Maintenance Cycle Extension in Advanced Light Water Reactor Plant Design

by

Mark Robert Galvin

Bachelor of Science in Nuclear Engineering
Oregon State University, 1992

Submitted to the Departments of Ocean Engineering and Nuclear Engineering in partial fulfillment of the requirements for the degrees of

Naval Engineer

and

Master of Science in Nuclear Engineering

at the

Massachusetts Institute of Technology

June 2001

© Mark Robert Galvin. All rights reserved.

The author hereby grants to MIT permission to reproduce and distribute publicly paper and electronic copies of this thesis document in whole or in part.

Author ....
Departments of Ocean Engineering and Nuclear Engineering
May 11, 2001

Certified by ....
Neil E. Todreas
Professor of Nuclear Engineering
Thesis Supervisor

Certified by ....
Clifford A. Whitcomb
Research Scientist
Thesis Reader

Accepted by ....
Sow-Hsin Chen
Professor of Nuclear Engineering
Chairman, Department Committee on Graduate Students

Accepted by ....
Henrik Schmidt
Professor of Ocean Engineering
Chairman, Department Committee on Graduate Students

MASSACHUSETTS INSTITUTE OF TECHNOLOGY
JUL 11 2001
LIBRARIES

BARKER

Maintenance Cycle Extension in Advanced Light Water Reactor Plant Design

by

Mark Robert Galvin

Submitted to the Departments of Ocean Engineering and Nuclear Engineering
on May 11, 2001, in partial fulfillment of the
requirements for the degrees of
Naval Engineer
and
Master of Science in Nuclear Engineering

Abstract

A renewed interest in new nuclear power generation in the United States has spurred interest in developing advanced reactors with features which will address the public's concerns regarding nuclear generation. However, it is economic performance which will dictate whether any new orders for these plants will materialize. Economic performance is, to a great extent, improved by maximizing the time that the plant is on-line generating electricity relative to the time spent off-line conducting maintenance and refueling. Indeed, the strategy for the advanced light water reactor plant IRIS (International Reactor, Innovative & Secure) is to utilize an eight year operating cycle.

This thesis has developed a formalized strategy to address, during the design phase, the maintenance-related barriers to an extended operating cycle. The top-level objective of this thesis was to develop a methodology for injecting component and system maintainability issues into the reactor plant design process to overcome these barriers. A primary goal was to demonstrate the applicability and utility of the methodology in the context of the IRIS design.

The first step in meeting the top-level objective was to determine the types of operating cycle length barriers that the IRIS design team is likely to face. Evaluation of previously identified regulatory and investment protection surveillance program barriers preventing a candidate operating PWR from achieving an extended (48 month) cycle was conducted in the context of the IRIS design. From this analysis, 54 known IRIS operating cycle length barriers were identified. The resolution methodology was applied to each of these barriers to generate design solution alternatives for consideration in the IRIS design.

The methodology developed has been demonstrated to narrow the design space to feasible design solutions which enable a desired operating cycle length, yet is general enough to have broad applicability. Feedback from the IRIS design team indicates that the proposed solutions to the investigated operating cycle length barriers are both feasible and consistent with sound design practice.

Thesis Supervisor: Neil E. Todreas
Title: Professor of Nuclear Engineering

Thesis Reader: Clifford A. Whitcomb
Title: Research Scientist

Acknowledgments

To acknowledge by name all the people who were instrumental in the completion of this thesis would require a separate chapter. Whether industry, government, or academia, I was always able to find people willing to assist. However, three people made substantial contributions to my effort and deserve my most sincere thanks:

- Professor Neil Todreas, who provided the influence, advice, and guidance to help me steer the course;
- Larry Conway of Westinghouse Science and Technology Division who critiqued my methodology and results, providing valuable feedback; and,
- My wife Jennifer, who kept me focused on the big picture.

I also want to thank the United States Navy for the opportunity to attend MIT. The experience was both professionally and personally enriching, in no small part due to Dr. Cliff Whitcomb and Captain McCord of the Navy Postgraduate Office.

To my children . . .
always learn

Contents

List of Tables

List of Figures

List of Acronyms

ANSI .... American National Standards Institute

ASME .... American Society of Mechanical Engineers

BOP........... Balance of Plant

CFR........... Code of Federal Regulations

DOE .... Department of Energy

EHC .... Electro-Hydraulic Control

FERC .... Federal Energy Regulatory Commission

GDC .... General Design Criteria

GL .... Generic Letter

IRIS........... International Reactor, Innovative & Secure

IST .... In–Service Testing

MIT .... Massachusetts Institute of Technology

MOV .... Motor Operated Valve

NERI .... Nuclear Energy Research Initiative

NRC .... Nuclear Regulatory Commission

NSSS .... Nuclear Steam Supply System

O&M .... Operations and Maintenance

PEBB .... Power Electronics Building Block

PRA .... Probabilistic Risk Assessment

PWR .... Pressurized Water Reactor

RCPB .... Reactor Coolant Pressure Boundary

RTNSS .... Regulatory Treatment of Non-Safety System

Chapter 1

Introduction

1.1 Impetus

The deregulation of the electric power industry is part of the ongoing national trend to deregulate major industries such as the airlines, telecommunications, and natural gas. The National Energy Policy Act of 1992 allows for the sale of electricity on the open market and for customers to choose their supplier. Also, Federal Energy Regulatory Commission (FERC) Order 888, "Promoting Wholesale Competition Through Open Access Non-discriminatory Transmission Services by Public Utilities, Recovery of Stranded Costs by Public Utilities and Transmitting Utilities," issued in 1996, requires that utility and non-utility generators have open access to the electric power transmission system. It is these stranded costs,[1] or more specifically the need to avoid them, that has motivated the nuclear power industry to develop strategies to improve its economic performance. Only by pursuing these strategies will the nuclear industry guarantee its short term survival, and position itself for long term growth in the deregulated environment.

Conventionally fueled power plants start with an immediate economic advantage over nuclear plants because of lower capital costs. Non–nuclear power plants typically have a shorter construction schedule and lower construction costs, allowing the investors to begin recovering their smaller capital investment sooner. Non–nuclear power plants also benefit from a lack of up front decommissioning costs, less regulatory costs, and (typically) much smaller plant staffing

[1] Stranded costs are investments or assets owned by regulated electric utilities that are likely to become inefficient or uneconomic in a competitive marketplace.

levels. Nuclear plants, however, have a clear advantage over all major electric power producing competitors: significantly lower fuel costs. But a nuclear power plant's lower fuel costs can only offset the higher capital costs if the amount of time spent on–line producing electricity at full capacity significantly exceeds the number of days spent shutdown.

The term typically used to measure the economic performance of a nuclear power plant is unit capability factor. Unit capability factor[2] is the percentage of maximum electricity generation that a plant is capable of supplying to the electrical grid, limited only by factors within plant management's control. Since U.S. nuclear power plants are typically operated at full power, the unit capability factor is directly related to the ratio of on–line days to on–line plus off–line days during any given period. Clearly, then, to improve the unit capability factor the on–line days must increase, the off–line days must decrease, or both. This can be accomplished by focusing on three general areas:

- Increasing the cycle length between refuelings,
- Minimizing refueling and planned maintenance outage times, and
- Reducing the frequency and duration of forced outages.

It should be noted that these three areas are not independent. Increasing the cycle length requires more maintenance to either be conducted on-line[3] or deferred to the refueling outage. However deferring maintenance actions increases the probability of a component failure (which might have otherwise been detected at a shorter maintenance interval) causing a forced outage.

Currently operating pressurized water reactor (PWR) plants are aggressively working to improve their economic performance by optimizing the operating cycle length. When these plants were built most operated on fuel cycles as short as 12 months. Today, many operate on an 18 month cycle and some are transitioning to fuel cycles as long as 24 months. The plant maintenance strategy was developed to support the initial shorter fuel cycle and then modified to support the longer cycle lengths. However, these plants were not built with components that support an extended fuel cycle since it was not foreseen that the nuclear power industry would struggle to

[2]As defined by the World Association of Nuclear Operators (WANO) and the Institute of Nuclear Power Operators (INPO).

[3]*On-line* performance of a maintenance or testing action on a component means that the plant is still at power. The component may be on-service, isolated from it's system, or secured during this period. *Off-line* performance of a maintenance or testing action on a component means that the plant is shutdown.

remain economically competitive. Regardless of advances in core design, which have been significant, the unit capabilty factor (and, hence, economic competitiveness) of currently operating PWR plants will be limited by the performance and maintenance requirements of the installed equipment.

A rapid increase and peak of conventional fuel costs in late 2000 and into early 2001 has been a significant factor in the renewal of interest in nuclear generation. Although there are certified PWR designs available (e.g., the Westinghouse AP600 and System 80+ pressurized water reactors), public concerns for improved passive safety, proliferation resistance, and spent fuel disposal have stimulated new advanced reactor plant designs which address these issues. One strategy to ensure proliferation resistance and potentially reduce the amount of spent fuel generated is to use a fuel cycle much longer than that of currently operating plants, on the order of five to ten years. Economically, there are advantages to matching the maintenance cycle to this longer fuel cycle.

1.2 The Need for a Design Methodology

Reactor plant designers working on the next generation of nuclear power plants must work aggressively to eliminate or mitigate the limitations of the currently operating (legacy) plants. Clearly, then, maintainability must be an important design objective. However, there is currently no methodology for integrating component and system maintainability issues into the reactor plant design process. This thesis will develop such a methodology, and the methodology will be applied to the selection and design of components whose maintenace requirements have been identified as potential operating cycle length barriers for an advanced light water reactor plant.

1.3 International Reactor, Innovative & Secure

The advanced light water reactor plant to which this design methodology will be applied is the International Reactor, Innovative & Secure (IRIS). IRIS is currently being developed by an international consortium, led by Westinghouse and including universities (University of California at Berkeley, Massachusetts Institute of Technology, Polytechnic of Milan), laboratories, industry (Bechtel, Mitsubishi Heavy Industries) and utilities (Japan Atomic Power Company, Tennessee Valley Authority). The nucleus of the effort was provided by the Department of Energy (DOE)

Nuclear Energy Research Initiative (NERI) program which funds the U.S. participants. The original NERI program has attracted international interest and with strong impetus from Italy and Japan has been transformed into a full-fledged international effort to develop a next generation reactor. The main characteristics of IRIS are:

- Enhanced safety systems. Utilization of a single, integrated, self-pressurized vessel and enhanced safety systems with passive safety features making severe accidents leading to core damage impossible. The integral configuration eliminates the possibility of loss of coolant accidents of significant entity, and the reactor is designed for a very high level of natural circulation, thus eliminating the loss of flow accident.

- Proliferation resistance. The core lifetime is projected to be on the order of eight years without fuel shuffling or refueling. Maintenance of the nuclear system is minimized and the goal is to design a reactor island which does not need to be accessed by the operator over the eight-year core lifetime.[4]

- Simple and economical. The capital cost is reduced because of the elimination of entire systems such as refueling, soluble neutron absorber, and emergency core cooling; the use of a single, integrated, self-pressurized vessel; and, simplifications throughout the plant, e.g. reduction in piping and valves. The operations and maintenance (O&M) cost is substantially reduced by the condition-based maintenance strategy, no partial refuelings (which will also increases the availability factor), and the use of modular, easily replaceable components.

- Environmentally friendly. Because of the very long life of the core the amount of radioactive waste spent fuel is drastically reduced (of the order of five times less than current reactors for the same power output). A possibility which will be considered is to dispose of the vessel 'in toto' (i.e., without removing the fuel) which would provide an additional barrier to the escape of radioactive products.

[4]In late 2000, partly based on input from this investigation, the IRIS operating strategy was changed from a ten year cycle to an eight-year cycle with, if necessary, a maintenance shutdown at the midpoint of the cycle. This strategy meets the NERI proliferation resistance objective of no operator access to the fuel since vessel head removal will only be required every other outage. In early 2001, the first core lifetime was changed from high enrichment to low (about 5%) enrichment due to enrichment facility licensing issues. As of April 2001, the first core lifetime is projected to be on the order of five years.

1.4 Goals and Objectives

The top-level objective of this thesis is to develop a methodology for injecting component and system maintainability issues into the reactor plant design process. However, it is recognized that the design process must consider many factors other than just maintainability. Therefore, the methodology developed must not simply identify the "best" design alternative based on maintainability considerations but rather must qualitatively rank proposed alternatives based on overall maintainability. Using this approach, the design methodology will find greater utility since other factors (such as cost) may have a higher design priority but knowledge of the impact of these other factors on maintainability will be possible.

The methodology is intended to be general enough to have broad applicability, yet descriptive enough to ensure that all relevant maintainability factors are considered. It cannot, nor is it intended to, replace the creative element in design. Rather, the methodology is intended simply to focus the creative design effort on those factors which are relevant to the process. Application of the methodology will be illustrated by considering several barriers identified in the IRIS concept design.

Chapter 2

Design Methodology Framework

2.1 Introduction

The design methodology which this thesis seeks to develop can be viewed as a four-step process, shown graphically in Figure 2-1. The first step is the synthesis of the general requirements that the component must satisfy, which is a non-trivial task based on both experience and judgement. The second step is the synthesis of the design objectives with the design requirements. The third step is to bound the solution space by application of suitable and relevant constraints. The final step is to develop design alternatives which meet the specifications of the synthesized design requirements, objectives, and constraints. This chapter presents a brief description of the components of the methodology. Following chapters explore the inputs, present the methodology 'engine' in detail, and demonstrate application of the methodology to several identified IRIS maintenance-related barriers.

2.2 Methodology Inputs

The primary inputs to the methodology 'engine' are shown graphically in Figure 2-1. These inputs are intended to encompass those factors most relevent to component and system design. Starting from *Functional Requirements* and working down (see Figure 2-1), the inputs are ordered such that each successive step in the requirements optimization engine serves to further define the solution space in terms of all previous inputs. The exceptions are the inputs *emerging materials* and

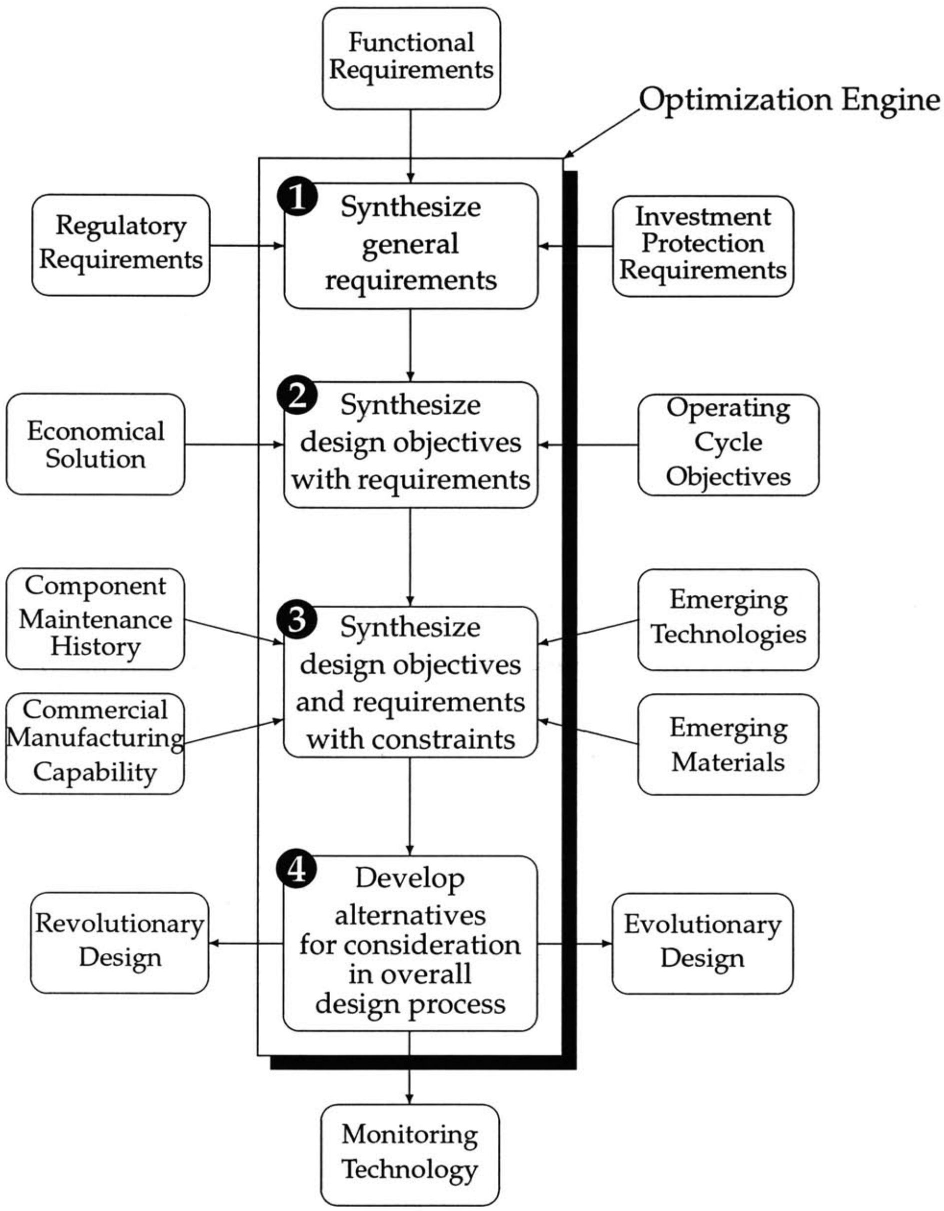


Figure 2-1: 'Requirements Optimization Engine' Concept

emerging technologies, which broaden the solution space by considering future capabilities rather than focusing solely on current capabilities.

2.2.1 Functional Requirements

First and foremost, the functional requirements that the component must meet need to be specified in the most explicit terms possible without introducing bias towards current solutions. If the functional requirements are presented in broad terms, then the solution space will be unmanagably large. If specifications are introduced based on currently used components (i.e., 'relieve pressure with zero seat leakage') then the solution space is artificially narrowed and the solutions which emerge will be biased towards the current component (in this case, a valve).

Early in the design process these functional requirements can often only be presented broadly, and so the solution space must be artificially constrained by making reasonable assumptions based on engineering judgement and experience. As an example, consider the general functional requirement to 'prevent reverse flow' in an arbitrary flow stream. There is no specification of the fluid, fluid conditions (temperature, pressure, and flowrate), conditions when reverse flow must be prevented, or upstream and downstream components. However, reasonable assumptions can be made as to the conditions under which this requirement must be met to further specify the functional requirement and bound the solution space. In the main feedwater supply line, for example, this requirement can be further specified as

- 'prevent reverse flow of high temperature and pressure water or water/steam mixture from the steam generator when the feedwater supply line pressure is less than steam generator pressure,
- allow forward flow of low temperature, high pressure water with minimum resistance, and
- perform functional requirement automatically and without an external energy input.'

Our design paradigms lead us immediately to a swing-type check valve as a design solution to these requirements. However, these requirements could also be met by either of the arrangements shown in Figure 2-2. By not artificially over-constraining the design space, innovative solutions meeting the functional requirements can be generated.

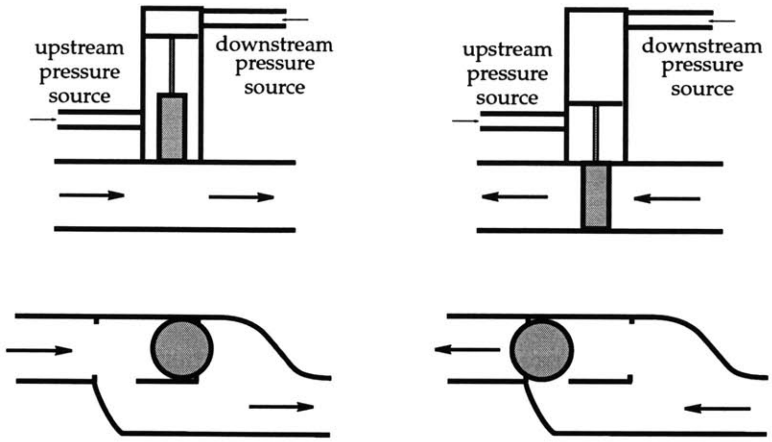


Figure 2-2: Conceptual reverse flow preventers meeting functional requirements. Hydraulically operated gate valve (top) uses differential pressure between the main feedwater pump discharge and steam generator inlet to move the piston. The valve on the bottom uses a ball which is moved by form drag caused by the flowing fluid. For both cases, forward flow is from left to right.

