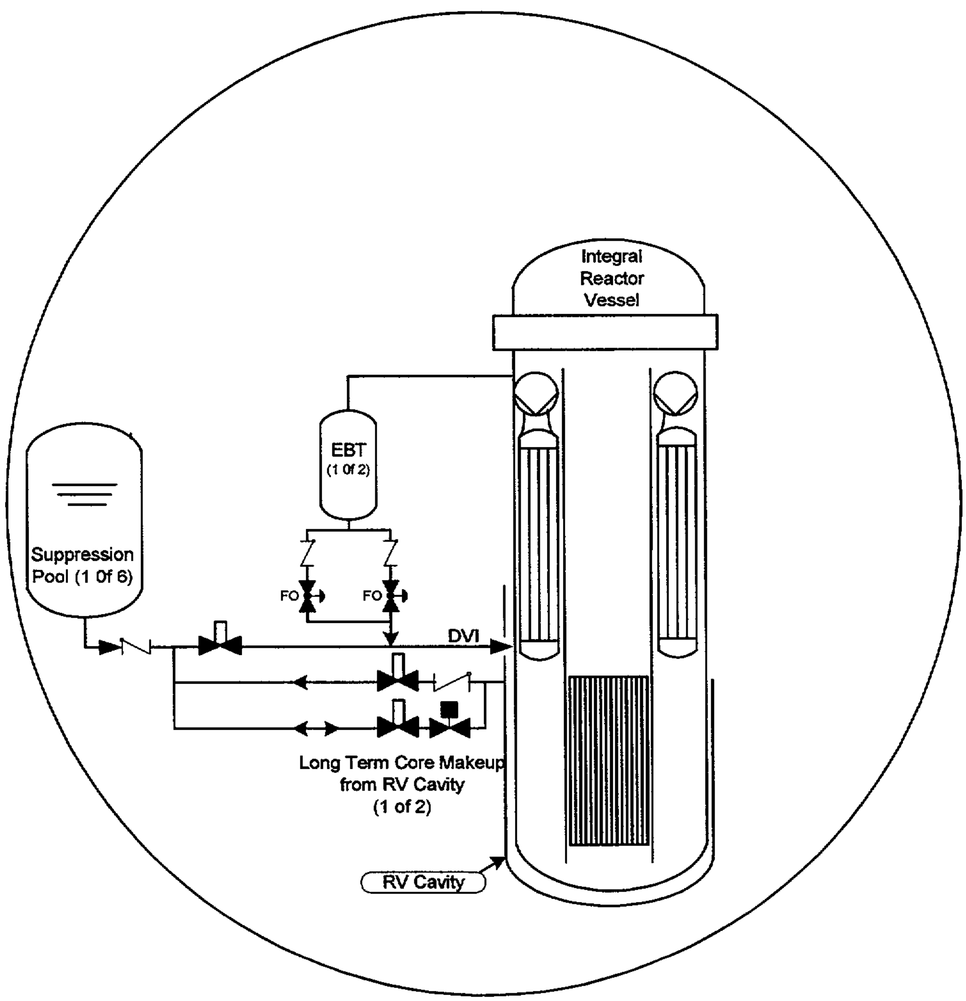

Figure 4.5-1 IRIS LGMS Simplified Sketch

4.6 CONTAINMENT PRESSURE SUPPRESSION SYSTEM

4.6.1 System Functions

The Containment Pressure Suppression System (CPSS) is designed to perform the following functions:

- Containment Pressure Control – Provides a means of condensing steam released into the containment and storing the containment atmosphere non-condensable gas such that the peak containment pressure following the worst postulated design basis event is less than the containment vessel design pressure;
- Long-term Makeup Water Supply – Provides a source of water that is available to drain by gravity into the reactor vessel when the reactor vessel internal pressure approaches containment pressure;
- Reactor Vessel Cavity Floodup – Provides a source of water sufficient to ensure that the reactor vessel cavity can be flooded with water;
- Long-term Makeup Water Boration – the CPSS stored water supply is sufficiently borated such that the core is maintained in a subcritical condition during all gravity makeup addition scenarios.

These protective functions provide safety-related plant protection, and therefore the CPSS is designed as a safety-related system. A Sketch of the system is provided in Figure 4.6-1.

4.6.2 System Description and Operation

The CPSS consists of 6 water tanks and a common tank for non-condensable gas storage. These tanks are located inside containment, and each suppression water tank is connected to the containment atmosphere through a vent pipe connected to a submerged sparger. Each water tank also has a pipe that connects from the top of the tank to the common gas storage tank. The CPSS is designed with the water tanks at a high elevation in the containment (above the direct vessel injection line connection to the reactor vessel) which allows the water in the CPSS to be available for gravity injection to the reactor coolant system through the LGMS lines. Due to space limitations within the containment, this results in the common gas storage tank being located at a lower elevation than the water tanks.

The water tanks employ a submerged sparger to ensure that the non-condensible gas and steam, which flows into the CPSS water tanks when the containment pressurizes, does not result in high mechanical loads. The spargers also ensure that the steam is effectively condensed in the CPSS water.

The ADS (Section 4.3) is also connected to a sparger in one of the CPSS water tank.

Normal Operation

During normal plant operations, the CPSS is maintained at ready to function state. The vent lines from the containment atmosphere into the water tanks contain no valves. The CPSS water inventory and its boric acid concentration is monitored and maintained using the CVCS. The water temperature is also monitored, and the water temperature can be reduced, if required, using the normal residual heat removal system (NRHRS).

Containment Pressurization Events

The CPSS is designed to limit the peak containment pressure, following the most limiting blowdown event, to <1.0 Mpa (130 psig), which is much less than the containment design pressure. The most limiting event for the release of mass and energy into the containment is a postulated break of a 4-inch diameter pipe connected to the reactor vessel. The rapid release of steam into the containment results in an increase in containment pressure, which forces containment non-condensable gas and steam into the CPSS water tanks. These gases are sparged into the water where steam is condensed and the non-condensable gas rises to the water surface. This gas then flows to the CPSS gas storage tank. As the containment pressure rises, this flow of air and steam continues until all/most the non-condensable gas that was originally in the containment is compressed in the storage tank. There is sufficient water in the CPSS water tanks to continue to condense the steam throughout the containment pressurization.

Following a LOCA, the IRIS EHRS removes heat from inside the reactor vessel by condensing steam on the large steam generator heat transfer surface. As the reactor vessel pressure decreases and containment pressure continues to increase due to the continued loss of mass and energy from the reactor, and as core decay heat decreases, the rate at which the EHRS condenses steam, matches and then exceeds the core steam generation rate. At this time the break flow from the reactor stops and then actually reverses, with steam flow occurring from the containment into reactor. This results in the containment (and reactor vessel) pressure decreasing. When the reactor pressure is equal to, or less than the containment pressure, the water in the CPSS is available to drain by gravity into the reactor vessel via the LGMS connections. This flow will initiate when/if the water level in the reactor is reduced and matches the water level in the CPSS water tanks. Because the CPSS water tanks and the LGMS connections are located above the reactor core, this water can drain completely to maintain core cooling and prevent uncovery.

Also, when the EHRS function reduces the reactor and containment pressures, the CPSS is available to ensure that the reactor vessel cavity is flooded. Because the CPSS gas storage tank was pressurized to approximately the peak containment pressure, the stored gas expands when the containment pressure decreases. This compressed gas acts to push the CPSS water up and out the vent lines into containment. This water flows into the reactor vessel cavity and helps ensure that the cavity always floods following a LOCA event. After a portion of the CPSS water has been pushed out of the tanks into containment, the sparger on the vent line partially uncovers and allows the remaining pressurized gas in the storage tank to vent back into the containment until they are at the same pressure.

Because the CPSS water is normally borated, its addition to the reactor vessel in response to a LOCA event helps to ensure that the reactor coolant is sufficiently borated to maintain the reactor in a shutdown condition, even when the reactor is cooled down.

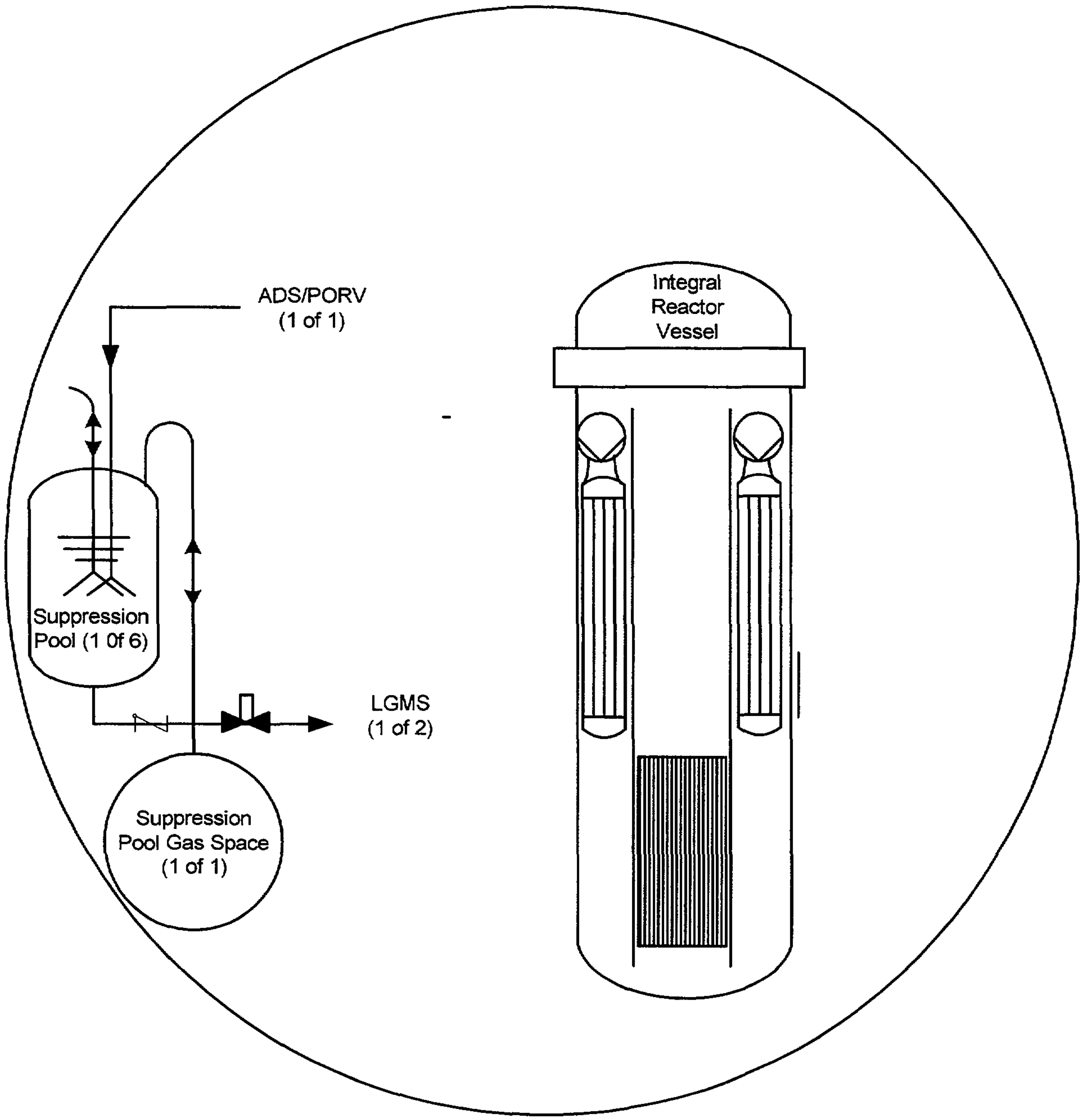

Figure 4.6-1 IRIS CPSS Simplified Sketch

4.7 PASSIVE CONTAINMENT COOLING SYSTEM

4.7.1 System Functions

The function of the Passive Containment Cooling System (PCCS) is to provide a non-safety grade ultimate heat sink for the removal of the core decay heat from the containment that is diverse from the EHRS. The PCCS, in performing this function, has the capability of removing sufficient energy from the reactor containment structure to prevent the containment from exceeding its design pressure.

4.7.2 System Description and Operation

The Passive Containment Cooling System (PCCS) is a non-safety grade system which is capable of transmitting heat directly from the reactor containment shell to the environment such that the containment design pressure (and temperature) are not exceeded following any postulated design basis event.

As illustrated in the sketch in Figure 4.7-1, the PCCS includes and makes use of the top portion of the steel reactor containment structure that is located within the fuel handling area. It additionally requires the use of the plant fire protection system water tank, and associated instrumentation, piping, and valves. Other alternative water sources can also be used to provide this function including the demineralized water system, and connections are provided to utilize water provided from offsite sources (e.g. fire trucks).

<u>Post LOCA Containment Heat Removal</u>

Following a postulated event that results in the pressurization of the containment, the EHRS provides the safety grade means of removing core decay heat. Should all four of the EHRS subsystems fail to function, the PCCS provides an alternate, diverse means of removing core decay heat. This is accomplished by simply flooding the refueling cavity located above the containment. This cavity, which is normally dry/empty during normal operation, contains the upper closure head for the containment vessel. The closure head has sufficient surface area to transfer core decay heat to the floodup water, such that the containment vessel pressure cannot exceed its design pressure of 13 MPa (188 psia).

