UNITED STATES

NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555

Solicitation No. RS-NRR-89-026

Nuclear Power Reactor Design
Inspection Services

PART III: Technical and Management Proposal

AECL Ref. 89-025
1989 June

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AECL TECHNOLOGIES A Division of AECL Inc.

SERVICES TECHNOLOGY & ENGINEERED PRODUCTS

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1-800-USA-AECL

Ref: 89-025
1989 June 15

Ms. Teresa McLearen
United States Nuclear Regulatory Commission
Room 1011
7920 Norfolk Avenue
Bethesda, MD 20814
U.S.A.

Dear Ms. McLearen:

Re: Solicitation No. RS-NRR-89-026
Nuclear Power Reactor Design Inspection Services

AECL Technologies is pleased to submit this proposal to provide nuclear power reactor design inspection services.

As requested in the RFP, we are submitting the following in three separate and distinct parts:

1) two (2) original signed copies of the Solicitation package;
2) one (1) original and five copies of the C[illegible] proposal; and
3) one (1) original and five copies of the Technical and Management Proposal.

We believe that our proposal is fully responsive to the intent of the request for proposal (RFP). The management and experience embodied in the proposed project team provide:

- A team well qualified to handle the tasks set out in the RFP.
- The ability to mobilize resources rapidly to complete the tasks to required schedule and budget.
- Flexibility to address new tasks as NRC's priorities may change in the future.

Furthermore, we are able to offer these extensive skills and capabilities clear of any conflict of interest.

Should you have any questions, or wish to have any supplementary data to facilitate evaluation, please do not hesitate to contact this office or our Mr. R.D. Gadsby, Deputy Director at 1-800-USA-AECL.

Yours truly,

D.R. Shiflett

D.R. Shiflett
Vice-President & General Manager
AECL-Technologies

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PART III

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PART III

1. INTRODUCTION

In accordance with instructions given in section L of the NRC Request for Proposal (RFP) No. RS-NRR-89-026 entitled Nuclear Power Reactor Design Inspection Services dated 1989 May 22, this proposal from AECL Technologies, a Division of AECL (and known henceforth as AECL), is being submitted in three (3) distinct and separate parts. The part herein is Part III - Technical and Management proposal.

AECL Technologies operates as a corporation under the laws of the State of Delaware. It can draw on the staff resources of over 4,500 highly trained people.

2. STATEMENT OF WORK

The objective of this proposal is to provide expert technical assistance in the fields of electrical power systems, instrumentation and controls, mechanical systems, mechanical components, civil and structuring engineering to the NRC inspection teams in the performance of design inspections. The design inspections will be carried out to assume that the design procedures used in constructing or modifying the operational nuclear plants have fully complied with the NRC regulatory requirements and licensee commitments.

The design inspections will provide a close examination of the design process and implementation for a selected, limited sample of the structures, systems and components. The selection of the samples will be arrived at in consultation with the NRC team. For the plants under construction the review will normally include the complete design process, i.e. from the formulation of principal architectural and engineering design criteria, through the development and translation of the criteria into the revisions and verifications of the design.

For operating plants, the design inspection will address the design aspects of the plant safety system functionality, including the review of the affects of the incorporated or proposed modifications.

As part of these inspections the lessons learnt will be documented in the final inspection reports.

It is noted that the efforts required in each of the disciplines for the annual design inspection or the overview of applicant design assurance is very similar to the efforts carried out by AECL for their CANDU Reactors that are either presently operating or are in the construction phases. Similar jobs/tasks have been carried for a number of our utilities. It is noted that our power plants are designed to meet the ASME, Canadian Standards Association (CSA), Ministry of Consumer and Consumer Relations (MCCR) and the Atomic Energy Control Board (AECB) requirements. Hence, our staff are fully familiar with the requirements for Design Inspections and/or reviews for the systems, structure and components for the operating and under construction.

It is also noted that our reviews and inspections in examining a system, structure, component or operational program at operating plants for our customers have also included, but not limited to the following topics:

- Validity of design inputs and assumptions
- Validity of design specifications
- Validity of analyses
- Identification of system interfaces requirements

- Potential synergistic effects of changes
- Proper component classification
- Revision control
- Documentation control
- Verification of as-built condition
- Testing and functionality of design product
- Past audit findings
- Known and suspected deficiencies and their solutions.

As mentioned earlier, these types of tasks have been carried out by our staff on a regular routine basis for a number of the utilities and other customers. It is noted that the standards and jurisdictional codes used are the same as those for the US Reactor. Appendix B includes a list of the typical tasks for two separate jobs. Also, included in this Appendix is a sample report.

The purpose o. ' .tachment is to give a sample of the kind of work undertaken by some of our staff offered in this proposal.

Typically, when the task order statement has been issued by NRC, the review/inspection specialists will perform the following activities:

(a) Reviewing background information.

(b) Selecting the plant aspects to be reviewed during the inspection.

(c) Reviewing implementation effectiveness of design quality assurance and quality control programs.

(d) Performing and documenting design inspection.

(e) Reviewing the final report of applicant/licensee self-initiated design verification programs for accuracy, completeness, and acceptability.

(f) Reviewing applicant/licensee responses to the design inspections and review findings and concerns, and providing written evaluations of the responses.

(g) Reviewing any additional information (not covered above), to evaluate the design/design process of the subject facility and/or to close out design inspection and review findings and concerns.

(h) Providing expert testimony on problems, issues, and allegations at public hearings which result from design inspections and reviews, as necessary.

(i) Documenting the results of design inspections and reviews and other assignments from the Project Officer, in reports to the NRC which are on software compatible with the IBM PC.

(j) Providing an internal quality assurance program to ensure contractor tasks are cionducted in an efficient and satisfactory manner.

3. WHY AECL IS QUALIFIED

1. The effort required in each of the disciplines for the annual design inspections of the overview of applicant design assurance is very similar to the efforts carried out by AECL for their CANDU Reactors that are eithe. presently operating or are in the construction phases. Our power plants are designed to meet the ASME, Canadian Standards Association (CSA), Ministry of Consumer and Commercial Relations (MCCR) and the Atomic Energy Control Board (AECB) requirements. Hence, our staff are fully familiar with the requirements for Design Inspections and/or reviews for the systems, structure and components for the operating and under construction plants. These reviews typically focus on the following aspects at various stages of the reviews:

- Validity of design inputs and assumptions
- Validity of design specifications
- Validity of analyses
- Identification of system interfaces requirements
- Potential synergistic effects of changes
- Proper component classification
- Revision control
- Documentation control
- Verification of as-built condition
- Testing and functionality of design product
- Past audit findings
- Known and suspected deficiencies and their solutions.

4. PROPOSED METHOD OF APPROACH

4. PROPOSED METHOD OF APPROACH

[redacted]

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5. SUPPORT PERSONNEL AND FACILITIES AVAILABLE

AECL has excellent human and technical resources at its disposal. It employs approximately 5,000 staff in two main operating divisions - CANDU Operations, (AECL-CO) and Research Company. As a result it can draw on these resources as and when required.

Members of our staff are very active in various nuclear and quality assurance standards activities at national and international levels.

AECL-CO maintains membership and participates actively in a large number of technical, professional and nuclear industry organizations. Some of these are:

- American Nuclear Society
- Atomic Industrial Forum
- Institute for Nuclear Power Operation
- International Atomic Energy Agency
- American Society of Mechanical Engineers.

AECL-CO maintains well staffed and equipped engineering laboratories. These facilities cover testing and development services, specialized tooling engineering, instrumentation services.

AECL can offer these extensive skills and capabilities clear of any conflict of interest.

6. KEY PERSONNEL

The following specialists have been identified as the key personnel:

1. Mechanical System

[redacted]

2. Mechanical Components

[redacted]

J.S. Panesar

[redacted]

3. Civil

[redacted]

4. Structural Engineering

[redacted]

J.S. Panesar

[redacted]

5. <u>Electrical</u>

[redacted]

6. <u>I & C</u>

[redacted]

7. RESUMÉS OF KEY PERSONNEL

The resumés of the key personnel identified in section 6 of this document are in the attached Appendix A.

LIST OF TASKS: JOB 2

The overall objective of the evaluation are to:

- Confirm the appropriateness and completeness of the design criteria established for the nuclear island design.
- Confirm that the design of the nuclear island complies with the design criteria established.
- Confirm that the safety analysis is both adequate and complete.
- Confirm that the reactor is stable and will operate as designed.
- Identify what additional engineering analysis and/or design modifications are essential to meet the above objectives taking due account of the construction status of the actual plant.

To accomplish the above objectives, the contract was divided into three phases, namely:

Phase 1: Review of Reference Codes, Regulations and Design Criteria

Phase 2: System and event Functional Analyses.

Phase 3: Additional Special Topics.

A summary of the recommendations and conclusions from all of the Products is included as Product 09040 (Recommendations Summary).

This Product Report is one of a series of reports prepared on the topics as indicated below in Phases 1, 2 and 3. Products marked with an asterisk(*) have microfiche copies of relevant computer data packages separately.

	PRODUCT	TITLE
Phase 1:	09011	Review of Principles, Criteria and Safety Methodology.
	09012	Review of Nuclear Codes and Standards.
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	09023	Preliminary Transient/Accident Analysis Validation.
	09024*	Transient/Accident Independent Analysis.
	09025	Process Systems Review.
	09026	I&C Systems Review.
	09027	Review of Reactor Core Systems and Fuel.
Phase 3:	09031*	Single Channel Event Analysis.
	09032	Extension to 09022 Event Sequence Review - Phase 3.
	09033*	Shutdown System Trip Coverage Assessment.
	09035*	Startup Simulations.
	09037*	Dryout Probability Assessment.
	09039*	Reactivity Insertions and Power Excursions.

Some tasks and the topics addressed in this report are as follows:

TASK # IN THE ATTACHMENT	GENERAL TASK
331001	Validity of design inputs and assumptions
331025	Validity of design specifications
331009	Validity of analyses
331006	Proper component classification
331067	Revision control
331075	Documentation control
331067	Verification of as-built condition
331068	Testing and functionality of design product
331058	Past audit findings
331037	Known and suspected deficiencies

8. SOURCE OF PERSONNEL

It is envisaged that all staff would be sourced from AECL and its various divisions. The key personnel have been identified in section 6.

AECL may substitute a person with equally appropriate qualification should a candidate become unavailable due to sickness, job termination, work load, etc.

9. MANAGEMENT ORGANIZATIONAL STRUCTURE

The project team will be assisted in the management function by various support groups, including planning and scheduling, project cost control, administrative services. Figure 9-1 shows the overall organization at AECL CANDU Operations.

The work will be executed through our Engineering Services Branch. J.S. Panesar will act as the NRC job Co-ordinator. He will review the tasks and select the appropriate personnel for the specified resource pool as shown in Figure 9-2.

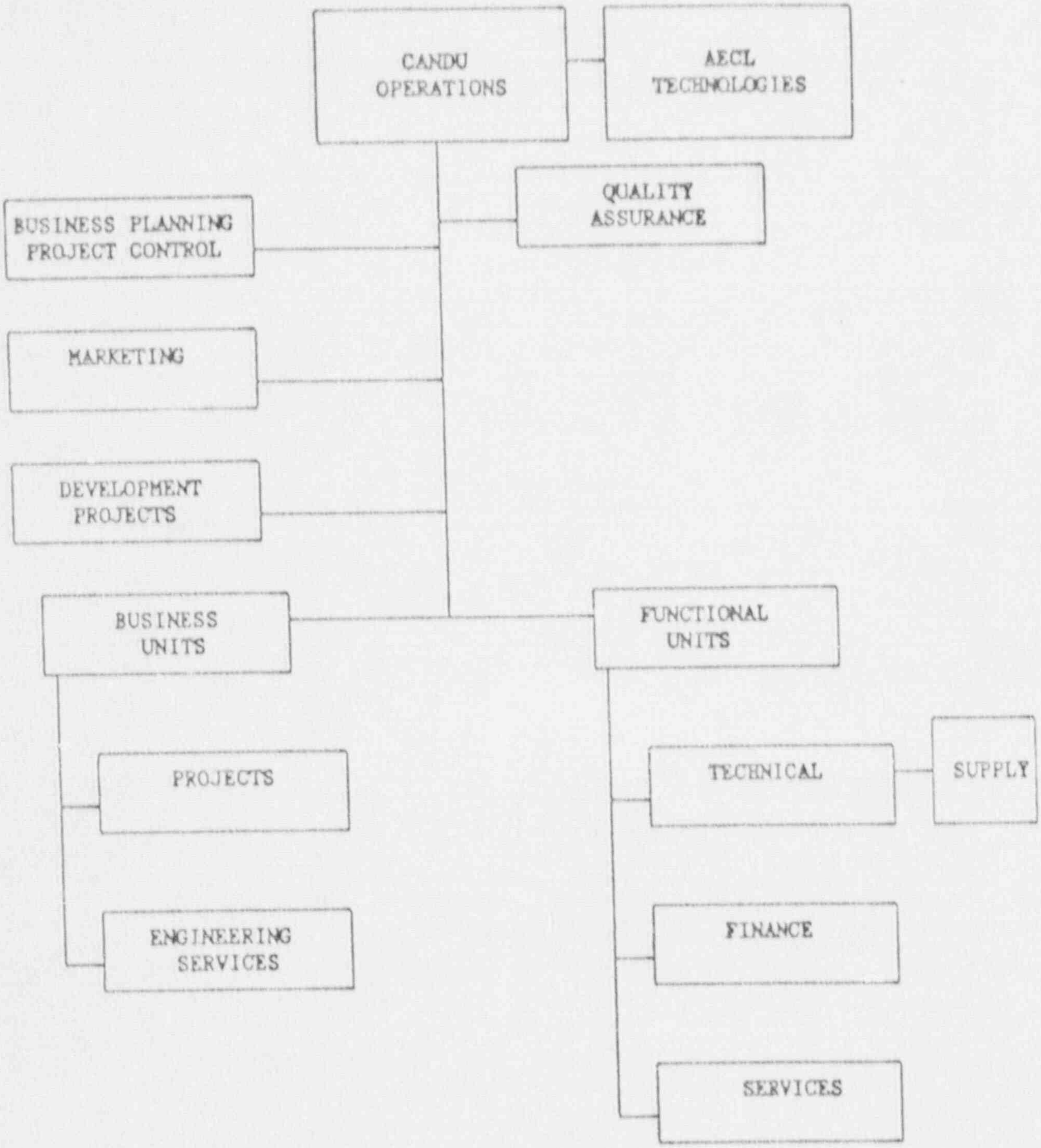
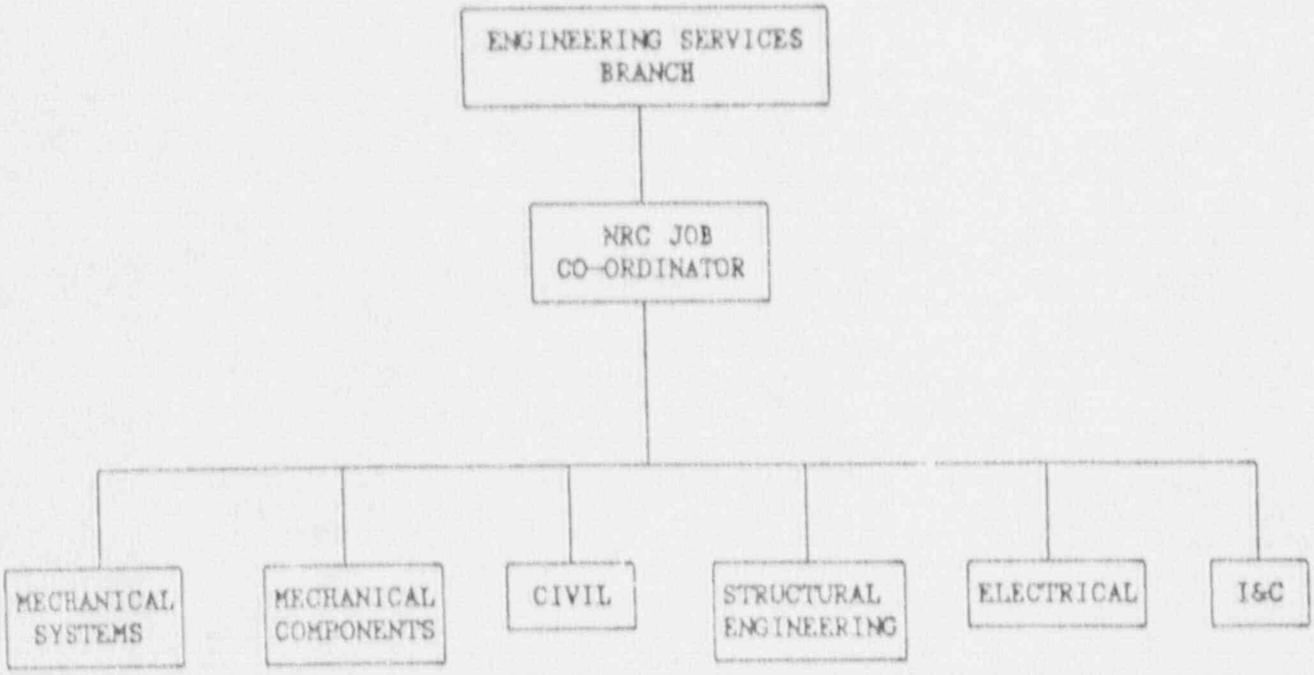


FIGURE 9-1

AECL - CANDU OPERATIONS ORGANIZATION



SUGGESTED STRUCTURE FOR NRC JOBS

FIGURE 9-2

10. MANAGEMENT SYSTEM SUPPORT

10.1 General: Project Cost Control and Monitoring System

The following systems exist for cost control, depending on the size of the tasks:

The cornerstone of Project Cost Control is the Work Breakdown Structure (WBS) which defines a contractual package of work into PRODUCTS (i.e. separate Client task orders) and TASKS.

