AECL Technologies

CANDU 3
Pressurized Heavy Water Reactor
Licensing Review Bases
November 1990

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TABLES



FIGURES

## 1.0 INTRODUCTION

As discussed in the May 25, 1989 letter from AECL Technologies to the Chairman of the Nuclear Regulatory Commission (NRC), AECL Technologies intends to submit the CANDU 3 Pressurized Heavy Water Reactor (PHWR) design for certification in accordance with the provisions of 10 CFR Part 52. By letter dated July 6, 1989, the NRC requested AECL Technologies to develop and submit a licensing review bases document. The document should address scope, content and format of the safety analysis report (SAR) to be submitted, submittal schedules, brief descriptions of selected aspects of the design, key design parameters and proposed acceptance criteria. The NRC recognized that the SAR would be submitted in stages. The NRC suggested that each submittal contain complete information and the number of submittals should be kept small.

The CANDU 3 is the latest (450MWe) version of the PHWR system developed in Canada. It is a generic standard design of an essentially complete plant using an envelope of site conditions which is compatible with most potential U.S. sites. Proven technology is used throughout the CANDU 3. All key components (steam generators, coolant pumps, pressure tubes, fuel, on-line fueling machines, instrumentation, etc.) are essentially the same as those proven in service on operating CANDU power stations.

The CANDU 3 design evolved from other CANDU PHWRs, including the CANDU 6 plants operating and under construction, and has retained the many features that have contributed to their successful safety record and operating history. Particularly noteworthy among these features are safety characteristics which are unique to the CANDU technology, namely: the use of two fully capable and independent fast-acting shutdown systems, a moderator heat sink capable of maintaining core coolability under accidents where normal and backup heat sinks are assumed unavailable, rapid cooldown capability without pressurized thermal shock, shutdown cooling system capable of high pressure, high temperature operation and entirely located inside containment, and on-power refueling which minimizes excess reactivity at all times.

To these traditional CANDU characteristics the CANDU 3 design adds several enhancements aimed at improving safety, reliability, operability, and maintainability, and reducing project schedule cost.

The key design parameters are set forth in Chapter 5 of the CANDU 3 Technical Description. This document was previously submitted to the NRC by letter dated November 14, 1989.

The proposed acceptance criteria will be provided in the appropriate chapters of the SAR. These criteria will be based on current NRC requirements (Regulations) and guidance (Standard Review Plan, Regulatory Guides, Policy Statements, etc.). A large portion of NRC requirements and guidance are applicable to the CANDU 3 PHWR because most of its design features are similar to those of a pressurized light water reactor. A few CANDU 3 design features (e.g., reactor assembly and on-line refueling arrangement) have no direct counterpart at light water reactors or other reactors licensed in the U.S.; consequently, the NRC's guidance does not address these features directly. Proposed acceptance criteria for these features will be based on Canadian licensing requirements and proposed to the NRC. Some NRC requirements are deemed to be not relevant to the CANDU 3 Standard Design Certification (SDC) because they are related to specific designs, licenses, or sites. Such requirements are listed on Table 4-2.

This document describes the agreements to be reached between the NRC and AECL-Technologies regarding the Standard Design Certification (SDC) process, review schedule, review criteria to be applied by the NRC, format of submitted information, and the approach to disposition of the technical issues for which NRC positions have not yet been established.

This document does not reiterate the requirements of 10 CFR Part 52. It is understood that compliance with the applicable portions of Part 52 is mandatory. Therefore, this document does not discuss those sections of Part 52 that deal with fees, administrative review of applications, referral to the Advisory Committee on Reactor Safeguards (ACRS), issuance of standard design certification, duration of certification, and certification renewal provisions.

# 2.0 OVERVIEW OF THE CANDU 3 DESIGN

## 2.1 Design Description

The CANDU 3 Nuclear Generating Station shown in Figures 2.1-1 and 2.1-2 consists of all the structures, systems and components required to generate power, achieve and maintain safe shutdown conditions and to prevent or mitigate the consequences of accidents. The structures are comprised of: reactor building, reactor auxiliary building, turbine building, group 2 service building, group 1 service building, maintenance building, main (group 1) pumphouse, group 2 pumphouse, and administration building. The distribution of equipment and services among the buildings is primarily by function. All structures, systems and components have been grouped according to the design basis events for which they have safety importance. Those important for operational states are in Group 1; those important for accidents and external events are in Group 2. Further definitions of these groups of systems are given below.

The CANDU 3 layout results from detailed study and review of station safety, constructability, maintainability, and operability. Specifically, the station layout maximizes safety and facilitates the CANDU two-group approach. This layout also shortens the construction schedule by: simplifying, minimizing and localizing interfaces; accommodating many contractors without interference; eliminating construction congestion; providing direct access to all areas; providing flexible equipment installation sequences; and, by minimizing material handling requirements. The layout also facilitates station operation and maintenance.

The principal buildings are connected via the reactor auxiliary building through enclosed umbilicals and personnel access routes at grade elevation. The main control room is located in the reactor auxiliary building. The secondary control area is located in the group 2 service building.

The reactor building houses and supports the nuclear steam supply system components such as the reactor, the moderator system, and the heat transport system as well as parts of the safety systems. The reactor building, which provides an environmental boundary, a post loss-of-coolant accident pressure boundary, and biological shielding, is a principal component of the containment system.

The reactor building is a reinforced concrete structure composed of a base slab, a cylindrical perimeter wall, and a dome. The structure has a steel liner on the inside to provide leak tightness. All cables and piping pass through the containment boundary via pressure-retaining penetrations which are embedded in the reactor building perimeter wall below the reactor auxiliary building roofline (except for the main steam and feedwater lines).

The reactor building perimeter walls are separate from the building internal structures to eliminate interdependence between the containment wall and the internal structures. The internal concrete structures include the reactor vault walls, the steam generator support walls, the heat transport pump support walls, and intermediate floors. These walls and floors are sized to support all imposed loads and to provide shielding. The internal steel structures include three major steel floors and various steel structures providing equipment support, crane runway support, pipe restraint, walkways, and stairs.

The internal structure is designed to minimize personnel exposure to radiation while maximizing the access for test and maintenance of components.