The fire protection system contains sufficient water, such that this diverse method of containment cooling can operate for at least seven (7) days. The fuel handling area is designed to provide vents for the boil-off of the PCCS water.

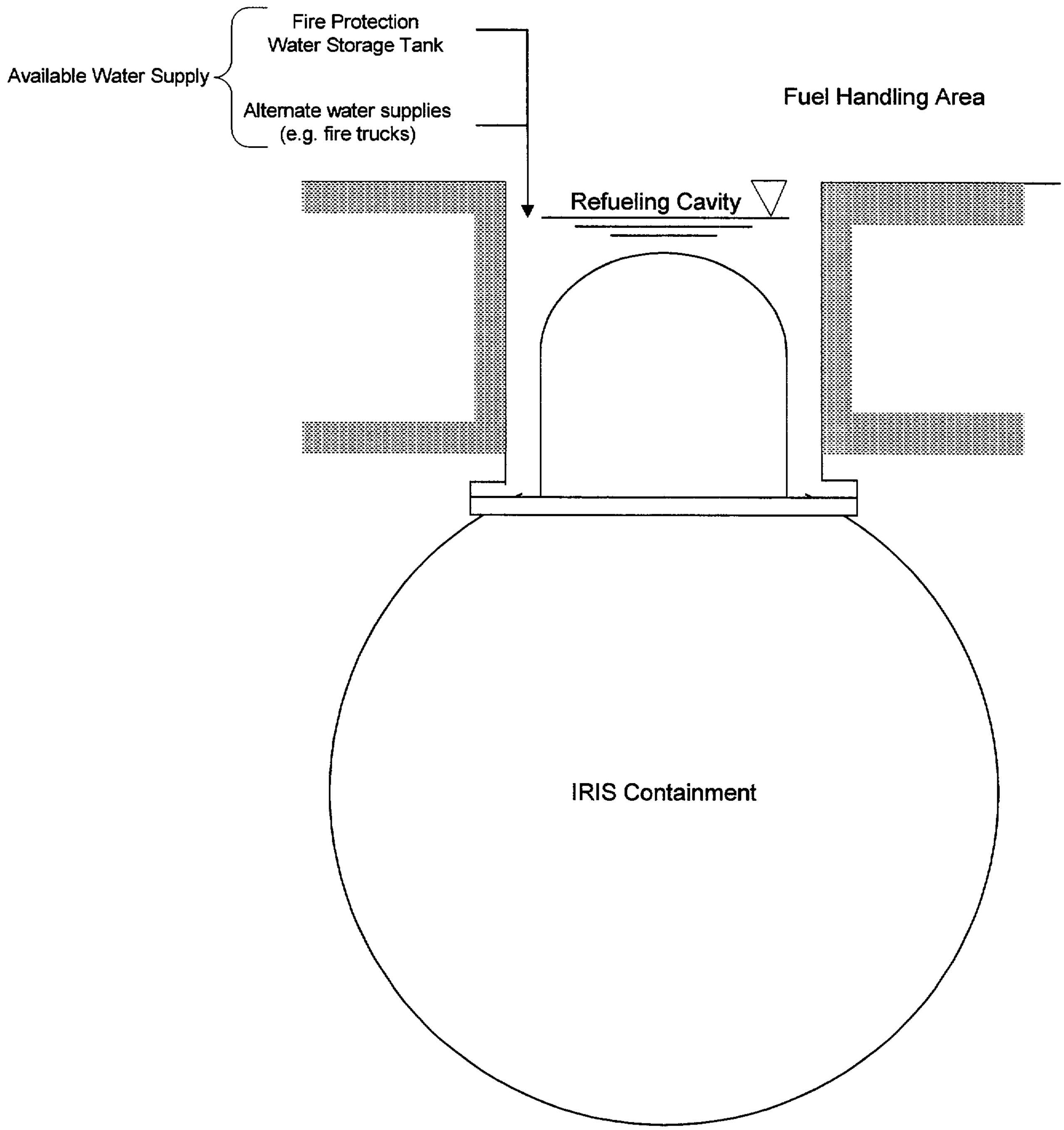

Figure 4.7-1 IRIS Passive Containment Cooling System Simplified Sketch

4.8 MAIN CONTROL ROOM EMERGENCY HABITABILITY SYSTEM

4.8.1 System Functions

The Main Control Room Emergency Habitability System (VEHS) is designed to operate following an accident that would disable or prohibit use of the normal HVAC system. The major functions of the VEHS are as follows:

- Ventilation. Provide forced ventilation to deliver an adequate supply of breathable air for the Main or Emergency Control Room (MCR/ECR) occupants.

- Pressurization. Provide forced ventilation to maintain the MCR/ECR at a slight positive pressure with respect to the surrounding areas to minimize the infiltration of airborne contaminants.

- Cooling. Provide cooling of the equipment and facilities that must remain functional during an accident. This includes the MCR/ECR, the Integrated Protection Cabinet Rooms (IPCRs), the Emergency Switchgear Rooms (ESWGRs), and the 1E Battery Rooms.

These protective functions provide safety-related plant protection, and therefore the VEHS is designed as a safety-related system.

4.8.2 System Description and Operation

The VEHS is made up of two redundant trains of emergency equipment consisting of compressed air storage tanks and associated valves, piping, and instrumentation. Each train consists of four compressed air storage tanks and is sized to deliver the required air flow to the MCR/ECR to meet the ventilation and pressurization requirements for a 3 day habitability time period. The VEHS is designed to remain functional during an SSE and other design basis site events such as hurricane or tornado. A simplified sketch is included as Figure 4.8-1.

When a source of ac power is available, the nuclear island non-radioactive ventilation system provides normal HVAC service to the main control room, technical support center (TSC), instrument and control rooms, dc equipment rooms, battery rooms, and the nuclear island non-radioactive equipment rooms.

When a source of ac power is not available to operate the non-safety HVAC ventilation system or unacceptable radiation levels are present, the VEHS is capable of providing required emergency ventilation and pressurization for the MCR/ECR and cooling for the safety related instrument and control equipment, battery rooms and the dc equipment rooms.

Automatic transfer of habitability system functions from the non-safety grade ventilation to the VEHS is accomplished by the reception of one of two signals:

- High-2 MCR Radiation
- Loss of AC Power Sources

VEHS operation can also be initiated by manual actuation; or, in the event of loss of ac power sources, the normal ventilation system isolation dampers automatically close and the VEHS isolation valves automatically open.

After the VEHS isolation valves are opened, the air supply pressure is regulated by a self contained control valve which maintains a constant downstream pressure regardless of the upstream pressure. A constant air flow rate is maintained by the flow control orifice located downstream of the pressure control valve. The storage tanks are sized to provide an air flow rate that prevents the MCR/ECR CO_2 concentration from exceeding 1% by volume assuming a maximum occupancy of 5 plant personnel. For this occupancy the VEHS compressed air storage tanks should provide 20 scfm of ventilation air. If VEHS operation is required after the 3 day period, the emergency air storage tanks can be refilled using portable compressed air equipment (bottles or a compressor).

The MCR/ECR is maintained at a slight positive pressure to minimize the infiltration of airborne contaminants from the surrounding areas. Pressurization of the MCR/ECR is achieved by limiting the amount of air leakage from the leak tight construction of the MCR envelope to less than the air supply. The air flow rate is sufficient to maintain the MCR/ECR envelope at 1/8 inch water gauge positive differential pressure with respect to the surroundings.

To ensure a suitable working environment for personnel, the temperature rise in the MCR/ECR envelope, following a loss of normal ventilation, is less than 8.33 °C (15 °F) over a 3 day time period. Sufficient thermal mass is provided in the walls and ceilings to absorb the heat generated by the equipment, lights, and occupants. The temperature in the instrumentation and control rooms and the dc equipment rooms following a loss of the normal ventilation function remains below 48.9 °C (120 °F) over a 3-day period. As in the MCR/ECR, sufficient thermal mass is provided surrounding these rooms to absorb the heat generated by the equipment. If VEHS cooling is required beyond 3 days, self-powered, on-site equipment can be employed to continue the required cooling functions.

4.8.3 VEHS Parameters

Minimum MCR/ECR air flow rate	20 scfm
Maximum number of occupants	5
Maximum MCR CO_2 concentration (by volume)	1%
Minimum MCR positive pressure	1/8 inch water gauge
Maximum MCR envelope temperature rise (in 3 days)	8.33 °C (15 °F)

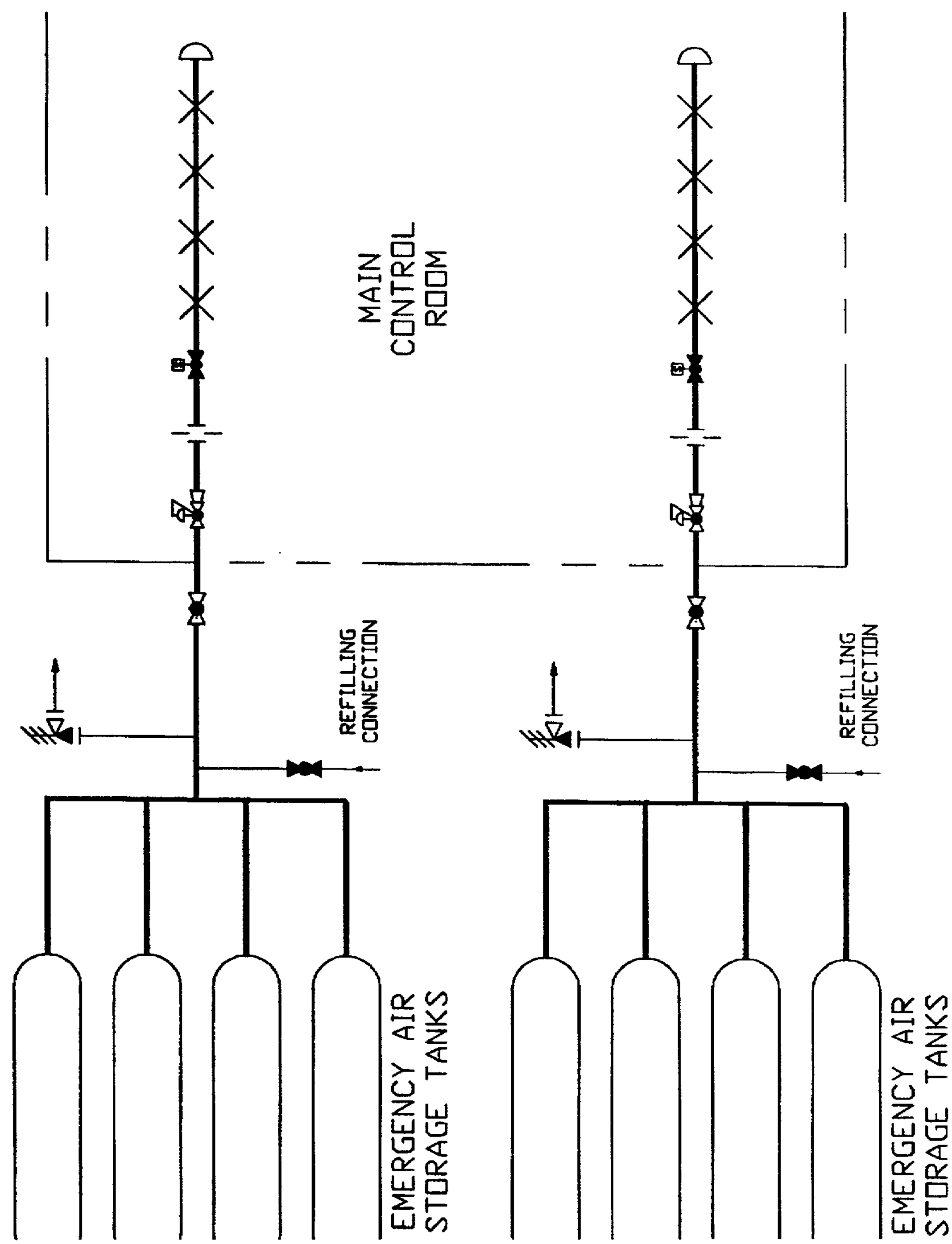

Figure 4.8-1 IRIS Main Control Room Habitability Simplified Sketch

4.9 CHEMICAL AND VOLUME CONTROL SYSTEM

4.9.1 System Functions

The Chemical and Volume Control System (CVCS) is designed to perform the following major functions:

- Purification - Maintain the Reactor Coolant System (RCS) fluid purity and activity level within acceptable limits.
- RCS Inventory Control and Makeup - Maintain the required coolant inventory in the RCS.
- Chemical Shim and Chemical Control - Adjust the reactor coolant boron concentration as required for all plant operating modes. Maintain the desired reactor coolant boron concentration and fluid chemistry conditions within acceptable conditions.
- Oxygen Control - Provide the means for reducing and maintaining the proper oxygen concentration in the reactor coolant.
- Filling and Pressure Testing the RCS - Provide the means for filling and pressure testing the RCS.
- Borated Makeup to Auxiliary Equipment - Provide makeup water to all of the auxilliary systems which require borated reactor grade water.
- Pressurizer Auxiliary Spray - Provide pressurizer auxiliary spray water in order to depressurize the RCS prior to refueling.