2.2.2 Regulatory Requirements

The purpose of regulatory requirements is to ensure the health and safety of the general public. The scope and periodicity of any regulation should be traceable back to it's role in ensuring public health and safety. For any design decision made, the potential impact of that decision on public health and safety must be assessed so that the regulatory impact can be estimated.

For the purposes of this thesis investigation, it is helpful to view regulatory requirements as being of two categories: those that currently exist and those that are likely to be generated as a result of design decisions which depart from current practice. It would be naive to assume that a creative design solution which satisfies the wording of a particular regulation will automatically satisfy the intent of the regulation. Most regulations are not developed proactively but rather reactively in response to a proposed design configuration or, in some cases, public perception of the risk associated with the proposed design.

The Nuclear Regulatory Commission (NRC) has embarked on a wide range of efforts to increase the effectiveness with which it regulates the nuclear industry. Key to these improvements are three specific initiatives: the Regulatory Excellence initiative; the overall movement toward a regulatory approach that is risk-informed and, where appropriate, performance-based; and the cost-beneficial licensing program.[1] Strategies to make the entire NRC regulatory framework more risk-informed (i.e., such that areas of highest risk receive the greatest focus) and, where appropriate, more performance-based (i.e., more results-oriented and more open to allowing licensee flexibility in how to meet NRC regulatory requirements) are being developed.

The NRC staff has developed generic regulatory guidance, in the form of regulatory guides and standard review plans, as well as on the use of probabilistic risk assessment (PRA) findings and insights in support of licensee requests for changes to their licensing requirements. Pilot applications have approved graded quality assurance requirements and increased allowed outage times for equipment in Technical Specifications.[2] Out of these pilots, application-specific regulatory guides and standard review plans are being developed and are under review by the Advisory Committee on Reactor Safeguards (ACRS).

The NRC cost beneficial licensing action program was established in 1994 to increase agency responsiveness to licensee requests for reduction or elimination of license requirements with small effects on safety but high economic burden. Although activity and involvement in this voluntary program has varied among licensees, the NRC staff has approved over 300 cost beneficial licensing actions. The licensees estimate that the savings resulting from these cost beneficial licensing actions exceed $799 million over the life of the facilities.

The cumulative effect of these NRC initiatives is to create a regulatory environment where the regulatory *intent* is being clarified and adherence to the *intent* of the regulations is being emphasized. Rather than dictate to the licensee how to meet the regulatory requirements, the NRC is shifting the burden to the licensee to determine (and demonstrate) the most appropriate method of ensuring that the regulatory intent is met. This change creates design flexibility, since significant departures from current design practice (such as is the case for IRIS) need only to demonstrate to the NRC what the safety role of the system/component is, the risk significance of the system/component, and what method(s) will be used to ensure that the system/component can perform the specified functions when required.

[1]Statement submitted by the United States Nuclear Regulatory Commission to the Subcommittee on Energy and Power, Committee on Commerce, United States House of Representatives by Shirley Ann Jackson, Chairman, USNRC, March 25, 1998

[2]Technical Specifications are part of an NRC license authorizing the operation of a nuclear production or utilization facility. A Technical Specification establishes requirements for items such as safety limits, limiting safety system settings, limiting control settings, limiting conditions for operation, surveillance requirements, design features, and administrative controls.

2.2.3 Investment Protection Requirements

Some components and systems such as the main turbine generator represent a significant capital investment by the owner/operator, and catastrosphic failure (and associated replacement costs) cannot be tolerated in a deregulated, economically competitive market. To ensure operability of these components the utility performs surveillances,[3] even if not required by regulation. The scope and frequency of these surveillances are determined by trading off the risk (and cost) of failure and subsequent downtime against the cost of performing the surveillances.

A subset of these investment protection requirements are surveillances performed to maintain the revenue stream. They typically are related to components that, if performance is degraded, directly impact the amount of electrical power generated. An example is the main condenser waterbox. Fouling of the condenser tubes by micro-organisms and other organic matter in the cooling water can significantly degrade the condenser heat transfer capability, reducing thermodynamic cycle efficiency. If the main condenser heat transfer capability is degraded, either the plant must be operated at reduced load or operate at full load and risk overheating and potentially severe damage to the condenser (including overpressurization and rupture). It is therefore in the economic interest of the utility to conduct periodic main condenser waterbox cleaning, and waterbox cleaning is currently performed at 18-24 month intervals (coincident with a refueling outage).

2.2.4 Economical Solution

An owner/operator operates for the sole purpose of generating revenue. It typically has no particular preference as to how the electricity is generated (coal, natural gas, or nuclear) as long as the plant meets environmental regulations and is economically competitive in the long-term market. As stated earlier, non-nuclear plants have significantly lower capital costs but nuclear plants incur significantly lower fuel costs. Therefore, to make a nuclear plant attractive to potential investors the capital costs must be reduced.

All design decisions have an impact, either directly or indirectly, on both capital and operating costs. However, potential investors are concerned about both the time to recoup the initial investment (which is directly linked to the capital costs) and the long-term profitability (which is

[3]The term 'surveillance' defines a variety of component tests, inspections, overhauls, and preventive maintenance actions.

directly linked to the operating costs). Necessarily, then, these design decisions must consider the impact on both capital and operating costs together to find an optimal point.

2.2.5 Operating Cycle Objectives

To place the functional requirements in context, an operating (or maintenance) cycle length must be specified. To meet the specified cycle length objective, the component must either require no maintenance for the entire cycle or be maintainable during the cycle. From a practical standpoint, many components (or the systems in which they operate) can often be secured for short periods for the performance of maintenance. However, some components are necessary for continuous plant operation or to ensure safety. These components cannot be secured unless their vital functions can be performed by another component or system.

The operating cycle length goal for IRIS is eight years. It is not an objective to eliminate all maintenance between IRIS maintenance outages, but rather to perform all surveillances which have a periodicity of less than eight years on-line. However, to ensure proliferation resistance no operator access to the reactor island[4] is permitted between refueling outages.

As a strategy to achieve the eight year maintenance cycle length objective IRIS will first look to design solutions which permit on-line maintenance using current techniques and then, if a suitable design solution cannot be found, to development of techniques which will permit on-line maintenance of the current component. The benefit in seeking design solutions first is two-fold: the design will be to current standards and thus less susceptible to regulatory challenge, and the cost of development of new maintenance monitoring and performance evaluation techniques (including costs associated with potentially required regulatory changes) is avoided.

2.2.6 Component Maintenance History

In general, the further a design departs from current practice the greater the risk in terms of both cost and performance. It is prudent, therefore, to evaluate the component which is currently used to meet the specified functional requirements to assess it's deficiencies. A minimal risk solution might be found which involves only a minor modification to the currently used component.

[4]Specifically, no fuel access is permitted by preventing access to the reactor vessel internals during non-refueling outages.

A thorough evaluation of the currently used component may also discover component undesirable attributes which cannot be removed simply by minor component modification. Identification of these components, and the particular attributes requiring redesign, is a critical step in narrowing the number of components needing redesign to a managable size. In his 1996 thesis, Moore[5] presented a strategy for a four year operating cycle at a commercial PWR plant. He concluded that to achieve a four year cycle at the plant being investigated significant modifications would be required, due in major part to a limited number of surveillances which could not be resolved to a four year operating cycle.

2.2.7 Commercial Manufacturing Capability

Implied in any design is the ability to manufacture the various components contained in the design. It is reasonable to assume that a currently manufactured component can be manufactured with minor modifications at roughly the same cost and on a similar manufacturing schedule. New components, on the other hand, require new machine tooling which adds significantly to the component acquisition cost and manufacturing timeline. Additionally, new components require testing and evaluation at much greater detail than modified components which also adds to the cost and procurement time.

2.2.8 Emerging Technologies

As owner/operators work toward a deregulated competitive marketplace, much effort has been expended examining the basis of current maintenance and operating practices. One area receiving considerable attention is reduction of outage duration by conducting maintenance on-line. The byproduct of this attention is research and development of advanced technologies which become on-line maintenance enablers.

The focus in applying these technologies is on currently installed components, since extensive back-fits to install new components which utilize these new technologies are generally not cost effective. But, as a result of these development efforts, undeveloped technologies may exist which

[5]Moore Jr., Thomas Joseph, "A Surveillance Strategy for a Four Year Operating Cycle in Commercial Pressurized Water Reactors," Massachusetts Institute of Technology Department of Nuclear Engineering, Nuclear Engineer's Thesis, May 1996.

would be beneficial to a modified component but were not pursued further since they were not relevent to any currently installed components.

2.2.9 Emerging Materials

For certain component attributes which contribute to shortened life, such as corrosion resistance and susceptibility to embrittlement, new materials may provide solutions where the original component is retained but fabricated from a 'better' material. New materials may allow the component to operate in an environment that the original material could not, saving considerable design effort and simplifying the integration of the component into the overall design. New materials, however, may not necessarily lead to cost savings since they may need to be proven in the anticipated operating environment.

2.3 Methodology

The simplified design resolution methodology is shown in Figure 2-3. The resolution methodology iteratively evaluates the current state of the design against the specified requirements until all the requirements have been met through component modification or redesign.

The resolution process begins with the fuctional requirement to be satisfied and the component currently used to satisfy that requirement. Successive iterations evaluate the design against the next performance requirement in an external process until all requirements have been satisfied. If the current state of the design does not meet a particular requirement, then the design is either modified (if possible) or a new design is generated (if necessary) by external processes.[6] These external processes, described in detail in Chapter 5, draw upon the judgement and experience of the engineer to move past current design paradigms and apply creativity to overcome the imposed barrier.

[6]Used in this thesis, *external* processes are those creative design processes which cannot be formally structured within the resolution methodology.

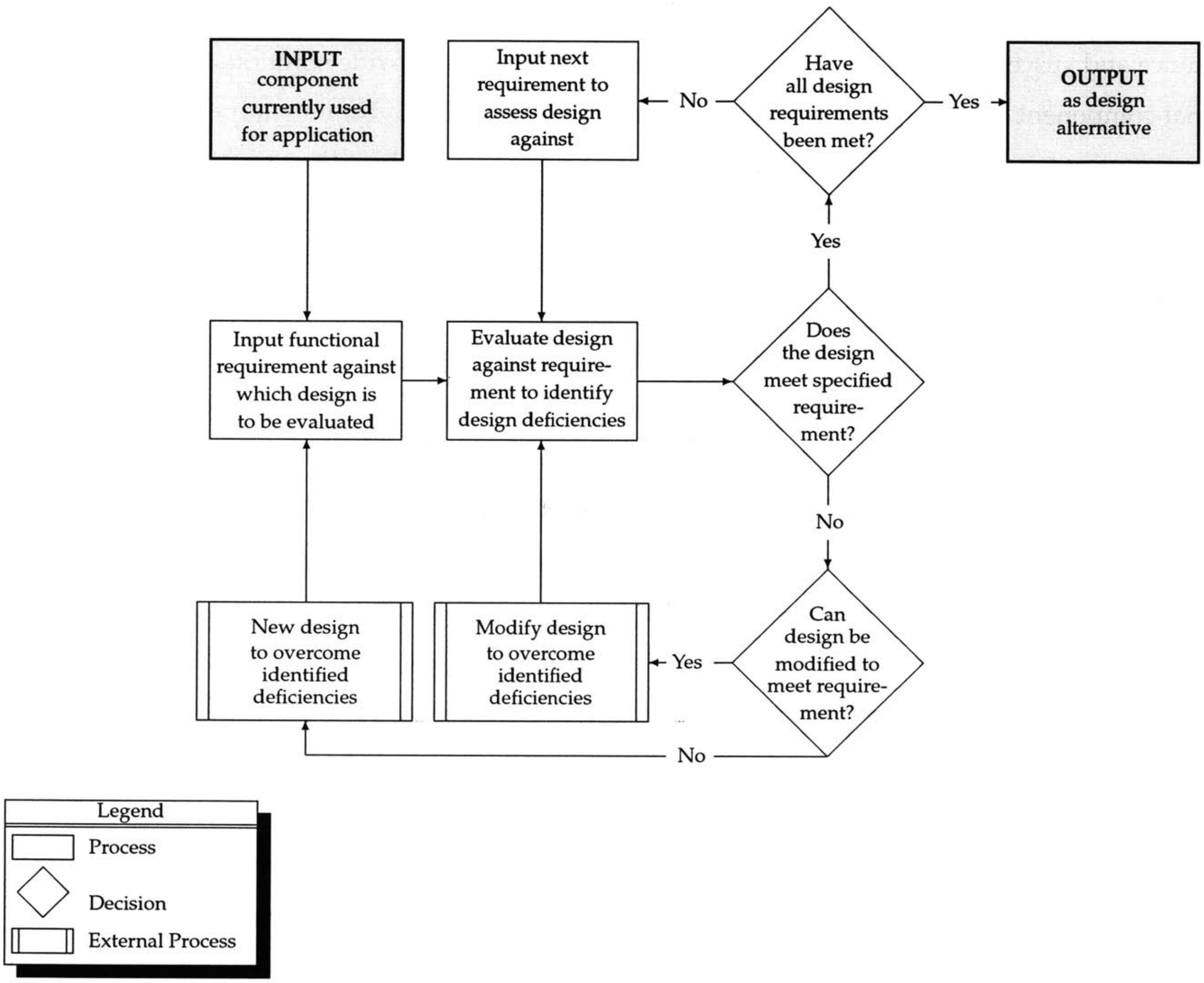


Figure 2-3: Simplified Design Resolution Methodology

2.4 Output

2.4.1 Evolutionary and Revolutionary Design Solutions

When a limitation is identified in a component or system, or an increase in performance above current capabilities is required, design changes are required. The designer can either improve the current component by modifying the component (*evolutionary design*) or by finding a completely different method to meet the design requirements (*revolutionary design*). In practice, all design processes involve a combination of the two. The designer must always evaluate whether a revolutionary design would better accomplish the prescribed function than an evolutionary design in order to ensure that the most cost–effective and best engineered solution is obtained.

Necessarily, IRIS will include both evolutionary and revolutionary design solutions. A truly revolutionary design is inherently unproven, and the economic risk is likely to be unacceptable. A truly evolutionary design is unlikely to significanly improve the performance (including maintainability) of current PWR designs, and therefore may not be economically desirable. The implicit goal for IRIS, then, is to utilize revolutionary design solutions where necessary and evolutionary design solutions where practical.

2.4.2 Design Alternatives

The output from the methodology consists of a set of design alternatives, all of which meet the specified requirements, ranked by maintainability. The methodology will only be a useful tool for design optimization if the maintainability optimization has already been conducted.

Chapter 3

Operating Pressurized Water Reactor Surveillance Program

3.1 Introduction

This chapter describes the cycle length barriers existing in a currently operating pressurized water reactor. The data utilized in this chapter is obtained from a recent thesis investigation into extending a candidate currently operating PWR from an 18 month to a 48 month operating cycle. The data has been reexamined, in the context of this thesis, to determine the types of maintenance related barriers for which a design solution must be found.

After presenting the barriers to the candidate PWR 48 month operating cycle, the implications of the candidate PWR cycle length barriers on the IRIS cycle length are discussed. This chapter concludes by describing where design effort should be focused to resolve the cycle length barriers which will likely exist in an IRIS maintenance program.

3.2 Basis for Operating Pressurized Water Reactor Surveillances

Surveillances are performed either because they are required to ensure safety or because they are prudent to protect capital investment. All surveillances, then, can be categorized into the following two broad categories:

- Regulatory Based: surveillances performed to meet technical specification requirements. In general, the scope of the surveillance and the performance interval are specified by regulatory authority.

- Investment Protection: non-technical-specification-based surveillances, including surveillances performed as a result of committments to agencies other than the NRC. In general the scope, performance mode, and periodicity are selected at the discretion of the owner/operator to protect those systems and components with significant investment costs.

The investment protection surveillances can be further broken down into the reactor and supporting components and systems, referred to as the nuclear steam supply system (NSSS), and all others, referred to as balance of plant (BOP). Since the safety function is primarily associated with the NSSS, most regulatory based surveillances apply to NSSS components and systems.

3.3 A 48 Month Operating Pressurized Water Reactor Surveillance Program

In 1996, Thomas Moore developed a surveillance strategy for a 48 month operating cycle in a commercial PWR.[1] Moore's investigation was part of the Massachusetts Institute of Technology (MIT) Extended Fuel Cycle Project which, under the auspice of the MIT Program for Advanced Nuclear Power Studies, investigated surveillance strategies for extending commercial pressurized water and boiling water nuclear reactor plant operating cycles to 48 months.[2] Moore analyzed the existing surveillance program at a candidate PWR plant to assess the impact of an operating cycle change from 18 months to 48 months. After appropriate justification, surveillances were placed in one of three categories: candidates for on-line performance (Category A); candidates for off-line performance interval extension to 48 months (Category B); and barriers to a 48 month cycle (Category C).

The 3108 surveillances considered at the candidate PWR were categorized as following:

[1]Moore Jr., Thomas Joseph, "A Surveillance Strategy for a Four Year Operating Cycle in Commercial Pressurized Water Reactors," Massachusetts Institute of Technology Department of Nuclear Engineering, Nuclear Engineer's Thesis, May 1996.

[2]McHenry, R.S., T.J. Moore, J.H. Maurer, and N.E. Todreas, "Surveillance Strategy for an Extended Operating Cycle in Commercial Nuclear Reactors," The Fifth International Topical Meeting on Nuclear Thermal Hydraulics, Operations, and Safety (NUTHOS-5), April 14-18, 1997, Beijing, China.

- 2673 in Category A (on–line), of which 67 require a reduced power condition,
- 381 in Category B (extended to 48 months), and
- 54 in Category C (incompatible with a 48 month cycle).

A breakdown of the candidate PWR surveillances is shown in Table 3.1.[3]

It should be noted that 690 electrical related investment protection surveillances were not explicitly analyzed, but were considered conducive to on–line performance and are included in the Category A total.

It should also be noted that many of the 381 Category B surveillances could have been placed in Category A rather than extending the surveillance periodicity. However, the goal of Moore's effort was to develop a balanced surveillance strategy and not simply to maximize on–line surveillance performance. For IRIS, maximizing the on–line surveillance performance will be a key enabler for the eight year operating cycle length objective.

3.4 Extending the 48 Month Surveillance Program to Eight Years

Although the MIT Extended Fuel Cycle Project team developed a four-year surveillance strategy, there has been no industry effort to achieve such an operating cycle length. A practical application of the proposed four-year surveillance strategy would have provided both validation of the methodology and historical data on the effectiveness of the methodology. In the absence of such data, resolving all surveillances at the operating PWR with respect to the baseline four year (or goal eight year) IRIS cycle length would require examination of each of the several thousand surveillances. It is not the intent of this thesis to resolve all the cycle length barriers, but rather to develop a methodology which will assist the reactor plant designer in designing systems which are not cycle length limiting.

Therefore, this analysis will begin with the results of Moore's investigation into a 48 month operating cycle. Moore's investigation identified the barriers to extending the operating PWR from 18 months to 48 months, and provided a methodology for developing the technical justification

[3]Moore Jr., Thomas Joseph, "A Surveillance Strategy for a Four Year Operating Cycle in Commercial Pressurized Water Reactors," Massachusetts Institute of Technology Department of Nuclear Engineering, Nuclear Engineer's Thesis, May 1996, Tables 3-12 and 3-39.

Table 3.1: Recommended Pressurized Water Reactor 48 Month Operating Cycle Surveillance Program[3]

Technical Specification	Surveillances Total	Cat A	Cat B	Cat C
Regulatory Based Surveillances:				
In–Service Testing	229	147	67	15
Reactivity Control	17	13	4	0
Power Distribution Limits	17	17	0	0
Instrumentation	436	390	44	2
Reactor Coolant	18	13	4	1
Emergency Core Cooling System	16	9	5	2
Containment	81	61	20	0
Plant Systems	51	51	0	0
Electrical Systems	846	824	22	0
Technical Requirements	61	61	0	0
Subtotal	**1772**	**1586**	**571**	**20**
NSSS[a] Investment Protection Surveillances:				
Component Cooling	21	8	4	9
Rod Control	22	6	16	0
Chemical Volume and Control	7	3	3	1
Nuclear Instruments	4	0	4	0
Reactor Coolant	84	39	37	8
Residual Heat Removal/Safety Injection	4	0	4	0
Miscellaneous NSSS	84	44	40	0
Subtotal	**226**	**100**	**108**	**18**
BOP[b] Investment Protection Surveillances:				
Auxiliary Systems	83	35	41	7
Condensate	17	14	0	3
Circulating Water/Service Water	19	14	5	0
Diesel Generator	28	26	2	0
Main Steam	39	21	12	6
Feedwater	60	45	15	0
Turbine Systems	72	49	23	0
Miscellaneous BOP	102	93	9	0
Subtotal	**420**	**297**	**107**	**16**
Total	**3108[c]**	**2673[c,d]**	**381**	**54**

[a]Nuclear Steam Supply System

[b]Balance of Plant

[c]Includes 690 electrical systems investment protection surveillances which, although not analyzed, are considered likely candidates for on–line performance

[d]67 at reduced power

for performance interval extension. After application of the methodology, 54 off-line surveillances were identified that would not be compatible with a 48 month operating cycle. An additional 381 off-line surveillances were either already compatible with the 48 month operating cycle or could have their performance interval extended, based on the performance interval extension methodology, to 48 months.