Each product has:

- a numerical identifier based on the Subject Index (SI)
- a title, generally similar to the SI title
- a scope of work
- a list of deliverables
- a schedule
- a budget (related to the schedule and to the deliverables)
- a record of accumulated expenditures.

Whereas the "PRODUCT" is a package of work, a TASK is a finite activity or chain of activities leading up to the production of a DELIVERABLE. For each one of the technical evaluations to be performed the Product will be broken down into a number of TASKS/DELIVERABLES.

10.2 Project Management and Implementation Plan

10.2.1 Management Controls

In a continuing effort to ensure that all major and minor projects undertaken by AECL are executed with the appropriate resources, on schedule and within budget, senior management conducts Operation

Review Meetings regularly. Through this information/feedback/control mechanism, the Project Team will have access to top management participation in the project. In emergency situations, however, the Project Manager is at liberty to consult with individual senior executives on specific issues and obtain guidance and/or resolution of conflicts that could arise in the course of project execution.

10.2.2 AECL Plan

(a) The overriding objective of the Project Implementation Plan is to ensure that Task Orders are executed and performed to high standards, within predetermined time frames, and within budget.

The Project Management and Implementation Plan describes the structure and functions of the Project Team and the methods, systems and controls that will be used to successfully manage and execute the Task Orders.

The concepts presented are proven and developed. They have been used successfully on numerous previous AECL-CO projects.

(b) On Notice of Award (NOA)

On receiving the NOA, the AECL-CO Project Manager will prepare a Project Plan which will define how the project is to be managed, controlled and administered. The Project Plan will include: scope; schedule; budgets; deliverables; project objectives and goals; project organization structure and resource plan; project policies and procedures; contractual terms and conditions; financial plan; QA plan; engineering plan and client interface requirements.

The Project Team members will be assembled and the basic project administration systems established.

(c) On Receipt of a Task Order RFP (TOR)

On receipt of a Task Order Request for Proposal (TOR) from the Client Contracting Officer, AECL will respond.

(d) On Receipt of Task Order Award (TOA)

Actual initiation of project activities will be triggered by the receipt of a Task Order award from the Contracting Officer. The major engineering implementation steps are as follows:

(i) The Project Manager will review the scope and intent of the Task Order, and draw the appropriate manpower resources from the functional disciplines.

(ii) The Task Orders will be subdivided into products. Each product will cover a typical area as described in the RFP e.g. planning and scheduling, Quality Assurance, etc. It is expected that each product will necessitate a multi-disciplinary participation.

(iii) Each product will be assigned a leader to be attached to the Project Manager's team. The leadership of the product will be assigned to an individual whose discipline is a large contributor to the product. The Product Leader will provide technical leadership as well as budget control and schedule monitoring to ensure the quality and timely completion of the Task Order deliverables. The Product Leader will seek and receive, as necessary, technical assistance from the appropriate Functional Engineering Managers in order to assure a quality execution of the work.

(iv) Each product will consist of one or several tasks addressing the specific topics representing specific issues, e.g. Performance objective, development of generic cost factors, replacement power cost analysis, etc. If the product has one task, the Product Leader may assume the task leadership as well. For a multi-task product, separate task leaders shall be chosen. Project engineers, nuclear physicists, etc. with the appropriate skills will be part of each task reporting to the Task Leader.

(v) Depending on the nature of the Task Order, i.e. its scope, budget allocation, schedule requirement, the Task Leader and the Task team member will be either attached to the Project Manager's team or operating from their home base in the various functional areas. In either case the Task team will still operate in a matrix organization.

(vi) The Task Leader will receive technical and project direction from the Product Leader to ensure that the work proceeds on a course satisfactory to the requirement of Client and its Project Officer.

(vii) The final report will receive approval from both the appropriate Functional Manager and the Product Leader after review by the Task Leader.

(e) Work Package System

The concept of the AECL Work Package System is to break the work down into a number of definable and manageable packages.

For this project, a Task Order will be called a Product and each Product may be assigned to a Work Package comprising a number of Products depending on the size, complexity, duration, etc.

The scope of each Product (Task Order) will be specified to the respective Product Leader by the Engineering Manager. The specification will include the following:

(i) The intent of the Task Order,
(ii) Defined Scope of Work,
(iii) Agreed budget and charge numbers,
(iv) Input documentation and output deliverables,
(v) Responsibilities between groups or individuals.

Following receipt of the initial Product, the Product Leader will, at a point in time consistent with the work schedule elaborate the scope of the product, subdivide the work content of his Product into Tasks with an estimate (per task) of the manpower and other costs required to carry out the work indicated.

In setting up these Tasks, the Product Leader will observe the following guidelines:

(i) Tasks should generally be associated with a deliverable (usually a report of its equivalent) having a clearly defined end date.

(ii) A separate budget and time schedule defined by the Product Leader should be allocated with each Task.

(iii) To facilitate monitoring, each Task should have only one deliverable.

(iv) To facilitate monitoring, only one group should be involved in any one task if practicable.

(v) The task definition should include a description of interfaces with other Products, type of interface, information expected or to be transferred, etc.

(vi) The task breakdown will facilitate the use of resource skills available and selected to participate in the work of the Product.

The Product Leader will be responsible to the Project Manager for the successful completion of the Product (Task Order) within the agreed budget and schedule.

(f) Project Progress Reporting

A monthly progress report will be prepared and issued to the Client, in accordance with the Client requirements. The report will include: schedule status, financial status, accomplishments, results, problem areas and planned corrective action. The report will be issued by the Project Manager.

10.3 Elements of the Accounting System

Elements of the accounting system used for Project Cost Control used within AECL are as follows:

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(a) Master Code File (MCF)

This system contains the Company's accounting structure and "dictionary" files. The other systems draw data from MCF as and when required.

(b) Time Capture System (TICS)

All time sheet data is entered into TICS for editing, correction and distribution. It feeds data, as required, to WPRS and PLRS.

(c) Work Package Reporting System (WPRS)

This system carries all project manhour budgets and expenditures to date (by Professional, Technical, Drafting, Administration), and generates costed manhours for billing purposes. Manhours are recorded at the W/P, Product (Job), and Task level.

(d) Project Ledger System (PLRS)

This system contains all fiscal $ budget and expenditure data. Manhours are also included at a summary level for information purposes.

(e) General Ledger System (GL)

This system contains all AECL financial transactions in support of the company's income statements and balance sheet.

Further information is provided in the following sections.

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10.4 Master Code File System (MCFS)

MCFS maintains AECL-CO's commonly used data lists and "dictionary" files. The system separates the file maintenance function from the processing and reporting functions performed by other systems. Typical files are as follows:

(a) Organization Analysis Table, ir luding employee numbers, Branch Account numbers, project n ers, W/P numbers, SCA numbers etc., and their associated names or titles. Also included are the necessary roll-up tables.

(b) W/P "Validest" file containing all W/P, Job, Task data.

(c) GL/PL validation file containing the account structures.

(d) TICS parameter file containing week ending dates, statutory holidays, etc.

(e) Other miscellaneous security files and table files.

10.5 Time Capture System (TICS)

TICS handles the input, edit, and correction of manhour data, and the feed of this data to other systems. Specific functions are as follows:

- Receive time sheet input data
- Generate time sheet data for those on "automatic" time sheets
- Generate missing time sheet eports
- Edit time sheet input data
- Generate time sheet error reports receive and edit time sheet correction data

- Provide corrected manhour transactions to WPRS and any other system which requires transaction data
- Product audit trail reports
- In future, generate "turn-around" tim heets

Some of those functions, such as error corr ion, can also be done on-line. In addition on-line enquiry .. provided to the:

- good time sheet file
- error file
- validest file
- adjustment file

10.6 Work Package Reporting System (WPRS)

WPRS stores and reports on actual manhour expenditures vs budgets, at the WP level and at various roll-up levels. Specific functions are:

- receive actual weekly manhour data from TICS
- receive manhour budgets from MCFS
- generate weekly reports of actual expenditures
- generate monthly W/P reports of current month actuals, cum-to-date actuals, budgets, and % spent (data sorted by Job/Task)
- generate similar reports summed up to the Branch, Dept., Division, and Company level, with each level containing summary data from the next lower level report.
- apply labour rates against manhour expenditures to produce monthly cost to labour data for PL & GL
- generate Statement billing reports
- generate SCA reports for manhour and billing purposes
- produce leave and overtime reports

10.7 Project Ledger System (PLRS)

This system collects all expenditure data, by budget category and reports against budgets on a fiscal year basis. It performs the following functions:

- receives total [illegible] and costed manhours from WPRS
- receives expense [illegible] tion data from the expense edit subroutine connected to the General Ledger System
- generates monthly transaction reports
- generates monthly W/P reports, sorted by budget category
- generates monthly/Branch reports, sorted by project, by budget category
- generates monthly SCA reports

The project ledger system contains the following data fields:

- Project
- Work Package
- Scope Change Application (SCA)
- Budget Category
- Branch Ø (Budget Acct Ø)
- Current Month and year-to-date data for total hours, expenditures, control budget, authorized budget
- Fiscal year data for control budgets and authorized budgets

10.8 Progress Report Format

Financial and technical assessment progress for each project task will be reported to the Client each month by means of three standard reporting formats:

1. Financial Summary
2. Monthly Progress Report
3. Technical Assessment

10.8.1 Financial Summary

The Financial Summary presents the total spends by labour category, billing rate, expenses, fee and totals for individual project tasks. The contract/budget dollar amount of the task is also displayed on this report.

10.8.2 Monthly Progress Report

The Monthly Progress Report displays the summary listing of total spends for labour, rates, expenses, fee and totals for all products (project tasks) worked on for the overall contract(s) awarded by the Client.

10.8.3 Technical Assessment

The Technical Assessment provides the summary provides the current month spends of labour, expenses and fee for individual products. Also provided on this form is the narrative of technical progress and forecast of work planned for the succeeding month.

11. OTHER WORK COMMITMENTS AND EXPERIENCE

11.1 Other Work Commitments

AECL has the resources and the flexibility to handle assignments as defined in this RFP. As mentioned earlier, AECL can draw on a staff of almost 5,000 through its various divisions. Therefore, NRC requirements can be met without adversely affecting other commitments.

11.2 Experience

Please find attached in Appendix B a summary of AECL work experience in addressing similar tasks.

APPENDIX "B"

SAMPLE WORK

LIST OF TASKS: JOB 2

The overall objective of the evaluation are to:

- Confirm the appropriateness and completeness of the design criteria established for the nuclear island design.
- Confirm that the design of the nuclear island complies with the design criteria established.
- Confirm that the safety analysis is both adequate and complete.
- Confirm that the reactor is stable and will operate as designed.
- Identify what additional engineering analysis and/or design modifications are essential to meet the above objectives taking due account of the construction status of the actual plant.

To accomplish the above objectives, the contract was divided into three phases, namely:

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	09035*	Startup Simulations.
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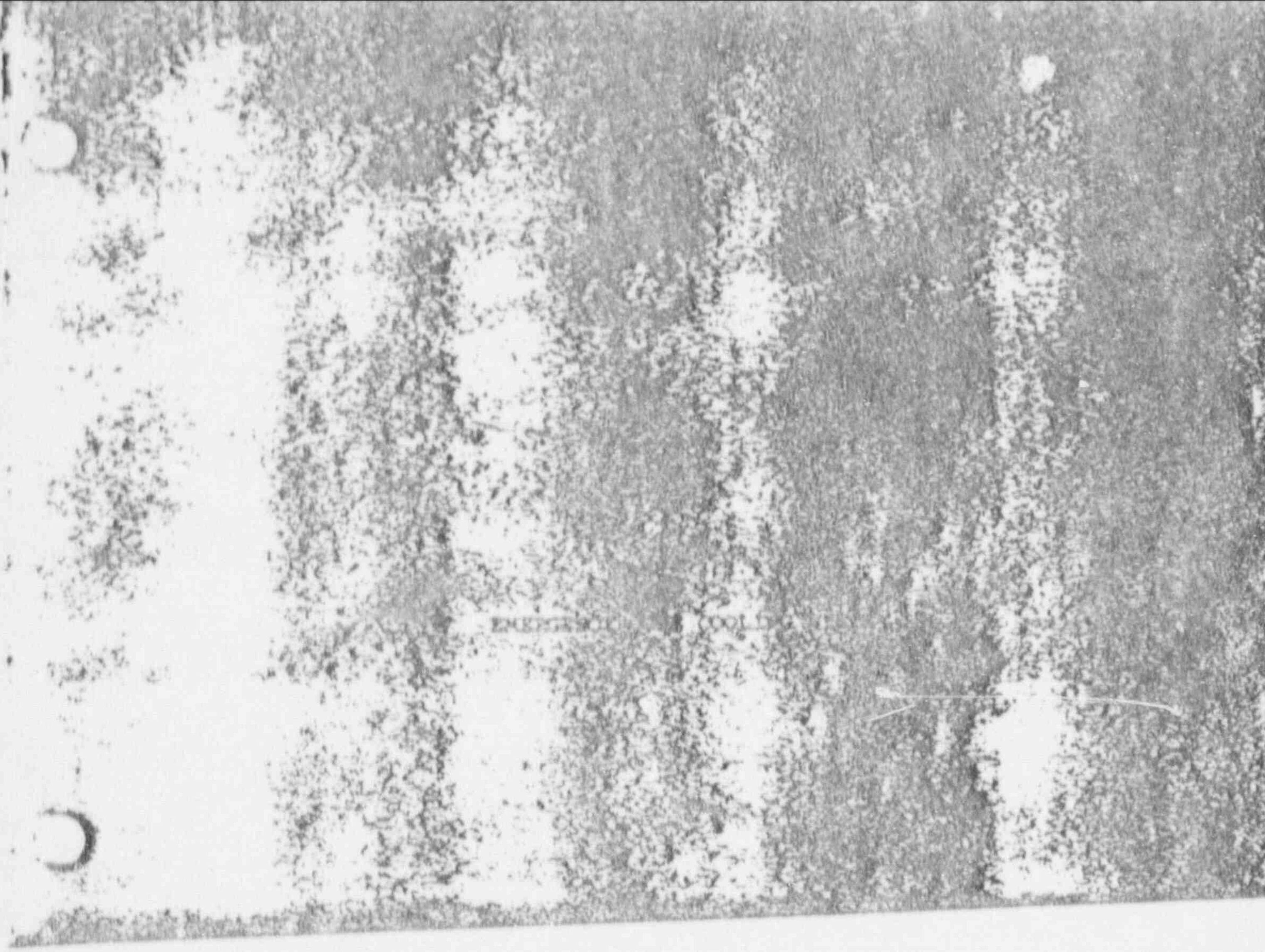

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2.0 REVIEW OF SYSTEM DESIGN AND COMPONENT PARAMETERS

2.1 System Description

A schematic diagram of the ECC system is shown in Figure 1.