A large portion of the reactor building is accessible when the reactor is operating, facilitating on-power maintenance, inspection, and testing. Shielding for personnel from steam generator radiation fields is provided by the concrete steam generator enclosure walls. Similarly, the reactivity mechanism floor, which is a concrete slab extending over the shield tank assembly, provides a shielded working floor for maintenance personnel.

There are two airlocks providing entrance/egress routes into the containment: An equipment airlock and an auxiliary personnel airlock.

Shielding doors located within the reactor building separate the reactor vault from the accessible fueling machine maintenance area.

Rooms containing potential heavy water leakage sources such as the fuelling machine and certain moderator system components, have controlled atmospheres. Doors within the reactor building have face seals, as required, to maintain isolation between the different reactor building atmospheres.

The reactor building crane, augmented by monorails and hoists, facilitates maintenance of equipment in the reactor building.

The major systems of the design are:

Reactor and Reactivity Control Systems

Moderator System and Moderator Auxiliary Systems

(Primary) Heat Transport System

Main Steam and Feedwater System

Containment

Reactor Shutdown Systems 1 and 2

Emergency Core Cooling System

Group 2 Feedwater System

Shutdown Cooling System

Fuel Storage and Transfer System

Electric Power Systems including the redundant Group 1 diesel generators, the redundant Group 2 diesel generators, and the switchyard

Instrumentation, Control and Monitoring Systems

Turbine Systems

Condensate Systems

Cooling Water Systems (Including Ultimate Heat Sinks)

Radwaste Systems

All systems in the CANDU 3 are assigned to one of two groups (group 1 or group 2). The systems within each group are capable of shutting the reactor down, cooling the fuel, and plant monitoring. Group 1 systems are those primarily dedicated to normal plant power production and coping with operational transients. The group 2 systems include safety and safety support systems. These maintain plant safety in the event of a loss or partial loss, of group 1 systems, and mitigate the effects of accidents and external events.

To guard against cross-linked and common mode events and to facilitate the comprehensive seismic design of the group 2 systems, the group 1 and group 2 systems are, to the greatest extent possible, located in separate areas of the station. The allocation of key systems to the two groups is shown in Figure 2.2-2.

All group 2 services, except for the low pressure emergency core cooling tank and the group 2 raw service water system, are totally accommodated within the group 2 service building and the group 2 portion of the reactor auxiliary building. All group 2 systems are seismically qualified and are protected from or hardened against the environmental consequences (pressure, temperature, humidity, and radiation) of accidents.

Group 1 services are placed in several locations, such as the turbine building, the group 1 pumphouse, the group 1 service building, and the group 1 areas of the reactor auxiliary building. In general, the group 1 areas are not seismically or environmentally qualified beyond local building code requirements.

Some exceptions to this are the main control room (which is tornado and seismically qualified to a sufficient extent to assure operator survival), the personnel route to group 2 service building, and the irradiated fuel bay. Other exceptions include the moderator system which is environmentally qualified to remain operable following a loss of coolant accident coincident with the unavailability of the emergency core cooling system, and the group 1 service systems (electrical power and cooling water supplies) which are protected from steam main breaks outside containment to the extent that these systems are required to mitigate the effects of such accidents.

Refueling operations are carried out on a regular basis with the reactor at full power. The number of fresh fuel bundles introduced into a channel is variable and the bundle shuffling pattern along a channel is flexible By adjusting the fuelling rate in various regions of the core, the power distribution in these regions is effectively controlled on a long-term basis.

Since power is not changed or interrupted for refueling, it is not necessary to tailor the refueling schedule to the utility's system load requirement.

The envelope of potential site characteristics used for this design certification are shown on Table 2-1. These characteristics are taken from the CANDU 3 design documents.

2.2 Design Objectives

The overall CANDU 3 design objectives are as follows:

a. To achieve safety, low radiation exposure to plant personnel, high capacity factor and ease of maintenance.

b. To provide a standardized plant design that is suitable for most U.S. sites without significant changes to design or documentation changes.

c. To employ state-of-the-art technologies, including design, construction, operation and project management technologies, consistent with construction in the 1990 to 2010 period.

d. To facilitate maintenance and in-service inspection. A short maintenance outage is planned every year, however components are designed to operate without major servicing for a minimum of two years, with minimal on-power maintenance. A special maintenance outage lasting up to 90 days is expected to be required no more frequently than every 20 years, for major equipment replacement, major system modernization, or major component refurbishing.

## 2.3 Operating Characteristics

The following is a summary of significant CANDU 3 operating characteristics:

* The unit is capable of sustained operation at a net electrical output of 100 percent of rated full power output.

* The overall plant control is normally of the reactor-following-turbine type.

* For power increases, the nuclear steam plant (NSP) portion of the plant is capable of maneuvering at a rate of 4 percent of present power per second in the range zero to 25 percent full power, and at 1 percent of full power per second in the range 25 percent to 80 percent of full power, and at 0.15 percent of full power per second in the range 80 percent to 100 percent of full power.

  The overall plant maneuvering rate is a function of turbine design and is typically 5 to 10 percent of full power per minute.

* During normal plant operation, assuming an initial power of 100 percent, the xenon load at a steady level, and a normal flux shape, the reactor power may be reduced to 60 percent of full power at rates of up to 10 percent of full power per minute. The power may be held at that new lower level, indefinitely. Return to high power (80 percent) can be accomplished within 60 minutes, or less, depending on the degree and duration of the power reduction. In most cases, a maximum of four hours is required to return to 98 percent of full power from 80 percent of full power.

* In the event of a temporary or extended loss of transmission line(s) to the grid, the unit can continue to run and supply its own power requirements.

  The turbine bypass system to the condenser is capable of accepting the entire steam flow during a reactor power setback following loss of transmission line or turbine trip. The steam flow is initially 100 percent, but decreases to a steady state value in the range of 60 percent after several minutes.

* The unit is capable of reaching 100 percent net electrical output, from a cold shutdown in about ten hours. If the pressurizer is at its normal operating temperature and pressure and the xenon level in the fuel is low, the unit is capable of reaching 100 percent

electrical output from a cold shutdown within three hours. These time intervals are for the nuclear steam supply system and may be extended by the turbine generator requirements, depending on the turbine design.