Because of his objectives, Moore's investigation did not consider either shutdown surveillances[4] or off-line surveillances[5] with a performance interval already compatible with a 48 month operating cycle. There are shutdown surveillances which cannot have their performance interval extended to eight years even though they are only required to be performed during an outage. An example is shutdown rod testing, which is required each outage in which reactor vessel head removal occurs. Although no performance interval is specified, it is unlikely that an eight year rod testing performance interval (i.e., the IRIS refueling interval) will be frequent enough to validate the reactor protection system assumptions regarding rod control system performance (e.g., position indication, rod speed, and rod motion without binding). Of greater potential impact are the off-line surveillances which were already compatible with a 48 month operating cycle (and neither identified nor investigated by Moore), but are unlikely to be compatible with an eight year operating cycle.

Assessment of the operating PWR surveillance program relative to the IRIS eight-year operating cycle length objective requires, in part, resolution of the 435 surveillances (54 Category C and 381 Category B) identified by Moore. It must be recognized that the technical justification Moore provided to extend the Category B surveillances to 48 months may not necessarily apply to an interval extension to eight years, resulting in IRIS cycle length barriers.[6] However, development of a methodology which will resolve the identified Category C barriers will likely provide a solution to the unidentified Category B barriers as well.

[4]Shutdown surveillances are those surveillances which are performed in conjunction with a planned outage, usually on components and systems which support the outage. These surveillances are not required to be performed when the reactor plant is at power.

[5]Off-line surveillances are surveillances on components and systems which support power operation, but cannot be performed at power.

[6]As stated in Section 1.3, the IRIS operating cycle objective is eight years. The BNFL economic model indicates that IRIS is still economically competitive with a maintenance outage at mid-cycle, but an eight-year maintenance cycle is preferred. The economic model also indicates that with more than one maintenance outage per eight-year refueling cycle, IRIS economic competitiveness drops considerably.

3.5 Potential IRIS Cycle Length Barriers

The following discussion describes the operating PWR Class C maintenance–related barriers which are potential barriers to the eight-year IRIS operating cycle length. This section only identifies the limitations of current PWRs if the operating cycle were extended to eight years, regardless of whether or not those components and systems would be utilized in IRIS. Although Table 3.1 breaks down the surveillances by system, the discussion here will focus on the general limitations by component type. As noted in Section 3.4 above, some Category B surveillances could also be limiting for an operating cycle length greater than 48 months. However, the Category C surveillances are representative of the maintenance-related barriers existing in the operating PWR surveillance program and a methodology to resolve the Category C surveillances should also lead to resolution of the Category B surveillances as well.

3.5.1 Regulatory Based Surveillances

Regulatory based surveillances are those surveillances performed to meet technical specification requirements. Administrative Technical Specifications and Refueling Technical Specifications were not included in the original analysis, and are not included here, since their specific requirements are independent of cycle length.

3.5.1.1 Relief Valves

At the candidate PWR, there are several regulatory based relief valve surveillances which are currently performed shutdown. The relief valves to which these surveillances apply cannot be tested on–line and, because of their performance history, testing cannot be extended to eight years. At the candidate PWR these 38 valves include the three American Society of Mechanical Engineers (ASME) Class 1 pressurizer relief valves and 35 ASME Class 2 containment pressure boundary relief valves.[7] Extrapolating the candidate PWR's valve performance history, and based on consistent but limited survey results, it appears likely that no relief valve used in

[7]Generally, ASME Code Class 1 includes all reactor pressure boundary components. ASME Code Class 2 generally includes systems or portions of systems important to safety that are designed for post-accident containment and removal of heat and fission products. ASME Code Class 3 generally includes those system components or portions of systems important to safety that are designed to provide cooling water and auxiliary feedwater for the front-line systems.

these applications (regardless of specific type or brand) has a performance history which supports an eight year testing interval.

The Operations and Maintenance of Nuclear Power Plants, ASME/ANSI, OM–1989,[8] Chapter 1, lists the requirements for in–service performance testing of nuclear power plant pressure relief devices. It requires all ASME Class 1 relief valves to be tested every five years <u>and</u> that at least 25% of each type of Class 1 valve be tested every 24 months, 50% every 36 months, 75% every 48 months, and every relief valve be tested at least once every 60 months. Plants have the option of testing the relief valves in place or replacing the relief valve with a bench tested spare. Relief valves which are replaced by bench tested spares are also required to be bench tested after removal to determine if a removed valve exceeds the ±3% set pressure criteria. For those relief valves failing to meet set pressure criteria, the causal effect must be evaluated to determine the need for additional testing. The candidate PWR conducts Class 1 relief valve testing at a shorter interval, coincident with refueling outages.

The Class 2 relief valves can also either be tested in place or replaced by a bench tested spare. Unlike Class 1 relief valves, Class 2 relief valves are only required to be tested every ten years with at least 25% of each type tested every 48 months. However, the performance of the candidate PWR's Class 2 relief valves has not proven historically to be good enough to suggest that testing at an eight year interval would be acceptable.

From the above discussion, it appears unlikley that any Class 1 or Class 2 relief valve can operate for an entire eight year maintenance cycle without testing. Therefore, to eliminate the need for frequent shutdowns a method to either remove or test these relief valves on-line must be developed.

3.5.1.2 Motor Operated Valves

The candidate PWR has surveillances involving motor operated valves (MOVs) which cannot be performed on–line and, based on industry experience with motor operated valve (MOV) performance and subsequent regulatory response, are unlikely to have their performance interval extended. Nuclear power plant operating experience, valve performance problems and MOV

[8]Subsequent updates to "The Operations and Maintenance of Nuclear Power Plants," have occured but the 1989 Edition of Section XI is referenced in 10 CFR §50.55a(b).

research have revealed that the focus of the ASME Code on stroke time and leak-rate testing for MOVs was not sufficient in light of the design of the valves and the conditions under which they must function. For this reason, on June 28, 1989, the NRC staff issued Generic Letter (GL) 89–10, "Safety-Related Motor-Operated Valve Testing and Surveillance." In GL 89–10, the staff requested that licensees and permit holders ensure the capability of MOVs in safety-related systems to perform their intended functions by reviewing MOV design bases, verifying MOV switch settings initially and periodically,[9] testing MOVs under design-basis conditions where practicable, improving evaluations of MOV failures and necessary corrective action, and trending MOV problems. Generic Letter 89–10 was superceded by GL 96–05, "Periodic Verification of Design–Basis Capability of Safety–Related Motor–Operated Valves," issued September 18, 1996.

The code states that the maximum inservice test frequency shall not exceed ten years. In GL 96–05, the NRC staff agrees with this condition of a maximum test interval of ten years based on current knowledge and experience. However, in addition to this maximum test interval, in the case where a selected test interval extends beyond five years or three refueling outages (whichever is longer), GL 96–05 states that the licensee should evaluate information obtained from valve testing conducted during the first five-year or three-refueling-outage time period to validate assumptions made in justifying the longer test interval. Based on performance and test experience obtained during the initial interval, a licensee may be able to justify lengthened MOV periodic verification intervals.

As discussed in GL 96–05, the NRC staff has long recognized the limitations of using stroke–time testing as a means of monitoring the operational readiness of MOVs and has supported industry efforts to improve MOV periodic monitoring under the in–service testing (IST) program and GL 89–10. As such, the staff would consider a periodic verification program that provides an acceptable level of quality and safety as an alternative to the current IST requirements for stroke–time testing and could authorize such an alternative, upon application by a licensee, pursuant to the provisions of 10CFR50.55a(a)(3)(i).

Licensees of several facilities (for example, Callaway, Monticello, and South Texas) have estab-

[9]No specific periodicity is established by the NRC. However, in GL 89–10 the NRC suggested that the MOV data be periodically examined (at least every 2 years or after each refueling outage after program implementation) as part of a monitoring and feedback effort to establish trends of MOV operability. These trends, according to the NRC, could provide the basis for a licensee revision of the testing frequency established to periodically verify the adequacy of MOV switch settings.

lished MOV periodic verification programs that the staff found acceptable during closure of its review of GL 89–10 programs. One approach to MOV periodic verification that the staff found acceptable is to diagnostically test each safety–related MOV every five years (or every three refueling outages) to determine thrust and torque motor–actuator output and any changes in the output. A specific margin to account for potential degradation such as that caused by age (in addition to margin for diagnostic error, equipment repeatability, load–sensitive behavior, and lubricant degradation) is established above the minimum thrust and torque requirements determined under the GL 89–10 program. The selection of MOVs for testing and their test conditions should take into account safety significance, available margin, MOV environment, and the benefits and potential adverse effects of static and dynamic periodic verification testing on the selected MOV sample. Measures such as grouping and sharing of valve performance between facilities are appropriate to minimize the need to conduct more rigorous periodic verification tests.

Two significant conclusions can be drawn from the above discussion. First, a periodic verification program that actually strokes the MOV is the minimum acceptable requirement to verify operability. Second, the longest periodicity deemed acceptable by the NRC is five years and this periodicity is based on utilization of historical performance of the actual MOVs for trending. Therefore, it is unlikely that the testing periodicity of a new MOV can be established at eight years until sufficient performance data can be collected in accordance with GL 89–10. The implication of these conclusions is that, for an operating cycle length greater than five years, an acceptable on-line MOV testing method which actually strokes the valve must be developed.

3.5.1.3 Other In–Service Testing

The ASME Boiler and Pressure Vessel Code, Section XI, "Rules for In–Service Inspection of Nuclear Power Plant Components," requires that all safety related pumps and valves be tested for operability on a quarterly basis. In some instances, performance of the required testing is either hazardous or impossible to perform on–line. The owner/operator may petition the NRC to defer the surveillance, using a risk–based argument. However, deferral has only been previously granted for Refueling and Cold Shutdown surveillances. Refueling surveillances are those which cannot practically be performed with the reactor core installed. Cold Shutdown surveillances are not as limiting as Refueling surveillances, but still cannot be performed with the plant on–line.

In both cases, the surveillances involve only standby systems such as the Residual Heat Removal System and the Safety Injection System.

If a plant incurs an unplanned outage during the operating cycle after more than three months from the last in-service testing period, the ASME code requires the following rules be followed for all surveillances designated as Cold Shutdown Tests:

- Testing is to commence as soon as practical when the Cold Shutdown condition is achieved, but no later than 48 hours after shutdown. Testing shall continue until all testing is completed or the plant is ready to return to power.

- Completion of all testing is not a prerequisite to return to power, and any testing not completed during one cold shutdown should be performed during any subsequent cold shutdown starting with those tests not previously completed.

- Testing need not be performed more often than once every 3 months.

- In the case of an extended cold shutdown, the testing need not be started within 48 hours, but all Cold Shutdown Testing must be completed prior to returning to power.

If a plant operates uninterrupted for an entire cycle, cold shutdown testing is only performed during refueling outages. The ASME code does not address an upper limit on the allowable length between Cold Shutdown Testing. Technical justification would be necessary to extend the permissible interval to eight years, to be consistent with the IRIS cycle length.

3.5.1.4 Engineered Safety Features

The candidate PWR has regulatory based surveillances involving three similar Engineered Safety Feature Actuation System tests which cannot be performed on–line and which are unlikely to be extendable to eight years. These are integrated tests which involve sensors, signal processing, and valve and pump actuation. The tests are:

- Diesel Generator Operability and Engineered Safeguards Pump and Valve Response Time Testing. It would be possible to devise a testing procedure which would test the integrated features of all the safety systems involved, with the exception of actually injecting water into the core. But since (cold) water injection to a critical reactor would risk an unacceptable power excursion, there are no testing scenarios which would allow this test to be performed safely at power. This test is central to proving that cooling water can be delivered to the core in sufficient quantities to mitigate postulated accidents, so the proof–of–flow portion is unlikely to be deferrable.

- Actuation of Auto Safety Injection, Containment Building Spray, and Control Building Air Systems. This surveillance verifies system actuation (and appropriate alarms) within allowable time limits upon receipt of a command signal. Because response time includes the time for the components to physically actuate (i.e., valves open and switches close), acceptable performance is unlikely to be demonstrated using signal monitoring only.

- Emergency Core Cooling Systems Automatic Actuation Test. This surveillance tests that the various Emergency Core Cooling System components will realign within specified time limits upon receipt of a Safety Injection signal including the initiation of feedwater isolation, diesel generator start, containment isolation, containment ventilation systems isolation, and primary component cooling water system realignment. This test cannot be conducted on–line due to feedwater isolation, and because of it's accident mitigation function is unlikely to be deferrable.

These surveillances are performed to ensure that necessary safety systems are operable and will perform when required. Based on their safety importance, they cannot be deferred eight years to the refueling outage. Therefore, a method to verify the operability of these systems on-line must be developed. Particularly challenging will be development of a safe, yet thorough and effective, method to conduct on-line testing of those components which involve physical operations that present a safety risk (such as valve actuation which would permit cold water injection to a critical reactor).

3.5.1.5 Steam Generator Eddy Current Testing

Current NRC inspection guidelines for steam generators require eddy current testing of the steam generator tube bundle at a periodicity of up to 40 months. After conducting an eddy current inspection, the allowed operating period until the next required inspection is established by the owner/operator after analysis of all previous inspection results. The 40 month periodicity can be utilized only after two previous successful inspections at shorter intervals indicate no tube degredation has occured which can potentially lead to tube failure. Although there are development efforts underway, there currently exists no method for on-line steam generator tube inspection. Based on previous experience within the nuclear industry with tube failures due to stress corrosion cracking and aging, steam generator eddy current testing is unlikely to be deferrable.

As a preliminary step in evaluating a transition to a 48 month fuel cycle, the candidate PWR completed a draft technical request to the NRC to extend the interval between steam generator tube inspections to 50 months. The technical evaluation concluded that tube degredation over the course of 50 months in the type[10] of steam generators used at the candidate PWR would not reduce the margins of safety required by NRC Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," August, 1976. No decision was made on this request, however, since the candidate PWR opted to pursue (for a variety of other reasons) a 24 month operating cycle as an intermediate step to an extended operating cycle.

Steam generator tubing makes up a significant portion of the reactor coolant pressure boundary (RCPB), and the industry-wide historical performance dictates that inspections be conducted to verify the integrity of this boundry. The inspection frequency has been established by consideration of this performance, and it is unlikely that it will be feasible to extend this frequency to eight years. Therefore, an on-line inspection method (and the means to conduct it) must be developed.

3.5.1.6 Rod Drop and Rod Position Indication Testing

Control and shutdown rod drop testing is currently performed at the candidate PWR following refueling to guarantee that the control rods have an unimpeded path to the bottom of the core and that maximum drop times are consistent with the assumed drop times used in the plant safety analysis.

The NRC Improved Standard Technical Specifications only require rod drop testing following vessel head removal, and no upper limit to the periodicity is currently specified. However, a senior NRC inspector indicated during an informal discussion that the decision not to place an upper limit on the periodicity was, in part, due to the fact that no current plant operates on a cycle

[10]The candidate PWR uses four Westinghouse Model F steam generators with 5626 Thermally Treated, Inconel 600 U-tubes (SB-163) hydraulically expanded into the tubesheet at each end. The tube bundle is supported by "V"-shaped Anti-Vibration Bars in the U-tube bend region and eight stainless steel tube support plates.

length where fuel swelling would be a significant concern.[11] Since IRIS will operate on a cycle that is more than twice the length of the longest current operating cycle, it is likely that an upper limit will be placed on rod drop testing periodicity and that it will be less than eight years.

Rod position indication testing is normally conducted in conjunction with rod drop testing, and is required every 18 months. For the candidate PWR, once at power the control and shutdown rods are fully withdrawn and remain fully withdrawn with long term reactivity control maintained by primary coolant boron inventory. Since the relationship between actual and indicated position does not change during the cycle (as long as the rods remain fully withdrawn), these checks can be deferred (if necessary) past 18 months to the scheduled shutdown period.

If control rod motion is used for long term reactivity control, instead of using boron inventory, then the assumption that the relationship between actual and indicated position does not change during the cycle is no longer valid. Under these conditions, it is unlikely that the rod position indication checks would be deferrable eight years to the scheduled shutdown period.

3.5.1.7 Reactor Coolant Pumps

Inspection of the reactor coolant pump flywheel bore and keyway are ultrasonically inspected for volumetric expansion in the areas of highest stress concentration every 36 months. Additionally, a complete surface examination of all exposed reactor coolant pump surfaces and a complete ultrasonic volumetric examination is conducted at ten year intervals. All of these inspections require the reactor coolant pumps to be secured.

3.5.1.8 Electrical Breaker Checks

There are several safety equipment breaker overcurrent relay checks which are currently performed at 36 month intervals but, due to their importance in safety assurance, are unlikely to be deferrable to eight years.

[11] In a typical three zone refueling scheme for a plant with a 24 month operating cycle, a batch would remain in the core for 72 months. However, demonstration that the controls rods have an unimpeded path to the bottom of the core is performed every 24 months.

3.5.2 Investment Protection Surveillances

Investment protection surveillances include all the non–technical–specification–based surveillances performed at the candidate PWR. A small number of these are performed as a result of commitments to agencies other than the NRC. In general, however, the investment protection surveillences are performed in the mode and at the interval selected by the utility to protect those systems and components with significant investment costs.

For this investigation, a large number of the Category B investment protection surveillances have been summarily dismissed from consideration as maintenance–related barriers to the eight year operating cycle. These surveillances involve components which can either be easily removed from the design (or replaced by a less maintenance intensive component) or for which an on–line method of performance could readily be developed. An example is installing an instrument bypass in the Reactor Trip System. This allows on–line testing of a single protection channel without the test signal being interpreted by the protective system as a genuine trip signal, and reduces the probability of receiving a spurious protective action.

3.5.2.1 Relief Valves

At the candidate PWR there are several relief valve investment protection surveillances which cannot be performed on–line and, based on component operating history, the testing interval cannot be extended. The relief valves are Class 2 containment boundary valves, of the same design as the regulatory based relief valves, and the same discussion applies.

3.5.2.2 Condenser Waterbox

The main condenser is the primary heat sink for the power plant. If the main condenser heat transfer capability is degraded, then the plant must either operate at reduced power or risk condenser damage due to overheating and overpressurization. The candidate PWR performs condenser waterbox cleaning every 18 months, during which all steam must be secured since the three tube bundles are not individually isolable. Based on material history, this interval cannot be extended.

3.5.2.3 Reactor Coolant Pumps

There are eight reactor coolant pump surveillances, all involving the reactor coolant pump lubrication oil system, which require the pumps to be secured and (based on material history) cannot be performed at a longer interval. These surveillances include checking the pump oil hi/lo level alarms, lube oil sampling, and lube oil change.

3.5.2.4 Main Turbine

The main turbine and generator represent a substantial capital investment, and the large number of surveillances on this machine reflect the magnitude of this investment. In general the main turbine system surveillances involve the speed governer, lubricating oil system, and generator electrical components. The generator surveillances are all performed at 72-96 month intervals, and in general surveillance results at the operating PWR have been satisfactory suggesting that all these surveillances could be extended to eight years. However, the main turbine speed governer and lubricating oil system surveillances are unlikely to be extended past 48 months.

3.5.2.5 Main Steam

There are several surveillances on the main steam isolation valves involving component replacement (software and solenoids) which could physically be performed on-line, but are necessarily performed off-line since isolation valve operation is prevented during performance of the surveillances. The main steam relief valves require periodic lift testing and, like all other relief valves, are unisolable from the system and thus must be tested off-line.

3.6 Summary

This chapter presented the maintenance related operating cycle length barriers existing in the surveillance program of a currently operating PWR. A tabular summary of these barriers in presented in Appendix A. After review of the off-line portion of the operating PWR surveillance program, 54 surveillances are identified as definite barriers and another 381 have been identified as potential barriers. Of these barriers, some will be eliminated by design differences between the operating PWR and IRIS while the rest will need to be eliminated be design.