2.1.1 High Pressure ECC Injection Flowpath

The high pressure injection portion of the ECC system consists of one accumulator gas tank, two accumulator water tanks and associated valves and piping. These three tanks are located in the Reactor Auxiliary Building. The gas tank which is normally pressurized to [redacted] is isolated from the water tanks by two pneumatic normally closed, fail open valves in parallel, which are referred to as the GAS TANK ISOLATION VALVES. The water tanks and the piping downstream are normally pressurized to [redacted]

The sixteen inch discharge line from the two water tanks splits into two twelve inch branches. This is the separation point for the two ECC subsystems, referred to as Train A and Train B. The two trains are conceptually identical. There are some minor differences in the pipe layout caused by relative locations of the ECC accumulator tanks and the recovery sump.

Downstream of the separation point, each branch contains two normally open, electrically operated valves and a check valve, all in series. These three valves, which are referred to as the ACCUMULATOR DISCHARGE VALVES and the ACCUMULATOR DISCHARGE CHECK VALVE, isolate the accumulator tanks at the conclusion of high pressure injection.

Immediately downstream of the three valves, each high pressure injection flowpath merges with the sixteen inch low pressure injection flowpath associated with the same train (ie. the train A high pressure flowpath joins with the train A low pressure flowpath and similarly for the B trains).

2.1.2 Low Pressure ECC Injection and Recovery Flowpath

The low pressure injection portion of the ECC system consists of the Emergency Cooling Water Storage (ECWS) tank, ECC pumps, heat exchangers and associated valves and piping. The ECWS tank is located outside the Reactor Auxiliary Building at grade elevation. The tank is common to both ECC subsystems and to both Reactor Containment Spray (RCS) subsystems.

All four subsystems share a common discharge nozzle from the ECWS tank. [redacted]

Downstream of the separation point there is a normally open, electrically operated valve in series with a check valve. These valves are called the ECWS TANK ISOLATION VALVE and the ECWS TANK CHECK VALVE. These isolation valves prevent air entrainment into the ECC pump suction when the water in the ECWS tank is depleted.

The low pressure ECC injection line then merges with the line from the recovery sump. Upstream of this connection, the recovery sump line is isolated by a normally closed, electrically operated valve in series with a check valve. These valves are known as the RECOVERY SUMP ISOLATION VALVE and the RECOVERY SUMP CHECK VALVE. The recovery sump isolation valve is required for containment isolation since the recovery sump is located in the basement of the Reactor Containment Building (RCB) and is open to the containment atmosphere. The recovery sump check valve prevents water from flowing into the recovery sump during the transition from low pressure short term injection to the recovery mode.

The common low pressure injection/recovery line is connected to both the ECC pump suction and the RCS pump suction. Downstream of the ECC pump, the ECC system is completely separate from the RCS system.

The sixteen inch ECC pump discharge line directs flow via a heat exchanger to the junction of the high and low pressure injection piping. A check valve and a normally closed, electrically operated valve on the low pressure injection line termed the LOW PRESSURE INJECTION VALVE and LOW PRESSURE INJECTION CHECK VALVE prevent the upstream piping and equipment from being overpressurized during high pressure ECC operation, testing of the accumulator water tanks or failure of either gas tank isolation valve.

There is a recirculation line from the ECC pump discharge to the pump suction. The maximum flow through this line is [illegible] m^3/s. The recirculation line permits testing of the ECC pumps during normal reactor operation. It also provides the minimum pump flow during high pressure ECC injection. The recirculation line contains a normally open, fail open pneumatic valve known as the PUMP RECIRCULATION VALVE.

2.1.3 Common High/Low Pressure Injection Flowpath

Downstream of the point where the high and low pressure lines join, the sixteen inch common injection line contains a normally closed, electrically operated valve referred to as the ECC INJECTION VALVE. This valve is located just outside the RCB wall and serves as containment isolation.

After penetrating containment, the ECC injection line splits into two branches. One branch provides ECC flow to the headers at the north end of the reactor and the other branch provides ECC flow to the headers at the south end of the reactor. A rupture disc is installed in each branch line to maintain separation between the light water used for ECC injection and the D_2O used in the Primary Heat Transport (PHT) system. The rupture discs are designed to burst open when the upstream pressure exceeds the downstream

Downstream of the separation point there is a normally open, electrically operated valve in series with a check valve. These valves are called the ECWS TANK ISOLATION VALVE and the ECWS TANK CHECK VALVE. These isolation valves prevent air entrainment into the ECC pump suction when the water in the ECWS tank is depleted.

The low pressure ECC injection line then merges with the line from the recovery sump. Upstream of this connection, the recovery sump line is isolated by a normally closed, electrically operated valve in series with a check valve. These valves are known as the RECOVERY SUMP ISOLATION VALVE and the RECOVERY SUMP CHECK VALVE. The recovery sump isolation valve is required for containment isolation since the recovery sump is located in the basement of the Reactor Containment Building (RCB) and is open to the containment atmosphere. The recovery sump check valve prevents water from flowing into the recovery sump during the transition from low pressure short term injection to the recovery mode.

The common low pressure injection/recovery line is connected to both the ECC pump suction and the RCS pump suction. Downstream of the ECC pump, the ECC system is completely separate from the RCS syste'.

The sixteen inch ECC pump discharge line directs flow via a heat exchanger to the junction of the high and low pressure injection piping. A check valve and a normally closed, electrically operated valve on the low pressure injection line termed the LOW PRESSURE INJECTION VALVE and LOW PRESSURE INJECTION CHECK VALVE prevent the upstream piping and equipment from being overpressurized during high pressure ECC operation, testing of the accumulator water tanks or failure of either gas tank isolation valve.

There is a recirculation line from the ECC pump discharge to the pump suction. The maximum flow through this line is 0.18 m^3/s. The recirculation line permits testing of the ECC pumps during normal reactor operation. It also provides the minimum pump flow during high pressure ECC injection. The recirculation line contains a normally open, fail open pneumatic valve known as the PUMP RECIRCULATION VALVE.

2.1.3 Common High/Low Pressure Injection Flowpath

Downstream of the point where the high and low pressure lines join, the sixteen inch common injection line contains a normally closed, electrically operated valve referred to as the ECC INJECTION VALVE. This valve is located just outside the RCB wall and serves as containment isolation.

After penetrating containment, the ECC injection line splits into two branches. One branch provides ECC flow to the headers at the north end of the reactor and the other branch provides ECC flow to the headers at the south end of the reactor. A rupture disc is installed in each branch line to maintain separation between the light water used for ECC injection and the D_2O used in the Primary Heat Transport (PHT) system. The rupture discs are designed to burst open when th upstream pressure exceeds the downstream pressure by [redacted] The differential pressure required to burst the discs in the reverse direction is slightly greater. The discs are scored so that they will not fragment, but petals will fold back with flow creating an opening with a cross-sectional area slightly smaller than the pipe. Tests were performed on the rupture discs used in other CANDU 600 ECC systems. These tests have shown that the additional resistance added to the ECC system by the rupture disc is the same order of magnitude as a fully open globe valve.

Downstream of the rupture disc, each branch line is further subdivided into four ten inch injection lines each eventually connecting to a different reactor header. Each injection path contains a check valve and a normally closed, electrically operated valve in series. These valves are called the D_2O CHECK VALVE and the ECC/PHT ISOLATION VALVE. Upstream of the D_2O check valve the pressure in the ECC lines is kept close to atmospheric. Downstream of the ECC/PHT isolation valve the pressure in the ECC lines is the same as in the reactor headers. The pressure in the interspace between the D_2O check valve and the ECC/PHT isolation valve will be at some intermediate value which depends on the relative leakage rates of the D_2O check valve and the ECC/PHT isolation valve.

The D_2O check valves prevent both core and boiler bypassing when the ECC/PHT isolation valves are opened following a LOCA. They also permit the ECC/PHT isolation valves to be tested during normal reactor operation without reverse rupture of the rupture discs.

A restriction orifice is provided in series with and upstream of each ECC/PHT isolation valve to limit the flow through this path in the event of an ECC line break downstream of the ECC/PHT isolation valve.

The train A and train B injection lines for a given reactor header combine downstream of the ECC/PHT isolation valves and the restriction orifices. These combined injection lines do not connect directly to the reactor headers but rather to the shutdown cooling system piping which in turn is connected to the reactor headers.

2.2 System Design Changes From Phase III Study

2.2.1 Number of D_2O Check Valves

In the Phase III study one D_2O check valve was provided upstream of a pair of ECC/PHT isolation valves. In the present design one D_2O check valve is provided upstream of each ECC/PHT isolation valve (Ref. 1). This prevents boiler bypassing following a real or spurious LOCA signal. This change results in a total of eight valves being added to the ECC system.

2.2.2 Containment Penetrations And Injection Valve Arrangement

In the Phase III study, the high pressure injection flowpath and the low pressure injection flowpath penetrate containment separately and then join together. [redacted] As part of this change, the accumulator discharge valves and the low pressure injection check valves were relocated outside containment which facilitates valve maintenance.

The valve arrangement at the containment penetration was modified slightly in the present design (Ref. 2). In the Phase III study, the accumulator discharge valves serve as containment isolation for the high pressure injection flowpath. Also in the Phase III study there was a second valve in series with the low pressure injection valve for containment isolation.

In the present design a new valve (the ECC injection valve) located in the common ECI injection line, just outside the containment boundary, serves as containment isolation. As a result, the valve in series with the low pressure injection valve was no longer required and was eliminated. Since the ECC injection valve is located downstream of the accumulator discharge valves and is normally closed, both accumulator discharge valves in each subsystem can be normally open. This reduces the operational requirements on the accumulator isolation valves - they are still required to be fast closing valves, but the opening time is no longer important. The ECC injection valves are required to be fast opening valves.

The accumulator discharge check valves were added in the present study (Ref. 2) to prevent water from being pumped into the accumulator tanks if the low pressure injection valves were accidentally opened during ECC pump testing.

2.2.3 ECC Injection Line Orificing

In the reference design, the ECC injection lines to the reactor outlet headers each contain a restriction orifice having a k-value of 20. The purpose of these orifices was to improve the ECC flow distribution between the inlet and outlet headers following a reactor outlet header break with the PHT pumps running. Increasing the resistance in the ECC lines to the outlet headers (ROH) increases the relative ECC flow to the reactor inlet headers (RIH) which are at a higher pressure because of the pump head and the break location.

It was assumed in the Phase III study that similar orificing would be provided in the ECC system design for [redacted] However, the safety analyses for the [redacted] study assume a loss of Class IV power when the reactor is tripped. Therefore the PHT pumps would not be running during ECC injection and it is likely not necessary to have a higher resistance in the ECC line to the ROH than in the line to the RIH.

In the present study orifices are provided in the ECC lines to the ROH and the lines to the RIH. These orifices are intended to limit ECC water wastage (ECC water flowing directly out the break without passing through the reactor core) in the event of an ECC line break. The DTE phase analysis based on the [redacted] reference design has shown that the ECC water wastage following a non-orificed (RIH) line break is significant, but acceptable. [redacted] However ECC injection line orificing will reduce the amount of wastage and increase the margin for high pressure ECC injection.

In this present design stage it is proposed that the restriction orifices provided in the lines to the inlet headers and in the lines to the outlet headers, have the same resistance value, ie., k equal to 20.

2.2.4 PHT Set Pressure For ECC Initiation

In the Phase III study the PHT set pressure for ECC initiation was [redacted] In the present study this set pressure has been increased to [redacted] This change is most beneficial for the small LOCA scenarios because the ECC initiation signal is generated earlier. This results in an earlier crash cooldown signal to the steam generators, faster depressurization of the PHT headers and hence earlier ECC injection into the reactor core.

2.2.5 Crash Cooldown Time Delay

In the Phase III study the time interval between the ECC initiating signal and the signal to open the steam relief valves on the steam generators is 30 seconds. This delay was to allow the stop valves to the turbine to fully close before the MSSVs are opened. It has since been determined that only 10 seconds need to be allowed for closing of the stop valves. Therefore, in the present design the time delay between the ECC initiating signal and the crash cooldown signal has been reduced to 10 seconds (Ref. 3). This change results in earlier ECC injection into the reactor core.

2.2.6 Sequencing of the Class II Powered Valves

All electrical valves required to operate to allow high pressure injection from the accumulator water tanks are supplied by Class II power. These valves include the ECC injection valves and the ECC/PHT isolation valves.

Operation of the above valves is staggered at an interval of [redacted] This is a change from the [redacted] reference design in which the Class II inverters were sized to permit simultaneous opening of all the ECC Class II powered valves. Staggered opening of the ECC valves offers a considerable economic advantage (Ref. 4).

Two independent inverters are provided, one is used for the Train A ECC valves and the other for the Train B ECC valves. There are a total of nine Class II powered ECC valves on each ECC train. Therefore the final ECC valve would start to operate [redacted] after the first valve had started to operate.

There is also a delay in operating each valve due to the control and instrumentation. This delay is estimated to be [redacted] In addition, there is a delay [redacted] in permitting first flow through the valve due to ineffective stem travel. These delays are consecutive. Therefore, with sequencing, the final ECC valve required for high pressure ECC injection would start to permit flow [redacted] after the ECC initiation signal (referred to as the LOCA signal) was generated.

Because the ECC injection valve is in series with the eight ECC/PHT isolation valves in each train, it should be opened first. The order in which the ECC/PHT isolation valves are opened is not important. The analysis will assume that the ECC injection valves start to permit flow [redacted] after the LOCA signal and that all eight ECC/PHT isolation valves start to permit flow simultaneously at [redacted] after the LOCA.

In the safety analysis for the [redacted] reference design, acceptable results were obtained assuming that [redacted] elapsed between the LOCA signal and the first ECC injection flow into the PHT system. Because the PHT set pressure for ECC initiation has been increased to [redacted] the LOCA signal will be generated slightly earlier for the [redacted] design. This would reduce the effect of the delay in opening the Class II powered ECC valves on the [redacted] safety analysis.

The LOCA signal is also used to close the valves in all the interconnection lines joining the heat transport loops. These include the pressurizer isolating valves 3332-MV1 and MV2, the D_2O feed valves 3331-MV13 and MV22 and purification system valves 3335-MV1 to MV4. The pressurizer isolation valves will also be sequenced at 0.2 second intervals after the ECC valves. The analysis will also assume a 0.5 second delay in the operation of the valves due to instrumentation and control. The remaining valves will be powered by Class III power and therefore are not sequenced.

2.2.7 Opening Signal For Low Pressure Injection Valve

In the Phase III study, the opening of the low pressure injection valve was controlled by the water level in the accumulator tanks. During high pressure injection, when the level in the tanks reached a low level setpoint, the low pressure injection valves would start to open. This was intended to prevent flow into the low pressure injection piping during high pressure injection.

In the present design, it was decided that the low pressure injection check valves and the ECWS tank check valves were sufficient to prevent flow into the low pressure injection piping. Therefore, it was decided to open the low pressure injection valve on the LOCA signal to promote a smoother transition from the high pressure stage to the low pressure stage of ECC injection.

The low pressure injection valve is operated by Class III power and is therefore not sequenced.

2.2.8 Operation of the ECC/PHT Isolation Valves

[redacted]

In the present study it was decided to discount failure of the D_2O check valves since these valves are normally in the closed position. Following a LOCA, the valves will open for high pressure ECC injection and the D_2O check valves to the unfailed loop should close when low pressure injection takes over. Since the interval in which the valves are open is very short, it is extremely unlikely that the D_2O check valves will stick in the open position. Therefore, the ECC/PHT isolation valves to the unfailed loop will remain open after a LOCA.