* The reactor and turbine are controlled by computer from zero to 100 percent of full power.

* Following a shutdown from sustained full power operation with equilibrium fuel, the reactor can be restarted within 25 minutes and returned to full power operation.

## 2.4 CANDU 3 Unique Technical Characteristics

AECL Technologies has provided the NRC with a report that identifies significant unique aspects of CANDU 3 to identify those areas on which the NRC may wish to concentrate in the early stages of the review. These are:

1. Reactor Physics
2. Reactivity Coefficients
3. Separation of Reactivity Devices for Control and Shutdown
4. Fuel Design
5. On-power Fueling
6. Control of Heavy Water and Tritium
7. Seismic Design
8. Protection Against Common Mode Events
9. Heat Sinks
10. Reactor Coolant Pressure Boundary
11. Classification of Pressure Retaining Systems
12. Codes and Standards
13. Safety Analysis
14. Computer Codes Used in Design and Analysis
15. Electric Power
16. Computer Control
17. "Two Group Approach"

Because of the unique technical aspects of the CANDU 3 design, the NRC staff anticipates that some new regulatory guidance will be necessary and that some regulatory guidance will be modified. For the purposes of licensing in the United States, the NRC staff considers the CANDU 3 to be an advanced reactor as defined in 10 CFR Part 52. The NRC staff does not anticipate that a prototype will be required because the design is based on proven heavy water technology (See Canadian CANDU 3 Design Certification (SECY-89-350) and Presenting Views on Prototype Requirements on CANDU 3 Design (SECY-90-133)). The advanced reactor category was assigned, by the NRC, recognizing that for them the CANDU technology is new. However, AECL Technologies is providing the CANDU 3 as an evolutionary plant based on proven operational CANDU experience.

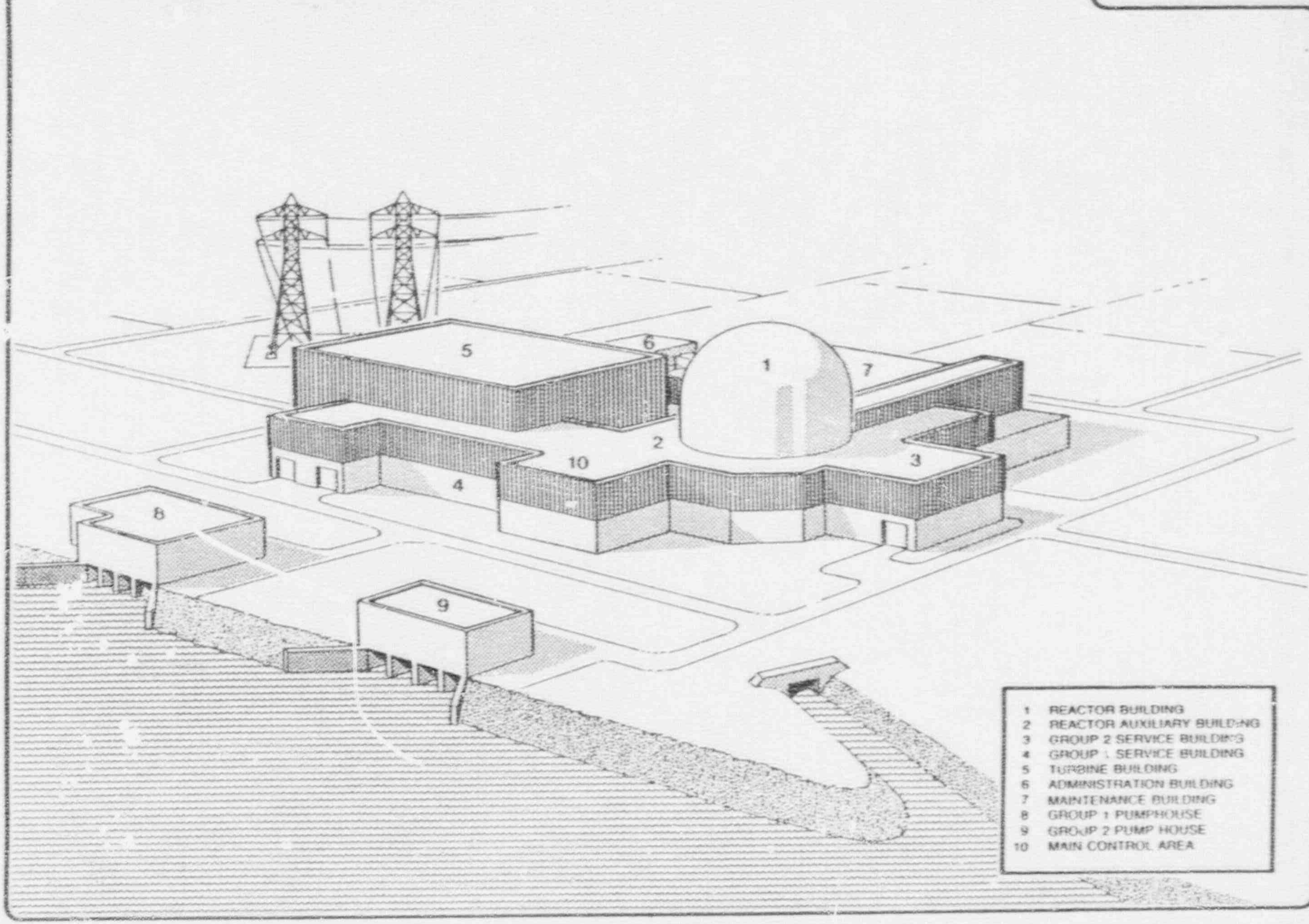
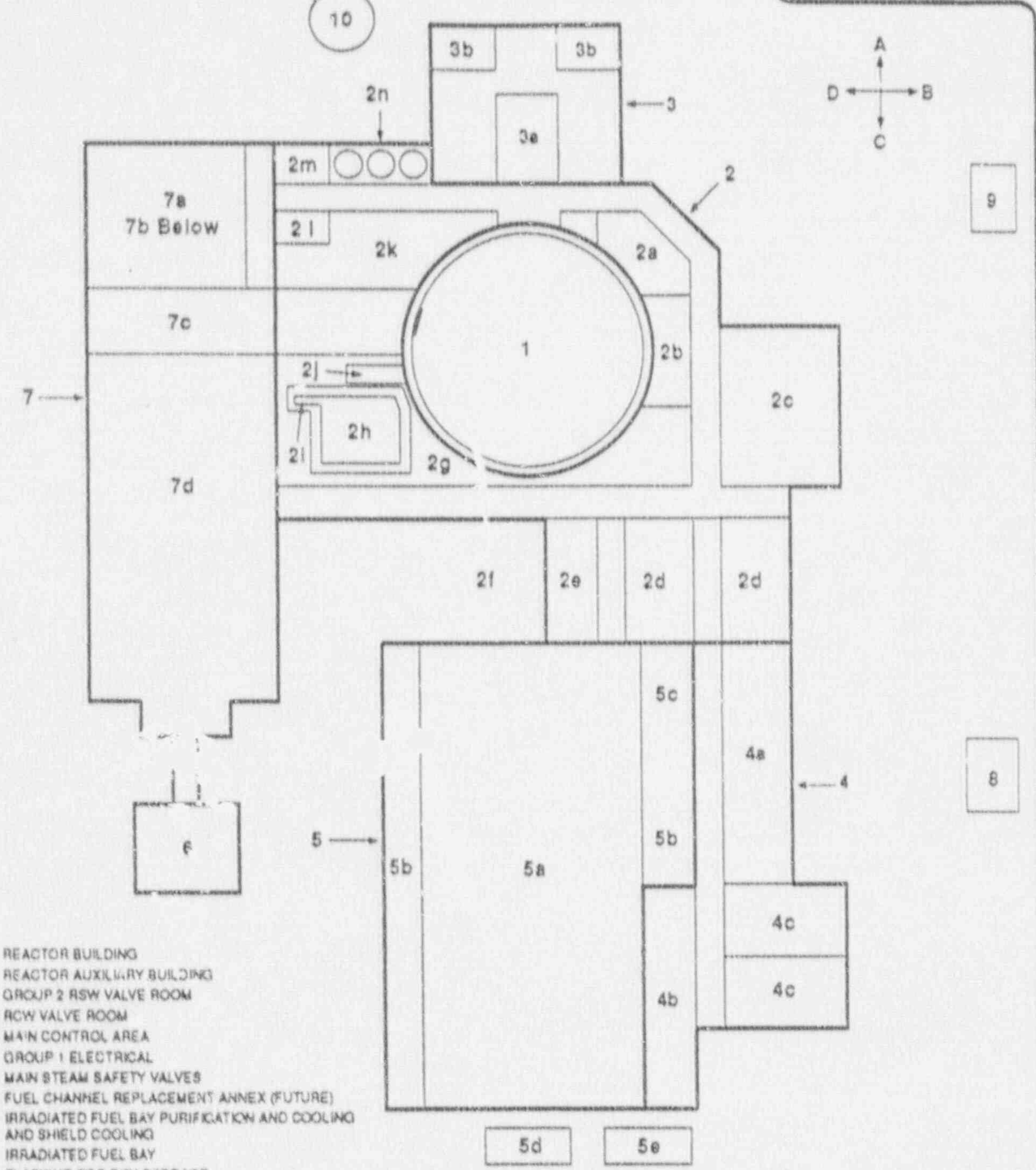