It is not the intent of this thesis to resolve all the cycle length barriers. However, this evaluation of the maintenance related operating cycle length barriers provides the foundation to develop a methodology which will assist the reactor plant designer in designing systems which are not cycle length limiting. To achieve the IRIS operating cycle length goal, all surveillances must either be conducted with the plant at power or have a maintenance periodicity at least as long as the refueling outage interval.

Chapter 4

Surveillance Resolution Strategy

4.1 Introduction

This chapter outlines the strategy used to resolve the maintenance-related barriers identified in the candidate PWR surveillance program. First, the surveillances are categorized to identify how they will be addressed relative to IRIS. Next, the methods which may lead to on-line performance or complete elimination of these surveillances are presented. After evaluating the surveillances for on-line performance or elimination, those surveillances which remain are those requiring design resolution.

4.2 Surveillance Categorization Relative to IRIS

With respect to IRIS, the operating PWR surveillances can be placed into one of four categories:

Category 1: On-line surveillances which will be performed on-line in IRIS;

Category 2: Candidate surveillances for design resolution to create an on-line performance mode in IRIS;

Category 3: Surveillances requiring further analysis to determine performance mode in IRIS; and,

Category 4: Off-line surveillances likely to have performance interval extended to at least eight years in IRIS.

The flowchart in Figure 4-1 shows how the operating PWR surveillances (Categories A, B, and C) are segregated into these four numerical categories. This thesis will focus on those surveillances requiring design resolution, Category 2. Beginning with all surveillances performed at the operating PWR, it can reasonably be expected that an on-line performance mode in IRIS can be found for those operating PWR surveillances currently performed on-line. It can also be expected that, although likely few in number, there exists surveillances that are likely to have their performance intervals extended to at least eight years. After removing these two groups, the operating PWR off-line and shutdown surveillances remain. These can be immediately segregated into those performed off-line at less than 48 month intervals (i.e., those analyzed by Moore) and those with a performance interval greater than 48 months or performed shutdown.

Although they are expected to be characterized by the Category 2 surveillances, those surveillances with a performance interval greater than 48 months or performed shutdown were not specifically analyzed by Moore and will not be analyzed here. These surveillances are immediately carried down to the 'further analysis' category. Of those surveillances performed off-line at less than 48 month intervals, many have an on-line performance mode and are thus placed in the 'IRIS online' category. The remaining 435 operating PWR surveillances (54 Category C plus 381 Category B) which are currently performed off-line and do not have an on-line performance mode are the potential IRIS cycle length barriers.

From this group of 435 surveillances a qualitative assesment will be made as to whether the surveillance plausibly could have an on-line performance mode in IRIS, based on the IRIS design goals and objectives. Included in this category are surveillances which are not required in IRIS based on expected configuration differences between the operating PWR and IRIS. However, as seen in Figure 4-1, these surveillances are carried into the 'further analysis' category since any change in the IRIS design goals and objectives may make them applicable. Those surveillances for which an on-line mode is not plausible are carried down to the 'design resolution' category.

The last category includes those surveillances which plausibly could have their performance interval extended to eight years. A separate category was created for these surveillances, but it is anticipated that they will be few in number.

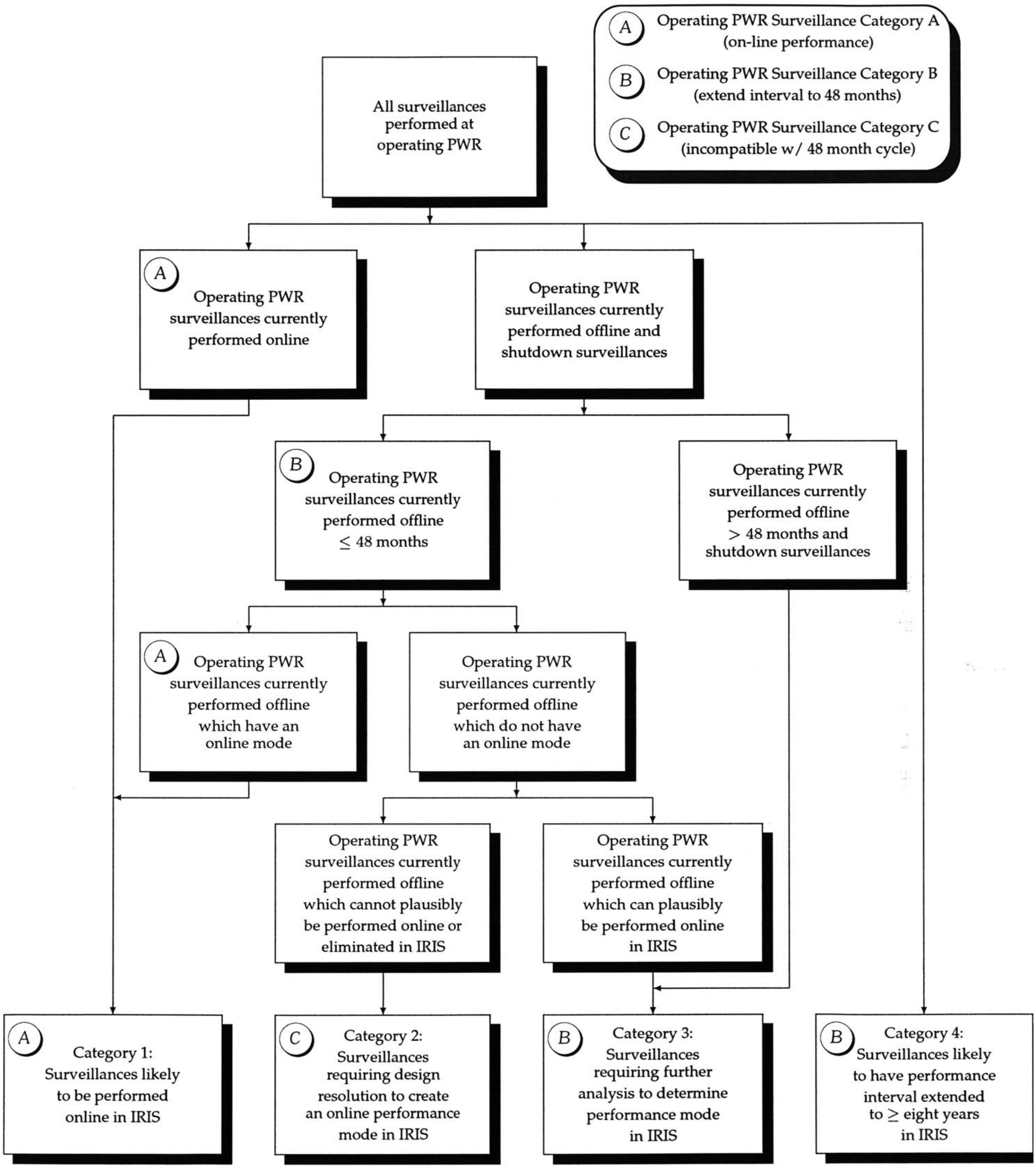


Figure 4-1: Operating Pressurized Water Reactor Surveillance Categories

4.3 Resolution Categorization Considerations

The remainder of this chapter describes in greater detail the considerations given when categorizing the operating PWR surveillances according to Figure 4-1. Although unable to be captured here, engineering judgement and experience are critical to proper evaluation and categorization of the surveillances. Costly design mistakes can often be avoided by having an seasoned designer compare the design team's evaluation against previous design efforts.

4.3.1 Plausibly On-Line or Eliminated Surveillances

All investment protection surveillances can be considered for elimination or performance interval extension, since their performance mode and interval are generally not established by regulatory authority. However, since the intent of these surveillances is to protect a large capital investment, categorically eliminating these surveillances is neither prudent nor responsible. Analysis must consider the cost of performing the surveillance (including planned outage downtime costs) against the cost of unexpected failure (include forced outage costs) if the surveillance is not performed. However, this investigation is based on the premise that the IRIS cycle length objective is a key factor in making IRIS desirable to a potential customer and thus economics will only be considered on a qualitative basis, and only when the cost impact of a particular decision is significant.

Regulatory surveillances, on the other hand, are unlikely to be eliminated since their basis is ensuring protection of the public. It is possible, however, that an acceptable alternative surveillance can be created which provides the same safety assurance. The NRC has indicated a willingness to consider alternatives as long as the proposed method demonstrates operability and is adequately supported by technical justification. For example, the NRC acknowledges the limitations of stroke-time testing of MOVs in assessing operational readiness. It has stated it would consider authorizing a testing program which provides an acceptable level of quality and safety in lieu of stroke-time testing to meet IST requirements.[1]

Because of their importance in demonstrating the ability to perform safety functions, regulatory based surveillances are unlikely to have their periodicity extended to be compatible with

[1] NRC Generic Letter 89–10

the IRIS operating cycle length until sufficient historical data is collected (at testing intervals less than eight years) to provide technical justification for extension. Therefore, unless the first several IRIS operating cycles are of the same length as currently operating PWR cycles (to collect in-situ historical data) these surveillances must either be performed on-line or eliminated by design. To provide for an on-line performance mode, the applicable system or component must either be able to be temporarily taken out of service for testing without compromising operational safety or be able to be tested without interrupting the operability of the component.

To eliminate a surveillance the function performed by the component or system must be performed by a different set of components for which the surveillance is not applicable, or the functional requirement must be eliminated completely. Although this objective is similar to the objective of the 'design resolution' category, these eliminated surveillances are those for which a readily apparent solution is available. Based on IRIS design objectives and goals, there will be no chemical and volume control system and therefore all operating PWR surveillances on this system are eliminated in IRIS. Note, however, that these surveillances are carried into Category 2 (design resolution) since elimination is the design resolution made at a particular stage in the design. If the design changes to include a chemical and volume control system, then the applicable surveillances must be resolved again in terms of the current design objectives and goals.

4.3.1.1 Advanced Monitoring Techniques

Advances in remote and on-line monitoring techniques now allow for conducting many inspections at power in locations which are, due either to environment or radiation, unaccessible by personnel. Examples of these techniques include robot assisted ultrasonic inspection, on-line motion and vibration monitoring, and radiation hardened infrared imaging. A common characteristic of all these techniques is that they are passive, non-destructive, and non-invasive.

Selection of representative indications which can be monitored by these advanced techniques to adequately characterize the condition of the component can produce a two-fold benefit. First, the investment is better protected by more frequent (or even continuous) assessment of component condition without requiring an outage. Second, these techniques are generally passive and no testing-induced failures (which can occur with a time-based surveillance program) are expected.

For microprocessor controlled components and systems, integral diagnostics can be included in the control logic which routes short duration (i.e., too short to cause component or system response) signals throughout the entire circuit to verify electronic continuity. Although this does not demonstrate component response to the applied signal, it does minimize the amount of the system for which assured operation is uncertain.

4.3.1.2 Improved Technologies

Many commercial industries have, over the past several decades, taken a critical look at their maintenance practices in a focused effort to reduce operating and maintenance costs. As a result, a new generation of highly reliable and more easily maintained components have emerged from vendors and manufacturers. In most cases the capital expenditure to backfit a operating PWR with these new components is not justified by the reduced maintenance return, since the operating PWR operating cycle length is currently short enough to perform effective (although frequent) maintenance on the older components. New design and construction, however, affords the opportunity to take advantage of these component improvements.

Switchboard and breaker technology, for example, has improved significantly. Fully-enclosed switchboards are now available which do not require frequent cleaning. These enclosed switchboards can also be fitted with infrared sensors and fire extinguishing agents to minimize the impact of electrical-related fires. The air circuit breaker will soon give way to the solid-state breaker, based on power electronics building block (PEBB) technology (Figure 4-2). Solid-state breakers under development contain the integral diagnostics discussed above which can verify the operability of breaker protective features without interrupting power to the load.[2] Solid-state breakers based on PEBB technology can also be used in an electrically-reconfigurable electric power distribution system, allowing for multiple power sources for a vital component without the need for relays or bus transfer switches.[3] Application of these technologies will eliminate the need to secure a load in order to inspect and verify proper operation of the load's power supply.

[2] U.S. Navy Office of Naval Research, http://pebb.onr.navy.mil.

[3] Borraccini, J., W. Ruby, T. Duong, D. Cochran, E. Roth, D. McLaughlin, and T. Ericsen, "Demonstration of Power Electronic Building Block (PEBB1) Function and Plans for PEBB2 and PEBB3," Government Microcircuit Applications Conference (GOMAC), Las Vegas, March, 1997.

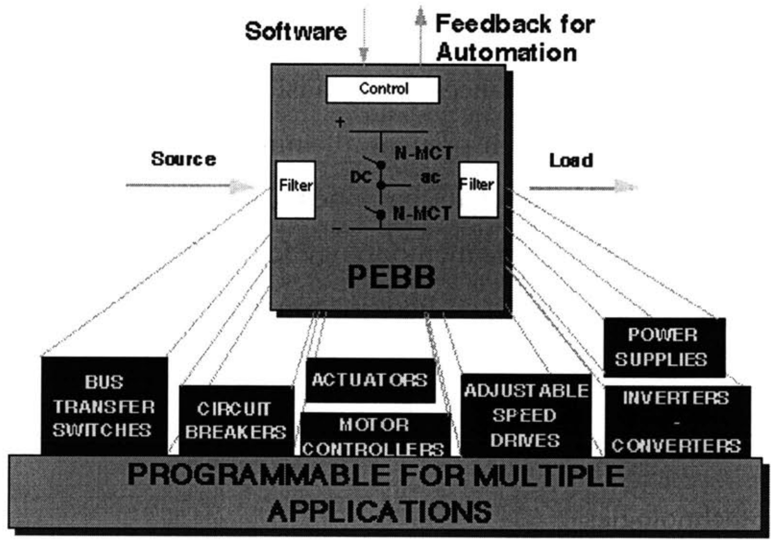


Figure 4-2: Power Electronics Building Block Functional Diagram

4.3.1.3 Redundant Capabilities

Many investment protection calibrations and alignments can have their periodicity extended by installation of diverse, reliable, and redundant monitoring capability. Methods selected should operate different enough (with different calibration curve slopes and failure indications) so that the instruments cannot drift the same way and provide a consistent inaccurate indication. This will ensure that consistent indication correlation between these redundant monitoring methods is accurate and reliable, eliminating the need for instrument calibrations and alignments until a divergence of these redundant monitoring methods is indicated.

4.3.1.4 Regulatory Change

As noted previously, the NRC has indicated a willingness to consider alternatives to current testing requirements as long as adequate technical justification is provided. However, since this technical justification is based on the performance of a particular component in a particular application, it is unlikely that a significant number of regulatory changes would be approved for simultaneous application in a new reactor plant design where no performance history exists.

4.3.2 Surveillances Requiring Design Resolution

After evaluating the off-line surveillances for a plausible on-line solution (or elimination), the remaining surveillances are those which prevent attaining the IRIS cycle length objective. Although this category is populated by discrete components, the aggregate set represents the general challenges to IRIS for which a systematic methodology for resolution must be found.

Where design is necessary to create an on-line performance mode, the preferred order of design is:

1) utilize existing components,

2) utilize existing technologies,

3) develop new components/systems, and

4) develop new technologies.

In order, each method involves increasing design effort and risk.

The reader is reminded that the objective of this thesis is not to resolve all 435 surveillances identified above, but rather to categorize the barriers in more general terms such as classes of components which share common limitations. It is from these generalizations that the resolution methodology will be developed, so that design resolution of particular IRIS maintenance-related barriers can be made.

4.4 Summary

This chapter has outlined the methods available to the designer which can be used to resolve identified operating cycle length barriers. The strategy for eliminating these barriers is "defer if practical, perform on-line when possible, and eliminate by design where necessary." Evaluating surveillances for deferral requires in-depth analysis of the surveillance basis and the component's maintenance history. This evaluation is outside the objectives of this thesis, but is necessary in any reactor plant design effort. Chapter 5, 'Eliminating the Maintenance–Related Barriers,' outlines the methodology used to address those surveillances categorized as requiring an on-line performance mode or elimination.

Chapter 5

Eliminating the Maintenance–Related Barriers

5.1 Introduction

This chapter presents the resolution methodology which is utilized to assist in resolving the identified operating cycle length barriers. It is structured as a flowchart, which methodically and systematically evaluates the current state of the design against the requirements, objectives, and goals. In Section 5.4, each of the decision points and process are described. The intent of developing this methodology is not to introduce new factors for the designer to consider, but rather to organize the relevant factors into a methodology which will assist in identifying where the design effort should be focused.

5.2 Establishing the Solution Space

To ensure that IRIS maintenance considerations are evaluated in the design process, the maintenance requirements must be identified prior to, or concurrent with, the design formulation. However, for a design such as IRIS which will deviate significantly from current commercial PWR practice, these requirements are not well known. These maintenance requirements, although not nearly as well defined as the known limitations of operating PWR components, represent potential barriers to attaining the IRIS objectives.

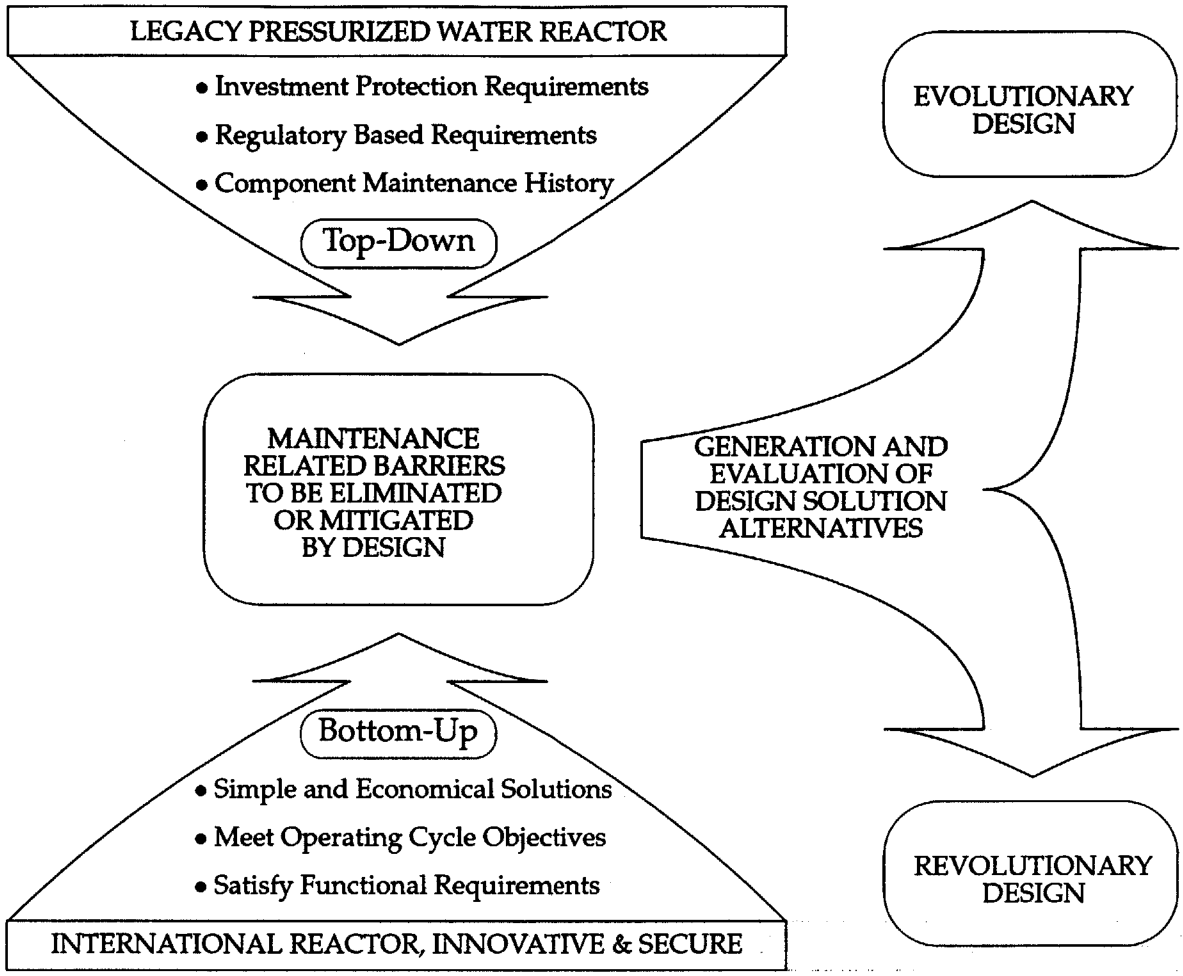


Figure 5-1: Resolution of Maintenance Related Barriers Relevant to IRIS Design

The maintenance–related barriers will be identified using a concurrent top–down and bottom–up approach. The top–down approach starts with the operating PWR and identifies barriers based on maintenance requirements and component operating history (Chapter 3, 'Operating Pressurized Water Reactor Surveillance Program'). The bottom–up approach starts with the IRIS design requirements to determine the best design solutions to meet the design requirements. It is the IRIS design requirements and solutions that determines which operating PWR systems and components could potentially find use in IRIS. From this aggregate set of components/systems and their accompanying maintenance–related barriers, the preferred method of resolution (evolutionary or revolutionary design) will be identified. This is shown graphically in Figure 5-1.