2.2.9 Accumulator Gas Tank Operating Pressure

In the Phase III study the accumulator gas tank operating pressure was [redacted] This value was taken from design information at an early stage of the reference ECC system design. Later, the safety analysis for the reference design demonstrated that an operating gas tank pressure of [redacted] was adequate and the reference design was revised.

In the present ECC design for [redacted] the operating gas tank pressure was also reduced to [redacted] same as the reference design. In addition a higher operating pressure in the gas tanks would result in a more abrupt transition from high pressure to low pressure ECC injection.

2.2.10 Accumulator Water Tank and ECWS Minimum Operating Temperature

[redacted] Heating requirements for the accumulator water tanks and the ECWS tank will be reviewed in the detailed design stage.

2.2.11 Changes in Equipment Specifications

During the present design phase, the specifications for the major process components were studied in more detail than in the Phase III study and some changes were made to optimize the ECC system design. [redacted] The ECC pump head was [redacted] from the Phase III study to provide a smoother transition from high pressure ECC injection to low pressure ECC injection. [redacted] Calculations have shown that the recovery sump and ECC injection temperature could still be kept within design limits using the smaller heat exchanger.

2.3 Component Parameters

2.3.1 Accumulator Gas Tank

The accumulator gas tank provides the pressure for high pressure ECC injection. The design specifications are listed below:

GENERAL:
  Equipment No.
  Type

MATERIAL:

CAPACITY:

TEMPERATURE:
  Operating

  Design

PRESSURE:
  Operating
  Design

APPLICABLE CODE:
SEISMIC CLASSIFICATION

[redacted]

2.3.2 Accumulator Water Tanks

During initial ECC injection, water stored in the accumulator water tanks is injected under pressure to the reactor headers. The design specifications are listed below:

GENERAL:
  Equipment No.
  Type

[redacted]

MATERIAL:

CAPACITY:
Effective Volume for Injection

TEMPERATURE:
Operating

Design

PRESSURE:
Operating
Design

APPLICABLE CODE:
SEISMIC CLASSIFICATION

[redacted]

2.3.3 Emergency Cooling Water Storage Tank

During the low pressure stage of short term ECC operation, wate- stored in the ECWS tank is injected to the reactor headers using the EC pumps. Because the opera:ion of the RCS system has been changed such t at the RCS recirculation mode will begin on the same low lev-l signal used to initiate the ECC recovery mode (Ref. 6', the capacity of the ECWS tank has been [redacted] from the Phase III study.

The ECWS tank is insulated to reduce heat losses. During the cold months the operating temperature of the tank is maintained [redacted] using an external heater in series with a recirculation pump. The principle purpose of the recirculation pump is to assist in the chemical corrosion control of the ECWS tank.

The design specifications for the ECWS tank are listed below:

GENERAL:
Quantity
Equipment No.

MATERIAL:

CAPACITY:
Effective Volume for ECC Injection
Effective Volume for RCS Injection

TEMPERATURE:
Operating

Design

PRESSURE:
Operating
Design

[redacted]

APPLICABLE CODE: MITI Class 3
SEISMIC CLASSIFICATION A

2.3.4 ECC Pumps

The ECC pumps are required to operate during the low pressure stage of short term ECC operation and during long term ECC operation. Thus one of the two ECC pumps must be operational for at least [illegible] months.

In the reference design, the low pressure stage of short term ECC operation is referred to as 'medium pressure injection'. This is because the source of water for injection is the dousing tank located [redacted] above the recovery sump. Thus the injection pressure during medium pressure injection is significantly higher than during the recovery mode.

This is not the case in the ECC design for EPDC. The ECWS tank is located at near grade level as is the recovery sump.

The ECC pump head for EPDC was [redacted] from the reference design to compensate for the use of the grade level ECWS tank instead of the higher level dousing tank. In determining the final pump specifications, it was necessary to achieve adequate injection flows with a minimal pressure reduction at the end of high pressure injection. At the same time, pump run-out had to be prevented and the ECC pump power requirements minimized.

The pump head vs flow curve assumed in the safety analysis is shown in figure 2. The actual pump curve will not be available until the pump manufacturer is selected, however there should be very little difference from the assumed flow curve. The EPDC pump design will be essentially the same as the reference pump design except a different impeller will be used and the pump internals may be slightly modified.

The design specifications for the ECC pumps are listed below.

GENERAL:
- Equipment No.
- Type
- Fluid pumped
- Rated Flow
- Head at Rated Flow
- Shutoff Head
- Operable Flow Range Required
- NPSH Required at Rated Flow

MATERIAL:
- Impeller, case, shaft

PRESSURE:
- Normal
- Design

[redacted]

TEMPERATURE:
Normal
Operating
Design

DRIVE MOTOR:
Type
Size

APPLICABLE CODE:

SEISMIC CLASSIFICATION

[redacted]

2.3.5 <u>Heat Exchangers</u>

The ECC heat exchangers are required to remove the reactor decay heat from the failed PHT loop following a LOCA. In the present study the duty of each ECC heat exchanger was [redacted] Calculations show that with one RCS system heat exchanger operating, the smaller ECC heat exchanger would still maintain the recovery sump temperature and the ECC injection temperature within design limits. These calculations are further described in Section 4.0 and are conservative.

The design of the heat exchanger was specified in detail using a simple computer programme. The primary and secondary side pressure drops across the heat exchanger were kept close to those in the Phase III study. [redacted] (see Section 4.0 for further details). [redacted] due to the mixing effect of the pump recirculation line. This change in the primary inlet temperature reduces the efficiency of the heat exchanger therefore it was decided not to reduce the secondary side flow.

A 1-shell pass, 2-tube pass type heat exchanger is considered the optimum and recommended for the present design. The required heat transfer area is approximately 1100 m^2. The design specifications for the heat exchangers are summarized below:

GENERAL:
Equipment No.

Type
Service
Capacity
Heat Transferred

[redacted]

SHELL SIDE DATA:

Fluid
Flow rate
Inlet Temperature
Outlet Temperature
Design Temperature
Design Pressure
Design Pressure Drop
Applicable Code
Seismic Class

TUBE SIDE DATA:

Fluid
Flow Rate
Inlet Temperature
Outlet Temperature
Design Temperature
Design Pressure Drop

[redacted]

2.3.6 Valves

It is recommended that the large valves in the low pressure portion of the ECC system which contains light water be specified as butterfly valves with a ring type seal. These valves should also be flanged for ease of maintenance.

The large valves in the high pressure portion of the ECC systems containing either light water or heavy water are required to be gate valves. [redacted] All high pressure valves and all heavy water valves are to be welded in order to minimize leakage.

[redacted]

3.0 SYSTEM OPERATION

The ECC system is on standby during normal reactor operation. It operates only in the event of a LOCA.

3.1 Reactor Trip

In the event of a LOCA, the pressure inside the RCB rises and the pressure in the PHT system drops. Before ECC system operation is initiated the reactor is tripped. The trip parameters are high neutron power and/or PHT low flow and/or high RCB pressure.

3.2 Blowdown

The time after the LOCA, until the PHT pressure reaches the ECC injection pressure, is known as the blowdown period. The length of time for the blowdown period is dependent on the break size.

During the blowdown period, the LOCA signal will be generated [redacted]

The LOCA signal initiates the following major actions:

- Open the gas tank isolation valves
- Open the ECC injection valves**
- Open all sixteen ECC/PHT isolation valves**
- Start the ECC pumps
- Start cooling water flow to the ECC heat exchangers
- Open the low pressure injection valves
- Open the SG safety relief valves after a 10 second delay
- Close the PHT loop isolation valves: pressurizer isolation valves,** D_2O feed valves and the purification system valves (This isolates the failed loop from the unfailed loop and isolates both PHT loops from the pressurizer, feed system and purification system.)

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* The reference CANDU 600 design has been changed such that it now uses a high moderator level signal as conditioning for in-core breaks. Thus retention of the high moderator cover gas pressure signal for EPDC constitutes a design difference from the reference design.

**The operation of these valves is sequenced at 0.2 second intervals to limit the size of the Class II power inverters, see also section 2.2.6.

3.3 Emergency Core Cooling

During the full sequence of emergency core cooling operation, decay heat removal is by transfer of heat to the steam generators or by discharge of fluid through the break. The latter mode predominates for the large break. The former mode predominates for small breaks.

3.3.1 Short Term Operation

Within a few seconds after the gas tank isolation valves first start to open, [redacted] because the piping between the accumulator water tanks and the gas tank isolation valves is gas filled and initially at low pressure.

ECC injection flow from the accumulator tanks into the PHT system will start at [redacted] after the LOCA signal (due to ECC valve opening delays) or when the PHT system pressure drops to [redacted] if this occurs after [redacted] There is sufficient coolant available in the accumulator water tanks for a minimum of [redacted] for a 100% header break size. For smaller breaks the time is extended.

As coolant is discharged from the water tanks, the coolant level in the tanks and the injection pressure fall. [redacted] For smaller break sizes the pressure and especially the temperature are higher. [redacted]

On low coolant level in the accumulator water tanks, the accumulator discharge valves are closed to prevent gas from being injected into the PHT system.

While high pressure injection is occurring, the ECC pumps have started and are recirculating water via the pump recirculating lines. The low pressure injection valve opens on the LOCA signal but the low pressure piping remains isolated by the check valve in the low pressure injection line. When the pressure downstream of the check valve falls below the pressure generated by the pumps, injection from the ECWS tank begins.

From this time, until the accumulator discharge valves are fully closed, ECC injection water is being supplied from both the accumulator water tanks and the ECWS tank. This helps smooth the transition from high pressure to low pressure injection.

There is sufficient water in the ECWS tank for 15 minutes injection for a large LOCA with both ECC trains operating. Short term ECC injection will therefore last a minimum of 17.5 minutes.

3.3.2 Long Term Operation

During short term operation the injection flow escapes from the break and collects in the RCB basement. When the ECWS tank reaches a pre-determined low level setpoint, the recovery line isolation valves from the RCB basement are automatically opened. Shortly after, when the ECWS tank reaches the very low level setpoint, the ECWS tank isolation valves are closed and the long term recovery mode of ECC operation begins. This mode of operation referred to as the recovery mode, continues until the decay power is very low. For design purposes, this period is assumed to be three months.

The recovery mode is essentially steady state recirculation of water discharged from the break back to the failed PHT loop. The water is heated in the core by decay power and is cooled by heat transfer to the reactor component cooling water system via the ECC heat exchangers and also by heat transfer to the steam generator secondary side.

Steam generated in the steam generator is rejected via the main steam safety valves. These have been opened after a LOCA for steam generator cooldown. The auxiliary feedwater system provides flow to the steam generator secondary side.

For a small break the ECC pump flow is small and heat transfer to the steam generators will be a significant fraction of the total decay heat removal, particularly when the water in the PHT system is hot. The ECC heat exchangers still remove heat from the recovered water. After approximately four hours, the core will be cooled to a low enough temperature that the ECC heat exchangers alone are adequate for cooling the fuel and feedwater flow to the steam generators in the failed loop is not required.

3.3.3 Cooling of the Unfailed Loop

The unfailed loop is cooled by thermosyphoning with heat removal by the steam generators. The natural circulation flow is sufficient to provide adequate cooling of the fuel. When the PHT coolant temperature drops to [redacted] the shutdown cooling system can be connected to the unfailed loop.

Makeup to the unfailed loop is provided whenever the loop pressure falls below that of the emergency core cooling injection pressure.

4.0 ECC WATER TEMPERATURE

In order to size the ECC heat exchangers, it was necessary to calculate the temperature of the water in the recovery sump at the start of the ECC recovery mode. Two cases were considered: a 100% ROH break and a feeder size break.

The data for these calculations was derived from AI-1088: "Station Data and Modelling Information for the 600 MW Containment Analysis" (Ref. 7). The method of calculation and the key assumptions used are described below:

1. The average ECC flow for each break size was determined from the AI-1088 data. These flows were adjusted to account for the higher pump head and different resistances in the EPDC design. For the feeder size break the [redacted] ECC flow is [redacted] and for the 100% ROH break the ECC flow is [redacted]

2. The time at which the recovery mode begins was calculated by dividing the total injectable mass from the accumulator water tanks and the ECWS tank (2.022×10^6 kg) by the combined RCS and ECC flow from one train. For the feeder size break the recovery mode begins at [redacted] and for the 100% ROH break recovery begins at [redacted]

3. It was assumed that all the energy discharged from the break was present in the recovery sump liquid. Thus the total enthalpy of the sump water was calculated by summing the initial enthalpy of the RCS water and the enthalpy of the fluid discharged from the break. The break discharge enthalpy from AI-1088 was corrected for the higher initial temperature of the ECC water [redacted] in the reference design) and the higher ECC flows. For the feeder size break the total enthalpy of the sump water is [redacted] and for the 100% ROH break the total energy is [redacted] The total enthalpy is higher for the feeder size break because of the longer time before the recovery mode begins.

4. The temperature of the recovery sump water was calculated by dividing the total energy of the sump water by the mass of the sump water (2.022×10^6 kg - no hold-up of the discharged water was assumed). For the feeder size break the average enthalpy of the sump water is [redacted] corresponding to a temperature of [redacted] For the 100% ROH break the average enthalpy of the sump water is [redacted] corresponding to a temperature of [redacted]

It should be noted that the assumption that all the energy discharged from the break is present in the sump liquid is very conservative since flashing of the fluid discharged from the break will occur. Some of the vapour will condense on the RCB walls, internal structures and equipment. The latent heat of condensation will be transferred to these surfaces and not to the sump water. Some of the vapour will not condense. This also reduces the total energy present in the sump water and hence the sump water temperature.

SHELL SIDE DATA:

Fluid
Flow rate
Inlet Temperature
Outlet Temperature
Design Temperature
Design Pressure
Design Pressure Drop
Applicable Code
Seismic Class

TUBE SIDE DATA:

Fluid
Flow Rate
Inlet Temperature
Outlet Temperature
Design Temperature
Design Pressure Drop

[redacted]

2.3.6 Valves

It is recommended that the large valves in the low pressure portion of the ECC system which contains light water be specified as butterfly valves with a ring type seal. These valves should also be flanged for ease of maintenance.

The large valves in the high pressure portion of the ECC systems containing either light water or heavy water are required to be gate valves. [redacted] All high pressure valves and all heavy water valves are to be welded in order to minimize leakage.

3.0 SYSTEM OPERATION

The ECC system is on standby during normal reactor operation. It operates only in the event of a LOCA.

3.1 Reactor Trip

In the event of a LOCA, the pressure inside the RCB rises and the pressure in the PHT system drops. Before ECC system operation is initiated the reactor is tripped. The trip parameters are high neutron power and/or PHT low flow and/or high RCB pressure.

3.2 Blowdown

The time after the LOCA, until the PHT pressure reaches the ECC injection pressure, is known as the blowdown period. The length of time for the blowdown period is dependent on the break size.

During the blowdown period, the LOCA signal will be generated. [redacted]

The LOCA signal initiates the following major actions:

- Open the gas tank isolation valves
- Open the ECC injection valves**
- Open all sixteen ECC/PHT isolation valves**
- Start the ECC pumps
- Start cooling water flow to the ECC heat exchangers
- Open the low pressure injection valves
- Open the SG safety relief valves after a 10 second delay
- Close the PHT loop isolation valves: pressurizer isolation valves,** D_2O feed valves and the purification system valves (This isolates the failed loop from the unfailed loop and isolates both PHT loops from the pressurizer, feed system and purification system.)

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* The reference CANDU 600 design has been changed such that it now uses a high moderator level signal as conditioning for in-core breaks. Thus retention of the high moderator cover gas pressure signal for EPDC constitutes a design difference from the reference design.

**The operation of these valves is sequenced at 0.2 second intervals to limit the size of the Class II power inverters, see also section 2.2.6.

3.3 Emergency Core Cooling

During the full sequence of emergency core cooling operation, decay heat removal is by transfer of heat to the steam generators or by discharge of fluid through the break. The latter mode predominates for the large break. The former mode predominates for small breaks.