4.9.2 System Description and Operation

The CVCS consists of a regenerative and a letdown heat exchanger, demineralizers and filters, circulating pumps, makeup pumps, tanks, and associated valves, piping, and instrumentation.

The normal CVCS purification loop is entirely inside containment and operates at RCS pressure. Due to the low developed head required for the RCPs, IRIS utilizes small canned motor pumps to circulate the purification flow to and from the reactor vessel. Two CVCS circulating pumps are provided, with one pump normally running and one pump as an installed spare. These pumps are designed to operate continuously.

The purification loop arrangement is such that the reactor coolant is first cooled in the regenerative heat exchanger, and then further cooled by component cooling water in the letdown heat exchanger. It is then purified in the mixed and (optionally) cation bed demineralizer, pumped through the reactor coolant filter, and then returned to the reactor vessel after passing through the opposite side of the regenerative heat exchanger.

Removal of radiogases from the RCS will not normally be necessary because the gases will not build up to unacceptable levels when fuel defects are within normally anticipated ranges. If radiogas removal were to be required because of high fuel defects, the CVCS could be operated intermittently in a semi-closed loop arrangement, with a portion of the letdown flow routed outside of containment through the Liquid Radwaste System (WLS) degasifier to one of the WLS holdup tanks, and then returned to the RCS via the CVCS makeup pumps.

Changes in the reactor coolant volume will be accommodated by the pressurizer level program for normal power changes, including transition from hot standby to full power operation and returning to hot standby. In addition, the pressurizer has sufficient volume, within the deadband of the level control program, to accommodate minor RCS leakage for some time. The CVCS provides inventory control to maintain the RCS inventory following minor leakage, and to accommodate expansion during heatup from cold shutdown, and contraction during cooldown.

One of the CVCS makeup pumps will start automatically on a low reactor vessel (pressurizer) level signal, taking suction from the boric acid tank and the demineralized water tank, in order to provide makeup to the RCS. The makeup water boron concentration is determined by positioning a three-way control valve to blend the proper boric acid concentration to match the RCS coolant boron concentration.

The other CVCS functions are listed and briefly described below:

- <u>RCS Reactivity Control and Chemical Shim</u> - The CVCS maintains the required reactor coolant boron concentration by providing the means to reduce the concentration of boron in the coolant in a controlled manner for plant startups, to reduce or dilute the coolant boron concentration during normal operation to compensate for fuel depletion, to increase the coolant boron concentration prior to RCS cooldown to ensure shutdown conditions.

- <u>RCS Chemistry Control</u> - The CVCS maintains the required reactor coolant water chemistry conditions by providing the means to control the RCS pH by maintaining the proper concentration of lithium hydroxide (LiHO), to minimize the radioactivity level in the reactor coolant by providing demineralization, to minimize the amount of particulate in the coolant by filtration, and to assist in the removal of radioactive deposits in the RCS by adding hydrogen peroxide (H_2O_2) during plant cooldown.

- <u>Oxygen Control</u> - The CVCS controls the concentration of oxygen in the RCS coolant, both during startup by introducing hydrazine and during power operations by maintaining excess hydrogen in the reactor coolant. This is accomplished by injecting high pressure gaseous hydrogen into the reactor coolant, as required. The excess hydrogen limits the amount of oxygen resulting from the decomposition of reactor coolant due to radiolysis in the core region during normal operation.

- <u>RCS Filling and Pressure Testing</u> - The RCS is filled and pressure tested using the CVCS makeup pumps to provide water at the proper boron concentration. The CVCS does not perform hydrostatic testing of the RCS, which is only required prior to initial startup and after major, non-routine maintenance, but provides connections for a temporary pump to do so.

- Borated Makeup to Auxiliary Equipment - The CVCS makeup pumps are used to provide makeup water at the proper boron concentration (approximately refueling water concentration) to the CPSS water tanks, the refueling water storage tank and the spent fuel pit.

- Pressurizer Auxiliary Spray - The CVCS provides auxiliary spray water to the pressurizer region steam space for depressurization of the RCS.

4.9.3 CVCS System Parameters

Purification flow rate	6.3 10^{-3} m^3/s (100 gpm)[1]
Normal boration flow rate	6.3 10^{-3} m^3/s (100 gpm)
Normal dilution flow rate	6.3 10^{-3} m^3/s (100 gpm)
Maximum makeup water flow rate (RCS pressurized)	8.52 m^3/s (135 gpm)
Temperature of reactor coolant entering CVCS	328.4 °C (623.1 °F)
Demineralizer resin volume (per demineralizer)	1.42 m^3 (50 ft^3)
Expected life of demineralizer resin	1 fuel cycle
Normal temperature of effluent to Liquid Radwaste System	54.4 °C (130 °F)
Normal flow rate to Liquid Radwaste System	6.3 10^{-3} m^3/s (100 gpm)

[1]All volumetric flow rates are based on 54.44 °C and 15.856 MPa (130°F and 2300 psia).

4.10 SPENT FUEL PIT COOLING SYSTEM

4.10.1 System Functions

The Spent Fuel Pit Cooling System (SFPCS) is designed to perform the following major functions:

- Spent Fuel Pit (SFP) Cooling - Remove heat from the stored fuel by cooling the water in the spent fuel pit during all modes of operation to maintain a desired pit water temperature.
- Spent Fuel Pit Purification - Provide purification and clarification of the spent fuel pit water during all modes of plant operation.
- Refueling Cavity Purification - Provide purification of the refueling cavity during refueling operations.
- Water Transfers - Transfer water between the refueling water storage tank (RWST) and the refueling cavity during refueling operations. Water can also be transferred to the cask loading and washdown pit after refueling operations are complete.
- RWST Purification - Provide purification of the refueling water storage tank during normal operation.

4.10.2 System Description and Operation

The SFPCS consists of two mechanical trains of equipment. Each train consists of a pump, heat exchanger, demineralizer and filter. The two trains of equipment share a common suction line and discharge line to/from the SFP. In addition, the SFPCS is comprised of piping, valves and instrumentation necessary for correct system operation

The SFPCS is designed such that either train of equipment can be operated to perform any of the required functions independently of the other train. One train is normally aligned to cool and purify the spent fuel pit water while the other train is available for water transfers, RWST purification, or aligned as a backup to the operating train of equipment.

The SFPCS is designed to remove heat from the spent fuel pit such that the spent fuel pit water temperature will be < 48.9°C (120°F) following a normal refueling. The heat load is based on the decay heat generated by the accumulated fuel assemblies stored in the fuel pit for 10 years plus 1/2 core placed into the pit beginning 120 hours after shutdown. The SFS can perform this function with only one SFP pump and heat exchanger operating.

Also, the SFPCS is designed to remove heat from the spent fuel pit such that the spent fuel pit water temperature will still be <48.9°C (120°F) following a full core off-load. The heat load is based on the decay heat generated by the accumulated fuel assemblies stored in the fuel pit for 12 years (including 1/2 core placed in the pit from the most recent refueling) plus 1 full core placed in the pit beginning at 120 hours after shutdown and being completed 72 hours later. Analysis has determined that the maximum heat load to the SFPCS resulting from this operation occurs from a full core off-load at the end of a 36 month fuel cycle. The SFPCS performs this function with both SFP pumps and heat exchangers aligned to cool the spent fuel pit.

The SFPCS is a non-safety related system. The safety function of cooling and shielding the fuel in the spent fuel pit is performed by the water in the pit. However, since the SFPCS is not safety related, the following criteria have been adhered to in the design of the spent fuel pit as well as the SFPCS:

- The spent fuel pit shall be designed such that a water level is maintained above the spent fuel assemblies for 7 days following a loss of the spent fuel pit cooling system. The assumed heat load is the heat load associated with a full core off-load.
- The radioactivity exposure to the public shall be shown to be within the dose limits specified in 10CFR100.
- The fuel handling area shall be designed to vent steam /boiloff from the SFP water to the environment.
- The SFP or SFPCS shall provide safety grade connections to allow makeup water addition to the spent fuel pit within 7 days following any design basis event including a seismic event.

4.11 PRIMARY SAMPLING SYSTEM

4.11.1 System Functions

The Primary Sampling System (PSS) is designed to perform the following major functions:

- Normal Operation Sampling - Collect and deliver representative samples of liquids and gases from various plant fluid systems, including the Reactor Coolant System, to sample stations for laboratory analysis. Monitor the plant and various system conditions and performance using the collected and analyzed samples. Provide both on-line monitoring and grab sampling capabilities.
- Post Accident Operations - Provide the means to perform post accident sampling as required by NUREG/CR 4330 and Regulatory Guide 1.97. Grab samples from various points in the plant as specified in NUREG-0737 are provided.

4.11.2 System Description and Operation

The PSS includes a shielded grab sampling unit which is common to both the normal and post-accident activities. All the valves (except for containment isolation) are manual and located at the grab sampling unit. Degassing and dilution equipment and piping are also located at the grab sampling unit.

It consists of two independent subsystems, liquid sampling and gas sampling capability, and each is capable of diluting high activity sample fluid.

The liquid sampling portion is used to collect and/or analyze samples at various locations. The gas sampling portion is used to analyze containment atmosphere at two different locations. All necessary controls and instrumentation are provided to assure safe and reliable operation.

The sampling fluid is directed to one of the two available destinations: the laboratory or the grab sampling unit for grab samples. Samples to determine the dissolved gas in the liquid can be obtained via a shielded syringe from the liquid sampling portion. Dilution of highly active sampling fluid can be performed as required, and a diluted liquid sample can be extracted utilizing a second shielded syringe.

During normal operation the RCS system pressure is used as the motive force to deliver the sampling flow to the grab sampling unit. In the event the RCS is depressurized, an eductor is used to draw the sampling flow. The eductor is an integral part of the PSS and contains no moving part; consequently, it is highly reliable.

Gas (containment atmosphere) samples are collected in one of the two sampling bottles reserved for containment air sampling. The motive force for this sampling process is the primary sample ejector. Nitrogen is used as the pressure source of the ejector. The mixture of the nitrogen and the containment atmosphere discharged from the ejector is directed back to the containment. The gas sample can be diluted as required in the PSS if the incoming gas flow is highly radioactive.

4.11.3 PSS System Parameters

Normal Operation:	
Liquid sampling flow rate	4.4 10^{-5} m^3/s (0.7 gpm) (1)
Liquid sampling flow temperature	51.7 °C (125 °F) (2)
Gas sampling flow rate	TBD
Gas sampling flow temperature	10 °C to 48.9 °C (50°F to 120°F)
Post Accident Operation:	
Liquid sampling flow rate	4.4 10^{-5} m^3/s (0.7 gpm) (1)
Liquid sampling flow temperature	51.7 °C (125 °F) (2)
Gas sampling flow rate	TBD
Gas sampling flow temperature	same as containment atm.

(1) Based on 51.7 °C and 15.513 MPa (125 °F and 2250 psia).
(2) Downstream of the sample cooler (outside containment)

4.12 NORMAL RESIDUAL HEAT REMOVAL SYSTEM

4.12.1 System Functions

The Normal Residual Heat Removal System (NRHRS) is designed to perform the following major functions.

- Normal Cooldown - Remove heat from the core and the Reactor Coolant System (RCS) during normal cooldowns.
- Refueling Heat Removal - Remove heat from the core and the RCS during refueling operations including when the level in the reactor vessel is reduced for removal of the RV head.
- Refueling Draindown - Provide connections to the RCS that facilitate reducing the reactor vessel level for head removal and other operations during refueling.
- Shutdown Purification - Provide RCS and refueling cavity purification flow to the Chemical and Volume Control System (CVCS) during refueling operations.
- RWST Cooling - Provide cooling for the Refueling Water Storage Tank (RWST) during operation of the Emergency Heat Removal System (EHRS) heat exchangers or during normal plant operations when required.
- CPSS Cooling - Provide cooling for the Containment Pressure Suppression System (CPSS) during normal plant operations when required.
- Post-Accident Closed Loop Cooling - Remove heat from the core and the RCS following successful mitigation of non-LOCA accidents by the Emergency Heat Removal System (EHRS).
- RCS Makeup - Provide low-pressure makeup to the RCS from the CPSS.
- LTOP Function - Provide low temperature overpressure protection (LTOP) function for the RCS during refueling and shutdown operations.

4.12.2 System Description and Operation

The NRHRS consists of two mechanical trains of equipment. Each train consists of one residual heat removal (RHR) pump and one RHR heat exchanger. The two trains of equipment share a common suction line from the Reactor Coolant System (RCS) and a common discharge line back to the reactor vessel. In addition, the NRHRS is comprised of piping, valves and instrumentation necessary for correct system operation.