5.3 Resolving the Barriers

The designer can deal with maintenance–related barriers in one of three ways:

1. Modification: Modify the component such that the barrier no longer exists (*evolutionary design*),

2. Substitution: Perform the limiting component's function using a different component that is not subject to the limitation (combination of *evolutionary* and *revolutionary design*), or

3. Replacement: Use an entirely different method to perform the functional requirement (*revolutionary design*).

The method to be utilized depends on a number of factors including cost, technical risk, engineering feasibility, and effectiveness. It is not possible to determine the best method without considering all of these factors.

5.4 Resolution Methodology

This section presents the resolution methodology used to synthesize the requirements into design solutions. It is, and is intended to be, general in nature for maximum applicability. Design inherently requires a high degree of creative thought and engineering judgment, and these intangibles cannot be captured in any methodology. What is asserted is that given a framework to guide this creativity, innovative solutions can more readily be developed.

With the inputs providing the requirements, the methodology must systematically address all the imposed requirements to generate design solution alternatives for consideration by the systems engineer in the overall design. This resolution methodology should perform as a transfer function, inputting the cumulative set of requirements and outputting possible solutions meeting the requirements. To be of utility to the systems engineer, the design solution alternatives must also be qualitatively ranked by maintainability.

The design resolution methodology flowchart is organized into three sequential figures, as shown in Figure 5-2. The flowchart is presented in Figures 5-3 through 5-5.[1] Conceptually, the

[1] The legend in Figure 5-3 applies to all three figures.

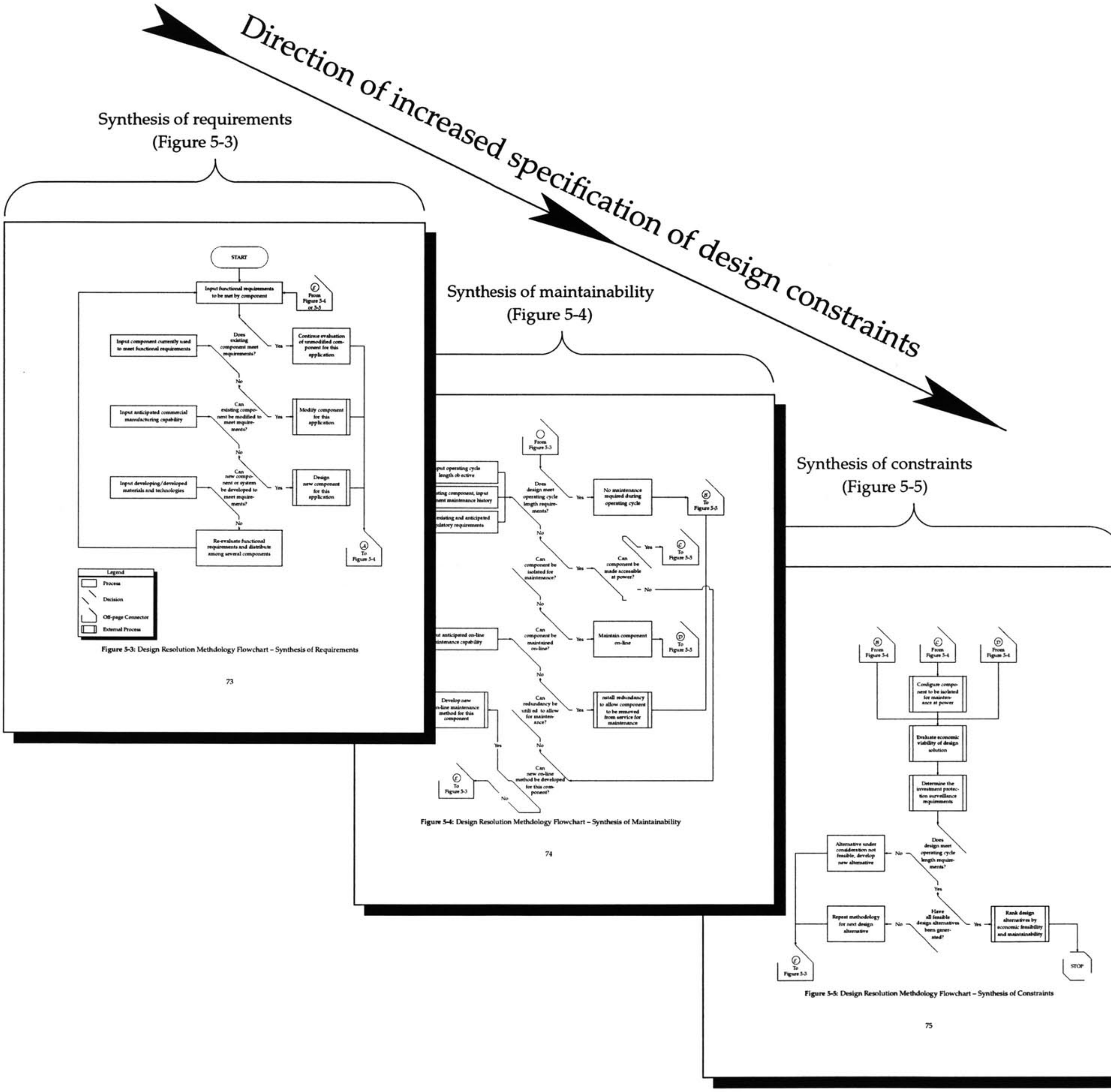


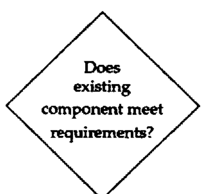
Figure 5-2: Design Resolution Methodology Flowchart – Layout

flowchart sequentially and logically steps through the inputs described in Section 2.2 ('Methodology Inputs') and identifies where design effort must be focused:

- Figure 5-3 synthesizes the requirements and evaluates what level of design effort is required (use existing component, modify component, or design new component).
- Figure 5-4 synthesizes maintainability into the design to resolve when and how the component will be maintained.
- Figure 5-5 synthesizes the economic and investment protection constraints to fully meet all design objectives.

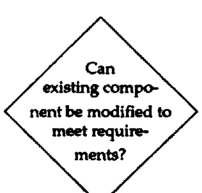
The various decisions and processes of the Design Resolution Methodology Flowchart are discussed in more detail below:

5.4.1 Decisions



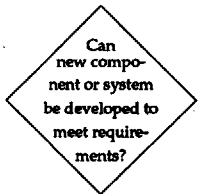
(Fig. 5-3)

This is the fundamental question that determines whether or not a currently used component can be used in the IRIS application. If this is the first iteration, the component currently used to meet these functional requirement is input for consideration. This is a logical starting point, since design effort should only be expended if necessary to minimize the number of unproven components in the design. In most cases, the component currently used will meet the functional requirements in the IRIS application leading to an attempt to use the component. Subsequent passes through the methodology will lead to modifications to the currently used component or a new component or system.



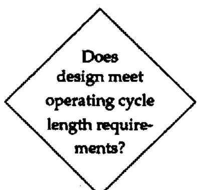
(Fig. 5-3)

If the currently used component fails to meet the functional requirements (first iteration) or result in a feasible design (subsequent iterations) then the next iteration looks for modifications to the currently used component. Since the limitations of the currently used component have been identified by the methodology (either upon entry to the flowchart or after a complete iteration loop), the scope of the necessary changes are apparent to the designer. The anticipated commercial manufacturing capability is used to determine feasible component modifications which will meet the requirements. This is the first point in the methodology where creative (evolutionary design) effort is required.



(Fig. 5-3)

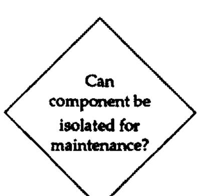
If the component under consideration cannot be modified to satisfy the requirements, then a new (revolutionary design) approach to the requirements is required. At this point, the designer must turn to emerging materials and/or technologies and seek a creative design solution. If a new component or system cannot be developed, then the functional requirements must be re-evaluated and distributed among several components.



(Fig. 5-4)

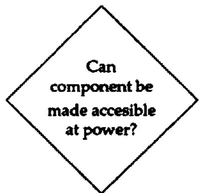
This is the first assessment of the design against the ultimate goal, operation throughout the entire cycle without requiring a plant shutdown for maintenance. To reach this point a component, group of components, or system has been conceived which meets the specified functional requirements. The component is evaluated against the regulatory requirements, with bias given (if an existing component is being used) to the maintenance history of the component. If the component is not substantially modified, then the existing regulatory requirements will likely still be applicable. However, if significant component modification has occured (or a new component designed) then the regulatory requirements must be postulated from the regulatory intent (Section 2.2.2, 'Regulatory Requirements').

If the design meets the operating cycle length requirement, then no maintenance is required for the cycle duration and the design proceeds to economic-related evaluation. If the component does not meet the operating cycle length requirements, due to either performance history or regulatory requirements, then a method of maintaining the component during the cycle must be developed.



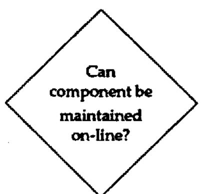
(Fig. 5-4)

If the role of the component in overall operations is such that the component can safely be isolated for maintenance, then the component will be evaluated for at-power accessibility. In this case, often times all that is required is to install sufficient capability to isolate the component from the system. However, some components cannot be secured at power and require plant shutdown. If the component cannot be isolated, then a method of maintaining the component on-line must be developed to avoid the necessity for plant shutdown to maintain that component.



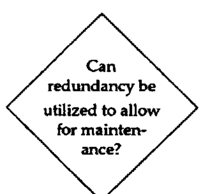
(Fig. 5-4)

Components that reach this point are those that can be safely taken out of service for maintenance. If the component is accessible then the component can be isolated and maintained on-line, and the design proceeds to economic-related evaluation. However many components, particularly those inside the containment building, will be inaccesible due to a high temperature or high radiation environment. For these components, an on-line maintenance capability must be provided.



(Fig. 5-4)

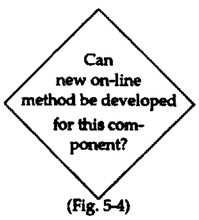
Components that reach this point are those that require an on-line maintenance method. On-line maintenance methods are input; existing methods for the first iteration and proposed methods for subsequent iterations. If a suitable on-line maintenance method is available (or proposed), then the design proceeds to economic-related evaluation. If, however, a suitable on-line maintenance method is not available then evaluation of other at-power maintenance methods is conducted.



(Fig. 5-4)

Installation of redundancy often solves the maintainability issue for small components which cannot be removed from service at power, such as pumps and valves. For larger components, such as heat exchangers and turbine generators, this becomes cost prohibitive. However, in the development of design alternatives, installation of redundancy may be the only solution for at-power maintenance.

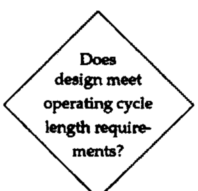
Note that installation of redundancy is only effective for components that are accessible at-power unless sufficient 'installed spares' are provided (to operate when running components are 'retired' in-place) to achieve the desired operating cycle length. If redundancy cannot be provided, then a new on-line maintenance method must be developed for this component.



(Fig. 5-4)

When all attempts at making the component maintainable at power through modifications and configuration changes fail, a new on-line method for maintaining the component must be developed. Like installation of redundancy, this path may lead to an economically non-viable solution.

Often, technologies which may lead to at-power maintainability are only in the early stages of development and thus require an investment (which is ultimately reflected in the total plant cost) to adapt these developing technologies for the required application. If an on-line method can be developed, then the design proceeds to economic-related evaluation. If not, then this design alternative is not viable in it's current state and so another iteration begins.



(Fig. 5-5)

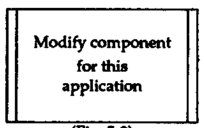
This is the final check of the design alternative to ensure that, after determination of the investment protection surveillance requirements, the desired operating cycle length can still be achieved. A component which reaches this point in the methodology can only fail to reach the desired operating cycle length due to investment protection concerns. If this is the case, then the alternative is not feasible in it's current state and so another iteration begins (with the a priori knowledge of the investment protection surveillance requirements).

The investment protection requirements potentially could be changed to be consistent with the desired operating cycle length. However, the investment protection surveillance requirements are determined based on risk to the owner and are independent of the desired operating cycle length. The designer must resist the temptation to modify these requirements to make the design compatible with the desire operating cycle length unless the risk to the owner is re-evaluated, and this should only be done after another iteration through the methodology (which will now consider the investment protection surveillance requirements).

(Fig. 5-5)

If all possible design alternatives have not been generated (i.e., consideration of existing component, modified component, and new component(s)) then the process is restarted without bias towards previously generated alternatives. If all feasible design alternatives have been generated, then the design alternatives are ranked by economic feasibility and maintainability and the procedure is exited.

5.4.2 External Processes

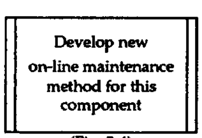


(Fig. 5-3)

Component modification seeks to make a minor change to the component that is with current manufacturing capabilities. The objective of this process is to make the changes necessary to meet the functional (first iteration) or cumulative (subsequent iterations) requirements specified for the component without also requiring a new manufacturing process. In some cases, the necessary changes can be met by an existing component developed for a different application. If not, then a custom manufacturing process (based on existing manufacturing practices) will need to be developed.

(Fig. 5-3)

If a current component cannot be modified to meet the specified functional requirements, then a new component must be designed. This typically requires significantly more creative design effort than simple component modification, but can result in a component better suited for the specific application than a modified existing component. Although design of a new component matches requirements to functionality, development and testing of the new component typically involve more time and cost than modification of an existing component.



(Fig. 5-4)

When a component simply cannot be isolated for maintenance during the operating cycle without a shutdown, then a means of maintaining the component on-line must be developed. In many cases, the maintenance actions required to be performed are inspections which allow predictive techniques to be used in estimating the remaining length of satisfactory operation of the component. Depending on the basis for conducting the maintenance (regulatory requirement or investment protection), development of an alternative inspection method which provides equivalent data on the overall health of the component may be acceptable.

Recent focus on predictive maintenance techniques has resulted in development of new methods for assessing component wear and performance. However many of these technologies, although promising for future use, are not sufficiently developed to transition into an actual field application. Therefore, utilization of these immature technologies will require time and money to research, develop, and test the technology for field application.

Install redundancy to allow component to be removed from service for maintenance
(Fig. 5-4)

Installation of redundancy is the simplest of the methods for creating an at-power maintainable component, so long as the component is accessible. The obvious drawback to this method is capital expenditure for installed spare components in the parallel path(es), especially for large components like heat exchangers and turbine generators. Some systems, such as those that are normally subject to large deviations from their nominal operating point, lend themselves well to installation of redundancy. An example is a cooling water loop, where the number of pumps required to be in operation is dependent on the temperature of the cooling medium.

Configure component to allow access for on-line maintenance at power
(Fig. 5-5)

Most components that can be taken out of service for at-power maintenance can be made accessible by physically moving the component to an accessible location. High temperature or radiation inside the containment vessel are the most common reasons that a component is inaccessible.[2] Moving the component to an accessible often requires only an additional length of piping or cabling, with appropriate consideration given to the impact of that addition to the overall design.

[2]The IRIS vessel is a large integral vessel with internal radiation shield plates located in a 1.5 *m* annulus. Preliminary calculations indicate that, due this thick shielded water annulus, the dose rate adjacent to the vessel during high power operation will be near background. The compact containment design, however, may result in high temperatures (especially at high elevations within the containment).

Evaluate economic viability of design solution
(Fig. 5-5)

Evaluation of the economics of a design decision is a complex process which involves both capital and O&M cost considerations. The viability of a design, and the design decisions made along the way, depend strongly on the owner's financial goals and objectives which are usually not well known during the design phase. Therefore, a baseline 'owner profile' must be established to place the other external factors (such as projected market conditions or the cost of borrowing money) in perspective.

This economic analysis is beyond the scope of this thesis, but is included in the methodology for completeness. What can be qualitatively asserted, however, is the relative economic risk and benefits of one design alternative over another. In later chapters of this thesis, this qualitative assessment will be substituted for detailed economic analysis when evaluating IRIS design alternatives.

Determine the investment protection surveillance requirements
(Fig. 5-5)

The investment protection surveillance requirements applicable to a particular design alternative, like economic viability, are strongly dependent on the owner. In this case, the owner's economic risk threshold directly influences the amount of investment protection maintenance to be conducted. In general, large capital expenditure components and those components whose operation is directly linked to plant output receive the most maintenance attention. For these components, failure typically results in plant down-time and high component repair or replacement costs. As with the economic analysis, this thesis can only qualitatively estimate the investment protection surveillance requirements that a baseline 'owner profile' would establish.

Rank design alternatives by economic feasibility and maintainability
(Fig. 5-5)

The methodology produces several design alternatives for consideration in the overall plant design. Ranking these alternatives by economic viability and overall maintainability is essential to identify the relative advantages of one alternative over another. However, like the economic viability and investment protection surveillance requirements assessment this ranking directly depends on the preferences of the prospective plant owner. Therefore, this ranking of alternatives (particularly the assessment of 'maintainability') will be only qualitatively performed.

5.5 Summary

The design resolution methdology described above and presented in Figures 5-3 through 5-5 systematically and methodically incorporates the design requirements, goals, and objectives into design alternatives which are then assessed against the specified constraints. The output from the methodology is a set of design alternatives which are ranked according to economic feasibility and maintainability.

The methodology presents a general framework of factors to be considered when resolving identified maintenance-related barriers to a specified operating cycle length. A nuclear reactor plant is a complex group of diverse systems and components. The methodology is therefore general enough to generate design alternatives to resolve a broad spectrum of barriers, yet structured enough to focus the design effort on the important factors and considerations.

In the remaining chapters, the methodology is applied to resolve the barriers identified in Chapter 3. To illustrate application of the methodology, Chapter 7, 'Application of Resolution Methodology–Reactor Vessel Overpressure Protection,' will explicitly step through the methodology flowchart presented in Figures 5-3 through 5-5 (pages 73 through 75). The other chapters present only a summary of the relevant factors and the methodology output.