FIGURE 2.1-1 STATION LAYOUT

CANDU 3




1 REACTOR BUILDING
2 REACTOR AUXILIARY BUILDING
2a GROUP 2 RSW VALVE ROOM
2b RCW VALVE ROOM
2c MAIN CONTROL AREA
2d GROUP 1 ELECTRICAL
2e MAIN STEAM SAFETY VALVES
2f FUEL CHANNEL REPLACEMENT ANNEX (FUTURE)
2g IRRADIATED FUEL BAY PURIFICATION AND COOLING AND SHIELD COOLING
2h IRRADIATED FUEL BAY
2i FLASKING FOR DRY STORAGE
2j NEW FUEL LOADING
2k ECC HEAT EXCHANGERS AND PUMPS
2l GROUP 2 FEEDWATER PUMPS
2m GROUP 2 FEEDWATER TANK
2n HIGH PRESSURE ECC TANKS
GROUP 2 SERVICE BUILDING
SECONDARY CONTROL AREA
GROUP 2 DIESELS
GROUP 1 SERVICE BUILDING
4a GROUP 1 ELECTRICAL
4b RCW PUMPS, HEAT EXCHANGERS
4c GROUP 1 DIESELS
5 TURBINE BUILDING
5a TURBINE HALL
5b TURBINE AUXILIARY BAYS
5c DEAERATOR
5d MAIN OUTPUT AND UNIT SERVICE TRANSFORMERS
5e STATION SERVICE TRANSFORMER
6 ADMINISTRATION BUILDING
7 MAINTENANCE BUILDING
7a DECONTAMINATION AREA
7b LIQUID WASTE TREATMENT AND $D_2O$ CLEANUP
7c CRANE HALL
7d CHANGE ROOMS, WORKSHOPS, LABS, STORES
8 GROUP 1 PUMP HOUSE
9 GROUP 2 PUMP HOUSE
10 LOW PRESSURE ECC WATER TANK

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FIGURE 2.1-2 STATION LAYOUT- PLAN