3.3.1 <u>Short Term Operation</u>

Within a few seconds after the gas tank isolation valves first start to open, [redacted] because the piping between the accumulator water tanks and the gas tank isolation valves is gas filled and initially at low pressure.

ECC injection flow from the accumulator tanks into the PHT system will start at [redacted] after the LOCA signal (due to ECC valve opening delays) or when the PHT system pressure drops to [redacted] if this occurs after [redacted] There is sufficient coolant available in the accumulator water tanks for a minimum of [redacted] for a 100% header break size. For smaller breaks the time is extended.

As coolant is discharged from the water tanks, the coolant level in the tanks and the injection pressure fall. [redacted] For smaller break sizes the pressure and especially the temperature are higher. [redacted]

On low coolant level in the accumulator water tanks, the accumulator discharge valves are closed to prevent gas from being injected into the PHT system.

While high pressure injection is occurring, the ECC pumps have started and are recirculating water via the pump recirculating lines. The low pressure injection valve opens on the LOCA signal but the low pressure piping remains isolated by the check valve in the low pressure injection line. When the pressure downstream of the check valve falls below the pressure generated by the pumps, injection from the ECWS tank begins.

From this time, until the accumulator discharge valves are fully closed, ECC injection water is being supplied from both the accumulator water tanks and the ECWS tank. This helps smooth the transition from high pressure to low pressure injection.

There is sufficient water in the ECWS tank for 15 minutes injection for a large LOCA with both ECC trains operating. Short term ECC injection will therefore last a minimum of 17.5 minutes.

3.3.2 Long Term Operation

During short term operation the injection flow escapes from the break and collects in the RCB basement. When the ECWS tank reaches a predetermined low level setpoint, the recovery line isolation valves from the RCB basement are automatically opened. Shortly after, when the ECWS tank reaches the very low level setpoint, the ECWS tank isolation valves are closed and the long term recovery mode of ECC operation begins. This mode of operation referred to as the recovery mode, continues until the decay power is very low. For design purposes, this period is assumed to be three months.

The recovery mode is essentially steady state recirculation of water discharged from the break back to the failed PHT loop. The water is heated in the core by decay power and is cooled by heat transfer to the reactor component cooling water system via the ECC heat exchangers and also by heat transfer to the steam generator secondary side.

Steam generated in the steam generator is rejected via the main steam safety valves. These have been opened after a LOCA for steam generator cooldown. The auxiliary feedwater system provides flow to the steam generator secondary side.

For a small break the ECC pump flow is small and heat transfer to the steam generators will be a significant fraction of the total decay heat removal, particularly when the water in the PHT system is hot. The ECC heat exchangers still remove heat from the recovered water. After approximately four hours, the core will be cooled to a low enough temperature that the ECC heat exchangers alone are adequate for cooling the fuel and feedwater flow to the steam generators in the failed loop is not required.

3.3.3 Cooling of the Unfailed Loop

The unfailed loop is cooled by thermosyphoning with heat removal by the steam generators. The natural circulation flow is sufficient to provide adequate cooling of the fuel. When the PHT coolant temperature drops to [illegible] the shutdown cooling system can be connected to the unfailed loop.

Makeup to the unfailed loop is provided whenever the loop pressure falls below that of the emergency core cooling injection pressure.

4.0 ECC WATER TEMPERATURE

In order to size the ECC heat exchangers, it was necessary to calculate the temperature of the water in the recovery sump at the start of the ECC recovery mode. Two cases were considered: a 100% ROH break and a feeder size break.

The data for these calculations was derived from AI-1088: "Station Data and Modelling Information for the 600 MW Containment Analysis" (Ref. 7). The method of calculation and the key assumptions used are described below:

1. The average ECC flow for each break size was determined from the AI-1088 data. These flows were adjusted to account for the higher pump head and different resistances in the EPDC design. For the feeder size break the ECC flow is [redacted] and for the 100% ROH break the ECC flow is [redacted]

2. The time at which the recovery mode begins was calculated by dividing the total injectable mass from the accumulator water tanks and the ECWS tank (2.022×10^6 kg) by the combined RCS and ECC flow from one train. For the feeder size break the recovery mode begins at [redacted] and for the 100% ROH break recovery begins at [redacted]

3. It was assumed that all the energy discharged from the break was present in the recovery sump liquid. Thus the total enthalpy of the sump water was calculated by summing the initial enthalpy of the RCS water and the enthalpy of the fluid discharged from the break. The break discharge enthalpy from AI-1088 was corrected for the higher initial temperature of the ECC water [redacted] in the reference design) and the higher ECC flows. For the feeder size break the total enthalpy of the sump water is [redacted] and for the 100% ROH break the total energy is [redacted] The total enthalpy is higher for the feeder size break because of the longer time before the recovery mode begins.

4. The temperature of the recovery sump water was calculated by dividing the total energy of the sump water by the mass of the sump water (2.022×10^6 kg - no hold-up of the discharged water was assumed). For the feeder size break the average enthalpy of the sump water is [redacted] corresponding to a temperature of [redacted] For the 100% ROH break the average enthalpy of the sump water is [redacted] corresponding to a temperature of [redacted]

It should be noted that the assumption that all the energy discharged from the break is present in the sump liquid is very conservative since flashing of the fluid discharged from the break will occur. Some of the vapour will condense on the RCB walls, internal structures and equipment. The latent heat of condensation will be transferred to these surfaces and not to the sump water. Some of the vapour will not condense. This also reduces the total energy present in the sump water and hence the sump water temperature.

Two criteria were used in designing the ECC heat exchanger:

- the ECC injection temperature (ie. the heat exchanger outlet temperature) must always be less than or equal to the design value of [redacted]

- in conjunction with the RCS heat exchanger the recovery sump water temperature must be reduced to and fall below design value of 65.5°C.

It was determined that maintaining the ECC injection temperature less than [redacted] for the 100% ROH break case was the limiting factor in the design. Consequently ECC heat exchanger was sized for the conditions corresponding to the 100% ROH break.

Once the recovery mode starts the ECC injection flow would drop slightly since the recovery sump is at a lower elevation than the ECWS tank. For the 100% ROH break size the ECC injection flow was estimated at [redacted] The primary side flow through the heat exchanger is equal to the ECC injection flow plus the corresponding ECC recirculation flow of [redacted] The primary inlet temperature is therefore equal to:

[redacted]

This data was fed into a simple heat exchanger design programme together with tube length and shell diameter restrictions taken from the Phase III study. The resultant heat exchanger is described in Section 2.3.5.

The temperature of the water in the recovery sump as a function of time was then calculated for the 100% ROH break and the feeder size break using the following assumptions:

- the initial sump water temperature is [redacted] for the 100% ROH break and [redacted] for the feeder size break.

- the heat input to the sump is equal to the decay heat of one reactor loop.

- there is one ECC and one RCS heat exchanger operational.

- the size and duty of the RCS heat exchanger is as described in the Phase III report.

- the Kays-London method (see Appendix A; Ref. 8) was used to determine the heat exchanger efficiency when operating at other design flows and inlet temperatures.

- all the water from the break is present at the recovery sump and there is uniform mixing.

The recovery sump temperature transients which result are presented in Appendix A. From the start of the recovery mode, the decay heat of one reactor loop is less than the heat removal capability of the ECC heat exchanger. Therefore the RCS heat removal capability does not affect the sizing of the ECC heat exchanger. For the 100% ROH break the ECC injection temperature decreases quickly demonstrating that very soon (ie. within a few minutes) after the recovery mode begins there is excess heat removal capability in the system. If the criteria for sizing the heat exchanger can be relaxed to permit the ECC injection temperature to remain above the design value of [redacted] provided that the heat removed by the ECC heat exchanger is at least equal to the decay heat of one reactor loop then the area of the ECC heat exchanger can be significantly reduced and consideration could be given to also reducing the cooling water flow through the ECC heat exchanger.

5.0 HYDRAULIC DATA

[redacted] The layout of the system is not completely finalized. However, in AECL's judgement, the hydraulic data used in the analysis is a reasonable representation of the overall ECC system flow resistance and of the relative distribution of the resistance between the various portions of the ECC system. It should be adequate for the present survey analysis. When the ECC system layout is completely finalized the hydraulic data will be updated for the future safety analyses. It is expected that any changes will be minor and will not affect the overall results of the present survey analyses.

In order to limit the number of cases to be studied in the present phase of the safety analysis it was necessary to symmetrize the ECC system hydraulic data. This treatment of the data was justified by the most recent proposals for the ECC system layout prepared by EPDC which are almost completely symmetric.

Consequently the safety analysis assumes that the resistance associated with the Train B flowpath to the ECC/PHT isolation valve station on the opposite side of the RCB from the Train B ECC injection containment penetration, applies to each of the flow paths from the accumulator tanks to the four ECC/PHT isolation valve stations. The resistance was selected because it was the largest resistance. To achieve symmetry in the actual ECC system layout each flowpath would have to have a similar resistance to the one selected. For the piping between the ECC/PHT isolation valve station and the reactor outlet headers, the average resistance of the four flowpaths was used. For the piping between the ECC/PHT isolation valves and the reactor inlet headers the same approach was used. An average value was used because the resistance of the piping was small relative to the restriction orifice in these lines. Thus the size of the orifice could be adjusted, if necessary, to make the resistance in each flow path the same.

The hydraulic data used as a basis for the safety analysis is included as Appendix 'B'.

6.0 SYSTEM EVALUATION

This section will be issued once the safety analyses involving the ECC system are complete.

7.0 CONCLUSIONS AND RECOMMENDATIONS

The ECC system will act following a LOCA to remove residual and decay heat from the reactor core and to limit fuel damage. The system is designed to tolerate any single active component failure during short term operation or either an active or a passive component failure during long term operation. The ECC system is automatically initiated and does not require operator action in the long term.

It is recommended that the material used for the ECWS tank be changed from carbon steel to stainless steel to simplify corrosion protection. At present, it is planned to use chemical control to prevent corrosion however this requires good circulation in the tank especially since the tank is vented to atmosphere. If a stainless steel tank were used the size of the pump used to recirculate the water can be reduced since the pump would only be required to maintain the temperature in the ECWS tank.

It is recommended that the design requirements for the ECC heat exchanger be reviewed to permit the ECC injection temperature to exceed [redacted] provided that the heat exchanger is always able to remove the decay heat of one reactor loop after the start of the recovery mode. This recommendation is reasonable given the conservatism of the assumptions used to calculate the recovery sump water temperature at the beginning of the recovery mode.

It is essential that the ECC hydraulic data be revised once the final ECC layout in the Reactor Containment Building and in the Reactor Auxiliary Building has been finalized. These data should be compared qualitatively to the hydraulic data used for the survey analyses.

Since the temperature of the accumulator water tanks and the ECWS tank is significantly increased above the reference design and Phase III study, the accumulator tanks will have to be insulated and/or a heater provided in series with the local recirculation pump. For the ECWS tank, a larger heater would be required. The heating requirement should be reviewed in the detailed design stage.

0.0 EXECUTIVE SUMMARY

BACKGROUND

[redacted]

As defined in the Contract the assessment is divided into three distinct phases, each with separate delineation of work and objectives:

Phase 1: Review of Reference Codes, Regulations and Design Criteria

Phase 2: System and Event Functional Analysis

Phase 3: Additional Special Topics

In Phase 1, the appropriateness and completeness of the safety design criteria established for the nuclear island design are reviewed. The Phase 1 summary report was issued in August [redacted] The report presents a general summary of the review of the criteria completed in Phase 1 with a brief outline of the most important conclusions and recommendations.

This report summarizes the work done in Phases 2 and 3. The overall objectives of Phase 2 work for the [redacted] technical evaluation are to:

(a) Confirm that the design of the nuclear island complies with the design criteria established in Phase 1.

(b) Confirm that the safety analysis is both adequate and complete.

and (c) Confirm that the reactor is stable and will operate as designed.

One objective of this Summary Report is to summarize how well the [redacted] nuclear power plant has met these three overall objectives. Information to support these summary statements is drawn from all the Phase 2 task reports for each of the products.

Phase 2 of the technical assessment contains the following products:

09021: Design Basis Events Review
09022: Event Sequence Review
09023: Preliminary Transient/Accident Analysis Validation
09024: Independent Transient/Accident Analysis
09025: Process Systems Review

09026: Instrumentation and Control Systems Review
09027: Review of Reactor Core Systems and Fuel

It should be noted here that in Product 09024 independent analysis of plant transients and accidents is done to supplement the Product 09023 review of the available safety analysis. Because this reviewed analysis was preliminary in nature, it was considered preferable to perform sufficient independent analysis in Product 09024 rather than wait for the final analyses to become available for review.

The objectives of Phase 3 work are to address special topics arising from the activities of Phases 1 and 2 or identified by ENEL. Phase 3 contains the following products.

09031: Single Channel Event Analysis
09032: Extension to Product 09022 Event Sequence Review
09033: Shutdown System Trip Coverage Assessment
09035: Startup Simulations
09037: Dryout Probability Assessment
09039: Reactivity Insertions and Power Excursions
09040: Recommendations Summary
09041: Shutdown Systems Reliability Assessment.

In this report, the results of the various reviews and analyses are discussed in the framework of certain "special topics". These are CIRENE plant issues that extend across the boundaries of the specific product work areas. This approach provides convenient access to the information and aids comprehension.

The presentation of information under these special topics is not intended to rank in order of importance any of the conclusions and recommendations made in the product reports. The choice of the special topics is also not claimed to be exhaustive but it does provide coverage of the more major issues identified.

Plant Stability and Control

The review of plant stability and control concluded that the reactivity control devices provided meet the intent of the relevant general design criteria. However, analysis of power manoeuvres and other plant transients in Product 09024 showed that the neutron power is not well controlled under these conditions. [redacted]

[redacted] However, the transients were run for too short a period of time, and the perturbations used may be too small to excite any inherent non-linearities. Analysis done by AECL indicated a potential steam drum level instability. The analysis also showed that the recirculation flow is unstable when a recirculation flow control valve time constant near the realistic value is used. Further analysis should be done to confirm that adequate margins to instability exist.

An assessment of the reactivity control devices reliability concluded that no unsafe loss of reactivity control is predicted at credible frequencies.

Core Heat Removal During Normal Operation

In normal operation, fuel cooling is provided by the primary coolant system and associated equipment and heat sinks. The review of the systems contributing to the function of heat removal from the core during normal reactor operation indicate that the function will generally be met. [redacted]

Core Heat Removal During Accidents

[redacted]

A number of fuel cooling concerns have been raised in the event sequence assessments done by AECL. The critical event sequences generally involve loss of heat sink in the medium to long term and can be divided into three general categories:

1. Event sequence limited by operator unreliability.

2. Event sequence limited by equipment unreliability.

3. Event sequence requiring analysis support to justify the stability of the event sequence end conditions or assess the consequences.

For sequences initiated by loss of feedwater, the main contribution to the loss of feedwater is the tripping of the normally operating feedwater or condensate extraction pumps. This high frequency is judged to place unacceptable demands on the operator. For sequences limited by the operator, it is recommended that a review of the alarm sequences be undertaken to determine if an increased reliance on timely operator action is justified. Appropriate training and procedures should also be considered. Total loss of Class 3 and 2 electrical power supplies is credible, based on an assumed unreliability of 8×10^{-3} for the standby diesel generator set. Class 2 system reliability improvements by design changes and/or more frequent testing are recommended.

A review of the preliminary safety analysis revealed a number of deficiencies in the thermohydraulic analyses codes which could affect fuel cooling results. Therefore, independent AECL analysis was done to clarify the concerns. The analysis results show that the fuel and channel related safety targets are satisfied in all cases but one. In the stagnation feeder break, the pressure tube is predicted to strain and contact the calandria tube in violation of a safety target. Recommendations to alleviate this problem include removing certain conservatism from the analysis, or accepting contact and analyzing its consequences.

Analysis of small break without high pressure makeup shows that when the emergency condenser depressurization is initiated, natural circulation fails and a prolonged fuel heatup occurs. Although the fuel-related parameters are acceptable, it is recommended that a general review of events which could lead to this type of behaviour should be conducted.