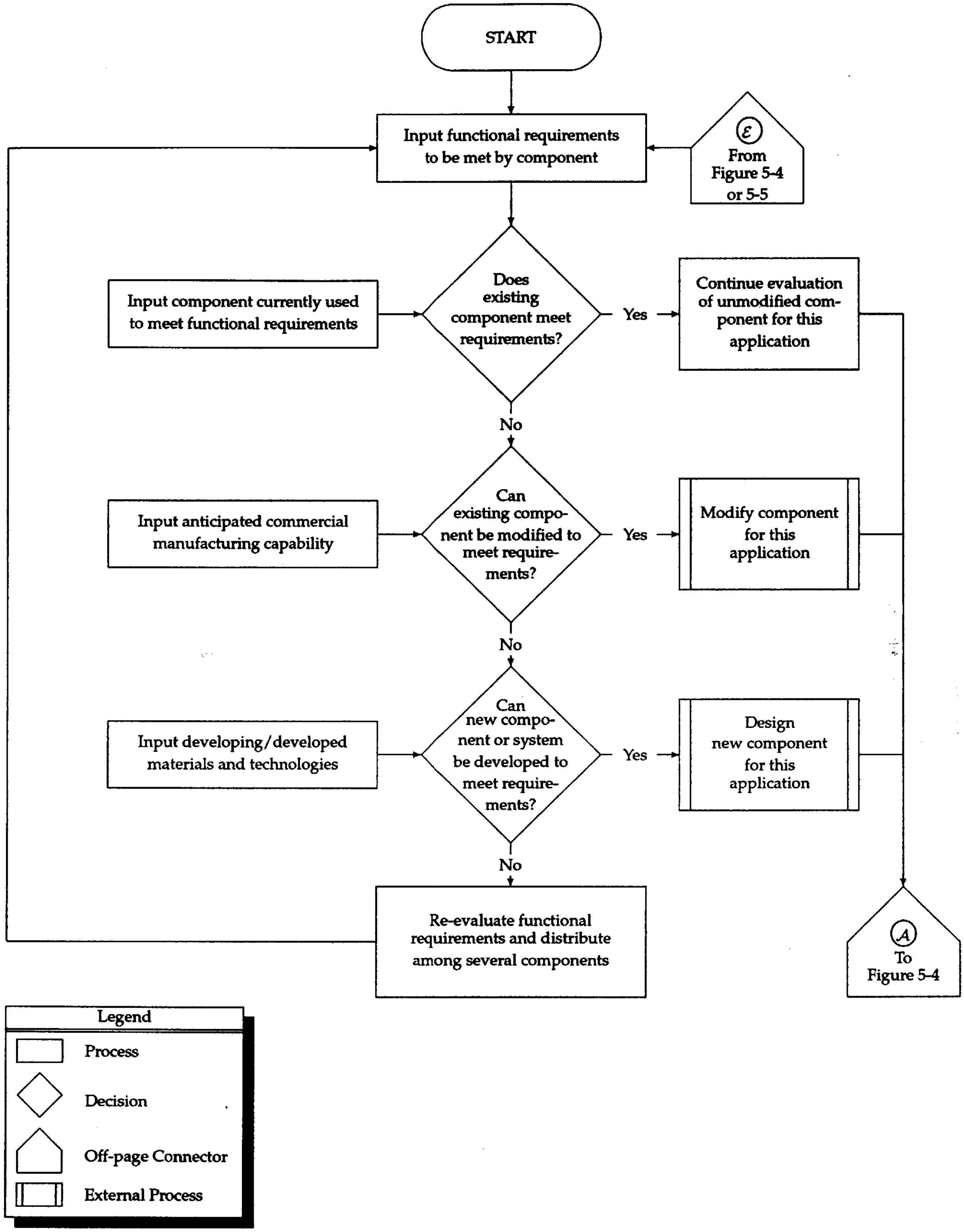


Figure 5-3: Design Resolution Methdology Flowchart – Synthesis of Requirements

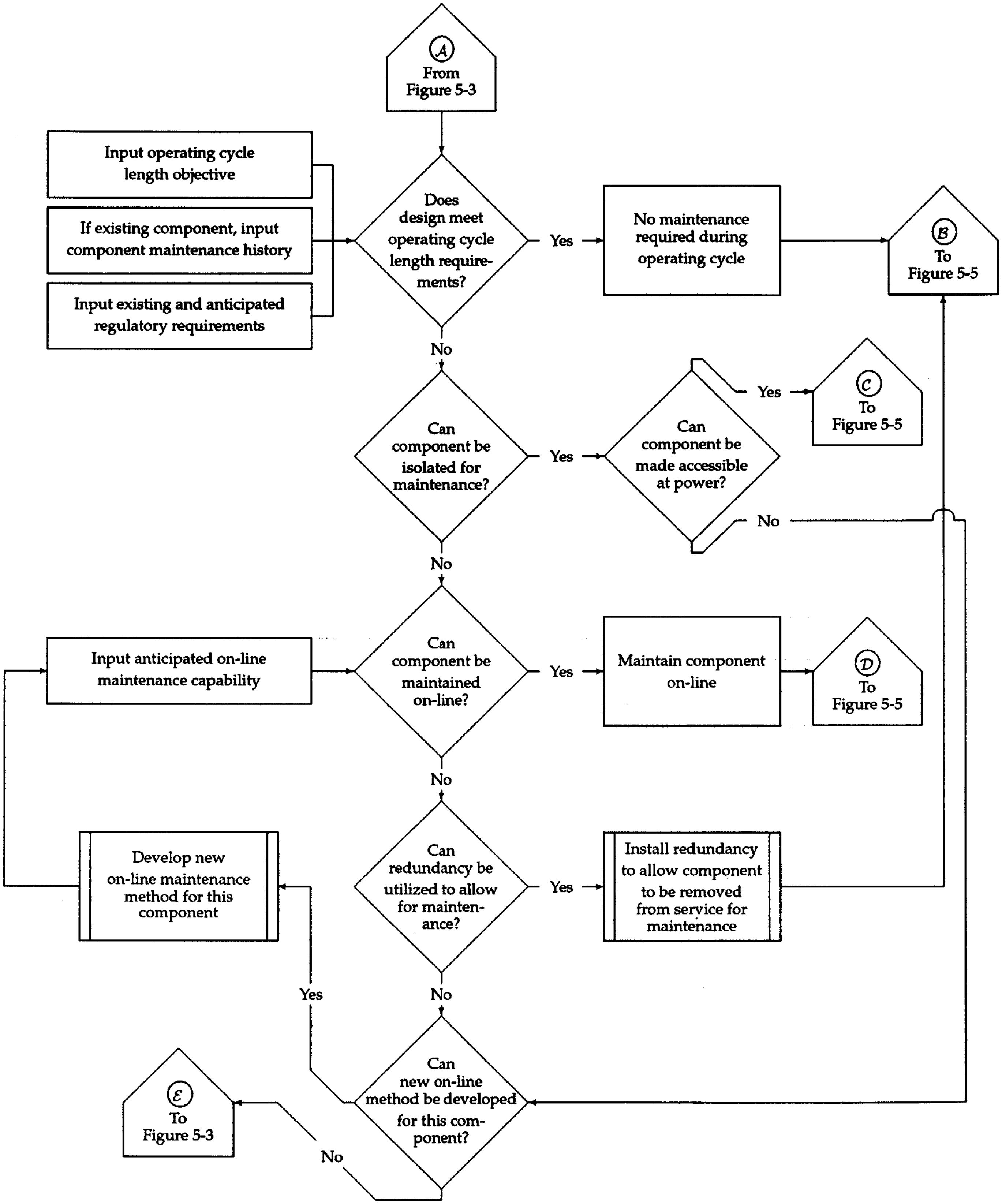


Figure 5-4: Design Resolution Methdology Flowchart – Synthesis of Maintainability

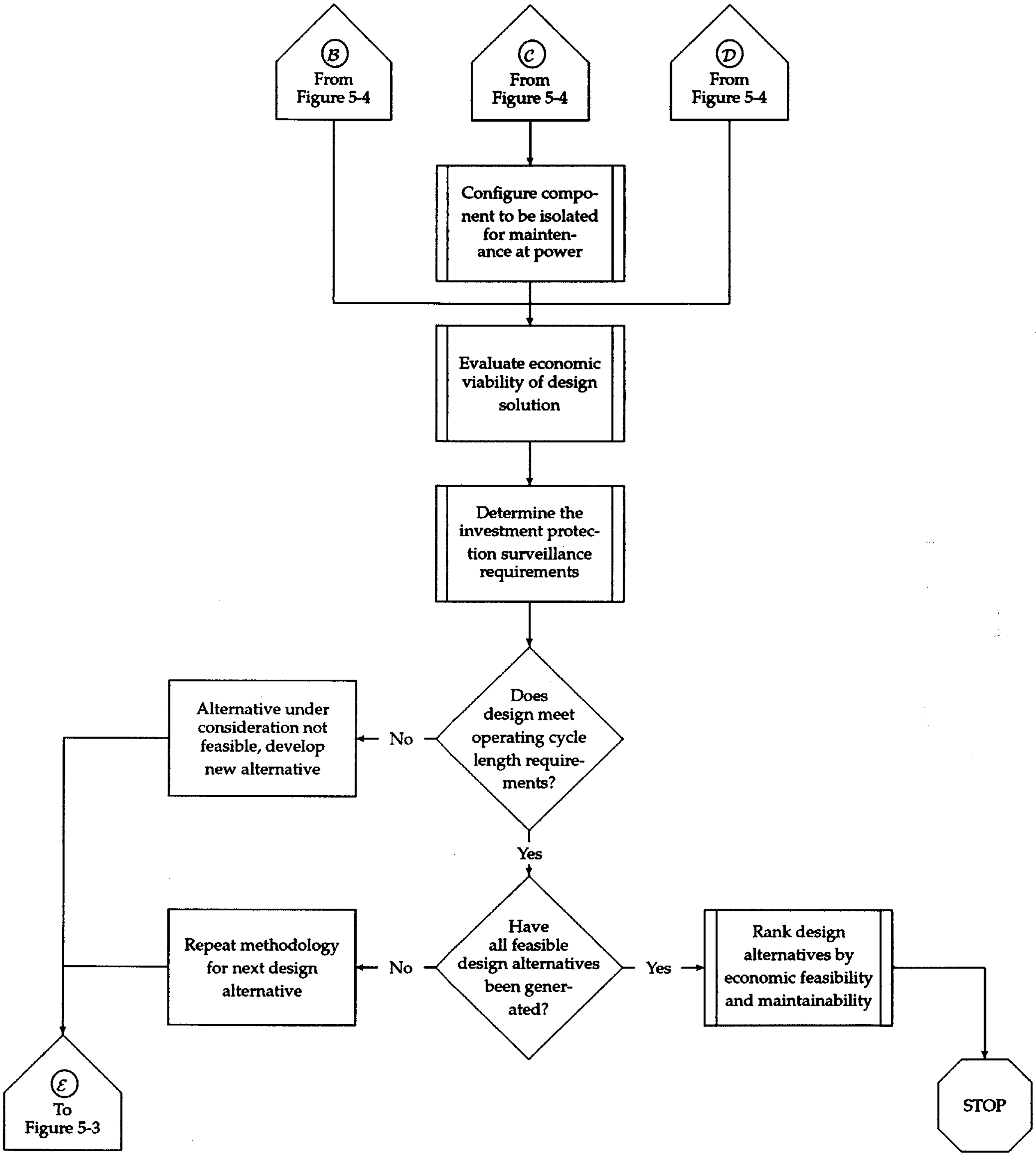


Figure 5-5: Design Resolution Methdology Flowchart – Synthesis of Constraints

Chapter 6

Resolution of Identified Barriers

6.1 Introduction

In Chapter 3, 'Operating Pressurized Water Reactor Surveillance Program,' the maintenance related barriers preventing an operating PWR from attaining an extended operating cycle were presented. This chapter addresses, in general terms and within the context of the IRIS design, the Category C surveillances identified by the MIT Extended Fuel Cycle Project which can neither be performed on-line nor have their performance intervals extended to 48 months. Where a potential solution is readily apparent, it is presented. Several barriers require additional design effort and are discussed separately in Chapters 7 through 11.

Many of the components and systems identified by the MIT Extended Fuel Cycle Project require testing and maintenance because of their role in ensuring safety. Evaluation of these components and systems using the methodology of Chapter 5 assisted in the conceptual development of a passive emergency heat removal system which could be tested at power. This system is described in Section 6.3, and the integrated testing and coordinated maintenance which it enables is described in Chapter 11, 'Application of Resolution Methodology–Reduced Power Window Surveillances.'

6.2 IRIS Resolution of Identified Surveillances Requiring Reduced Power or Plant Shutdown

6.2.1 IRIS Resolution of Regulatory Based Surveillances Requiring Plant Shutdown

6.2.1.1 In-Service Testing

Reactor Vessel and Primary System Component Relief Valve Testing Overpressure protection for the IRIS vessel is described in Chapter 7, 'Application of Resolution Methodology–Reactor Vessel Overpressure Protection.'

Operability and Engineered Safeguards Response Time Testing Integrated safety system time response testing will be required in IRIS. The conceptual IRIS passive cooling system (Section 6.3, below) is designed to be tested on-line (at reduced power) for 100% system operability demonstration. This passive cooling system is similar to the AP600 passive cooling loop, but is connected to the secondary loop rather than the primary loop. The specified testing periodicity for AP600 passive safety systems is off-line every two years, with quarterly operability checks deferred to the off-line period. With an on-line testing method it is anticipated that the quarterly operability checks will not be deferrable. Some of these operability checks, however, will require reduced power and are thus undesirable.

An integrated testing program is proposed (Chapter 11, Section 11.3) that performs limited operability testing quarterly and complete system operability testing every four years.[1] Quarterly assurances (which are currently deferred) that key components are functional reduce the uncertainty that the system will perform when required. This strategy of more frequent limited testing allows a longer 100% system operability demonstration performance interval, possibly as long as eight years.

Safety System Valve Operability Checks Valve operability checks will be conducted via the comprehensive safety system operability testing of Chapter 11, Section 11.3. AP600 safety sys-

[1]The IRIS strategy is to perform more thorough quarterly checks (which are not power limiting) and defer the complete system operability test as long as feasible. However, as will be seen in Chapter 8, 'Application of Resolution Methodology–Steam Generator Tube Inspection,' steam generator tube integrity inspections are the limiting inspections which make deferral of the complete safety system operability testing longer than the steam generator inspection interval unnecessary.

tems have a limited number of MOVs, utilizing air operated and squibb (explosively actuated) valves where MOVs have been traditionally been used. IRIS will also apply this design practice, minimizing or eliminating the use of safety grade motor operated valves.

It should be noted that during the regulatory review of the AP600 design, a new regulatory category was created for non-safety systems. This category, Regulatory Treatment of Non-Safety Systems (RTNSS), applies regulatory controls to non-safety systems which are used preferentially to safety systems when available. As there has been no AP600 plant built, it is unclear as to what the ultimate scope of the RTNSS program will be and how it will impact IRIS.

6.2.1.2 Containment Safety Features Response Time Testing

The IRIS containment design, which is largely borrowed from AP600, allows complete on-line integrated containment safety feature operability testing. This testing will be integrated into the comprehensive IRIS passive safety systems operability testing program described in Chapter 11, Section 11.3.

6.2.1.3 Steam Generator Eddy Current Testing

Steam generator tube integrity inspection is described in Chapter 8, 'Application of Resolution Methodology–Steam Generator Tube Inspection.'

6.2.1.4 Emergency Core Cooling Systems

These operability tests will be performed on-line (at reduced power) by the integrated 100% passive safety system operability demonstration of Chapter 11, Section 11.3.

6.2.2 IRIS Resolution of Nuclear Steam Supply System Investment Protection Surveillances Requiring Plant Shutdown

6.2.2.1 Component Cooling System Relief Valve Testing

Overpressure protection of individual components in the component cooling system will be required to prevent over-pressurizing an isolated component due to thermal expansion. These components can be protected by having a thermal relief check valve in parallel with the downstream isolation valve to ensure that isolated component pressure never exceeds component cooling water system pressure. This thermal relief check valve arrangement eliminates the need for individual component relief valves, and ensures that the component cannot be inadvertently overpressurized if it is isolated from the component cooling water system. For components which are connected to a higher pressure source, such as the reactor coolant pump stator jacket, the component must be manually isolable from the higher pressure source to prevent a leak into the component cooling water system from over-pressurizing the entire system.

6.2.2.2 Chemical and Volume Control System Relief Valve Testing

This particular surveillance is eliminated by design. Component overpressure protection will be provided as described in Section 6.2.2.1.

6.2.2.3 Reactor Coolant Pump Lubricating Oil

These surveillances are eliminated by design. The reactor coolant pumps will be of a sealed motor design and are lubricated by primary coolant.

6.2.3 IRIS Resolution of Balance of Plant Investment Protection Surveillances Requiring Plant Shutdown

6.2.3.1 Auxiliary Systems Relief Valve Testing

Component overpressure protection will be provided as described in Section 6.2.2.1.

6.2.3.2 Condenser Waterbox Cleaning

Condenser waterbox cleaning is described in Chapter 9, 'Application of Resolution Methodology–Main Condenser.'

6.2.3.3 Main Steam Safety Valve Testing

The main steam safety valves are ASME Class 1 valves, and will be resolved in a similar manner to reactor vessel overpressure protection (Chapter 7).

6.2.4 IRIS Resolution of Surveillances Requiring Reduced Power in the Extended Fuel Cycle Project

Resolution of surveillances requiring reduced power are described in Chapter 11, 'Application of Resolution Methodology–Reduced Power Window Surveillances.'

6.3 IRIS Emergency Heat Removal System

Implicit in the resolution of the regulatory based surveillances is the observation that, at some point, a 100% demonstration of the operability of all safety features must be performed. A passive cooling system for use in IRIS, similar to the AP600 passive cooling loop, is shown in Figure 6-1. Four loops will be utilized, with each loop's heat removal capability roughly equal to one-third of the total heat removal burden. This allows one passive cooling loop to be retired in-place during the operating cycle if necessary due to either failure of the cooling loop or failure of the corresponding steam generator.

Each passive cooling loop consists of isolation valves, an expansion tank, and a heat exchanger. When the isolation valves are opened, natural circulation causes water to be forced into the steam generator through the feed header. The cooling water is heated (or boils) and flows, due to the natural circulation head developed, back to the heat exchanger. Heat is transferred to the heat exchanger tank, which is vented to atmosphere and contains sufficient water to prevent evaporation from lowering the tank water level to the tube bundle. The steam generator water addition tank supplies makeup water to the passive cooling loop (which will be necessary as the bulk water temperature is reduced) to prevent loss of natural circulation head.

The strategy for testing the IRIS passive cooling loops is presented in Chapter 11, Section 11.3, 'IRIS Integrated Testing and Coordinated Maintenance.'

6.4 Summary

All of the four year cycle length barriers identified by the MIT Extended Fuel Cycle Project have been addressed in the context of IRIS. Although all solutions have not been addressed by detailed design, discussions with IRIS design engineers indicate that the problem is now suffi-

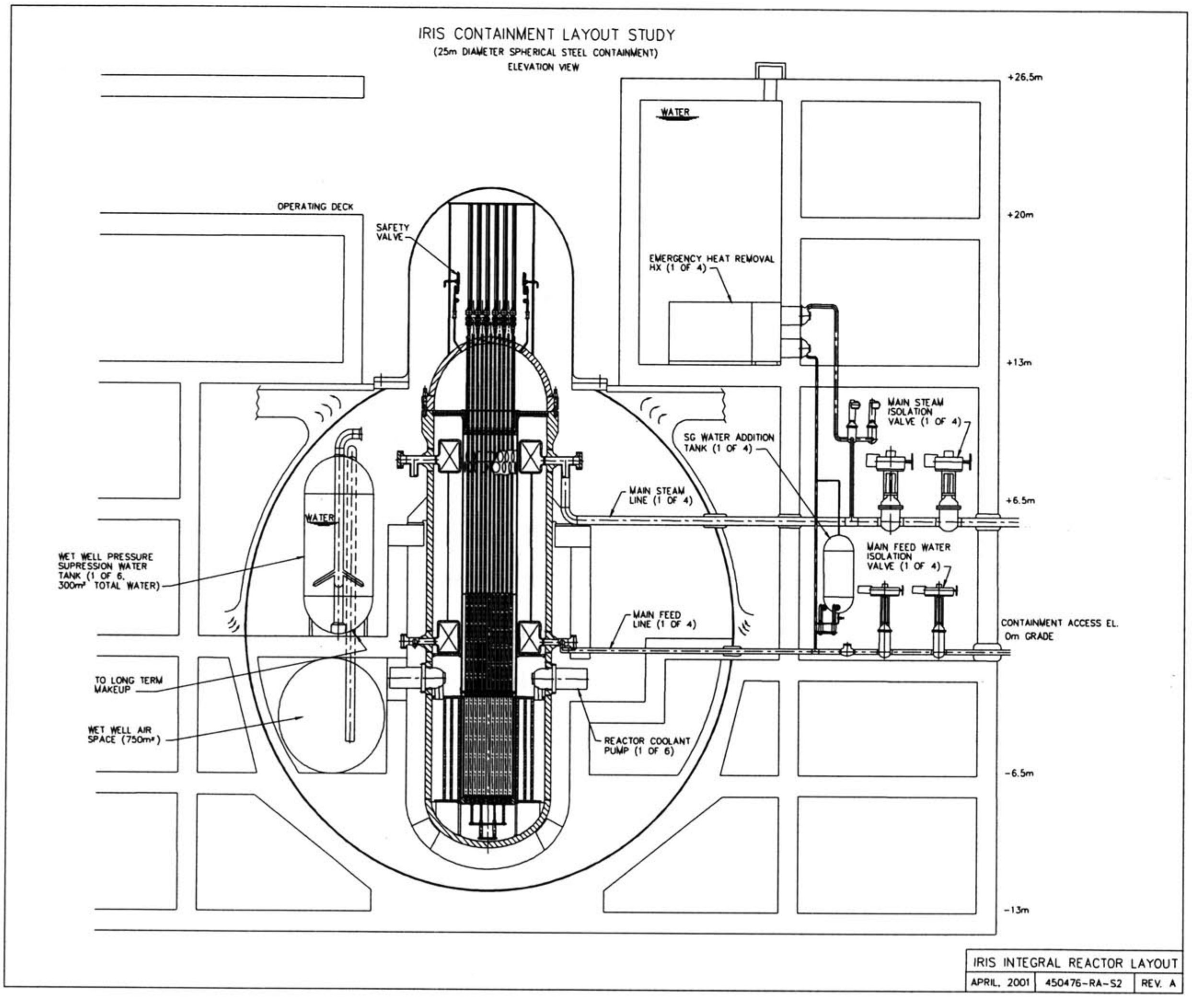


Figure 6-1: IRIS Containment Arrangement Showing Passive Cooling Loop (1 of 4 loops shown)

ciently bounded to readily and efficiently develop viable design solutions. The only cycle length barriers which cannot be readily solved are:[2]

- Primary relief valve testing,
- Steam generator eddy current inspection,
- Condenser waterbox maintenance,
- Main turbine throttle control maintenance,
- Safety system testing, and
- Reduce power window items.