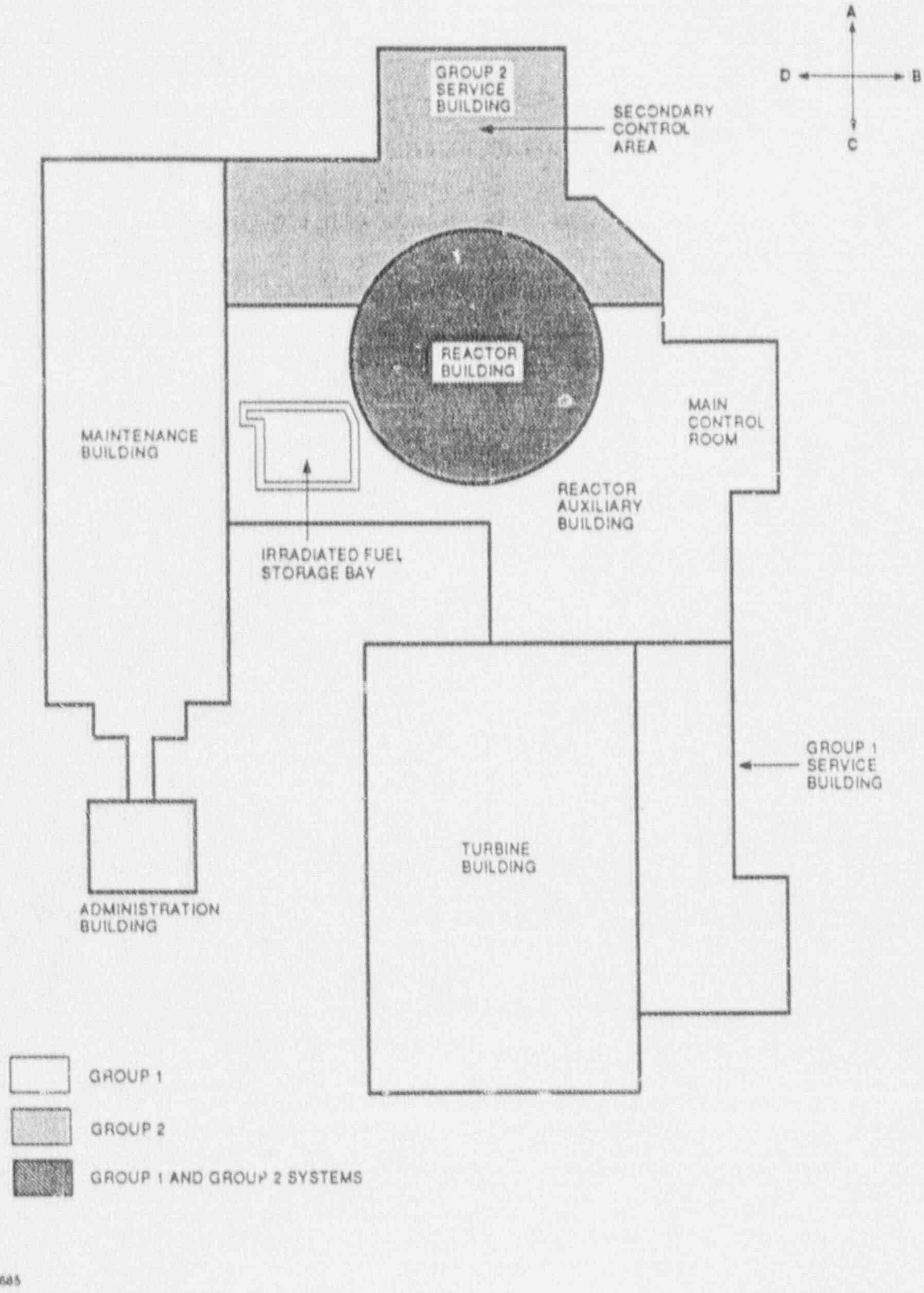


FIGURE 2.1-3 TWO-GROUP SEPARATION PHILOSOPHY

TABLE 2-1.A

SUMMARY OF GENERIC SITE PARAMETERS

| APPLICABLE DESIGN AREAS/FEATURES | VALUES OF GENERIC SITE PARAMETER |
|---|---|
| 1. Seismic Design(1)(2) | |
| (i) of safety related buildings, modules, systems and equipment | DBE = 0.3g Peak Horizontal Acceleration<br>SDE = 0.15g Peak Horizontal Acceleration |
| (ii) of non-safety related buildings, systems and equipment | Zone 3 of the 1985 National Building Code of Canada (Zonal Ratio of 0.15) |
| 2. Building Structural Design and Analysis | - Soil/rock shear modulus of (5 to 100) x $10^3$ kg/$cm^2$<br>- Max. Design Temperature<br>2 1/2% dry 46° C<br>2 1/2% wet 40° C<br>- Min. Design Temperature<br>2 1/2% -20° C<br>1% -32° C<br>- Wind Speed (1/100)<br>50 m/s<br>- Design Basis Tornado (DBT) parameters (See Table 2-1.C)<br>- Rain 15 minute rainfall 36 mm<br>24 hour rainfall 150 mm<br>- Snow Load 2.7 kPa<br>- Humidity up to 85% |
| 3. Heating, Ventilation and Air Conditioning Design (3) | - Max. Air Temperature (cold site/warm site)<br>2 1/2% dry 30° C/46° C<br>2 1/2% wet 22° C/40° C<br>- Min. Air Temperature (cold site/warm site)<br>2 1/2% -29° C/3° C<br>1% -32° C/0° C |

*Notes (1), (2), and (3)--See notes on page 16

TABLE 2-1.A (Continued)

SUMMARY OF GENERIC SITE PARAMETERS

| APPLICABLE DESIGN AREAS/FEATURES | VALUES OF GENERIC SITE PARAMETER |
|---|---|
| 4. Layout and Structure size. The weight and size of equipment used (i) for space requirement, access requirements for maintenance, and (ii) building size and floor loading design. | Use sizes and weights for:<br>- Larger 50 Hz Motors ***<br>- Additional space for 380V and 220V cable trays and conduits.<br>- Larger heat transfer equipment based on cooling water temperature(4) of<br>RCW* 35° C<br>RSW** 30° C |
| 5. Standard plant design including system, equipment, documentation, analysis and CADDS model | - Cold site<br>- Cooling Water Temperature<br>Min. Max.<br>RCW 5° C 22° C<br>RSW 1° C 17° C<br><br>Electrical Voltage and Frequency(5)<br>A.C. 24,000 V Main Generator<br>13,800 V<br>4,160 V<br>600 V<br>208 V<br>120 V<br><br>D.C. 250 V<br><br>Frequency 60 Hz |

* RCW Recirculating Cooling Water
** RSW Raw Service Water
*** Conservative Space-Weight Allocation