The NRHRS is designed to perform the normal plant cooldown function. The NHRS removes both residual and sensible heat from the core and the Reactor Coolant System (RCS) and reduces the temperature of the RCS during the second phase of plant cooldown. The first phase of cooldown is accomplished by transferring heat from the RCS via the steam generators to the Main Steam System (MSS). The normal RHR function is performed as it is in reactors with a traditional loop-type layout. The RHR pumps take suction from the core exit side of the in-vessel flow path. This "hot" water is cooled via the RHR heat exchanger and the cooled water is returned to the vessel downcomer to re-enter the core. The NRHRS cools the RCS from 177 °C (350 °F), at four hours after reactor shutdown, to 48.9 °C (120 °F) within 96 hours after reactor shutdown. The NRHRS performs this function with both trains of pumps and heat exchangers operating.

The NRHRS is also designed to perform its heat removal function during plant operations performed in preparation for and during plant refueling. These operations include reducing the level in the reactor vessel to facilitate removal of the reactor vessel head or removal of a reactor coolant pump. Design features aimed specifically at ensuring NRHRS operation at reduced vessel level include:

- Reactor Vessel Barrel - The reactor vessel barrel contains holes that allow the core exit flow to pass into the annular space between the barrel and the reactor vessel. These holes are located approximately at the middle of the steam generator tube bundle length. These holes ensure that a flow path from the core exit to the vessel piping nozzles, through which the NRHR pump suction flow is provided, is maintained even when the vessel level is reduced to drain the pressurizer region.

- Reactor Vessel DVI Nozzle – The return flow from the NRHRS to the reactor vessel is through the two DVI nozzles. The DVI penetrations into the vessel include a downturn that directs the NRHR return flow to the downcomer region, and back to the core inlet region. This downturn together with the NRHR pumps taking suction flow from the core exit, establishes a forced circulation flowpath through the core when the NRHR pumps are operating. Thus, the NRHRS can establish and/or maintain a subcooled condition in the vessel, during head removal operations when the vessel level must be reduced and no reactor coolant pumps can be operated. This flowpath is maintained for NRHR heat removal during the refueling operations and all expected outage inspection and maintenance operations, when the reactor coolant pumps are not operating and/or heat removal via the steam generators is not provided.

- Self-Venting Suction Line - The RHR pump suction line is sloped continuously upward from the pump to the vessel piping connections with no local high points. This eliminates the problems associated with refilling the RHR pump suction line in case of an RHR pump trip due to cavitation or excessive air entrainment. If the RHR pump suction line becomes filled with air causing the pump to cavitate and trip, the line will refill and the RHR pump can be immediately restarted once an adequate level in the vessel is re-established. Traditionally this has been a problem because the suction line often contains local high points that become filled with air and will not refill without local venting.

- RCS Integral Layout - The IRIS integral reactor vessel layout design is such that the level to which the water must be drained to in order to facilitate removal of the vessel head, or for removal of a reactor coolant pump is much higher than the NRHR pump suction line connection. This helps to ensure that adequate suction head to the NRHR pumps is always provided, even if it is assumed that the reactor water is at saturated (boiling) conditions.

These features all add to the reliability of the NRHRS to perform its RHR function during reduced water level operations. This design eliminates the problems that current plants have during mid-loop operations such as air ingestion into the RHR pumps which leads to a loss of RHR cooling, RHR throttle valve cavitation at mid-loop level, and restarting of RHR flow once air binding has occurred.

The IRIS NRHRS is designed to minimize the potential for an "inter-system LOCA" by both increasing the number of isolation valves between the RCS and the NRHR, and also by design of its equipment not to fail even if exposed to RCS pressure conditions.

4.12.3 NRHR System Parameters

System Design Pressure	6.2 MPa (900 psig)
System Operating Pressure	3.1 MPa (450 psig)
Cooldown Time to 120-F	96 hours
Total RHR Flow Rate (2 Pumps)	0.05 m^3/s (800 gpm)
RHR Relief Valve Setpoint	3.88 MPa (563 psig)

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5.0 STEAM AND POWER CONVERSION SYSTEMS

5.1 MAIN TURBINE SYSTEM

5.1.1 System Function

The Main Turbine System converts the thermal energy of the steam produced by the reactor into rotational mechanical work that rotates the electrical generator to produce electric power. This conversion is performed by the steam turbine.

5.1.2 System Description and Operation

Turbine

The turbine is designed to change the thermal energy of the steam produced by the reactor into rotational mechanical work that rotates an electrical generator to provide electrical power. The turbine consists of a double flow high pressure rotor and cylinder and a double flow low pressure rotor and cylinder which exhaust to the condenser. The Main Turbine System includes the steam turbine and all stop, control and intercept valves directly attached to the turbine, and the crossover and crossunder piping between the turbine cylinders, and the Moisture Separator Reheater (MSR). The system is designed to operate in conjunction with the reactor and other plant systems to produce approximately 335 MWe output to the grid, with all house loads supplied by the plant.

The high pressure portion of the turbine has extraction connections for two stages of feedwater heating. The high pressure turbine exhaust steam provides steam for one stage of feedwater heating in the Deaerator. The low pressure turbine has extraction connections for four stages of feedwater heating.

The Moisture Separator Reheater (MSR) is an integral component in the Turbine System which extract moisture from the steam and reheat the steam to improve the overall performance of the turbine. The MSR is located between the high pressure turbine exhaust and the low pressure turbine inlet, and has a single stage of reheat using the high pressure steam supply.

During normal turbine operation, the turbine will be on automatic control and will require minimal operator attention. The turbine control will operate in the 50 to 100 percent load range on a daily basis as required by the load schedule.

Turbine Auxiliaries

Digital Electro-Hydraulic (DEH) Control System - The turbine is equipped with a DEH control system for turbine control and overspeed protection. The system operates the turbine steam flow control valves, the turbine stop valves, reheat stop valves and intercept valves and interfaces with controls of other Steam and Power Conversion System valves to actuate the turbine trip actions.

The DEH control system performs the following functions:

(1) Turbine speed and acceleration control for wide range speed control and automatic synchronization during startup,
(2) Turbine load control either manually or automatically from the load dispatching center,
(3) Load limit control, load runback capability and load rate control,
(4) Automatic and manual capability for transfer from full arc to partial arc admission,
(5) Turbine trip anticipator and overspeed protection controls, and
(6) All turbine protective devices for low bearing oil pressure trip, low vacuum trip, thrust bearing trip, air relay dump valves for tripping extraction steam reverse flow valves, etc.

Turbine Lube Oil System - The Turbine System includes a complete turbine lubricating oil system. This is a closed circuit system, which circulates lubricating oil between the turbine oil reservoir, turbine bearings, and lubrication oil coolers. The Turbine Lube Oil System provides efficient oil filtration and purification and oil cooling to promote high turbine reliability.

The lube oil pumps, which take suction from the turbine lube oil reservoir tank, include a turbine shaft driven pump and auxiliary AC and DC motor driven pumps to guarantee lubrication and cooling lube oil during all modes of turbine operation. In addition, turning gear and turbine bearing lift pumps are provided for turbine start-up. Two turbine oil coolers and one turbine oil reservoir vapor extractor are part of the system. Interconnecting piping, controls and instrumentation complete the system.

This system must always be operational during turbine operation, and prior to turbine start.

Turbine Gland Sealing System

The turbine is provided with steam sealed glands on shaft packings to prevent steam leakage to atmosphere or air leakage into the turbine or condenser. The gland seal regulator automatically controls steam flow and pressure to the gland seals. The gland steam condenser and exhauster receive gland steam and entrained air from the low pressure turbine glands; there the steam is condensed and returned to the condenser, while the air is discharged by the exhauster to the atmosphere. The Gland Steam Seal System controls the gland steam pressure to maintain adequate sealing under all conditions of the turbine operation. This system must always be in service during turbine operation, and prior to turbine restart.

5.2 MAIN STEAM SYSTEM

5.2.1 System Function

The Main Steam System consists of primary steam piping involved in the power conversion systems and consist of the main steam, bypass and reheat steam piping used to supply steam to the turbine, the moisture separator reheaters, and the extraction steam piping used for regenerative feedwater heating; as well as the associated valves and instrumentation.

5.2.2 System Description and Operation

The Main Steam System consist of the main, reheat, and turbine bypass steam piping including the piping, valves, and instrumentation and controls to supply and distribute main steam inside the Turbine Building. Main steam is received from the Steam Generator System beyond the Main Steam Isolation Valves and is distributed to the Main Turbine, Moisture Separator Reheaters (MSR), and the Condenser steam dump via the bypass valves. The extraction steam piping includes the piping, valves, and instrumentation and controls needed to transmit the steam extracted from the main turbine to the feedwater heaters for regenerative feedwater heating. In addition, the system provides automatic feedwater heater isolation from the turbine to prevent water induction into the turbine in the event of high feedwater heater levels.

One main steam equalizing header is provided in the turbine building that connects the four steam supply lines coming from the eight steam generators in the Containment Building. At the turbine, the header branches into four turbine inlets and a line connecting to the turbine bypass dump valves. A steam line is also provided from the turbine header to the reheater of each of the MSRs.

The main turbine has two separate stop and control valve assemblies which are mounted on the turbine foundation on each side of the high pressure turbine. These valves are part of the main turbine system and are controlled from the main turbine control system. Main steam flow to each MSR reheater is regulated to control the temperature of the reheated steam, which is supplied to the low pressure turbine. The turbine bypass (or steam dump) has the capacity to dump 100 percent of the Turbine Guaranteed condition steam generator flow into the condenser following a reduction of external electrical load.

The Main Steam System includes the line drains and vents required to remove any accumulated condensate from the main steam lines and to maintain the turbine bypass header at operating temperature during plant operation. The drains are designed to accommodate drain flows during startup, shutdown, transient, and normal operation to protect the turbine and the dump valves from water slug damage. The drains are piped to the main condenser via the turbine island vent and drain piping.

5.3 CONDENSATE SYSTEM

5.3.1 System Functions

The Condensate System collects and condenses steam from the low pressure turbines and turbine steam bypass, then transfers this condensate from the main condenser to the deaerator. Condensate pumps take suction from the condenser hot wells and develop sufficient head to overcome the system resistance. This resistance includes system static head, pressure drop across the condensate polishing equipment, the gland steam condenser, and the four stages of closed feedwater heating while supplying adequate feedwater flow to the deaerator during all modes of plant operation.

The Condensate System serves as a cooling water source for the gland steam condenser, as a seal water source for the feedwater and condensate pumps, a source for turbine exhaust hood sprays, and provides the motive force and condensate supply through the condensate polishing system (CPS).

5.3.2 System Description and Operation

The Condensate System originates at the main condenser and includes three 50 percent condensate pumps, two 50 percent low pressure feedwater heater trains with two closed feedwater heaters per train, two 100 percent closed feedwater heaters, and a deaerating heater with a storage vessel. The system includes provisions for pump recirculation and component bypassing, and associated piping, valves and instrumentation.

Exhaust steam from the low pressure turbine is condensed in a two shell multi pressure condenser and collected in the condenser hotwells along with condensate from secondary cycle dumps and drains. Hotwell inventory is maintained by a level control system. Cycle makeup is provided by two parallel split ranged regulating valves which draw from the condensate storage tank. Excess inventory is pumped back to the same tank utilizing condensate pump pressure and is also regulated by two parallel split ranged valves.

Condensate from the hotwell of the lowest pressure condenser shell is piped in a single header to the suction of two operating condensate pumps (one pump in stand by). Condensate pump sealing water is provided by a pressure regulating station that takes its supply from downstream of the gland seal condenser. Two of the pump discharge valves have bypass valves to facilitate initial filling and operation of the system. The pumps discharge to a common header.

The main run of condensate flows through a full flow gland steam condenser. Branch lines supply turbine exhaust hood sprays and feedwater pump seal water. Another branch line provides the means for recirculation back to the condenser. The minimum recirculation flow is controlled to maintain an adequate flow through the gland seal condenser to assure adequate cooling rate and to maintain adequate condensate pump flow rate.

Another branch line from the recirculation line provides for dumping excess water back to the condensate storage tank. The level control valves in the branch line are modulated by a condenser hotwell level controller to control the quantity of water dumped to the condensate storage tank.

5.3.3 Main Condenser

The primary function of the condenser is to serve as a heat sink for the turbine exhaust steam and bypass valves steam dumps. As an intermediate heat exchanger between the cycle steam side and the circulating water system, it transfers cycle heat to the atmosphere through the cooling tower. It provides a low heat rejection temperature and pressure for the turbine cycle to improve the cycle heat rate and changes the steam to condensate that is collected for reuse in the plant cycle. In addition, it serves as a collection point for normal heater drains, other condensate drains and emergency drains from all feedwater heaters, the moisture separator reheater shell drains, and reheater steam drain tanks. The condenser also acts as a collection point for noncondensable gases so that they can be conveniently removed from the cycle. Condensate is deaerated and stored in the condenser for supply to the condensate/feedwater systems.

5.3.4 Main Condenser Description and Operation

The condenser is sized to condense and cool the maximum turbine steam exhaust flow and cycle drains. In addition, the condenser provides the heat sink for the turbine bypass system steam dumps during sudden load reductions whenever the NSSS steaming rates exceed the turbine steam demand during plant load ramp down. The steam dump quantity in this mode exceeds the normal maximum steam condensing flow and is equivalent to the turbine exhaust steam flow at 50 percent turbine load plus 50 percent of the normal main steam flow directly dumped to the condenser.