These items are discussed in the following chapters.

[2]Note that rod control system testing also presents a significant operating cycle length barrier. However, the IRIS core design has not been completely specified and therefore adequate rod control system requirements cannot yet be specified.

Chapter 7

Application of Resolution Methodology–Reactor Vessel Overpressure Protection

7.1 Introduction

This chapter discusses the impact of providing overpressure protection on the IRIS operating cycle length goal. The methodology of Chapter 5 will be explicitly applied to demonstrate it's applicability. The design alternatives proposed were not generated by a single pass through the methodology flowchart, primarily due to the high impact of regulatory requirements on the design. Therefore, this chapter begins with a discussion of the regulatory requirements so that the reader can observe this impact on the alternatives generated.

7.2 Regulatory Requirements

The requirement to provide system overpressure protection is given by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, hereafter referred to as the *Code*. Section III, Division 1–NB, Article NB–7000 of the Code provides requirements for Class 1 components with similar requirements existing under the Code for all other classes of components. Overpressure protection is currently provided for reactor vessels using two (or more) pressure relief valves.[1]

Pressure relief valve testing is conducted in accordance with the "Operations and Maintenance of Nuclear Power Plants, ASME/ANSI (American National Standards Institute), OM–1987" Chapter 1. Under OM–1987, the owner/operator has the option of testing the pressure relief valve in-place or replacing the pressure relief valve with a bench tested spare. Most plants opt to replace the pressure relief valve with a bench tested spare during a scheduled outage rather than test in-place because it places the individual system out of service for a shorter period of time. In general, the time required for testing (setup, conducting the test, and restoring the system) is comparable to the time required for pressure relief valve replacement (setup, valve removal, valve installation, and restoring the system). However, using the bench tested spare (which has already passed it's lift test) removes any scheduling uncertainty associated with the repair or replacement of a pressure relief valve which fails it's in-place test. Currently, all pressure relief valve testing on active systems (those required to be active when the reactor is on-line, such as pressurizer relief valves) is conducted with the reactor shutdown.

The operation of a pressure relief valve is characterized by the Code using three parameters: set, lift, and blowdown. *Set* is the set pressure at which the pressure relief valve begins to open. The Code specifies a set pressure tolerance based on operating system pressure, and for typical PWR conditions (including IRIS) the tolerance is $\pm 3\%$. *Lift* pressure is the pressure at which the pressure relief valve is fully open, and rated relief capacity is attained. The Code specifies that pressure relief valves shall attain rated lift at a pressure which does not exceed the set pressure by more than 10%. *Blowdown* is the pressure at which the pressure relief valve fully reseats. The Code does not specify a value for blowdown. Rather, the Code requires the blowdown not to exceed that

[1]The following definitions are provided from Section III, Division 1–NB, Article NB–7000 of the Code:

- A *pressure relief valve* is a pressure relief device which is designed to reclose and prevent the further flow of fluid after normal conditions have been restored.
- A *safety valve* is a pressure relief valve actuated by inlet static pressure and characterized by rapid opening or pop action.
- A *safety relief valve* is a pressure relief valve characterized by rapid opening pop action, or by opening generally proportional to the increase in pressure over the opening pressure.
- A *relief valve* is a pressure relief valve actuated by inlet static pressure and having a gradual lift generally proportional to the increase in pressure over opening pressure.
- A *pressure relief device* is designed to open to prevent a rise of internal fluid pressure, greater than a specified value, resulting from exposure to pressure transient conditions. It may be a pressure relief valve or a nonreclosing pressure relief device.

value which the designer has determined (and specified in the Overpressure Protection Report) to be the minimum reseat pressure.

ASME/ANSI, OM–1987 requires that all Class 1 Pressure Relief Devices (which includes the pressurizer relief valves) be tested:

- Prior to initial installation.
- Within the initial 5 year operating period according to the following schedule:

Time Period	Minimum Cumulative % of Valves of Each Type and Manufacture to Be Tested
Startup — 12 months	0
13 months — 24 months	25
25 months — 36 months	50
37 months — 48 months	75
49 months — 60 months	100

  Additionally, a minimum of 20% of the valves of each type and manufacture shall be tested within any 24 months. This 20% shall be previously untested valves, if they exist.

- During subsequent 5 year periods such that all valves of each type and manufacture shall be tested with a minimum of 20% of the valves tested within any 24 months. This 20% shall be previously untested valves, if they exist.

7.2.1 Eliminating the Need for Overpressure Protection by Design

Section NB–7110 of the Code, "General Requirements: Scope" specifies that "a system shall be protected from the consequences arising from the application of conditions of pressure and coincident temperature that would cause either the Design Pressure or the Service Limits specified in the Design Specification to be exceeded." Specifically excluded from the scope of the Article are the effects of extremely short duration pressure increases (such as water hammer) and the design of reactor shutdown systems.

Within NB–7110, there exists the possibility that a system could be designed that would not be subject to conditions which lead to exceeding either the Design Pressure or the Service Limits specified in the Design Specifications. The IRIS design is a large integral reactor vessel with a large steam space, and design solutions could be sought which meet the Code requirements. However, it is unlikely that a nuclear power plant could be designed in this manner since the energy stored in the fuel can, under certain conditions, be released in a rapid enough manner as to cause an unacceptably high peak pressure. Other plausible conditions, such as a sudden loss of steam demand, can create a rapidly rising pressure condition which may require intervention to prevent exceeding the Design Pressure or Service Limits.

Acknowledgement of the need for overpressure protection in accordance with the Code, then, requires compliance with the Code. As written, Article NB–7000 is almost entirely devoted to pressure relief valves. Within the category of pressure relief devices required by the code are both pressure relief valves and nonreclosing pressure relief devices. Within nonreclosing pressure relief devices, rupture disk devices are the only devices addressed. However, rupture disk devices are not permitted to be used as the sole pressure relief device. It appears, then, that the intent of the Code is to ensure that a pressure relief valve is used in the overpressure protection scheme. Therefore, the opinion of the IRIS design team is that amount of effort required for the development, testing, and validation of a pressure self-mitigating vessel would be better spent seeking other (less revolutionary) design solutions.

7.3 Synthesis of Requirements

7.3.1 Functional Requirements

To protect the reactor vessel and attached piping from potential overpressure conditions (which could ultimately lead to catastrophic failure), overpressure protection is required. The capacity of the overpressure protection device must be great enough to arrest the design basis pressure rise and ensure that the design maximum pressure is not exceeded. For the IRIS design resolution we seek, the overpressure protection device must either be maintainable on-line or not require maintenance for the entire eight year operating cycle.

7.3.2 Currently Used Component — Pressurizer Relief Valve

All currently operating PWR plants use a pressurizer, located above the reactor vessel elevation and typically connected to one of the reactor coolant cold legs, to maintain reactor coolant system pressure. The pressurizer is a heated vessel that acts as a head tank or surge volume to mitigate system pressure transients. Overpressure protection for the reactor coolant system is provided by two or more relief valves directly connected to the pressurizer steam space. The reason for using a steam relief valve (vice a water relief valve) is two-fold: (1) less mass is lost from the system for a given pressure reduction upon actuation, and (2) there is less chance of fouling the valve seat (by corrosion products) during reseating.

Does existing component meet requirements? Based on the experience at the candidate operating PWR, as well as limited interviews with personnel at other plants, the relief valves in service today are unlikely to operate reliably for the entire IRIS cycle length. The pressurizer relief valve meets the IRIS overpressure protection functional requirements but, based on the above regulatory testing requirements, cannot meet the IRIS cycle length objective. To conduct pressurizer relief valve testing with the reactor at power requires either of the following:

- the capability to isolate and remove the valve from the system for bench testing,
- the capability to test the valve in-place (either isolated or unisolated from the primary system), or
- testing the unisolated valve in–place by raising system pressure to the valve lift point.

Employment of any of these methods is likely to require submittal of a Code case to the ASME, since these methods are not explicitly permitted by the code. However, Article NB–7142 does specify requirements to be met if an isolation valve is to be utilized and hence a solution may be found that is within the scope of the current Code.

7.3.3 Component Modification

Can existing component be modified to meet requirements? There are modifications which can be made to the pressurizer relief valve to allow for on-line testing. Potential modifications are described below.

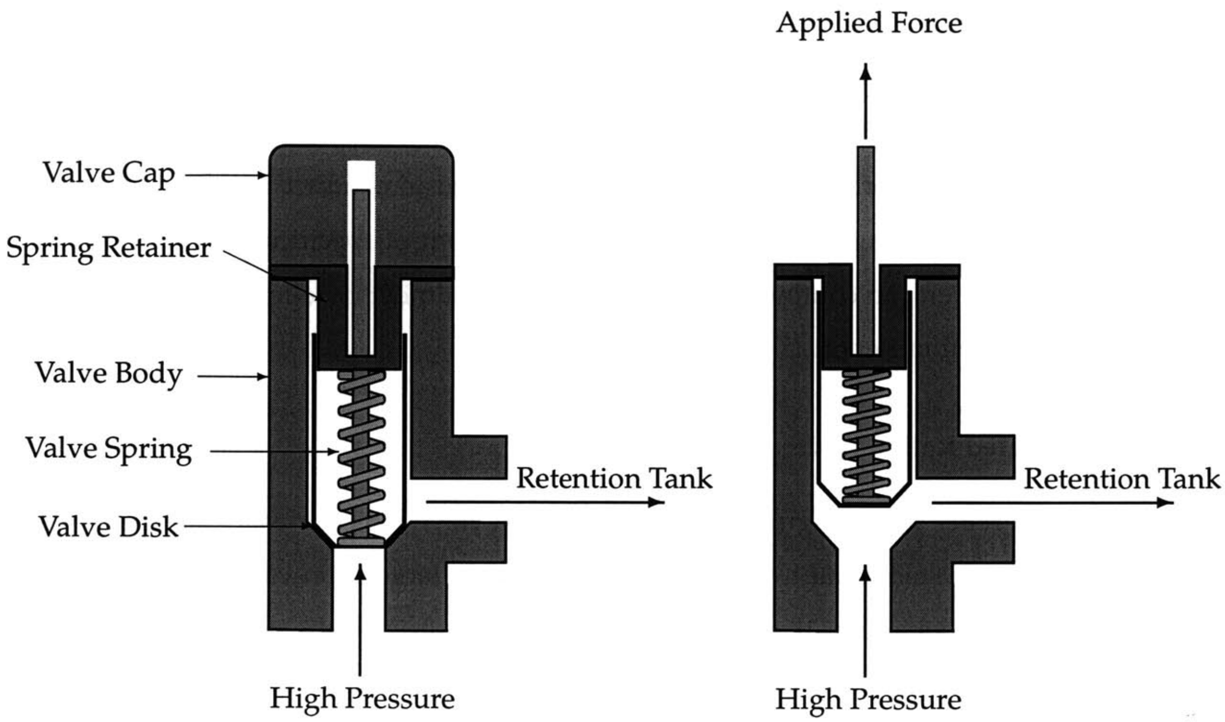
7.3.3.1 Spring-Loaded Relief Valve

The spring-loaded relief valve, shown in Figure 7-1(a), is the most commonly used overpressure protection device. An unmodified spring-loaded relief valve meets the functional requirements to provide overpressure protection for the reactor vessel but cannot meet the IRIS eight year cycle length objective due to the regulatory requirements which specify more frequent testing (Section 7.2). The design issue to resolve is whether the relief valve can be taken out of service for on-line testing. Based on the safety role that the valve performs, overpressure protection cannot be suspended even for short time periods. Therefore, a method to test relief valves while maintaining reactor vessel overpressure protection must be utilized.

7.3.3.1.1 On-line Spring-Loaded Relief Valve Testing Several Engineering Services companies[2] provide on-line testing of simple spring-loaded safety and relief valves under normal operating pressure and temperature. It is primarily utilized in the nuclear industry for shutdown in-place testing of main steam safety valves where it is not feasible (due to time and expense) to remove the valve from the system, and full test pressure cannot be obtained to lift the valve. In-place assisted lift testing is limited to those applications where the valve is accessible for testing, the valve outlet can be monitored for leakage, and a lift assist device with reduced system pressure can be used to actuate the valve (since system pressure can not be used to achieve the set pressure of the valve). Assisted lift testing is currently only practical for valves in systems with highly compressible media, such as gas or steam. This method is not currently used to test water valves since the outlet of the valve usually cannot be observed for flow and determination of valve set and lift to the required accuracy is not possible.

Using the simplifed spring-loaded relief valve drawing of Figure 7-1, assisted lift testing is conducted according to the following (simplified) procedure. First, the valve set pressure and current system pressure are compared to determine the expected pressure difference. Using valve nameplate data, this pressure is converted to a force (pull) and the appropriate range load cell is selected and calibrated. The valve cap is removed, and the lifting mechanism (with load cell) is attached to the valve stem. Flow sensors, which allow detection of the valve set, lift, and blowdown, are attached to the discharge piping near the valve body. While monitoring the discharge piping for flow, the lift mechanism pulls the disk (via the valve stem) against the valve spring until set is detected. Once set is determined, the lift mechanism opens the valve further until full flow (lift) is detected. Finally, the valve is unloaded and blowdown is measured. Again using the valve nameplate data, the measured forces are converted to pressures and added to the measured

[2]For example, Furmanite.



(a) Simplified Spring-Loaded Relief Valve

(b) Simplified Assisted Lift Test Setup

Figure 7-1: Assisted Lift Relief Valve Testing — Simplified Drawing

system pressure to determine the valve set, lift, and blowdown.

Assisted lift testing has two distinct advantages. First, since the testing is conducted at an actual system pressure less than the set pressure there is a large differential pressure across the valve disk once the lift mechanism releases the valve stem. This ensures that the valve disk reseats quickly and positively, without the valve chatter normally experienced when the applied pressure approaches the blowdown pressure. Second, the lift mechanism can force the valve closed if the valve sticks open after the valve stem is released by the lift mechanism. Although assisted lift testing of pressurizer relief valves has never been conducted at power, these advantages could mitigate the risk of excessive coolant loss for a stuck open relief valve. For IRIS, the relief valves would either need to be made accessible (so personnel could connect the lift mechanism) or redesigned to incorporate the lift mechanism in the valve. Even if the lift mechanism is permanently installed, the process cannot be completely automated since calibration of the lift mechanism load cell is required prior to testing.

At least one additional relief valve should be installed in excess of the number specified in the Overpressure Protection Report to eliminate the potential need to conduct relief valve setpoint adjustments at power. If a single relief valve is found to be out-of-specification, then it can be gagged shut and operation can continue[3] with the minimum number of relief valves specified in the Overpressure Protection Report.

7.3.3.2 Pilot Operated Relief Valve

A pilot operated relief valve (Figure 7-2) is a compound valve which uses a small pilot valve to direct high pressure system fluid to the operating piston of a large main valve. When the pilot valve opens and the underside of the operating piston is pressurized, the main valve spring is compressed allowing fluid flow through the main valve. The operation of the pilot valve is similar to the spring-loaded relief valve, but typically uses a corrugated bellows instead of a spring. When system pressure is reduced and the pilot valve closes, the high pressure in the main valve operating cylinder bleeds off allowing the main valve spring to force the main valve closed. The primary advantage of the pilot operated relief valve is that, unlike the spring-loaded relief valve, the main valve is not subject to near-zero differential pressure. Therefore, a large differential pressure always exists to rapidly close and seat the main valve. Testing of a pilot operated relief valve consists of determining the pilot valve operating characteristics and verifying that the main valve is not physically bound.

7.3.3.3 Improved Relief Valve

The primary limitation of current pressure relief valves (whether simple spring-loaded or pilot valve actuated) is setpoint drift. Current pressure relief valves operate by generating a force to compress a spring and lift the main disk off it's shut seat. Changes in material properties, corrosion

[3] OM-1987 has specific requirements for bench-tested relief valves which are found to be out-of-specification, but not for on-line tested valves. It does specify that the valve shall be repaired/replaced, the cause of failure shall be determined and corrected, and the valve shall be satisfactorily retested prior to returning to service. For this configuration, with valve(s) installed in excess of the requirement, the out-of-specification valve will not be returned to service until the next operating cycle. However, the cause of failure will need to be determined to ensure that a common-cause failure will not disable all overpressure protection.

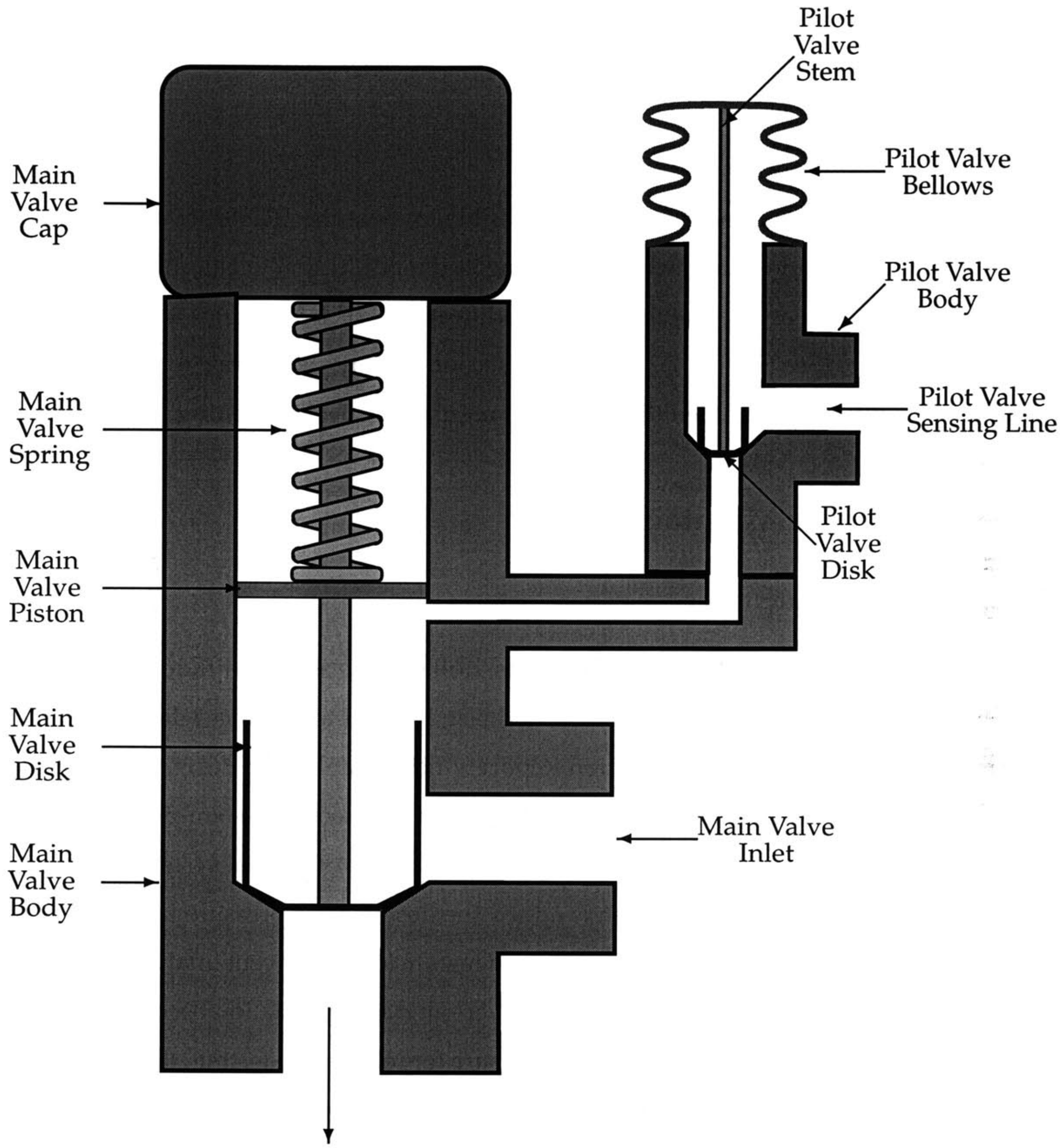


Figure 7-2: Pilot Operated Relief Valve — Simplified Drawing

buildup, and thermal effects all contribute to changes in valve actuation characteristics. Only through material advances can these deficiencies be corrected. However, since the industry does not have a need for these advanced relief valves there is no industry impetus to invest in the research and development necessary to field a relief valve which overcomes these limitations.