Notes (4) and (5)--See page 17

TABLE 2-1.B

GEOTECHNICAL DESIGN PARAMETERS

| | |
|---|---|
| Stratigr.phy | Uniform homogeneous elastic half space, competent rock at a depth of 2m |
| Foundation Medium Properties: | |
| Shear Modulus | (5 to 100) x $10^3$ kg/$cm^2$ |
| Poisson's Ratio | 0.3 to 0.4 |
| Unit Weight (wet) | 2.0 to 3.0 g/$cm^3$ |
| Allowable Bearing Capacity | 10 kg/$cm^2$ (Static) |
| Groundwater Level | At the ground surface |
| Granular Backfill Material Properties: | |
| Unit Weight wet | 2.1 g/$cm^3$ |
| saturated | 2.2 g/$cm^3$ |
| submerged | 1.2 g/$cm^3$ |
| Angle of Internal Friction | 30 degrees |

TABLE 2-1.C

## DESIGN BASIS TORNADO CHARACTERISTICS

### WIND CHARACTERISTICS

a. Maximum windspeed: 420 km/h (260 mph)

b. Translational windspeed: 92 km/h (57 mph)

c. Rotational wind radius: 138 m (453 ft)

d. Maximum pressure drop: 10 kPa (1.46 psi)

### Tornado Missile Spectrum

(Based on a maximum horizontal windspeed of 420 km/h)

| Missile | Mass (kg) | Dimensions (m) | Velocity | |
|---|---|---|---|---|
| | | | km/h | m/s |
| A. Automobile* | 1810 | 5 x 2 x 1.3 | 162 | 45 |
| B. Utility Pole* | 510 | 0.343 dia x 10.68 | 137 | 38 |
| C. Steel Pipe (12 inch) | 340 | 0.32 dia x 4.58 | 50 | 14 |
| D. Steel Pipe (6 inch) | 130 | 0.168 dia x 4.58 | 94 | 26 |
| E. Steel Rod | 4 | 0.025 dia x 0.915 | 86 | 24 |
| F. Wood Plank | 52 | 0.092 x 0.289 x 3.66 | 223 | 62 |

* Maximum altitude: 9 m above site grade level
Vertical Velocity: 70% of horizontal velocity shown above

Table 2-1.A-C Notes

(1) Seismic Design

A complete list of structures and systems to be seismically qualified to these earthquakes, along with the definition of the design basis earthquake (DBE) and the site design earthquake (SDE) ground response spectra and the description of methods for seismic design and analysis, will be provided in the CANDU 3 Safety Analysis Report.

The proposed design earthquake levels reflect the current international practices of regulatory agencies, potential client requirements and are in line with other vendors' approaches. In addition, they represent a reasonable balance between the increase in the capability of the design against the increase in plant cost. The standard CANDU 3 plant when designed to these earthquake levels would be suitable for all areas of low to medium seismic activities.

For the design of all non-safety related buildings, systems and equipment supports, a design earthquake level associated with zone 3 of the 1985 National Building Code of Canada (zonal ratio of 0.15) are used. The seismic design and analysis of non-safety related structures and systems are in accordance with "Seismic Design Requirements-Application of the National Building Code of Canada."

(2) Geotechnical Parameters

The geotechnical design parameters, necessary for the seismic analysis of the plant and the design of the foundations of the structures for the standard CANDU 3, are given in Table 2-1.B.

(3) Cold Site and Warm Site Parameters

For the analysis and design of the buildings, an envelope of both Cold Site and Warm Site temperatures are used.

For the design of the heating, ventilation and air conditioning systems, the CANDU 3 Standard Product design are based on Cold Site conditions. However, space allocation for equipment installation and maintenance as well as building design are suitable to accommodate equipment and systems designed for either Cold or Warm Site conditions.

Equipment is sized and space assigned for a hot water heating system (supplemented by electric heaters) capable of maintaining suitable internal temperatures for the minimum air temperature defined in Table 2-1.A. For adaptation to a warm site, the hot water heating system will be deleted and local electric heating units will be used if required.

(4) Cooling Water Temperatures

For the Standard Product design of the CANDU 3 plant, the system and equipment design are based on the Cold Site conditions. However, for space allocation, equipment installation, maintenance access and floor loading for structural design and analysis, equipment sizes and weights based on Warm Site conditions are used.

For the Standard Product design of the CANDU 3 plant, a salt water site is assumed. The condenser cooling is based on a maximum cooling water temperature rise of 14 degrees C.

(5) Electrical Parameters

Allowances are made in the Standard Product design for larger cables and consequently larger conduits, more cable trays and additional switchgears associated with 380V and 220V designs to facilitate conversion to sites with these voltage levels.

## 3.0 STANDARD DESIGN CERTIFICATION REVIEW

AECL Technologies will submit an application for Standard Design Certification (SDC) in accordance with the requirements of Subpart B -- "Standard Design Certification" of 10CFR52.

The SDC will be accomplished in three major phases: (1) the Technology Background Phase, (2) the Review Phase, and (3) the Design Certification Rulemaking.

### 3.1 Technology Background Phase

This phase is ongoing It will continue until AECL Technologies completes its application for Standard Design Certification.

The purpose of this phase is to (1) identify design issues and NRC policy issues, (2) establish an approach for resolving such isues, (3) formalize agreements between AECL Technologies and NRC on the key ground rules for the review, and (4) establish the application submittal schedule.

Design and policy issues may be identified during the review of the previously submitted CANDU 3 Conceptual Design. These issues may result because:

A. Some CANDU 3 features that meet NRC requirements are different from those described in NRC guidance.

B. Some CANDU 3 features have never been reviewed by the NRC staff.

C. In Evolutionary LWR Certification Issues and Their Relationships to Current Regulatory Requirements (SECY-90-16), the NRC staff identified several issues that go beyond 10CFR52 requirements. Other such requirements may be identified prior to the CANDU 3 application.

A process for identifying, tracking and resolving the design and policy issues will be established. As a minimum, an approach to resolving each issue will be established, prior to establishing the application schedule.