The condenser is mounted on a spring foundation with a welded joint between the turbine exhaust flange and the condenser neck. The unit condenser is installed so that the tubes are perpendicular to the turbine shaft.

The low pressure Feedwater Heaters No. 1 and 2 are located in the condenser neck with steam extraction piping between the turbine and the heaters.

The bypass valve main steam discharge piping includes a steel pipe manifold inside the condenser, which is designed to uniformly distribute the steam dump into the major condenser steam lanes for condensation.

The condenser hotwell is divided into four separate condensate collecting zones to prevent a condenser tube leak from contaminating the entire hotwell storage capacity and to allow each section of the hotwell, with its associated water pass, to be isolated. The total hotwell water storage volume is equivalent to three minutes of condensate demand at maximum load.

The condenser has an integral central core air off-take system in the coldest circulating water pass tube bundle. Air and other noncondensible gases are removed by the condenser vacuum pumps through piping connected to the condenser shell.

Each condenser has two tube bundles, each with separate inlet-outlet and return water boxes. The inlet/outlet water box circulating water lines are provided with motor operated isolation butterfly valves, and the water boxes are provided with drains. This arrangement permits in-service repair of a leaking condenser tube by removing half of the condensing surface from service.

The condenser during various operating modes is designed to accept steam flows from the main turbine exhaust, bypass valve steam dump, turbine ventilator valve discharges, and gland seal spillover steam. In addition, the condenser has connections for and is designed to accept high energy drains from the plant's steam systems, emergency drains from feedwater heaters and MSR shell and heaters, and low energy drains from the low pressure heaters and gland condenser. The condensate/feedwater clean-up recirculation, condensate pump recirculation and condensate makeup are also added to the condenser. Heater vents for the removal of noncondensible gases are collected in the condenser.

5.4 MAIN AND STARTUP FEEDWATER SYSTEM

5.4.1 System Functions

The Main Feedwater System (FWS) is designed to take suction from the deaerator storage vessel and supply the steam generators with adequate feedwater during all modes of operation including transient conditions.

The FWS is designed to meet flow requirements of the steam generator at maximum guarantee conditions and to provide stable operation during the most severe transient. The most severe transient condition occurs on a 100 percent load drop while maintaining the turbine system with sufficient steam for easy restart and dumping approximately 100 percent of the steam capacity to the condenser with the by-pass valves. For plant power levels between 25 percent and 100 percent, the Main Feedwater System is designed to respond to a 10 percent step increase or decrease without steam dump and ramp load changes of 5 percent per minute.

The function of the startup feedwater pumps is to provide feedwater to the steam generators under low flow conditions such as startup, hot standby, and shutdown when the main feedwater/condensate pumps may be unavailable or oversized for the required flow rates. These pumps are designed to deliver feedwater to the steam generators using the Deaerator as the normal suction source. The pumps automatically start following reactor trip, loss of main feedwater and other anticipated transients with feedwater provided at a sufficient flow to remove reactor core or decay heat and to cool the plant down at a rate of 50°F/hour via the steam generators.

5.4.2 System Description and Operation

The Main Feedwater System includes two nominal 50 percent capacity, variable speed, motor driven, centrifugal, low NPSH booster, and main feedwater pumps. The booster pumps take suction from the deaerator storage tank and supply water to the main feed water pumps, which in-turn supply water to the steam generators. Whenever the feed water flow measured at the main feed water pump discharge falls below a minimum value , the normally closed recirculating line (mini-flow) control valve will modulate to maintain the pump flow above the minimum flow required to prevent pump damage.

Both the booster pumps and main feedwater pumps are sized to provide the flow and head individually to operate the plant at 70 percent power level in the event that any one booster and main pump pair is out of service.

A single pipe connects the feedwater pump discharge header with feedwater heaters to ensure stable pressure and temperature conditions. Two 100 percent high pressure feedwater heaters with a common string bypass is provided to allow either heater to be isolated while maintaining full flow to the steam generators.

Downstream of the feedwater heaters, the single feedwater pipe branches to four separate feedwater control stations, each regulating the flow of feedwater to their respective steam generator pairs.

There are two start-up feedwater motor driven pumps, each taking suction from the Deaerator and discharging into the Main Feedwater piping. An alternate suction feed is provided from the condensate tank. The combined capacity of both pumps is equivalent to approximately 5 percent of the normal feedwater flow. Each startup feedwater line contains one control valve. During normal startups and shutdowns, the control valve automatically controls the flow to one SG pair. This automatic flow control also functions in post reactor trip situations when the Startup Feedwater System is activated. Startup feedwater discharges into the main feedwater supply in the containment building area.

5.5 HEATER DRAIN SYSTEM

5.5.1 System Functions

The function of the Heater Drain System is to provide a method for draining, collecting, and returning to the feedwater and condensate systems the condensed extraction steam used for feedwater heating in a regenerative feedwater heating cycle. The system consists of two distinct parts: the high pressure heater drain and the low pressure heater drain portions of the system.

The function of the high pressure heater drains portion of the system is to collect condensed fluid known as "drains" from the moisture separator tubes and the two high pressure feedwater heaters into the deaerator (an "open" heater) from where it is pumped forward as feedwater to the steam generators. Drains from the moisture separator shell are normally pumped to the deaerator and the high pressure feedwater heaters normally cascade drain from the highest pressure heater (heater number 7) through the lower pressure heater (heater number 6) and then to the deaerator (heater number 5). This action of cascading high pressure drains is performed in order to extract residual heat from condensed steam and improve the plant's thermal efficiency. During certain operating modes, the drains from the moisture separator and the high pressure feedwater heaters may be dumped to the condenser.

The function of the low pressure heater drain portion of the system is to collect the condensed extraction steam from each low pressure heater. Drains are cascaded from the highest pressure to the lowest pressure heater (heater numbers 4 to 3 to 2 to 1) through a series of heater integral drains and finally dumped to the condenser. This action of cascading low pressure drains is performed in order to extract residual heat from condensed steam and improve the plant's thermal efficiency. During certain operating modes, these drains may also be dumped to the condenser.

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6.0 PLANT ARRANGEMENT AND CONSTRUCTION

6.1 INTRODUCTION

A study on the general arrangement of a complete IRIS electrical power plant was performed as requested by U.S. utilities for consideration in the early site permit (ESP) siting evaluation program. The IRIS plant site plot plans and the preliminary general arrangement drawings developed in this study were largely based on the assumption that the IRIS control room, emergency power supply batteries, non-safety related batteries, reactor control and protection cabinets and switchgear areas would be similar in size to the corresponding areas provided in the AP600 plant design. As such, it is expected that more detailed layout development effort will result in a significant reduction in the IRIS building volumes and the overall plant footprint. This layout does however provide a starting point for future optimization and illustrates the application of the small, spherical IRIS containment integrated with the fuel handling area and innovative safety systems, into an overall plant arrangement. Two plant arrangements were considered in this initial IRIS site layout study - three independent single unit plants, and an arrangement with two independent twin-unit plants. The three single unit plants provide a net electrical output of 1005 MW and the two twin-units provide a net electrical output of 1370 MW, in order to meet US utilities' request for 1000 MWe minimum output.

Future optimization would include the results of on-going studies on small reactors, where potential optimizations will include increasing the amount of shared equipment between reactors and even establishing a single, centralized, control and protection building for all the units on a given site.

6.2 SITE PLAN

The study on the general arrangement of the IRIS reactor developed two site plans: one for the construction of multiple, single unit plants; and a multiple, twin unit arrangement. Figures 6.2-1 and 6.2-2, show three, independent, single units and two, independent, twin unit IRIS plant arrangement options, respectively. In both arrangements, the auxiliary building which forms the seismic block has a common base mat at elevation –15 meters (-49'-2"). The auxiliary building (Item 1) contains the control room and all safety related equipment, as well as the containment and shield structure (Item 2), and the fuel handling area (Item 3). The plant grade level is at 0 meters. The roof elevation of the auxiliary building, over the fuel handling area, which is located over the containment and shield building is at +32 meters (+105') above grade. The fuel handling area occupies the upper, southern portion of the auxiliary building and extends over the containment such that the containment and RV closure heads, the reactor vessel internals, and the fuel can be lifted vertically into the fuel handling area during refueling operations. Should the need arise, the reactor coolant pumps, steam generators, and lower core internals can also be removed from the integral reactor vessel and stored, inspected, repaired, or loaded onto a rail-car in the fuel handling area.

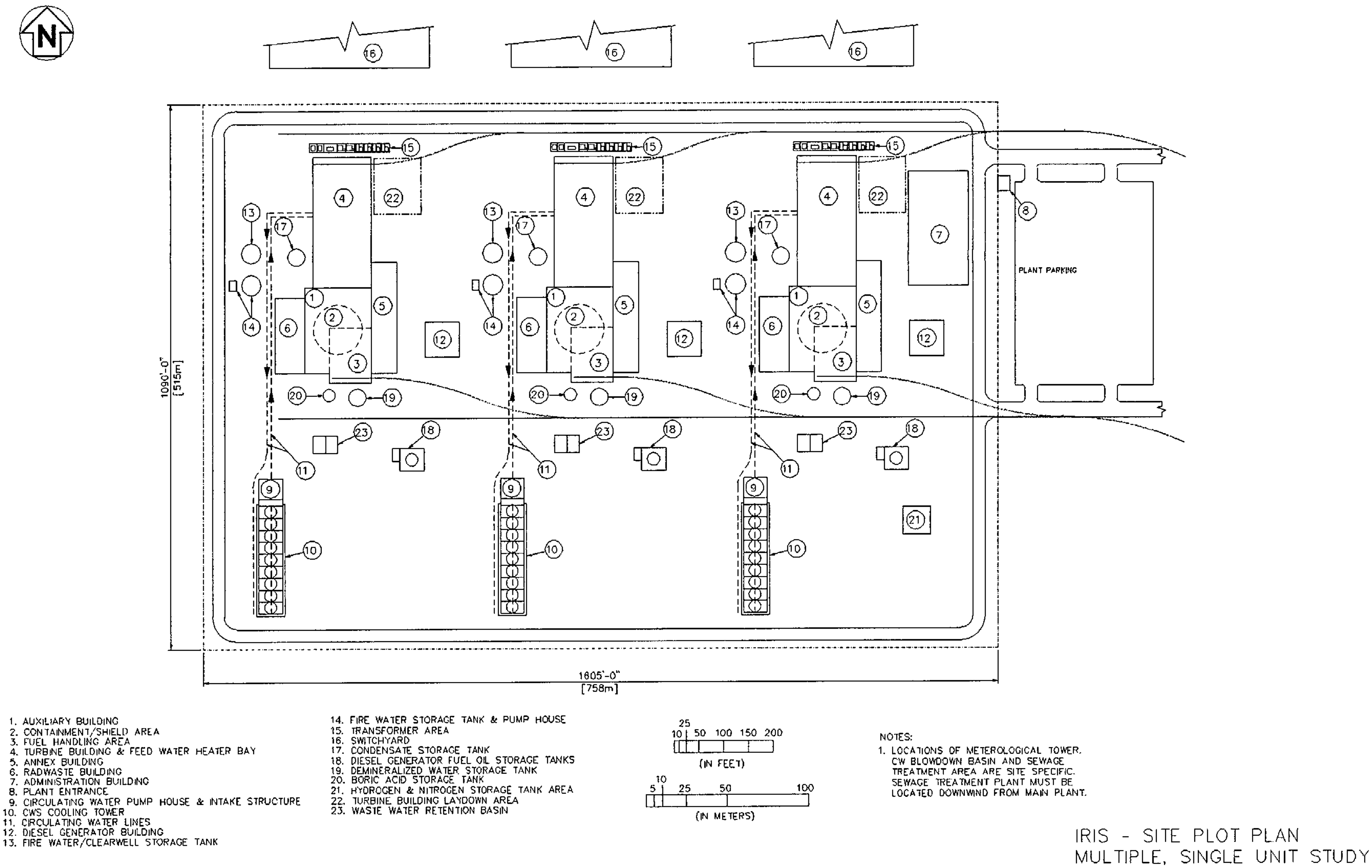
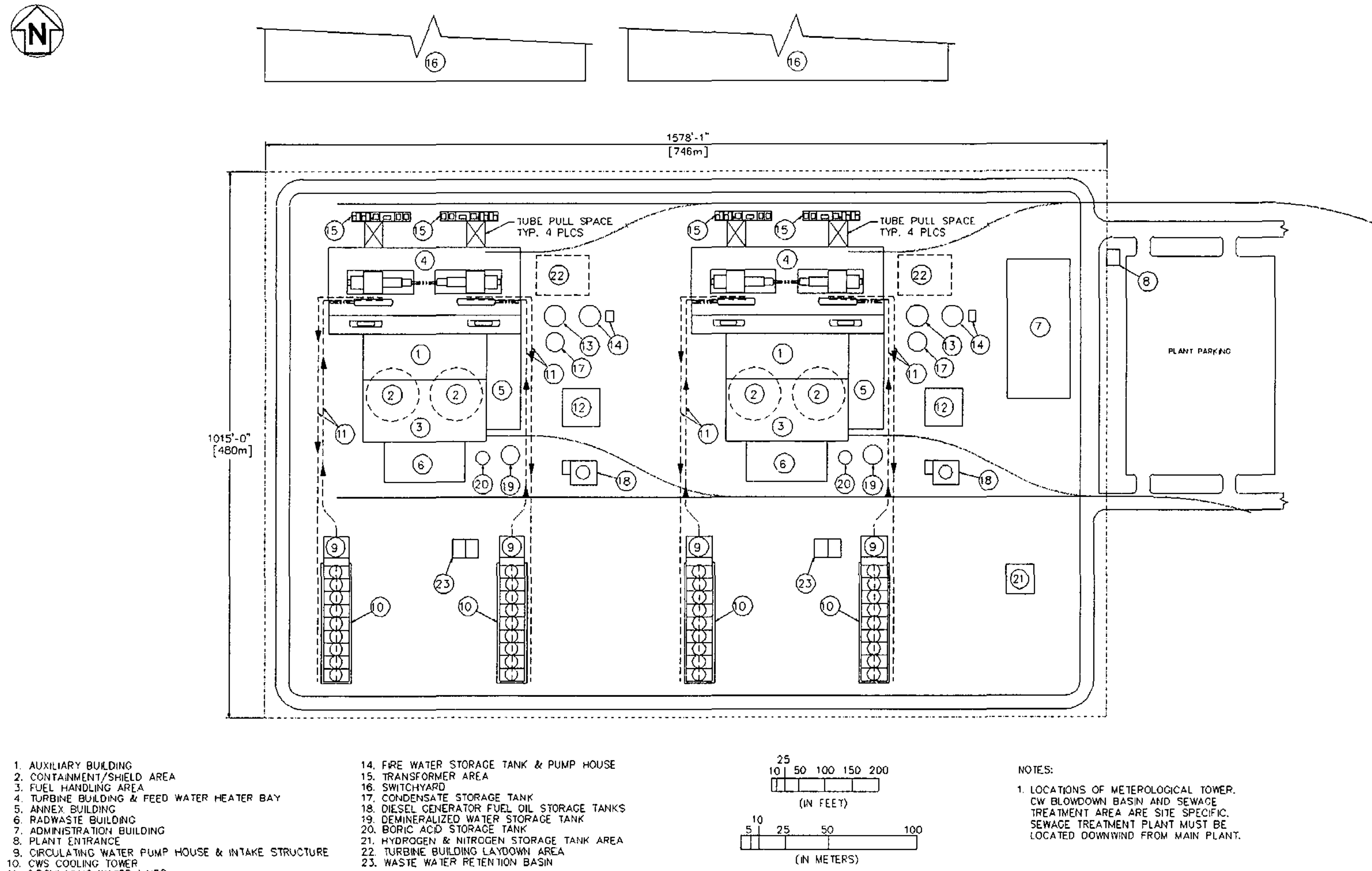