The analyses also indicate that the emergency injection system is effective in cooling down the fuel. Two specific situations are identified which are not addressed in the analysis: effectiveness of recovery injection to cool the affected channel for the header break with a feeder check valve stuck open, and the effectiveness of injection for a feeder break.

Core Heat Removal During Shutdown

The systems required for core heat removal under shutdown conditions include the main circulation system, the main steam and feedwater system, the emergency condenser system, the residual heat removal system and the main component cooling system. A number of recommendations are made to improve the function of these systems under shutdown conditions.

Initiating Events

The postulated initiating events considered in the [redacted] design documentation have been compared against a list of postulated initiating events compiled using Canadian philosophy. [redacted] Notable among the frequent events missing are malfunctions of control computers, spurious opening of the emergency condenser condensate line valves, and inadvertent opening of the lines to the residual heat removal system.

A number of infrequent events that are not addressed in the [redacted] documentation are loss of service systems, moderator system failures and end shield failures. Based on [redacted] experience some of these system failures can result in severe challenges to the integrity of the primary circuit boundary and the structural integrity of the reactor assembly.

Single Channel Events

The comparison of limiting initiating events in the [redacted] and AECL lists identified eight events not considered in the [redacted] analyses. Three of these are single channel events which are the subject of AECL analyses. The assessment of pressure tube rupture showed that [redacted] Assuming the pressure tube rupture occurs, the consequences are acceptable.

The flow blockage analysis showed that the most likely blockage would not be sufficient to cause fuel sheath dryout in the affected channel. To further strengthen the plants' capability to deal with flow blockage, it is recommended that local measurements for all channel flows be made available to the operator in the main control room.

End fitting failure analysis was done assuming a range of fuel geometries after the fuel bundles are ejected into containment. In the case of the most likely geometry after the end fitting failure (scattered elements), the oxidation release is calculated to be about one third of the total channel inventory. It is recommended that releases from containment and public doses be calculated, taking into account the containment response to the event.

Event sequence analysis for the single channel events revealed that the loss of Class 3 and 2 power following a pressure tube rupture is credible.

Mitigating Systems

Systems required to stabilize the plant following an initiating events are called mitigating systems. Their unreliability dictates the course of the event sequences and the event sequence end point frequencies. The [redacted] In particular, a detailed AECL assessment of the shutdown systems reliability was done in Product 09041, and confirmed the unavailabilities calculated by [redacted] It was noted however that the common cause failure rates (which were also adopted for the AECL analysis) were insufficiently justified.

A variety of analyses have been performed by AECL which support the ability of each shutdown system acting alone to terminate design basis power transients. Large downcomer and header breaks have been analyzed in Product 09024 assuming only the moderator dump system is available. The margins to fuel breakup and prompt criticality are substantial. The functional review of the shutdown systems identified several issues which need to be resolved. In particular, the measurement of neutron flux by the in-core and out-of-core power range detectors does not adequately take into account the effects of moderator level changes.

The review of the preliminary safety analysis revealed several deficiencies in terms of completeness. One of these is a lack of analysis over a range of initial powers and event severities. Therefore, AECL performed a trip coverage assessment, as described below. [redacted] but which have been identified for further consideration include a small break with Class 3 power available and a LOCA with a recirculation pump discharge check valve stuck open.

Dryout Analysis

[redacted] Among the concerns raised were the lack of a hot channel analysis, and concerns about the [redacted] The AECL analysis showed that the criterion is exceeded for slow loss of flow control and slow loss of power control cases. Design solutions involving changes in low flow and overpower setpoints, and in the nominal full power recirculation flow have been suggested.

Point Kinetics

An important issue is the use of point kinetics by [redacted] to analyze transients and accidents. To evaluate the adequacy of the point kinetics approach, a postulated LOCA was analyzed by AECL using both a 3D dynamic code and point kinetics. Good agreement was obtained as long as the trip time determined with spatial kinetics was used in the point kinetics simulation. It was concluded that point kinetics is adequate to analyze accident scenarios provided that the delay time of the accident scenario of interest is known a priori.

In the overall technical assessment, the positive void reactivity coefficient was not considered a detrimental feature. Once the plant control problems outlined above are resolved, and, in particular, the thermal calibration problem, it is expected that the control of power and other key parameters can be shown to be adequate.

Trip Coverage

The shutdown system trip coverage assessment done in Product 09033 made use of AECL and [redacted] analysis results to construct trip coverage maps for the various limiting events. The majority of the most significant findings relate to weaknesses in trip coverage below [redacted] Generally, analyses or further reviews have been recommended to resolve these issues. To improve primary trip coverage for loss of regulation events resulting in slowly increasing power when startup steam is used, an overpower trip with trip setpoint just above [redacted] on the moderator dump system is suggested. To provide backup coverage for slow loss of flow events and slow loss of power control with a cross link failure of the recirculation flow control, addition of a low flow trip on the liquid rods system is suggested. In addition, eliminating the sharing of flux detectors between primary and backup trips, and between shutdown systems is recommended. It was also recommended that a trip coverage assessment of loss of moderator and end shield cooling events be made in conjunction with analyses recommended in Product 09022.

Severe Reactivity Insertion Analysis

Because of the Chernobyl accident, assessment of severe reactivity insertion events has become desirable. An AECL analysis of reactivity insertions beyond the design basis was performed under Product 09039. As part of the study, it was established that during a power excursion, the only mechanism for fuel and channel failure would be fuel breakup due to average fuel enthalpy in excess of 300 cal/g. The main finding of the analysis is that either shutdown system would effectively terminate a guillotine break of two headers, or a simultaneous guillotine failure of more than four steam mains. Since these events are much more severe than any of the design basis accidents there are no recommendations or actions required.

Recommendations Summary

All of the Technical Assessment conclusions and recommendations are summarized system-by-system in Product 09040. Actions are suggested to aid the implementation or dispositioning of the recommendations. A cover sheet is provided with each system package listing any recommendations which are judged to have a high probability of a major effect on the plant cost or schedule.

It is concluded that, implementation of the changes recommended in this report would strengthen a design which is already basically sound.

[redacted]

RECOMMENDATIONS SUMMARY - PRODUCT 09040

RECOMMENDATION #: 331001

SYSTEM(S): 331/332

REFERENCE: Product 09025, Task 4.1/5.1, Section P.2.3.

DESCRIPTION OF RECOMMENDATION OR CONCLUSION: Safety Class Criterion Based on High Pressure Makeup System

[redacted]

CATEGORIZATION: A

REQUIRED ACTION AND/OR COMMENTS: [redacted]

IMPLEMENTATION REQUIRED PRIOR TO PLANT STAGE NO: 3

DISPOSITION

[redacted]

RECOMMENDATIONS SUMMARY - PRODUCT 09040

RECOMMENDATION #: 331006

SYSTEM(S): 331

REFERENCE: Product 09025, Task 4.1/5.1, Section C.2.2.1.

DESCRIPTION OF RECOMMENDATION OR CONCLUSION: Safety Class Upgrading - Feeder Flow Measurement

[redacted]

CATEGORIZATION: A

REQUIRED ACTION AND/OR COMMENTS: [redacted]

IMPLEMENTATION REQUIRED PRIOR TO PLANT STAGE NO: 4

DISPOSITION

[redacted]

RECOMMENDATIONS SUMMARY - PRODUCT 09040

RECOMMENDATION #: 331009

SYSTEM(S): 331

REFERENCE: Product 09025, Task 4.2/5.2, Section P.2.5.4, Task 4.1/5.1, Section P.2.5.

DESCRIPTION OF RECOMMENDATION OR CONCLUSION: Plant Process Condition Transients

[redacted]

CATEGORIZATION: A

REQUIRED ACTION AND/OR COMMENTS

IMPLEMENTATION REQUIRED PRIOR TO PLANT STAGE NO: 5

DISPOSITION:

[redacted]

RECOMMENDATIONS SUMMARY - PRODUCT 09040

RECOMMENDATION #: 331025

SYSTEM(S): 331

REFERENCE: Product 09025, Task 4.2/5.2, Section C.2.3.1c/d.

DESCRIPTION OF RECOMMENDATION OR CONCLUSION: Alarm Setpoint Changes for Safety Valves

[redacted]

CATEGORIZATION: B

REQUIRED ACTION AND/OR COMMENTS: [redacted]

IMPLEMENTATION REQUIRED PRIOR TO PLANT STAGE NO: 4

DISPOSITION:

[redacted]

RECOMMENDATIONS SUMMARY - PRODUCT 09040

RECOMMENDATION #: 331037

SYSTEM(S): 331/I

REFERENCE: Product 09024, Task 1.0, Section * 0

DESCRIPTION OF RECOMMENDATION OR CONCLUSION: Flow Stability in Primary System

The recirculation flow is unstable if a recirculation flow control valve time constant close to the realistic value is used. It is recommended that further studies of the flow stability be made to ensure that sufficient margin to the instability exists.

CATEGORIZATION: B

REQUIRED ACTION AND/OR COMMENTS: This is a potentially serious problem which needs to be investigated in more detail. Much can be learned from commissioning tests, particularly hot commissioning tests with startup steam injected into the core. However, it is not clear that this will provide enough confidence about operation at power. [redacted] Code predictions played a very important role in the discovery of, and the development of a design solution to the "figure of eight" flow instability in the [redacted] design.

IMPLEMENTATION REQUIRED PRIOR TO PLANT STAGE NO: 4

DISPOSITION:

[redacted]

RECOMMENDATIONS SUMMARY - PRODUCT 09040

RECOMMENDATION #: 331058

SYSTEM(S): 331

REFERENCE: Product 09025, Task 11.6, Section 4.5.1.

DESCRIPTION OF RECOMMENDATION OR CONCLUSION: Oxygen Suppression in Steam

[redacted]

CATEGORIZATION: B

REQUIRED ACTION AND/OR COMMENTS: [redacted]

IMPLEMENTATION REQUIRED PRIOR TO PLANT STAGE NO: in-service

DISPOSITION:

1

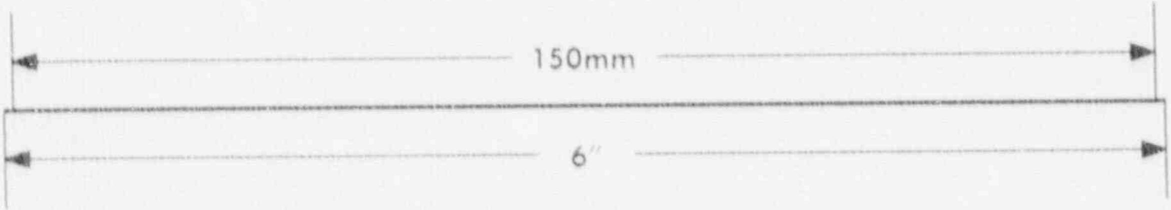
IMAGE EVALUATION TEST TARGET (MT-3)

150mm

6"

1

IMAGE EVALUATION TEST TARGET (MT-3)



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[redacted]

RECOMMENDATIONS SUMMARY - PRODUCT 09040

RECOMMENDATION #: 331067

SYSTEM(S): 331

REFERENCE: Product 09025, Task 11.5, Section 4.6.

DESCRIPTION OF RECOMMENDATION OR CONCLUSION: Editorial Changes to NIRA Document S-33100-PP-100-001

[redacted]

CATEGORIZATION: D

REQUIRED ACTION AND/OR COMMENTS

IMPLEMENTATION REQUIRED PRIOR TO PLANT STAGE NO:

DISPOSITION

[redacted]

RECOMMENDATIONS SUMMARY - PRODUCT 09040

RECOMMENDATION #: 331075

SYSTEM(S): 331

REFERENCE: Product 09025, Task 11.3, Section 4.3.

DESCRIPTION OF RECOMMENDATION OR CONCLUSION: Editorial Comments to NIRA Document S-33100-VA-910-001 and S-33100-YS-000-002

[redacted]

CATEGORIZATION: D

[redacted]

REQUIRED ACTION AND/OR COMMENTS: [redacted]

IMPLEMENTATION REQUIRED PRIOR TO PLANT STAGE NO:

DISPOSITION:

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1.0 SUMMARY

This report presents a summary of the review conducted on the Codes and Standards employed for the design of process systems, pressure vessels and heat exchangers in the [redacted] Nuclear Power Plant.

The review determined the adequacy and completeness of the Codes/Standards (1975 edition*) used in the [redacted] plant design. [redacted] They are complete and generally adequate for use in the design of process systems, pressure vessels/heat exchangers, for the selection and use of materials, and for the inservice inspection of components. But some significant differences and changes exist between the two editions of the Codes/Standards which [redacted]

These differences/changes are identified to be in the areas of:

- design for system overpressure protection;
- review and certification requirements of design reports for pressure vessels and heat exchangers;
- rules for design and analyses of critical components of pressure vessels and heat exchangers.
- inservice inspection of metal containment components

[redacted]

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* For [redacted] Code, it is [redacted] edition up to and including [redacted] winter addenda.

C[redacted]0097/C051

2.0 INTRODUCTION

This report deals with the review of Codes and Standards conducted as Task 05 work in Product 09012 of the [redacted] Technical Assessment. Basically, the scope and the objectives of this review are as follows:

1) [redacted]

2)

3)

The methodology adopted for the review is outlined in Section 3 of this report. Discussions of salient points of review results are presented in Section 4. The conclusions reached from the review and the recommendations are summarized in Section 5.

In Section 6, the [redacted] documents referred in the review are listed. In Section 7, the comprehensive reports from the three review parts are appended.

Throughout this report, wherever reference is made to [redacted] the code implied is [redacted] edition up to and including winter [redacted] addenda for the [redacted] application or the [redacted] edition, with no addenda, currently in use internationally.

3.0 METHODOLOGY

3.1 General

[redacted] These terms are defined as follows:

- [redacted] In these cases, although recommendations are made to use the current version of the standard to review these topics, the [redacted] (The terms applicable, suitable, adequate, appropriate are considered to have the same meaning.)

- [redacted]

In carrying out the review of adequacy and completeness the following steps were followed:

1. The design requirements which should typically be addressed for each type of pressure retaining component were identified.

2. [redacted] and their coverage of the design requirements was assessed.

3. The standards currently used internationally for the design of [redacted]

4. The major changes between the current [redacted] version and the version used on [redacted] were identified and assessed.

3.2 Process Systems

A "process system" means a system associated with the operation of a nuclear power plant, and consists of an assembly of items including supports connected to perform one or more designated functions with the boundaries extending either to a component that provides isolation from the adjacent system or to a discharge or vent port.

In the design phase, classification of process systems into safety classes provides criteria for the process system design from the pressure boundary integrity viewpoint, and, the basis for the design and construction requirements to assure the desired system/component integrity and reliability.

[redacted] their acceptability to the process system design.

Finally a comparison is made of the Task 03 and Task 04 Lists (Ref. 1 and 2) of the Codes and Standards to highlight the differences and omissions of significance to the system design from the safety viewpoint and to outline the significance of the differences between the 1975 and 1983 editions of the Codes and Standards to the process system design.

A comprehensive report of the review is attached as Appendix A to this report.

3.3 Pressure Vessels and Heat Exchangers

In current design practice, the basic design requirements for a typical pressure vessel, as an example, a tube-in-shell heat exchanger, cover the following areas:

i) Calculations for the pressure vessel sizing, including the vessel wall thickness, and for its external supports,

ii) Analysis of critical joints in the vessel assembly,

iii) Selection of materials for the vessel;

iv) Calculations and analysis for designing the internals of the vessel.

The Codes and Standards used in [redacted] for the design of pressure vessel (Ref. 1) are evaluated for their acceptability to the basic requirements defined above.

The final step is to identify the Codes and Standards (Ref. 2) employed internationally in the current pressure vessel design practices. [redacted]

- the [redacted] Codes/Standards are complete for the pressure vessel design
- there are differences/omissions between the [redacted] editions of the Codes/Standards and the significance of these differences that could affect the pressure vessel design in terms of current design practices.

A comprehensive report of the review as above is appended as Appendix B, Table 5 of this report.

3.4 Selection, Welding and Inspection of Materials, and Inservice Inspection

[redacted] These review findings are summarized in Appendix D in Section 7 of this report.