Because the CANDU 3 is presently in the Canadian licensing process, the NRC certification of the CANDU 3 will follow the Canadian licensing by several months. Issues will be raised in both the Canadian and the USA review. Issues raised in Canada will be resolved in Canada before they are addressed in the USA. Issues raised in the USA which are related to NRC requirements will be resolved on a schedule that is consistent with AECL Technologies request for the SDC.

The application submittal schedule will be established after the design and policy issues have been identified.

AECL Technologies will provide NRC with technical reports describing the technology and acceptance criteria for CANDU design features and analytical techniques germane to the CANDU PHWR not presently addressed in NRC guidance.

A list of these reports and their submittal dates are provided in Table 3-1.

NRC will provide timely review and evaluation of these reports and document their comments on the technology and on the approach proposed to resolve the issues.

3.2 Review Phase

This phase is initiated by the formal submittal of a licensing review basis document (LRBD).

Within appropriate Safety Analysis Report (SAR) chapters, AECL Technologies will discuss the CANDU 3 design principles and will provide a comparison of the CANDU 3 design with the licensing guidance deemed applicable to CANDU 3, e.g., General Design Criteria, regulatory guides, standard review plan, and generic letters. If alternatives are proposed to NRC guidance, information will be provided to justify an equivalent level of safety.

AECL Technologies will establish the CANDU 3 compliance with applicable NRC requirements.

NRC will provide draft Safety Evaluation Reports (SERs) during the review process to identify the issues that require resolution.

As part of the application, AECL Technologies will submit a design specific probabilistic safety assessment (PSA). Since all key components in CANDU 3 are essentially the same as those proven in service on operating CANDU power stations, it will be possible to reduce the uncertainties associated with system/component reliability data over more generic data commonly used. During the design of CANDU 3 the system/component reliability data will be incorporated into the design process to provide reliability design goals. This PSA will provide the NRC with an additional perspective in the resolution of safety issues to confirm the acceptability of the final design. NRC will complete the design review and issue its Safety Evaluation Report.

The result of the application review will be design acceptance in the form of a Final Design Approval.

3.3 Desi... ...tification Rulemaking

The Commission will initiate the design certification rulemaking after an application has been filed. The rule will be issued in accordance with the provisions of Subpart H of 10 CFR Part 2 as supplemented by the provisions of 10 CFR Part 52, §52.51. The specific procedures are yet to be determined by the Commission.

3.4 Application

3.4.1 Scope

AECL Technologies will comply with the requirements of 10 CFR Part 50, Appendix O regarding application for Final Design Approval (FDA) of the CANDU 3 PHWR design. The CANDU 3 PHWR design consists of all major structures and equipment required to generate power, achieve and maintain safe shutdown conditions and prevent or mitigate the consequences of accidents. The structures and systems provided are described in section 2 of this document.

The application will delineate those portions of the plant for which the application seeks certification. For such portions of the plant, a complete detail design will be provided. For those portions of the plant for which the application does not seek certification, the design will be described in sufficient detail to permit the staff's review of the final safety analysis report, probabilistic safety assessment, and interface requirements.

Interface requirements will be defined for those portions of the plant which are site dependent. Reference site parameters which constitute essential input to the standard product design, will also be specified.

3.4.2 Format

AECL Technologies will submit the information identified in 10 CFR Part 52, §52.47. Technical information will be organized in accordance with Regulatory Guide 1.70, Revision 3, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants." This information will be supplemented by topical reports and other submittals as required to address all the information requirements of §52.47. Where possible, AECL Technologies will reference, within the information format of Regulatory Guide 1.70, existing reports and documents that provide the necessary information.

AECL Technologies will provide an evaluation of the CANDU 3 standard design against the Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, NUREG-0800, in accordance with 10 CFR Part 50, §50.34 (g)(1)(ii).

### 3.4.3 Metric Units

The technical information discussed above will employ metric units with the corresponding English units, in parentheses, following the metric value. AECL Technologies will also provide appropriate tables to convert from one system of units to the other.

### 3.4.4 Information Form

AECL Technologies will provide a copy of the SAR on a diskette suitable for use on an IBM (or compatible) personal computer (except for drawings and graphs that are not amenable to such portrayal).

AECL Technologies will provide the requisite number of hard copies of the SAR specified in 10 CFR Part 50, §50.30 (a), (c)(1) and (3).

TABLE 3-1
TECHNICAL BACKGROUND
SUBMITTAL SCHEDULE

| Title | Submittal Date |
| --- | --- |
| List of CANDU Documents Requested by NRC | 7/8/89 |
| Unique Aspects of the Technical Characteristics of CANDU 3 | 7/27/89C |
| CANDU 3 Technical Outline | 7/27/89C |
| CANDU 3 Conceptual Safety Report (Vols. 1 & 2) | 8/29/89C |
| Canadian Codes & Standards (Canadian Standards Association) (2 vols) | 8/29/89C |
| CANDU 3 Conceptual Probabilistic Safety Assessment | 11/14/89C |
| CANDU 3 Technical Description (Vols. 1 & 2) | 11/14/89C |
| Atomic Energy Control Board Regulations and Supporting Documents Applicable to CANDU 3 | 11/14/89C |
| Operating Policies and Principles - CANDU 6 at Point Lepreau (Unit 1) | 11/14/89C |
| CANDU 6 Probabilistic Safety Study Summary (July 1988) | 12/19/89C |
| CANDU Fuel Channel Technology (Abstract) | 2/7/90C |
| CANDU LOCA Analysis Technology (Abstract) | 10/90C |
| CANDU 3 Licensing Review Basis Document (LRBD) | 1/91C |
| CANDU Fuel Channel Technology (Report) | 2/91 |
| CANDU Shutdown Systems Technology (Report) | 3/91 |
| CANDU On-Power Fueling Technology (Report) | 2/91 |
| CANDU LOCA Analysis Technology (Report) | 3/91 |

* C = completed action

4.0 NRC STAFF REVIEW

The NRC staff will review the application for compliance with the standards set out in 10 CFR Part 20, Part 50 and its Appendices, Part 73 and Part 100 as they apply to applications for construction permits and operating licenses for nuclear power plants and are technically relevant to the CANDU 3 design. The portions of these requirements that AECL Technologies believes are not technically relevant are listed on Table 4-2.