Figure 6.3-1, IRIS, Three Single Unit Site Plot Plan



1. AUXILIARY BUILDING
2. CONTAINMENT/SHIELD AREA
3. FUEL HANDLING AREA
4. TURBINE BUILDING & FEED WATER HEATER BAY
5. ANNEX BUILDING
6. RADWASTE BUILDING
7. ADMINISTRATION BUILDING
8. PLANT ENTRANCE
9. CIRCULATING WATER PUMP HOUSE & INTAKE STRUCTURE
10. CWS COOLING TOWER
11. CIRCULATING WATER LINES
12. DIESEL GENERATOR BUILDING
13. FIRE WATER/CLEARWELL STORAGE TANK
14. FIRE WATER STORAGE TANK & PUMP HOUSE
15. TRANSFORMER AREA
16. SWITCHYARD
17. CONDENSATE STORAGE TANK
18. DIESEL GENERATOR FUEL OIL STORAGE TANKS
19. DEMINERALIZED WATER STORAGE TANK
20. BORIC ACID STORAGE TANK
21. HYDROGEN & NITROGEN STORAGE TANK AREA
22. TURBINE BUILDING LAYDOWN AREA
23. WASTE WATER RETENTION BASIN

NOTES:

1. LOCATIONS OF METEROLOGICAL TOWER. CW BLOWDOWN BASIN AND SEWAGE TREATMENT AREA ARE SITE SPECIFIC. SEWAGE TREATMENT PLANT MUST BE LOCATED DOWNWIND FROM MAIN PLANT.

IRIS - SITE PLOT PLAN
MULTIPLE, TWIN UNIT STUDY

Figure 6.3-2, IRIS, Two Twin-Unit Site Plot Plan

6.2.1 Independent Multiple Single Unit Arrangement

The three single unit arrangement (Figure 6.2-1) shows three independent IRIS reactor plants that are completely independent with their own non-safety related service water and main circulating water mechanical draft cooling towers. This arrangement is based on the assumption that the units would be constructed in series in a "slide-along" manner. The units would be started up in sequence as construction, pre-operation testing, fuel load, and startup testing are all completed for a unit. The right-most completed unit could be operated while construction of the subsequent left-most unit(s) is still in progress, by establishing a temporary exclusion zone between the operating unit(s) and the unit(s) under construction. This arrangement and construction sequencing is aimed at minimizing the construction time of a unit and at providing the utility with generating capability as soon as possible. Other advantages of this slide-along construction method are envisioned to be shorter construction time for the subsequent units by taking advantage of the experience of the work force. In order to accomplish this in-series construction, the units are spaced sufficiently apart so that the exclusion zone associated with the operating unit(s) can be established.

The overall site currently shown has a north – south dimension of 332 m (1090'), with the switch yards located north of the exclusion fencing. The site east-west dimensions are 489 m (1605') for the nuclear related exclusion area.

The IRIS single unit arrangement includes the following major building structures:

- Auxiliary Building (Item 1) – The IRIS auxiliary building encompasses the containment and shield structure as well as the fuel handling facilities and equipment and is founded on a common basemat with the containment/shield. It also contains typical auxiliary building features such as the main control room, steam and feed water piping and isolation valve room, safe shutdown control area, and all safety related equipment including batteries for electrical power. The auxiliary building occupies an area of 58 x 41 meters (190' x 135') and it extends from the basemat bottom at elevation -15 meters (-49'-2") to a roof elevation of +32 meters (105').

- Containment and Shield Structure (Item 2) – The IRIS spherical steel containment is 25 meters (82') in diameter and is surrounded by a cylindrical concrete shield building, which has an OD of 30 meters (98'-5"). This shield structure extends from the top of the basemat at elevation –13 meters (-42-8") to +13 meters (42'-8"). The latter is the elevation of the bottom of the refueling cavity.

- The Fuel Handling and Storage Area in the auxiliary building (Item 3) occupies most of the southern, above-grade, half of the auxiliary building. This area includes the refueling cavity above the reactor/containment closure head, the spent fuel pit, the cask loading and washdown pits, the refueling machine, the new fuel storage area, heavy lift overhead crane, laydown area, and a rail-car loading bay.

- Turbine Building (Item 4) – The IRIS turbine building contains all the equipment associated with the power plant steam and feed water systems and power generation equipment. It is a non-seismic building and contains no safety-related equipment. The turbine and generator have been sized based on the 1002 MWt (335 MWe) reactor power. The building dimensions are 80 X 36 meters (260' X 118').

- The Annex Building (Item 5) – The IRIS annex building is a non-seismic, non-safety related structure that houses access control for both the auxiliary and turbine buildings, health physics, technical support center, and non-safety related equipment. This building is constructed at grade and its dimensions are 84 X 15 meters (275' X 50').

6.2.2 Independent Multiple Twin-Unit Arrangement

The two twin-unit arrangement (Figure 6.2-2) shows two independent, twin unit reactors. This arrangement is aimed at maximizing shared components between the two reactors comprising one twin-unit, yet maintaining the ability to initiate operation of a completed twin-unit while construction of subsequent twin(s) proceeds in a "slide-along" manner. Each twin-unit is completely independent from the subsequent twin(s) and each reactor within a twin has its own turbine generator, condenser, and feed and steam systems, contained in a single T/G building with their own non-safety service water and main circulating water mechanical draft cooling towers. However, within a twin-unit, many systems, functions, and physical facilities are shared including: (back to back) control rooms, fuel handling area with refueling machine and spent fuel pit and cask loading facility, radwaste treatment, support systems, and switchyard. Within the twin-unit, separate safety grade power supplies, protection cabinets and switchgear, and electrical systems are maintained.

The overall site with two twin-units currently shown has a north – south dimension of 309 m (1015'), with the switch yards located north of the exclusion fencing. The site east-west dimensions are 482 m (1580') for the nuclear related exclusion area. The IRIS twin unit arrangement includes the following major building structures:

- Auxiliary Building (Item 1) – The IRIS twin-unit auxiliary building encompasses the two containment and shield structures as well as the shared fuel handling facilities and equipment and is founded on a common basemat. It contains typical auxiliary building features, including the shared back-to-back main control room, a steam and feed water piping and isolation valve room for each reactor, safe shutdown control areas, and all safety related equipment including batteries for electrical power. Separation between the safety related equipment for the two reactors is maintained throughout the building; this equipment can only be accessed via the main control room area. The twin-unit auxiliary building has a base of 60 x 70 meters (200' x 250') and it extends from the basemat bottom elevation of -15 meters (-49' 2") to a roof elevation of +32 meters (105').

- Containment and Shield Structure (Item 2) – Each IRIS reactor is located in its own spherical steel containment that is 25 meters (82') in diameter and is surrounded by a cylindrical concrete shield, which has an OD of 30 meters (98'-5") and extends from the top of the basemat elevation of -13 meters (-42' 8") up to the bottom of the refueling cavity at elevation +13 meters (42' 8"). The two reactor containment/shield structures are founded on the auxiliary building shared common basemat.

- The Fuel Handling and Storage Area (Item 3) occupies most of the southern, above grade, portion of the auxiliary building and includes a refueling cavity above each containment closure head, while all other capabilities are shared. They include the spent fuel pit, cask loading and washdown pits, refueling machine, new fuel storage area, heavy lift crane, laydown area, and rail-car loading bay.

- Turbine Building (Item 4) – The twin-unit IRIS has a single turbine building that contains all the equipment associated with the power plant steam and feed water systems and power generation equipment. It is a non-seismic building and contains no safety-related equipment. Separate turbine generators and steam and feed systems are provided for each reactor and each is sized based on the single 1002 MWth (335 MWe) reactor power. The building dimensions are 110 X 50 meters (361' x 164'). The TG building is arranged such that the two generators are facing and their electrical output equipment is centrally located. Two separate power conversion systems are provided in order to maintain the operation of one reactor while maintenance is performed on the unit's power conversion system.

- The Annex Building (Item 5) – The twin-unit IRIS annex building is a non-seismic, non-safety related structure that houses access control, health physics, technical support center, and non-safety related equipment. This building is constructed at grade and its dimensions are 60 x 15 meters (197' x 50') and extends to the turbine building.

6.3 GENERAL ARRANGEMENT

Preliminary general arrangement drawings (GAs) of the auxiliary building, including the fuel handling area, were made in order to establish the site plot plans described above. These GAs also provided a base-line for the overall arrangement of major equipment and for establishing separation requirements. In addition, these GAs established the fuel handling area arrangement, with its unique vertical refueling method and the location of the emergency heat removal system and its integration with the main steam and feedwater lines. As mentioned in section 6.1, this layout is preliminary and is expected to be optimized. Below is provided a brief description of the layout features for two elevations of the single unit IRIS auxiliary building, which illustrate the integration of the spherical containment, the cylindrical concrete shield structure surrounding the containment, and the fuel handling and refueling portion of the auxiliary building.

Elevation –13 m (-42' 8") – (see Figure 6.3-1) This is the lowest elevation of the IRIS auxiliary building, and has at its center the base of the concrete pedestal which supports the containment shell and its internal structures. This is surrounded by the base of the vertical, cylindrical, shield structure and shows equipment spaces formed under the containment shadow area. This elevation also illustrates that the entire auxiliary building is founded on a common basemat slab. This common basemat concept provides IRIS with as large and stable foundation as possible, which maximizes the plant's ability to withstand large uplift and overturn forces. The foundation depth of –15 m (-49' 2") is typical of a common plant site, assuring that the soil can support the plant weight while also assuring that the plant remains stable for postulated high water conditions.

This elevation is divided between the north and south halves, where the south half contains safety grade batteries and other equipment devoted to the safety grade DC electrical system. Note that two way separation is provided for the two divisions of safety equipment at this elevation. The remaining two divisions of similar equipment are located at the next higher elevation, with vertical separation provided. The north half is devoted mainly to non-safety related mechanical components including the liquid radwaste tanks, normal residual heat removal pumps, and the component cooling water pumps and heat exchangers. The south half contains all "clean" (non-radioactive) equipment, while the north half of the plant contain systems and equipment that would handle normally radioactive water. This split between "dirty" and clean sides of the building will be continued on subsequent elevations in order to establish personnel access requirements through the health physics areas.