4.0 RESULTS AND DISCUSSIONS

4.1 Process Systems

The salient results of the review performed per Section 3.2 are listed below. Discussions, as appropriate, follow each set of results. For additional details, references are made to the comprehensive report Appendix A, appended in Section 7.

4.1.1 Codes and Standards of [redacted] Edition employed in [redacted]

Codes and Standards [redacted]

- [redacted]
- Standards of the Hydraulic Institute
- Power Test Code PTC-32.1, Nuclear Steam Supply System
- [redacted] N-193, Overpressure Protection of Low Pressure Systems connected to the Reactor Coolant Pressure Boundary.

[redacted]

Other Codes and Standards listed in the Task 03 list as employed for the design of [redacted] Plant are not applicable to nuclear process systems. These are generally applicable to equipment design and materials. Examples of these Codes/Standards are [redacted] Standards (Ref. 1, 2). Some of these Codes and Standards are reviewed in Appendices B and C of this task report, and in Task 06 from the equipment design viewpoint. The Italian Codes and Standards are not reviewed in this part of Task 05.

The [redacted] Code Class is applied to the process system in its design on the basis of the safety class of the system. The safety classification is based on the safety function the system is required to perform. There are four safety classes for the systems, namely SC-1,SC-2, SC-3 and Conventional.

[redacted]

The design of a nuclear system and its components to the requirements of the ASME Code Section III assures an acceptable degree of structural and pressure boundary integrity. The approach taken in [redacted] A similar approach prevails internationally, but the method in which these code requirements are complied with varies.

From the above, it is concluded that the Codes/Standards employed in [redacted] are acceptable.

4.1.2 Codes and Standards of 1983 Edition Employed Internationally

[redacted] (See Section 4.2 of Appendix A). For the [redacted] are used. However, the [redacted] (Div. 1), V, [redacted] That is, the [redacted] are similar to the [redacted] Code. Therefore, it is concluded that the codes and standards employed in [redacted] for process systems design are complete in terms of the Codes and Standards employed internationally.

A comparison of the [redacted] of the Codes and Standards with those used in [redacted] identifies a significant difference between the two editions of the Codes/Standards that may impact on the overpressure protection design for the systems. This difference arises from the revisions to the [redacted] codes during the period 1975 to 1983 due to changes in design philosophy towards better design/protection for Safety Systems and Components. Details of this difference are described in Section 4.4 of Appendix A. Briefly, this difference relates to:

- acceptability of certain types only of overpressure protection devices, e.g relief valves are more acceptable than rupture discs;
- requirements for overpressure protection reports;
- requirements for connections between the protective device and the adjoining system component;
- requirements on relief capacities of the protective device.

4.2 Pressure Vessels/Heat Exchangers

The salient results of the review performed per Section 3.3 are outlined below with the discussions, as necessary, following each set of results. Appendix B in Section 7 is referred to for details as appropriate.

4.2.1 Codes and Standards of [redacted] Edition Employed in [redacted]

Codes and Standards, of effective year [redacted] employed in the [redacted] plant for the design of pressure vessel/heat exchangers are (Ref. 1):

- [redacted]
- [redacted] Standards
- [redacted] Standards (1968 edition)
- [redacted]
- [redacted] Standards

The [redacted] standards are used widely in the U.S.A. and Canada. The [redacted] Standards are extensively used in the U.S.A. and Canada in the design of conventional, non-nuclear pressure vessels. The [redacted] Codes and [redacted] Standards are used extensively for nuclear pressure vessels in the current international design practices. Therefore, [redacted] It also specifies the design stress limits on the basis of design category, safety classification and "essential" status of the pressure retaining components (see Section 3.2.4 of Appendix B). The adequacy of the Codes and Standards listed above are determined based on the criteria and classifications described in the NIRA document.

Verification of the Codes and Standards employed in [redacted] against the basic design requirements for pressure vessels/heat exchangers shows that:

- basic calculations for vessel sizing, wall thickness and its external supports are covered by [redacted]

- the pressure test requirements are covered by [redacted]

- analysis of critical joints is done by using [redacted]

- [redacted] (Ref. 3) and are verified to be conservative with respect to the [redacted] requirements from the pressure boundary integrity viewpoint, The design stress limits for components in design category I are the same as those specified for Level A service conditions [redacted]

- the requirements for the design of the vessel internals and for protection against flow induced vibrations are not specified.

From the above, it is shown that, for the project specific design categorization and safety classifications of the pressure vessels/heat exchangers, the codes and standards employed are generally adequate, except that the requirements for vessel internals design, and protection against flow induced vibrations are not specified. It is noted that these two requirements are not covered by the Codes and Standards but are based on the wide usage in current design practices.

4.2.2 Codes and Standards of 1983 Edition Employed Internationally

The codes and standards employed internationally in current design practices for the design of pressure vessels and heat exchangers are listed in Section 3.3 of Appendix B. Briefly, in addition to those used in the [redacted] design (Section 4.2.1 above), the following are used:

[redacted]

The [redacted] are basically the same as the [redacted] Codes and, hence, their requirements are met through the [redacted] Code.

[redacted] There are no differences in content between [redacted] except in numbering of the standards.

These changes/differences are detailed in the Tables 5 to 7, of Section 4.2.3, Appendix B. These changes/differences are grouped into:

- Significant: A change which may result in a significant increase in the structural integrity or the reliability of the pressure retaining boundary to an extent that the existing [redacted] design may be inadequate in maintaining its safety function. [redacted]

- Not Significant: [redacted] That is, the existing design is adequate to maintain the safety function without incorporating the change.

Briefly, some of the significant changes that adversely affect the existing design are:

- the owner's (third party) review and jurisdictional authority's certification of the Design Reports for each code class component and supports [redacted]

- the rules governing the assumptions for and analysis of stresses in the critical parts of the pressure vessel (Subsection NB-3200);

- the detailed analysis in local stress regions, for example, in the vicinity of openings (Subsection NB-3300);

- the restrictive rules governing the dimensions of the openings relative to the vessel; similar rules for the spacing of the openings, and for the vessel thickness and reinforcements based on the allowable stresses (Subsections NB-3200 and NC-3200)

Another significant change, not listed in the Table 5 to 7, is related to the mandatory requirements for impact testing of [redacted] The impact testing is subject to several exemptions based on size, wall thickness, and type of materials used. In [redacted] design, components in Class 2 and 3 systems are reviewed and impact tests specified, where appropriate, based on similar considerations and on the safety functions. A review of this aspect should be done for the [redacted] design, using the rules of the [redacted] code to identify the components involved. The review should

C50097/C051

examine the material properties and physical dimensions to confirm that the pressure boundary is adequately reliable. Where impact testing is judged necessary and was not performed, the effects of a potential failure should be evaulated.

Significant changes arising from a comparison of the 5th (1968) and 6th editions of the [redacted] Standards relate to:

- the limits on size, maximum operating and design pressures;
- the limits on unsupported tube length.

4.3 Selection, Welding and Inspection of Materials, and Inservice Inspection

The important results of the review per Section 3.4 are discussed below. Appendices C and D in Section 7 of this report are to be referred to for details.

The Codes and Standards of [redacted] edition used in the [redacted] design for the [redacted] are:

- [redacted]
-
-
-

The first three standards are used widely in the U.S.A. and Canada, and hence are also acceptable for use in [redacted] The [redacted] Standards, which are also used extensively in the U.S.A. and Canada, are suitable for use in the design of conventional, non-nuclear quality components. The changes/differences between the [redacted] editions of the Codes and Standards involve:

- [redacted]
-
-

A review of these changes/differences indicate that they do not have an adverse effect on the existing [redacted] selection and use of materials which are based on the [redacted] edition of Codes and Standards.

The exceptions to the Codes and Standards cited by [redacted] (Ref. 4) for the selection and substitution of materials, and the conditions/criteria under which the individual exceptions may be utilized are acceptable because, in general, similar practice of qualified substitution of materials prevails in Canada. However, in Canadian practice, the justification for substitution of materials is only in terms of technical superiority and is not based on economic or supply reasons.

For the [redacted] design has employed the following code: [redacted] For [redacted] plants, even though the objective for inservice inspection is similar to that of the [redacted] is followed. The basic differences between the [redacted] code and the [redacted] are described in detail in Appendix D of this report. Briefly, the main differences are related to [redacted]

The differences/changes between the [redacted] are briefly as follows: (for details see Section 4.1 of Appendix D):

- different levels of visual inspection;
- eddy current examination of steam generator tubes;
- qualification of inspection personnel;
- alternatives in inspection cycles;
- repair procedures for [redacted]
- changes to exemption of components from inspection;
- requirements for more inspection of residual heat removal and emergency core cooling systems;
- new requirements for visual inspection of metal containment components and the acceptance standards for inspection;
- reduced test frequency for inservice testing of pumps;
- temperature criteria for system pressure tests;

A review of these differences/changes between the [redacted] code, and of the requirements of [redacted] Standards for [redacted] plants indicates that the following aspects have an [redacted]

- [redacted]

C0097/C051

[redacted]

5.0 CONCLUSIONS AND RECOMMENDATIONS

5.1 Conclusions

The following conclusions are reached from the review.

5.1.1 Process Systems

The Codes and Standards listed below of [redacted]

5.1.2 Pressure Vessels/Heat Exchangers

The following Codes and Standards, of [redacted] edition, used in the design of the [redacted]

- [redacted]
-
-
-
-

- review and certification requirements of design reports; (see note below)

- rules for stress analysis of critical components;

- rules governing the dimensions of the openings relative to the vessel.

- requirements for impact testing of Class 2 abd 3 Components.

On the review and certification requirements of design reports, it is to be noted that these requirements were added to ensure that, by review, the design was done properly, and by certification, the accountability/responsibility for the design is placed on the designers. [redacted]

It is concluded that the above [redacted]

5.1.3 Selection, Welding and Inspection of Materials, and Inservice Inspection

[redacted] For [redacted] plants, the [redacted] is followed. Although the objective in the [redacted] is similar to that in the [redacted] Code, there are major differences between the two with regard to the inspection details and requirements.

5.2 Recommendations

Based on the above conclusions, the following recommendations are made:

For Process Systems

[redacted]

For Pressure Vessels and Heat Exchangers

1. Guidelines for the design of heat exchanger tubes against flow induced vibrations should be provided.

2. Guidelines for the design of internals in the pressure vessel/heat exchanger should be established.

3. Design reports for all code class components and supports should be submitted for a third party review and certification by the jurisdictional authority.

4. Review of the component design details should be made to assess whether any design modifications are necessary in order to satisfy the changes to the Codes/Standards in the area of local stress regions, allowable stresses, stress type classification, stress concentration factors and the location of nozzles in the vessels.

For Inservice Inspection

1. [redacted]

2. Whether additional inspections are required or not should be determined for components of the residual heat removal systems and the emergency core cooling systems.

3. The inspection program based on [redacted] should be reviewed to ensure that it can be performed with a total radiation exposure which would not require an unnecessarily large number of inspection personnel. If not, measures to limit the extent of inspection should be considered to minimize radiation exposure of inspection personnel.

For Materials

1. The Class 2 and Class 3 components which would require impact testing according to [redacted] should be reviewed to ensure that the pressure boundary of the component is adequately reliable, or that potential failure would not result in unacceptable safety consequences.

6.0 REFERENCES

1. [redacted]

2.

3.

4.

7.0 APPENDICES

Appendix A: Review of Codes and Standards for [redacted] Process Systems Design

Appendix B: Review of Codes and Standards for Pressure Vessels/Heat Exchangers Design

Appendix C: Review of Materials Standards

Appendix D: Review of [redacted]

UNITED STATES
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555

Solicitation No. RS-NRR-89-026

Nuclear Power Reactor Design
Inspection Services

Addendum

PART III: Technical and Management Proposal

AECL Ref. 89-025 Rev. 1
1989 October 20

A/2

TABLE OF CONTENTS

AECL TECHNOLOGIES
SERVICES TECHNOLOGY & ENGINEERED PRODUCTS

A Division of AECL Inc

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Rockville MD 20855
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Telex 303442
Fax (301) 738-2246
1-800-USA-AECL

1989 October 20
File Ref. 89-025-1

Ms. Teresa McLearen
United States Nuclear Regulatory Commission
Room 1011
7920 Norfolk Avenue
Bethesda, MD 20814
U.S.A.

Dear Ms. McLearen:

Subject: Solicitation No. RS-NRR-89-026
Nuclear Power Reactor Design Inspection Services
"Best & Final Offer" Addendum Document

AECL Technologies is pleased to submit this "Best and Final Offer" for the above solicitation.

In accordance with instructions provided at a meeting between representatives of the USNRC and AECL Technologies on 1989 October 16 AECL Technologies is submitting as a supplement to the original proposal, three additional documents consisting of:

Part I: Solicitation Proposal - Addendum No. 1 Best and Final Offer
Part II: Cost Proposal - Addendum No. 1 Best and Final Offer
Part III: Technical and Management Proposal - Addendum No. 1 Best and Final Offer

These three Addendum parts are to be read in conjunction with our previous submission dated 1989 June 15.

As my staff must have mentioned during their discussions with you on 1989 October 16, that I attach great importance to the success of this job and our organization shall endeavour to provide the best resources to support your work.

I would also like to reiterate that we can offer these extensive skills and capabilities, clear of any conflict of interest.

Should you have any questions or wish to have any supplementary data, please do not hesitate to contact this office.

We thank you for this opportunity to be of service.

Sincerely,

D.R. Shiflett

D.R. Shiflett
Vice-President & General Manager
AECL Technologies

cc: Ralph Brittelli Jr. - Atlanta

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00048
89-10-18

PART III: TECHNICAL AND MANAGEMENT PROPOSAL

1.0 INTRODUCTION

In accordance with instructions provided at a meeting between representatives of the USNRC and AECL Technologies on 1989 October 16 AECL Technologies is submitting as a supplement to the original proposal, three additional documents consisting of:

Part I: Solicitation Proposal - Addendum No. 1 Best and Final Offer

Part II: Cost Proposal - Addendum No. 1 Best and Final Offer

Part III: Technical Management Proposal - Addendum No. 1 Best and Final Offer

The part contained herein is Part III: Technical and Management Proposal.

AECL Technologies is a division of AECL Inc. with offices in Rockville, Maryland and Atlanta, GA. AECL Technologies utilities the resources of Atomic Energy of Canada Limited (AECL). AECL Technologies operates under the laws of the State of Delaware.

The organization chart showing the relationship of AECL Technologies to the total corporation is included overleaf.

AECL is one of the largest and most diverse engineering, research and development companies in Canada. AECL is a government-owned (Crown) corporation established formally in 1952.