The Standard Review Plan (SRP) is the basic document to be used by the NRC staff in the SDC review of CANDU 3.

The staff will follow its review procedures in the SRP, supplemented and modified as follows:

(1) The CANDU 3 SAR is to be submitted in six groups of chapters, over a period of about 30 months. Correspondingly, the staff SER will be issued in draft form, in sections in accordance with the schedule shown in Table 4-1. Draft SER sections will be made publicly available.

(2) At the completion of the review of the individual SAR chapters, the staff will perform an integrated review of the application. This review will include the review of the CANDU 3 PSA. The PSA will provide the basis and perspective to evaluate broad issues of reactor safety and bring these issues to closure, as well as to assess the overall acceptability of the design. The staff will issue a composite final SER in accordance with the schedule.

(3) Each draft SER section will contain a description of open or unresolved issues that may be identified early in the review process, but which cannot be resolved until the completion of the review of later chapters. In addition, with the submittal of each chapter of the SAR, AECL Technologies will provide an updated check-list which identifies outstanding issues and the future chapter(s) in which resolution is anticipated.

(4) Each draft SER will contain a target schedule for closing outstanding SER issues that is compatible with the target FDA decision date.

Certain design features of the CANDU 3 are sufficiently different from existing U.S. nuclear plants that early interaction between the plant designers and the NRC staff would improve the efficiency of the SDC review. We propose one or more Design Review Meetings, following each submittal. Each meeting will be conducted using the process discussed in Use of Independent Design Reviews in the Regulatory Process (SECY-81-161) of March 12, 1981. The record of each of these meetings will form part of the basis for the draft SER to be issued by the NRC staff.

For each Design Review Meeting, the staff will provide a timely identification of issues that require resolution to permit the review to progress in an orderly manner.

TABLE 4-1
CANDU 3 REVIEW SCHEDULE

| Review Element (SAR Chapter/Appendix) | AECL T Submittal | NRC SER Complete | Cumulative Months |
|---|---|---|---|
| SAR Submittals | | | |
| Submittal No. 1 | 0 | 6 | 6 |
| 1. Intro & Gen'l Description of plant | | | |
| 2. Site Characteristics | | | |
| 3. Design of Structures, Components & Sys. | | | |
| 4. Reactor | | | |
| Submittal No. 2 | 6 | 12 | 12 |
| 5. Reactor Coolant System & Connected Systems | | | |
| 6. Engineered Safety Features | | | |
| Submittal No. 3 | 12 | 18 | 18 |
| 7. Instrumentation and Controls | | | |
| 8. Electric Power | | | |
| 9. Auxiliary Systems | | | |
| Submittal No. 4 | 18 | 24 | 24 |
| 10. Steam and Power Conversion | | | |
| 11. Radioactive Waste Management | | | |
| 12. Radiation Protection | | | |
| 15. Accident Analysis | | | |
| Submittal No. 5 | 24 | 30 | 30 |
| 13. Conduct of Operations | | | |
| 14. Initial Test Program | | | |
| 16. Technical Specification | | | |
| 17. Quality Assurance | | | |
| 18. Human Factors Engineering | | | |
| 19. Interfaces | | | |
| Appendices | | | |
| C. Comparison with Standard Review Plan | | | |
| D. CANDU 3 Inservice Inspection Program | | | |
| E. CANDU 3 Fire Hazards Analysis | | | |

TABLE 4-1
CANDU 3 REVIEW SCHEDULE

| Review Element (SAR Chapter/Appendix) | AECL T Submittal | NRC SER Complete | Cumulative Months |
| --- | --- | --- | --- |
| Submittal No. 6 | 30 | 36 | 36 |
| A. Probabilistic Safety Assessment/Failure Modes and and Effects Analysis | | | |
| B. Proposed Resolution of Generic Safety Issues | | | |
| Integrated Review/Final SER | | 42 | 42 |
| ACRS Review | | 48 | 48 |
| Proposed Decision Date for FDA | | 54 | 54 |
| Design Certification Rulemaking[1/] | | 66 | 66 |

1./ The schedule for the Design Certification Rulemaking phase depends on, among other factors, the type of rulemaking proceeding selected by the Commission.

TABLE 4-2

NRC REQUIREMENTS NOT TECHNICALLY RELEVANT TO CANDU 3 SDC

| Section | Subject | Basis |
|---|---|---|
| 50.34(f)(1)(v) | HPCI/RCIC Evaluation | BWR specific |
| (vi) | Relief valve challenges | BWR specific |
| (vii) | Automatic depressurization system | BWR specific |
| (viii) | Core-cooling studies | BWR specific |
| (ix) | Space cooling for HPCI/RCIC | BWR specific |
| (x) | Auto. depressurization sys. study | BWR specific |
| (xi) | Evaluation of depressurization methods | BWR specific |
| (2)(i) | Simulator capability | Site specific |
| (xvi) | Accumulation cycles of ECCS | B & W specific |
| (xxi) | Auxiliary heat removal systems | BWR specific |
| (xxii) | Integrated control system | B & W specific |
| (xxiii) | Anticipatory Reactor Trip | B & W specific |
| (xxiv) | Reactor Vessel Water Level Recorder | BWR specific |
| (3)(v)(B)(1) | Containment loads due to inerting | No inerting |
| (vi) | External hydrogen recombiners | No external recombiners |
| 50.36b | Environmental Conditions | Site specific |
| | 10 CFR Part 50 Appendices | |
| C | Financial Data | Licensee specific |
| D | | Reserved |
| E | Emergency Planning | Site specific |
| F | Fuel Reprocessing Sites | No fuel reprocessing |
| H | Reactor Vessel Material Surveillance | No reactor vessel |
| L | Antitrust Info | Licensee specific |
| M | Manufacturing Licenses | Licensee specific |
| N | Duplicate Designs at Multiple Sites | Site specific |
| P | | Reserved |
| Q | Early Site Reviews | Site specific |
| R | Fire Protection | Applies to pre 1979 operating reactors |
| 73.40 | Physical Protection | Site specific |
| | 10 CFR Part 100 Appendix | |
| A | Site Evaluation | Site specific |