Elevation +20.5 meters (67' 3") – (see Figure 6.3-2) This elevation is mostly occupied by the fuel handling area, with the +20.5 (67' 3") elevation being the operating floor elevation for the fuel handling and refueling activities. At the center of the building are the rails for the refueling machine, which can traverse over the reactor and the spent fuel pit and spent fuel cask loading area. This elevation also shows the refueling cavity over the containment at the center of the building. The flange for the containment vessel head is located at the +13.0 m (42' 8") elevation, which is the bottom of the cavity. The refueling cavity has room for the storage of the RV internals during the refueling operations. Two storage stands are provided for the upper internals and one stand for the lower internals. This cavity is ~7.5 m (24' 7") deep, which is sufficient to keep water above the irradiated fuel during its transfer between reactor vessel, the spent fuel pit, or the cask loading pit. This water depth is also sufficient to keep the long IRIS RV internals under water during the refueling operations. The refueling cavity is connected to the spent fuel pit, and the spent fuel pit is connected to the cask loading pit by narrow canals. These canals are normally closed by redundant gates that have seals pressurized with air. The gate seal air pressure as well as the water level between the gates is monitored and alarmed to assure that the gates maintain the water level in the spent fuel pool. The spent fuel is located in the lowest portion of the spent fuel pool, which cannot drain even in the event of the postulated failure of both redundant sealed gates.

The east side of the fuel handling area provides a large laydown area for storing the containment closure head and the reactor vessel upper head package. The building area over the containment and reactor closure heads, to and including the laydown areas, is traversed by a large 200 ton capacity over-head crane that is used to lift these large components. The over-head crane travel will be restricted such that it can not travel over the operating reactor. Also the handling of heavy loads by the over-head crane is restricted such that these loads cannot travel over the stored spent fuel or the new fuel storage area. Hatches are provided in the floor of this elevation to provide access to the rail car bay both for new fuel delivered to the site, and for the removal of loaded spent fuel casks.

This arrangement of having the over-head crane in the fuel handling area and this area also serving as the access to the reactor vessel allows the crane to be used for the installation and removal of the in-vessel components. They include the reactor coolant pumps and steam generators in addition to the reactor vessel internals. The large laydown area can be used to do equipment inspection and repairs out of the vessel. Also, equipment can be moved to and from this operating deck level to the rail car or to the grade elevation.

The north side of the building is used to house the refueling water storage tank, which contains the water needed to fill the refueling cavity above the reactor after the containment closure head and reactor vessel head have been removed. The tank also contain the Emergency Heat Removal Heat Exchangers that provide the safety grade means of removing heat from the reactor vessel via the in-vessel steam generators. Thus, this water serves the function of being the safety grade heat sink and therefore the tanks are vented to the atmosphere to allow them to steam to the environment in the event that extended heat removal from the reactor vessel is required. This elevation also shows the continuation of the steam and feed water penetration area vents, which continue up to elevation +25 m (82') where they vent to the environment. This portion of the building also contains the HVAC and other equipment for the main control room, which is located directly below.

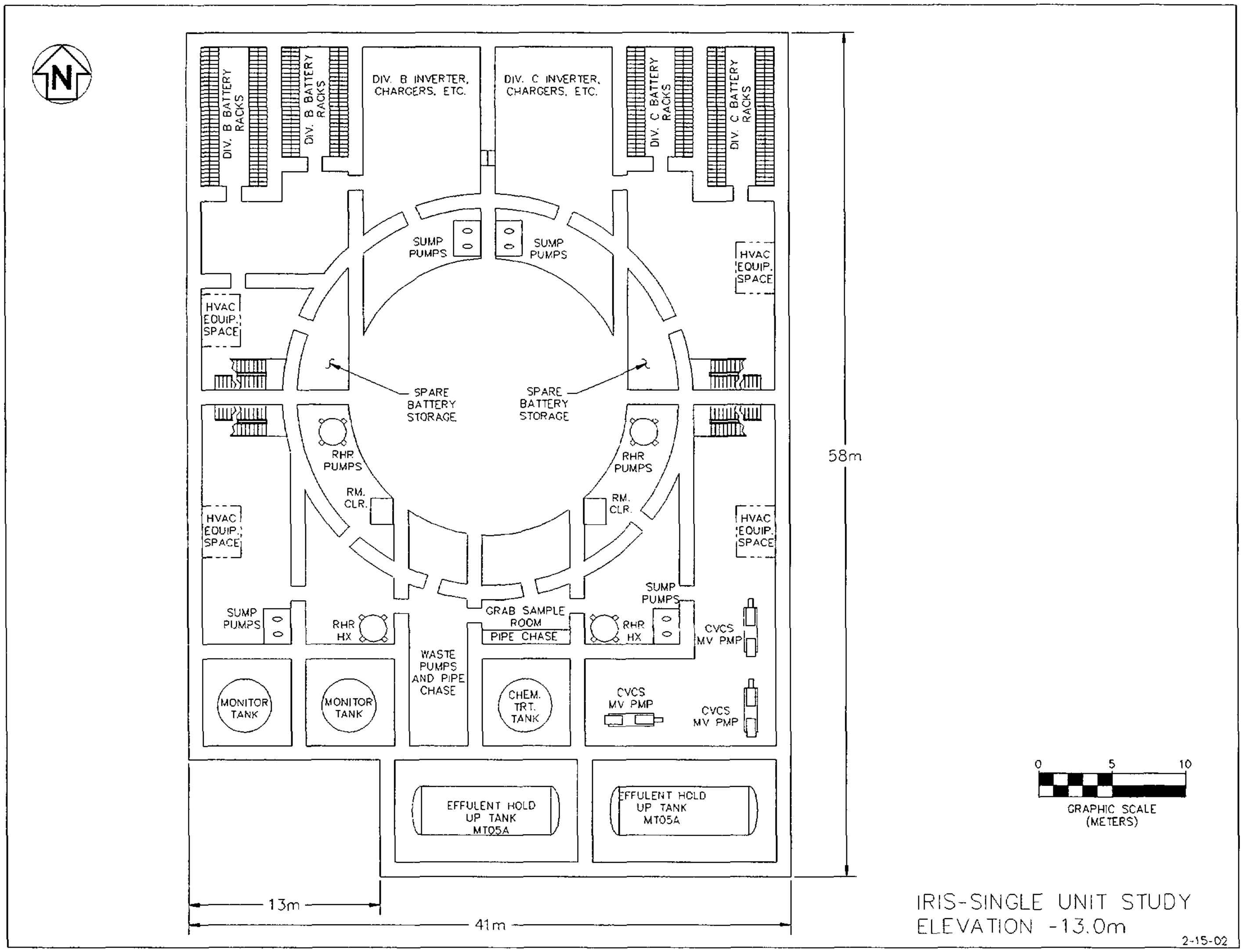


Figure 6.3-1, IRIS Auxiliary Building General Arrangement at Elevation -13 Meters

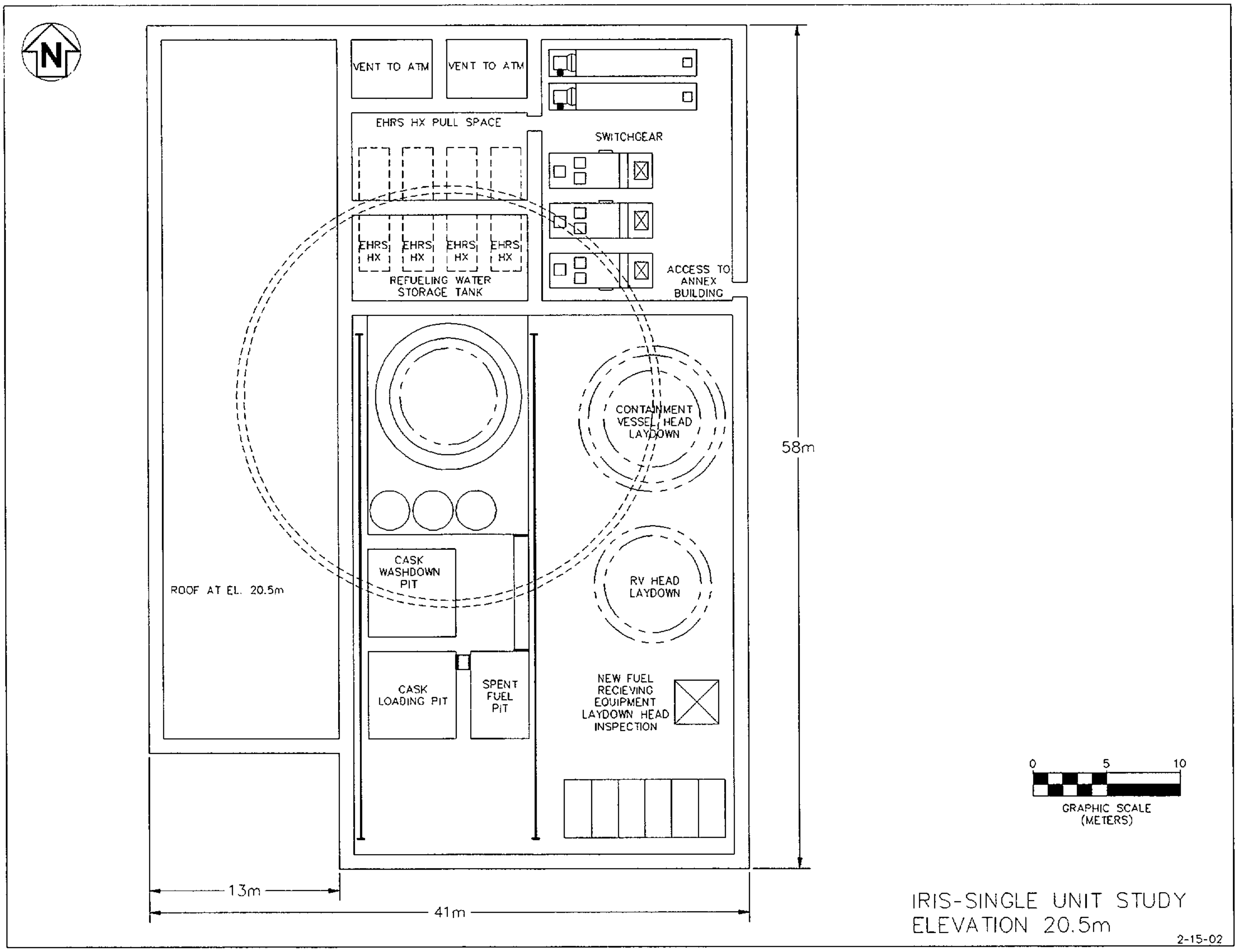


Figure 6.3-2, IRIS Auxiliary Building General Arrangement at Elevation +20.5 Meters

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7.0 OPERATIONS AND CONTROL CENTERS

The IRIS Operations and Control Centers Architecture is based on current Westinghouse PWRs experience, but IRIS specific details have not yet been defined.

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8.0 INSTRUMENTATION AND CONTROL SYSTEMS

The IRIS instrumentation and control systems will be similar in function to the instrumentation and control systems provided on the latest licensed Westinghouse plants. However, since the technology in this area of plant design is rapidly changing, it would be premature to specify these systems in detail at this early date. The detailed design of the IRIS instrumentation and control system will incorporate the latest technology available, and will address any IRIS specific features required, as well as licensing requirements established for small plant/multi-unit siting.

The functions of these systems include: to monitor the plant parameters sensed by the plant instrumentation, to automatically trip the reactor when needed, to automatically actuate the plant safety features when needed, to automatically control the plant normal operation, to provide data displays to the plant personnel, and to provide the ability to control the plant equipment. In current Westinghouse plants these functions are performed by the following major instrumentation and control systems:

- Protection and Safety Monitoring System (PMS)
- Plant Control System (PCS)
- Operation and Control Centers System (OCS)
- Data Display and Processing System (DDS)
- Diverse Actuation System (DAS)
- Special Monitoring System (SMS)

Provided below is an outline of the function of these systems.

8.1 PROTECTION AND SAFETY MONITORING SYSTEM

The Protection and Safety Monitoring System (PMS) has four principal functions:

- To provide an automatic reactor trip when plant conditions reach safety related limits.
- To actuate the Engineered Safeguards Features (ESF) to limit the consequences of any abnormal or accident condition.
- To provide information to the Data Display and Processing System and other monitoring systems so that these are able to alert the operator to any abnormal plant conditions and corrective action can be initiated.
- To provide data from protection grade sensors to the Plant Control System.

The PMS typically includes the following major components/features:

- Integrated protection cabinets
- Engineered safety features actuation cabinets
- Protection logic cabinets
- Qualified data processing cabinets
- Qualified data processing Input/Output (I/O) cabinets
- Qualified displays
- Reactor trip switchgear sensors
- Main control room and remote shutdown workstation multiplexers
- Main control room/remote shutdown workstation transfer panels

8.2 PLANT CONTROL SYSTEM

The Plant Control System provides for overall plant control during startup, ascent to power, powered operation, and shutdown. It provides the logic for the non-safety automatic and manual control of the nuclear steam supply system and the balance of plant functions that are operated from the main control room, or the remote shutdown area. The Plant Control System provides the instrumentation necessary to monitor the plant non-safety functions, and also provides signal conditioning of sensor signals and passes these signals to other instrumentation and control systems via the monitor bus.