AECL corporate office is located in Ottawa, Ontario. Day-to-day activities are carried out through two operating companies:

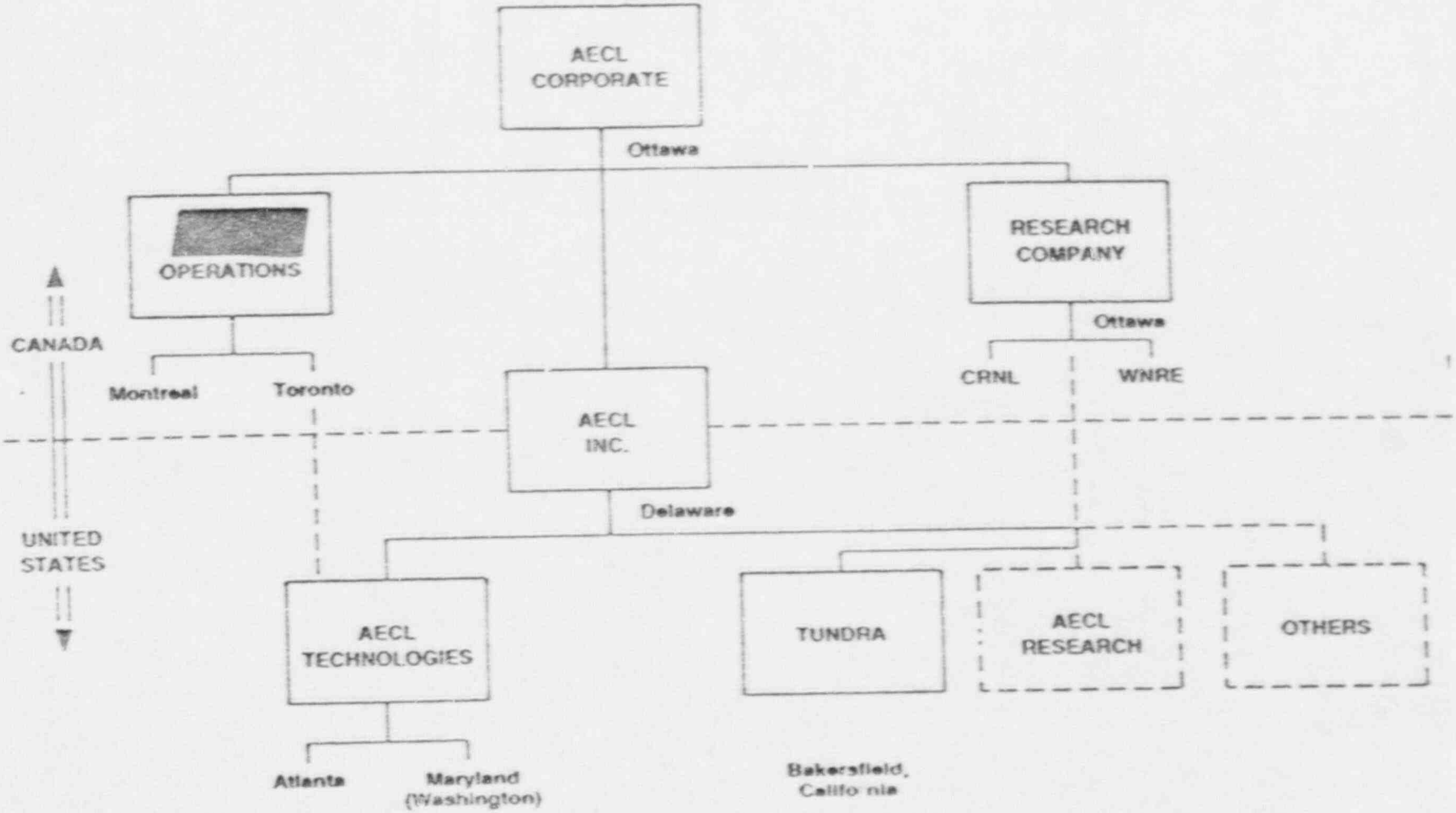
AECL [redacted] Operations,
AECL Research Company.

[redacted] Operations is located in Mississauga, (near Toronto) Ontario. It has a design, test, and commissioning focus for reactor design. [redacted] Operations is a profit-making entity based on projects and services.

The Research Company, a major research and development organization is located at two main sites: the Chalk River Nuclear Laboratories located at Chalk River, Ontario and Whiteshell Nuclear Research Establishment at Pinawa, Manitoba.

AECL ENGINEERING SERVICES

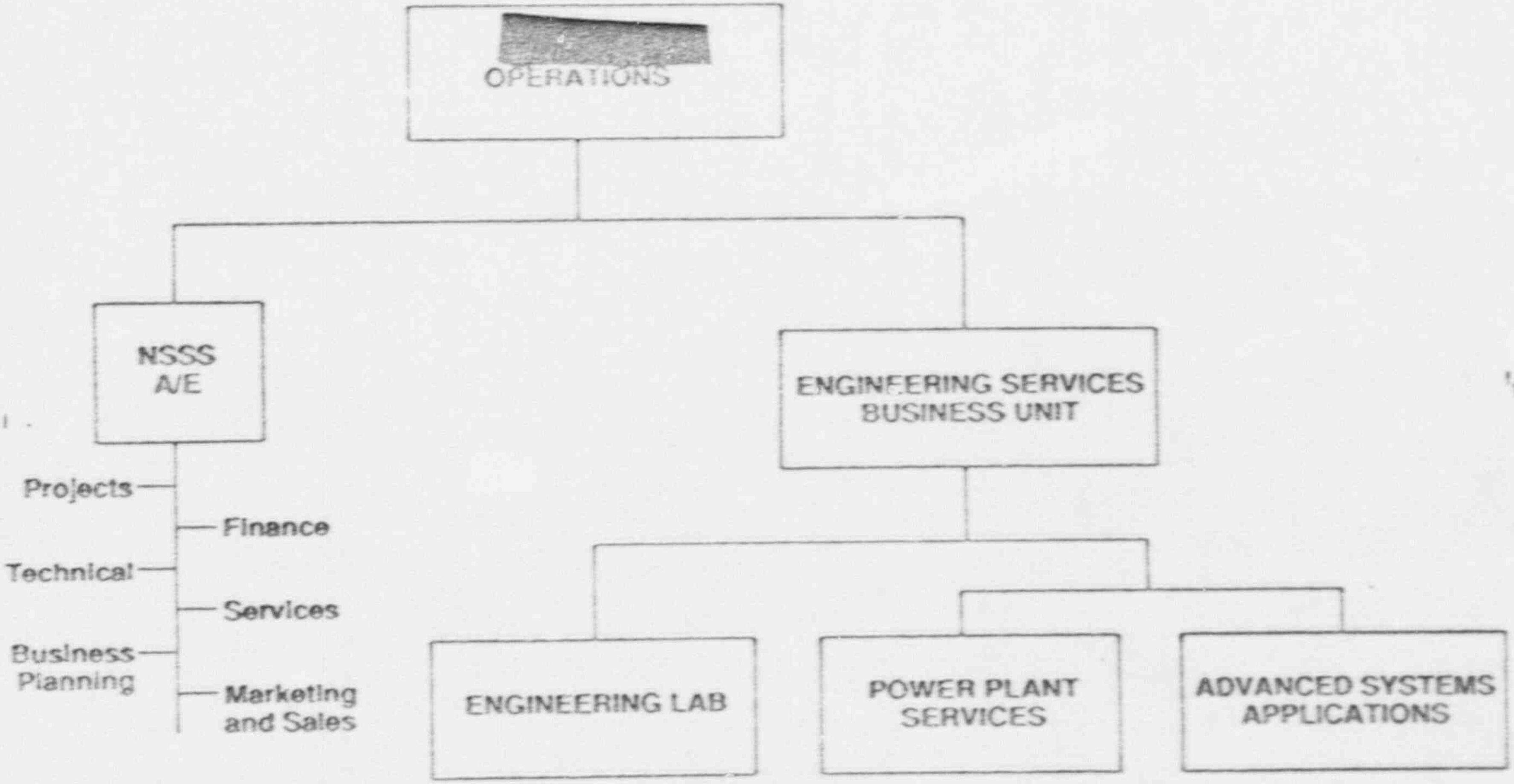
SERVICES, TECHNOLOGY & ENGINEERED PRODUCTS



AECL TECHNOLOGIES

SERVICES, TECHNOLOGY & ENGINEERED PRODUCTS

AECL ENGINEERING SERVICES
SERVICES, TECHNOLOGY & ENGINEERED PRODUCTS



AECL TECHNOLOGIES
SERVICES, TECHNOLOGY & ENGINEERED PRODUCTS

2.0 ITEMS REQUIRING FURTHER DISCUSSION BY AECL

RS-NRR-89026

NUCLEAR POWER REACTOR DESIGN INSPECTION SERVICES

a. AECL should discuss whether the personnel proposed for the subject contract are familiar with IEEE and ANSI standards.

a. *The personnel proposed for the subject contract are familiar with IEEE and ANSI standards.*

For example, the nuclear reactor design of the [redacted] *is based on meeting the requirements of the* [redacted] *Design Manuals and Design Requirement documents which in turn reference the pertinent IEEE and IES standards.*

Another example of AECL familiarity with the standards is the [redacted] *standards*

The [redacted] *design also requires the use of the ANSI standards for the design of* [redacted] *Another example is the use of ANSI standards for* [redacted] *which are extensively used in our designs.*

The personnel on the proposed subject contract have an average of approximately [redacted] *over this period.*

AECL understands that plants of various vintages will be reviewed and inspected by the design inspection team. This means that AECL will be reviewing plants whose applicable IEEE and ANSI standards are different than todays standards. AECL is particularly competent in handling this problem for NRC.

AECL has been designing nuclear power plants for over 30 years. As a result, our personnel have been associated and worked with these ever changing ANSI and IEEE standards. For the less experienced engineering staff members, AECL maintains state-of-the-art information center which enables the quick and efficient retrieval of applicable standards. This information system enables our highly experienced engineering staff to keep abreast of changes in applicable standards.

AECL does not feel that the changes in IEEE and ANSI standards over the years is a problem. We have experience engineering staff and information systems to deal with this issue.

b. AECL should discuss the availability of all personnel identified in Section 6 of the proposal. AECL should clarify the percentage of time the personnel would be committed to other projects and the percentage of time committed to the subject contract.

b. All the names identified in the proposal shall be available to work on the NRC task orders. The plan is to use the staff identified in the proposal as the resource pool from where the staff would be drawn. Should unusual circumstances arise, AECL can also draw on its corporate-wide labor pool of over 4,500 highly trained employees.

Based on a lead time of two or three weeks, all the people nominated are expected to be available to work on NRC task orders. The attached sheet Appendix I provides the percentage of time available to be committed to the NRC task orders [redacted]

To reinforce our resources capability, we have included [redacted] *provided in our earlier proposal of 1989 June 15.* [redacted] *We would like to indicate that among the* [redacted]

We do not anticipate any security or drug related problems based on our previous association with nuclear utilities or other security conscious organizations.

c. AECL should provide details of its understanding of the U.S. regulatory process or exhibit some familiarity with Regulatory Guides, Bulletins and Generic Letters. The offeror needs to address its understanding of U.S. Codes and Standards (e.g ASME and IEEE Standards) in Section 4 of the proposal.

c. *AECL is familiar with the NRC regulatory process. This is primarily based on an in depth internal AECL project study which assessed the US NRC regulatory process requirements in order to prepare a Preliminary Standard Safety Analysis Report (PSSAR) for the Nuclear Steam Supply System (NSSS) of a* [redacted] *under the NRC licensing requirements. The key technical issues (differences) were addressed in this study. One section of that report described the "formal licensing process" of NRC.*

It is to be noted that there are a number of similarities between the Regulatory Processes in Canada and the U.S.A. For example, both processes involve steps leading to:

- *a site approval*
- *a construction permit*
- *an operating licence*

The scope of the technical review and inspection is similar for the two systems.

Talks are presently being held with NRC regarding licensing a [redacted] *plant to NRC requirements. This has also provided AECL an opportunity of learning more about the NRC regulatory process.*

AECL is familiar with the Regulatory Guides and Bulletins. Some of these (e.g. Regulatory Guide 1.29 Seismic Design Classification) has been used routinely in some of our designs. For some areas, we have our own Design Guides, however, a number of the [redacted] *Design Guides make reference to the NRC Regulatory Guides. Most of the people are familiar with and have used the NRC Regulatory Guides in their* [redacted] *design.*

The AECL technical library maintains current versions of US NRC Regulatory Guides and Bulletins.

The proposed personnel have had an extensive working knowledge of the ASME Code [redacted] It should be noted that the [redacted] plants are essentially designed to the ASME codes. There are some [redacted] specific design aspects which are not covered by the ASME Code and designed to the Canadian (CSA) standards.

Another example of our familiarity with the US piping system analysis (including seismic) is that [redacted] Their work was judged to be professional and they made a significant contribution to the project.

d. AECL needs to further discuss its understanding of how the project organizational structure interacts with the NRC team leader to effectively complete a design inspection. This is discussed on page III 15 of the proposal, and in Section L. 16 of the RFP.

d. Upon the award of a task order, AECL will assign a staff member to the work. This representative will meet with the USNRC team leader to discuss scope of work, objectives and schedule. The timing of the initiating meeting will be governed by the urgency of the work as defined by the team leader.

AECL has identified Jarnail S. Panesar as the overall project manager. Mr. Panesar will assist the NRC technical leader with the identification and selection of an appropriate person from AECL's resource list.

In accordance with the team leader directions, AECL shall perform the required scope and report results to the utility, contractor and/or USNRC in accordance with the mandate provided at the initial interview.

Follow up inspections shall be performed as and when required and directed by the USNRC team leader.

The AECL representative shall be responsible to and report to the USNRC team leader.

e. AECL did not address Section L. 16.E of the RFP regarding corporate conflict of interest. This information must be provided for complete evaluation of the proposal and should include [redacted] AECL should also discuss if its licensing application for review of the advanced reactor design presents a conflict.

e. The identification of prior, current and planned work for nuclear utilities are attached in Appendix II.

AECL Technologie [redacted]

i [redacted]

ii [redacted]

iii [redacted]

The AECL Technologie [redacted] *AECL may provide other services consistent with a large design organization but these are not the subject of a focussed marketing plan — AECL has strong qualifications in these areas but these markets are generally well serviced already.*

[redacted]

---

* ATWS Mitigating System Activation circuit
ATWS = Anticipated Transient W/O Scram

Aside from the [redacted] a [redacted] reactor is largely similar to Pressurized Water Reactors — many of the systems are identical in function [redacted]

In the area of decommissioning, AECL has already decommissioned two major power reactors and is just finishing decommissioning, a third prototype reactor. Based on this experience, AECL has developed a full range of decommissioning services [redacted]

We do not see any conflict of interest situation between our licensing application for review of the advanced reactor design and this design review job. We are sure that NRC would have others review that application.

[redacted]

g. AECL notes in Section 2.1.4 of the proposal noted that travel, per diem and accommodations will be in accordance with company standard policy and procedures. The estimated travel costs must use U.S. Federal Travel Regulation per diem rates for travel destinations and allowable Government car rental rates. AECL should re-estimate its costs based on these guidelines.

g. AECL will invoice US NRC for reimbursement of travel costs in accordance with the rates and limits prescribed by the U.S. Federal Travel Regulations.

The re-estimate of costs for the Best and Final offer is being prepared using these regulations.

h. Confirm your understanding that the contract will be on a task order basis and only work that is specifically authorized by task order can be charged. Also, confirm your understanding that only those individuals named in your proposal in response to a task order request are authorized after NRC approval, to perform work. Thus, when people are not engaged in specific work authorized by a task order, their time cannot be charged.

h. We confirm our understanding that the contract will be on a task order basis.

We also confirm that only those individuals named in the task order proposal will be allowed to charge to the NRC job.

i. Confirm your understanding that the total estimated cost for each task order will be negotiated. Also, confirm your understanding that the fee for each task order will be negotiated. HOWEVER, the fee will not exceed the fee ceiling which will be established in the contract.

i. AECL understands and agrees that the total estimated cost and fee for each task order assigned to AECL will be negotiated.

AECL also confirms our understanding that the sum of all fees for services performed will <u>not</u> exceed the fee ceiling to be established in this contract.

j. Be prepared to negotiate a ceiling fee for task orders issued under this contract (see Sections B.2.c and L.16.j).

j. *AECL is prepared to negotiate a ceiling fee for task orders issued under this contract.*

k. In your Best and Final, please extend the time for acceptance of your proposal through December 31, 1989.

k. *The Best and Final offer prepared to be submitted shall be valid for acceptance by US NRC through 31 December 1989.*

APPENDIX I: KEY PERSONNEL

		% Of Time For NRC Work
1.0	Manager/Co-ordinator	
	J.S. Panesar	[redacted]
2.0	Mechanical System	
	[redacted]	33
		33
		33
		33
		25
		25
		25
		25
		33
		33
		33
		33
		33
		33
3.0	Mechanical Component	
	[redacted]	33
		33
		33
		25
	J.S. Panesar	[redacted]
	[redacted]	25
		33
		25
		25
		25
		50
		33
		25
		25
		25
4.0	Civil	
	[redacted]	25
		50
		40
		50

5.0 Structural Engineering

[redacted]

50
33
33
25
[redacted]

J.S. Panesar

6.0 Electrical

[redacted]

25
25
25
50
50

7.0 I & C

[redacted]

50
50
25
50
25