The major control functions provided in the Plant Control System are:

- Rod Control (Reactor Power)
- Pressurizer Pressure and Level Control
- Steam Generator Feed Water Flow Control
- Steam Dump Control
- Individual control of non-Class 1E components (i.e., pumps, valves, fans, blowers, etc.)

The major PCS components/features typically include:

- Integrated Control System
- Rod Control System
- Rod Position Indication System
- Rod Drive Motor-Generator Set Control
- Pressurizer Heater Control Cabinet

8.3 OPERATION AND CONTROL CENTERS SYSTEM

See Section 7.

8.4 DATA DISPLAY AND PROCESSING SYSTEM

The Data Display and Processing System supports plant operations and plant performance analysis by providing the following functions:

- An Operational Display System to display plant status, plant parameters, and operational information in the main control room, onsite remote shutdown area, technical support center, and other required locations.
- An Alarm System to provide dynamic alarm management to prioritize alarms, eliminate nuisance indications, and in general organize and structure information presented to the operator
- A Distributed Computer System to analyze plant data to provide operational information to operators and other plant personnel, as well as logging and historical storage/retrieval functions.
- An Interactive Plant Procedures System to assist plant operators in executing normal operating procedures more efficiently and cost-effectively, by helping them access and follow procedures in an easy and logical manner.
- A Monitor Bus to function as a communications link among elements of the integrated instrumentation and control system architecture for transmission of plant parameters, plant status, displays and alarms, as well as to provide links to stand-alone Balance-Of-Plant instrumentation and control systems.
- Remote I/O cabinets to input non-safety signals to support special data collection requirements.

The components/features of the PPDS typically include:

- Operational Display System
- Plant Alarm System
- Distributed Computer System
- Interactive Plant Procedures System
- Monitor Bus (data highway system)
- Remote Data Acquisition

8.5 DIVERSE ACTUATION SYSTEM

The Diverse Actuation System (DAS) is a nonsafety-related system that provides a diverse backup to the Protection and Safety Monitoring System. This backup is to support the aggressive IRIS risk goals by reducing the probability of a severe accident caused by the unlikely coincidence of a postulated transient combined with common mode failure in the protection and control systems. The DAS functional requirements will be based on an assessment of the protection system instrumentation common mode failure probabilities combined with the event probability.

The Diverse Actuation System provides automatic actuation signals, manual actuation signals, and indications for the plant operators. The typical PWR diverse actuations include:

- Trip rods via the motor generator set,
- Trip the turbine,
- Initiate emergency heat removal,
- Actuate emergency boration,
- Trip the reactor coolant pumps,
- Initiate automatic depressurization and long-term gravity makeup,
- Start passive containment cooling,
- Isolate critical containment penetrations (those lines that connect directly to the reactor coolant system, the containment atmosphere, or the containment sump).

To support the diverse manual actuations, sensor outputs are displayed in the main control room by the Diverse Actuation System in a manner that is diverse from the Protection and Safety Monitoring System display functions.

8.6 SPECIAL MONITORING SYSTEM

The Special Monitoring System comprises the following systems, whose functions are self-explanatory:

- Metal Impact Monitoring System
- Acoustic Leak Monitoring System
- Core Barrel Vibration Monitoring System
- Sensor Response and Automatic Rod Drop Test Set

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9.0 ELECTRICAL SYSTEMS

The IRIS Electrical Systems Design will follow the same approach and design criteria used in other Westinghouse Advanced Passive Plants like AP600 and AP1000. A detailed design has not been completed at this moment, and only a short overview is provided in this report.

9.1 CLASS 1E DC AND UPS SYSTEM

The Class 1E DC and UPS system (IDS) provides 125 volts power for safety related and vital control instrumentation loads including monitoring and control room emergency lighting. It is required for safe shutdown of the plant during a loss of offsite power coincident with the generator breaker trip and during a design basis accident with or without concurrent loss of off-site power.

The Class 1E loads pertaining to each of the four protective divisions (A through D) are powered from their own ungrounded 125 VDC Class 1E battery subsystem located in the Auxiliary Building. The Class 1E DC batteries are sized to power Class 1E DC loads from the DC Motor Control Centers (MCC) and DC distribution panels and the loads from the Uninterruptible Power Supply (UPS) Inverter Subsystem. The UPS provides the vital Class 1E 120 VAC uninterruptible power for the plant instrumentation, control, and monitoring. The system includes:

- Safety grade components that are essential for safe shutdown of the plant.
- Radiation monitoring.
- Emergency lighting in selected areas of the plant.

9.2 NON-CLASS 1E DC AND UPS SYSTEM

The non-Class 1E DC and UPS system (EDS) is designed to supply non-Class 1E DC and AC loads for:

- plant instrumentation, control and monitoring functions during all plant operating modes,
- plant computer and operator display systems,
- plant investment and personnel protection,
- plant fire protection system.

The EDS is not required for safe shutdown of the plant.

9.3 MAIN AC POWER SYSTEM

The Main AC Power System (ECS) is non-safety-related and is not required to perform any safety function. It is designed to supply power to:

- The plant non-safety loads required during normal plant operation, startup, and normal shutdown.
- The low voltage 480 VAC distribution system used to supply power to the Class 1E battery chargers, 120 VAC vital instrument panel and emergency lighting loads.

9.4 MISCELLANEOUS ELECTRICAL SYSTEMS

The design of the miscellaneous electrical systems conforms to the EPRI ALWR Utility Requirements Document. The following systems are included in the Miscellaneous Electrical Systems:

- Plant Lighting Systems, composed of the Normal Lighting Subsystem, Emergency Lighting Subsystem and the Security Lighting Subsystem
- Grounding System and Lighting Protection (EGS)
- Cathodic Protection System
- Heat Tracing System

9.5 ONSITE STANDBY POWER SYSTEM

The Onsite Standby AC Power Systems (ZOS) is designed to supply AC power to the selected plant permanent loads in the event of loss of offsite power concurrent with the main turbine generator trip. The selected loads include:

- the Class 1E DC and UPS system,
- the electrical components of the plant defense-in-depth non-safety related systems that enhance orderly plant shutdown under emergency conditions and the other non-safety systems that provide personnel and plant investment protection.

Operation of ZOS is not required to ensure nuclear safety.

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10.0 HVAC SYSTEMS

The IRIS HVAC Systems will conform to the Plant Design Criteria (Section 1) as well as the EPRI ALWR Utility Requirements Document. A design of IRIS HVAC systems has not yet been performed. The experience acquired by Westinghouse in the design of other advanced passive plants shall be used as reference for the development of the IRIS HVAC systems design.

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11.0 AUXILIARY FLUID SYSTEMS

The IRIS auxiliary fluid systems are very similar to other PWR auxiliary systems, and in particular will rely on Westinghouse experience in the design of the AP600/AP1000. This section briefly outlines the systems that will be included in the IRIS design, and additional information will be provided once the design is completed.

11.1 COMPONENT COOLING WATER SYSTEM

The Component Cooling Water System (CCWS) is a non-safety-related, closed loop cooling system designed to transfer heat from various plant components to the Service Water System during normal phases of operation. The CCWS provides cooling water for the following functions:

- Decay heat removal via the Normal Residual Heat Removal System (NRHRS) heat exchangers
- Spent fuel pit cooling via the Spent Fuel Pit Cooling System (SFPCS) heat exchangers
- Chilled water cooling via the Chilled Water System chillers
- Reactor Coolant System letdown flow cooling via the Chemical and Volume Control System (CVCS) letdown heat exchanger
- Primary Sampling System sample heat exchanger cooling
- Liquid Waste Processing System reactor coolant drain tank heat exchanger cooling
- CVCS makeup pump miniflow heat exchanger cooling
- NRHRS pump motor cooling
- Compressed and Instrument Air Systems air compressor cooling

11.2 SERVICE WATER SYSTEM

The Service Water (SW) System is a nonsafety related closed circuit cooling water system that provides cooling for other plant closed loop systems using a wet cooling tower for heat dissipation to the atmosphere. The SW system provides cooling during several phases of plant operation. In addition, the SW system provides sufficient cooling to maintain the plant in hot standby condition following a loss of offsite AC power. Each pump is connected to a diesel-backed bus.

The system shares the water inventory, pump bays, and cooling tower with the circulating water system.

The SW system is designed to provide a reliable cooling system to various closed loop plant cooling systems. The SW system provides cooling water for the following components:

- The component cooling water (CCW) heat exchangers

- The turbine building closed cooling water (TCS) heat exchangers.

The service water system is designed to operate during normal operation, normal plant cooldown, blackout, and refueling.

11.3 CIRCULATING WATER SYSTEM

The plant Circulating Water System (CWS) is designed to provide cooling water to remove heat from the main condensers and from the components of the service water system under varying conditions of power plant loading and design weather conditions.

11.4 CENTRAL CHILLED WATER SYSTEM

The function of the Central Chilled Water System (VWS) is to provide chilled water to all air handling units of the plant HVAC systems. It also supplies chilled water to any other plant equipment requiring chilled water services. The Central Chilled Water System includes two subsystems, a high capacity subsystem and a low capacity subsystem. The low capacity subsystem is dedicated to the Nuclear Island Non-Radioactive Ventilation System (VBS).

11.5 FIRE PROTECTION SYSTEM

The fire protection system (FPS) is designed to provide fire detection, alarm and suppression. The FPS performs the following functions:

- To detect and locate a fire by fire zone and alarm promptly,
- To quickly suppress those fires which do occur, thereby minimizing the adverse effects of fire on structures, systems and components important to safety and to production of electricity,
- To provide manual backup to automatic fire suppression systems,
- To limit the spread of fires,
- To ensure that a sufficient number of the redundant electrical divisions necessary to achieve safe shutdown are free of fire damage,
- To minimize radioactive exposure to personnel and radioactive releases to the environment as a result of a fire.
- To provide a source of water for the non-safety grade Passive Containment Cooling System (PCCS).

11.6 CONTAINMENT HYDROGEN CONTROL SYSTEM

The Containment Hydrogen Control System (VLS) monitors and controls the hydrogen concentration inside containment following a design basis accident (DBA) or severe accident. Following an accident, hydrogen is principally produced by radiolysis of the core and sump solution and by reaction of the zircaloy fuel cladding with water. Since the IRIS containment is inerted, the VLS has no short-term functions, such as recombining released hydrogen inside the containment.

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12.0 RADIOACTIVE WASTE SYSTEMS

IRIS radioactive waste systems are very similar to other PWR radioactive waste systems, and in particular will rely on Westinghouse experience in the design of the AP600/AP1000. This section briefly outlines the systems that will be included in the IRIS design, and additional information will be provided once the design is completed.

12.1 LIQUID RADWASTE SYSTEM

The Liquid Radwaste System (WLS) performs the following major functions:

- Receives all liquid wastes that may contain radioactive materials.
- Removes gases from influents which have the potential of containing high levels of hydrogen or fission gases.
- Processes the liquid wastes to remove radioactivity.
- Discharges the processed waste to the environs in a controlled and monitored fashion.

12.2 GASEOUS RADWASTE SYSTEM

The Gaseous Radwaste System (WGS) is designed to provide for the collection of gaseous wastes generated during plant operation which are potentially radioactive and hydrogenated, and process and discharge the waste gas within acceptable limits.

12.3 SOLID RADWASTE SYSTEM

The solid radwaste system functions to collect and process wet and dry solid wastes originating in the radiologically controlled area including protective clothing, respirators and potentially clean wastes. The system is designed to maximize recovery and recycle and to minimize the volume of radioactive waste for disposal.

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13.0 MECHANICAL HANDLING SYSTEMS

A design of IRIS Mechanical Handling Systems has not yet been performed. The experience acquired by Westinghouse in the design of other advanced passive plants, and specific considerations related to the adopted integral layout shall be used as reference for the development of IRIS Mechanical Handling Systems design.

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14.0 WATER AND WASTE TREATMENT SYSTEM

14.1 DEMINERALIZED WATER SYSTEM

The Demineralized Water Transfer System (DWS) receives and stores deaerated reactor makeup grade water from the Makeup Purification System (MPS), maintains deaeration of the water, provides reactor makeup water by gravity feed to the Chemical and Volume Control System (CVCS) makeup pumps, and provides a pressurized supply of demineralized water for use upon demand for various purposes throughout the plant. It is a non-safety related system except for piping and valves associated with one containment penetration.

The demineralized water tank has an operating band of approximately 36,000 gallons. This is adequate to supply the CVCS makeup pumps for at least 6 hours under normal conditions and 3.3 hours under maximum makeup conditions. At a maximum expected usage rate of 200 gpm, three hours of demineralized water supply are available. A makeup pump in the MPS can refill the tank in about three hours. The DWS is designed to supply demineralized water by gravity feed to the CVCS makeup pumps at up to 180 gpm. The DWS is also designed to supply pressurized demineralized water for various service functions throughout the plant at up to 100 gpm.

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