7.3.3.4 Summary of Modifications

Only two relief valve modifications are possible to meet the IRIS operating cycle length requirements: modification to allow on-line testing and modification to eliminate the need for testing. The first is most feasibly met through a modification which integrates an assisted lift mechanism into the valve (See Section 7.3.3.1.1). The second requires advances in materials technology to address the limitations that led to the current regulatory specified testing periodicity.

7.4 Synthesis of Maintainability

7.4.1 On-Line Maintenance

Can component be maintained on-line? Yes, utilizing the assisted-lift device of Figure 7-1.

System pressure, without assisted-lift, could also be used to conduct relief valve testing. The Code requires an Overpressure Protection Report which describes the design basis pressure transient upon which the total relief capacity and setpoint is based. If system pressure is to be used for in–place relief valve testing, then the lift pressure (instead of normal system pressure) now becomes the starting pressure onto which is added the design basis pressure transient. This results in a lower relief valve setpoint which provides an insufficient to prevent inadvertent lifting during normal operating transients. Additionally, inadvertent depressurization becomes a greater risk if the relief valve fails to reseat. Using system pressure for relief valve testing, therefore, should only be considered if no other on-line testing method can be developed.

7.4.1.1 Installing Redundancy to Permit or Defer Testing

Can redundancy be utilized to allow for maintenance? Although neither the ASME Code nor OM–1987 requires the reactor to be shutdown during relief valve testing, the rules do not explicitly permit isolation valves to be installed in the path of the relief valve. To the contrary, NB–7142 of the Code prohibits stop valve installation unless "such stop valves are constructed and installed with controls and interlocks so that the requirements of NB–7300 are met under all conditions of operation of both the system and the stop valves."

Section NB–7300 of the Code addresses the required relieving capacity of installed pressure relief devices, which includes consideration of all relevant design and operating factors which may contribute to an overpressure condition.

Figure 7-3 shows an arrangement which could meet the Code requirements while providing adequate isolation to conduct, via a test fitting, in-place testing. The key to this arrangement is the three-way valve which cannot simultaneously isolate both relief valves from the reactor vessel, even if the valve is inadvertently placed in a mid-position. The internal flow path is shown in Figure 7-4. Only when the valve is correctly aligned to one relief valve is the other isolated. To meet the Code requirements both valves must either be operational (since it is possible that an inoperable relief valve may be placed in service) or the valve physically prevented (via interlocks) from aligning the reactor vessel to an inoperable relief valve.

With the addition of suitable interlocks, the arrangement of Figure 7-3 can be used to allow the second relief valve to act as an installed spare. Since there is no regulatory prescribed shelf-life for a tested relief valve that is not in service, the second valve can remain isolated from the reactor vessel until the first valve requires testing. Then, rather than testing the first valve, it is isolated and the second valve is placed in service. Suitable interlocks could consist of a stem locking device or weld. However, as with the three-way valve design a Code case will likely need to be submitted to ASME for evaluation.

Can the component be isolated for maintenance? Yes, utilizing the arrangement of Figure 7-3. If the system is a high energy system, as is the case for the primary coolant system, then two upstream isolation valves and one downstream isolation valve (if the potential exists for reverse flow in the downstream piping from another source) are required for personnel safety.

Isolation (or, in the nomenclature of the Code, *stop*) valves are not specifically prohibited by the Code. Rather, specification is made that these stop valves shall be constructed such that during normal operation the pressure relief device cannot be rendered inoperable. This, however, is exactly what is intended by the isolation valve described in Section 7.4.1.1. But, the isolation valve has two relief valves attached and one is always on service. Therefore, the intent of the Code is

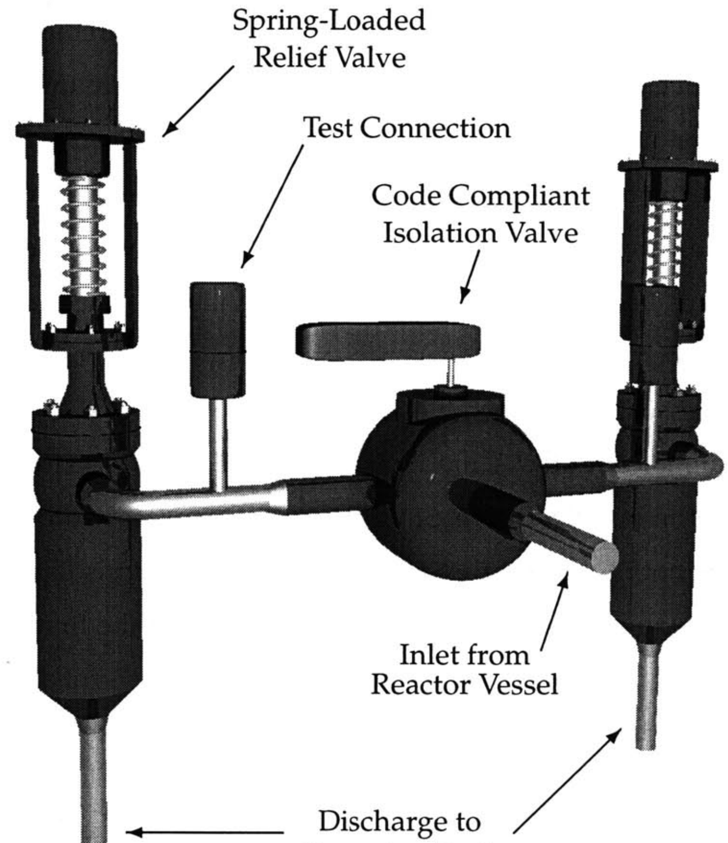
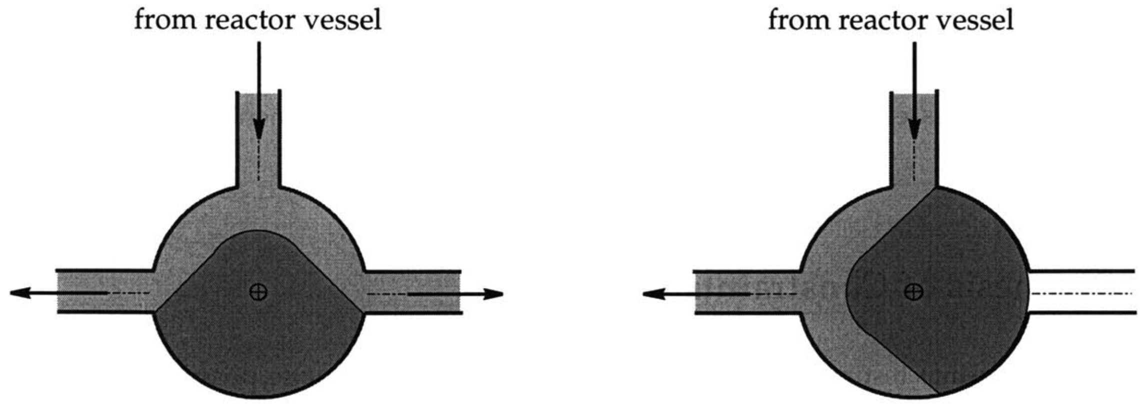


Figure 7-3: Redundant Spring-Loaded Relief Valves with ASME Boiler and Pressure Vessel Code Compliant Isolation



(a) Aligned to both relief valves (b) Aligned to one relief valve

Figure 7-4: ASME Boiler and Pressure Vessel Code Compliant Isolation Valve Flow Path. In (a), flow is directed to both relief valves. In (b), flow is directed only to one relief valve with the other isolated for in-place testing. Angular travel of the valve ball is mechanically limited to prevent simultaneously isolating both relief valves from the reactor vessel.

met since the isolation valve cannot isolate both relief valves simultaneously and the subsystem consisting of one isolation valve and two relief valves is considered a single pressure relief device.

7.4.1.2 On-Line Testing of Isolated Relief Valves

The Code compliant isolation valve described above can be used to enable on-line testing of either the simple spring or pilot operated relief valve. With the relief valve isolated, an external test device can be connected to the main valve inlet (for spring-loaded relief valves) or to the pilot valve inlet (for pilot operated relief valves). An external test device provides it's own testing medium, eliminating the need to use system fluid.

For the pilot operated relief valve, both the pilot sensing line and main valve inlet line should be isolated from the system during testing. This permits determining the pilot valve operating characteristics and manually exercising the main valve stem without requiring the main valve to pass system fluid. Note that by installing the isolation valve only on the main valve inlet line, assisted lift testing can be conducted on the pilot valve. However, since little system fluid flows through the pilot valve (only enough to pressurize the main valve operating cylinder) there is little risk of pilot valve damage during blowdown. Therefore, the added expense and complexity of installing an assisted lift device is not justified.

Can component be made accessible at power? Yes, based on the low anticipated dose rate (enabled by the large vessel annulus and shield plates) inside the containment vessel during normal operation.

7.5 Synthesis of Constraints

For all the on-line testing methods described, accessibility is made possible by the low anticipated dose rate inside the containment vessel. Failure to provide overpressure protection for the reactor vessel is an unacceptable risk, and so investment protection concerns dictate that the testing frequency should be at least as frequent as the regulatory requirement. However, the regulatory specified frequency is based on a significant amount of performance history, so the investment protection testing frequency is likely to not be more frequent. Therefore, the testing frequency of OM-1987 (and not more frequently) should be adopted for IRIS.

Does design meet operating cycle length requirements? The methods described enable the target operating cycle length of eight years by providing a means for conducting testing with the reactor at power.

Have all feasible design alternatives been generated? Creative design can continue to develop alternatives, but these are likely to be more complex and require accepting a higher technical risk.

7.6 Summary

This chapter has demonstrated the resolution methodology and presented several alternatives, for consideration in the IRIS design, which meet the eight year operating cycle length objective. The first two, assisted lift testing and installation of redundancy, utilize a current technology relief valve and conduct testing on-line and at power. The third alternative is to develop an improved relief valve that will perform satisfactorily for the entire IRIS operating cycle. The last alternative is to design IRIS to be pressure self-mitigating, and thus a pressure relief device would not be required.

Table 7.1 summarizes the alternatives qualitatively ranked by maintainability, economic viability, and technology risk. For the four-year IRIS maintenance cycle one additional alternative exists, regulatory change, but is not included here since the objective is to design systems which can be maintained at power allowing much longer operating cycles. Within the alternatives generated, installation of redundancy presents the most feasible and cost effective solution.

Table 7.1: Overpressure Protection Alternatives Summary

Design Alternative	Report Section	Maintain-ability	Anticipated Cost	Technology Risk
Assisted lift testing of current technology relief valve	7.3.3.1.1	medium	medium	low
Install redundancy and a code compliant isolation valve to permit in-place testing of isolated relief valve[a]	7.4.1.1	medium	low	low
Utilize installed spares and a code compliant isolation valve to allow deferral of testing	7.4.1.1	high	medium	medium
Design advanced relief valve which does not require maintenance during the operating cycle	7.3.3.3	high	medium	high
Design reactor vessel system such that overpressure protection device is not required	7.2.1	high	high	high

[a]Recommended alternative for use in IRIS.

Chapter 8

Application of Resolution Methodology–Steam Generator Tube Inspection

8.1 Introduction

Steam generator tubing constitutes a significant portion of the reactor coolant pressure boundary (RCPB). The design of the RCPB for structural and leakage integrity is addressed in either Title 10 of the Code of Federal Regulations (CFR), Part 50 (10 CFR Part 50), Appendix A or the licensing basis of a facility. The General Design Criteria (GDC) of Appendix A state that the RCPB shall "have an extremely low probability of abnormal leakage" (GDC 14), "shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation" (GDC 15), and "shall be designed to permit periodic inspection and testing of important areas and features to assess their structural and leaktight integrity" (GDC 32).

The first two requirements will not be explicitly considered here, but certainly must be met by any steam generator design. IRIS is considering a modular helical-coildesign (eight modules arranged in the reactor vessel annulus) with u-tube configuration as the backup design.[1] Regardless of the configuration, the steam generator will have inlet and outlet channel heads (or a single inlet/outlet head) which are mounted to the reactor vessel for accessibility to meet the third requirement. This arrangement will allow access to the steam generator from the secondary side either through a manway or by removal of the entire channel head cover plate. Figure 8-1 shows the IRIS reactor vessel design with straight-tubed steam generators. The proposed mounting method is the same for all steam generator design types.

Gaining access to the steam generator tubes is not a significant design obstacle. The design challenge is to create a steam generator which can be inspected at power, and ideally while in service. This chapter investigates this design challenge and potential solutions.

8.2 Requirements

The structural and leakage integrity of steam generator tubing is maintained through several defense-in-depth measures, including in-service inspection, tube repair criteria, primary-to-secondary leak rate monitoring, water chemistry control, operator training, and analyses to ensure that safety objectives are met. The degraded tubes must be removed from service (by plugging) or repaired if detected indications (flaws) exceed 40 percent of the nominal tube wall thickness as required in plant technical specifications. The indications are detected by periodic inspections using qualified nondestructive testing as required by Criterion IX in Appendix B to 10 CFR Part 50. Eddy current technology, one method of nondestructive testing, is the primary means used by the industry to assess the condition of steam generator tubing.

The eddy current inspection technique correlates the depth and length of an indication to signal responses received by probes passing through the inside of the tube. Although the eddy current method is a proven technique for detecting the length of indications, there has been lim-

[1] In April 2001, the IRIS team evaluated several steam generator options (u-tube, c-tube, helical-coil, modular helical-coil, and straight-tube) and selected the modular helical-coil design as the primary design and the u-tube design as the backup design. Where IRIS is exploring innovative technologies which have not previously been used for pressurized water reactor application, such as the modular helical-tube steam generator, the technology risk is mitigated by concurrently developing a backup design which is based on current technology.

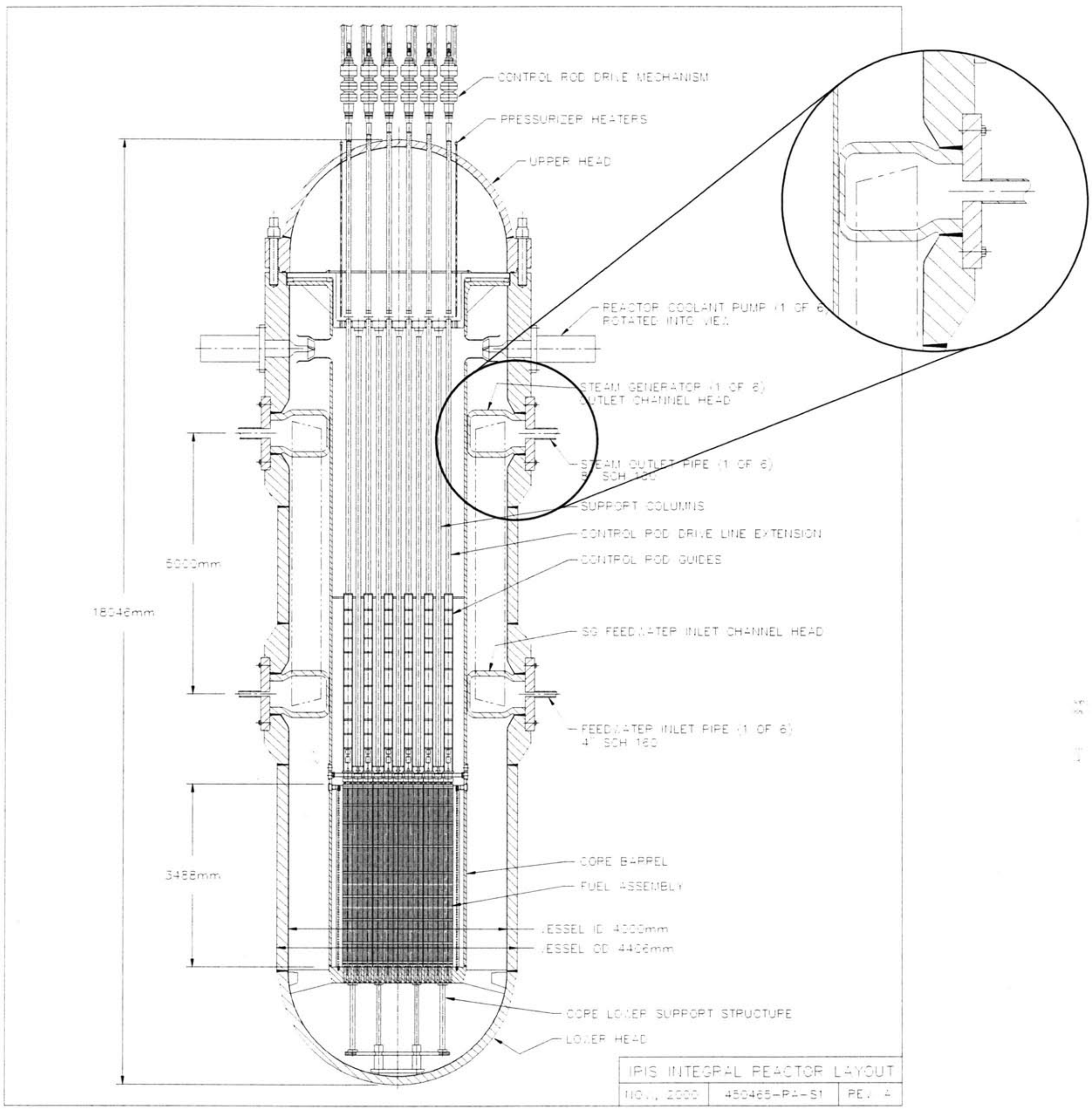


Figure 8-1: IRIS Reactor Vessel Drawing

ited success in demonstrating its capability to accurately measure the depth of certain types of steam generator tube indications. Specifically, indications caused by intergranular attack (IGA) and stress-corrosion cracking are difficult to size with eddy current techniques because of a number of complicating variables, such as oxide deposits, material properties and geometry, crack morphology, human factors, data analysis, and data acquisition practices. In one recent instance, a licensee sized the depths of IGA indications and removed from service those tubes with IGA indications exceeding the 40 percent through-wall repair limit. Data from subsequent destructive examinations of several degraded tube specimens removed from the licensee's steam generators during the outage indicated that the estimated through-wall extent of degradation in these specimens, based on eddy current, was significantly less than the true depth of the IGA indications.[2]

In order to successfully disposition steam generator tube degradation in accordance with the repair limits in the technical specifications and Appendix B to 10 CFR Part 50, the inspection process must be capable of (1) detecting indications of tube degradation, (2) characterizing the indications, e.g., cracklike, IGA, manufacturing burnish mark, or wear and the orientation for cracklike degradation, and (3) accurately sizing the depth of degradation. The term "inspection process" refers to the use of one or a combination of nondestructive inspection techniques to evaluate a specific mode of steam generator tube degradation. This evaluation could potentially include three inspection methods (e.g., eddy current probes)–one for detection, one for characterization, and a third to size the indication. However, the successful qualification of the inspection process requires a qualification of each method (i.e., probes, cables, software, etc.) for the mode of degradation being evaluated in the steam generator tube examinations. Experience has demonstrated that for effective qualification the data set demonstrating the capability of the inspection process should consist, to the extent practical, of service-degraded tube specimens (i.e., specimens removed from operating steam generators), supplemented, as necessary, by tube specimens containing flaws fabricated using alternative methods provided that the nondestructive examination parameter responses from these flaws are fully consistent with actual in–service degradation of the same flaw geometry.

[2]Proposed NRC Generic Letter, "Steam Generatur Tube Inspection Techniques," SECY-97-280 ,December 3, 1997. NRC Generic Letter 97-05, "Steam Generatur Tube Inspection Techniques," was subsequently issued December 17, 1997.

8.3 Currently Used Component — Westinghouse Model F Steam Generator

The candidate PWR uses Westinghouse Model F steam generators with 5626 Thermally Treated Inconel 600 U-tubes (SB-163) that are hydraulically expanded into the tubesheet at each end. The current NRC inspection guidelines for steam generators allow for periods between steam generator eddy current testing of up to 40 months. This interval is allowed only after two previous successful inspections at shorter intervals. However, rather than accomodate a changing inspection interval in maintenance planning most PWRs inspect all their steam generators every refueling outage regardless of the maximum permitted inspection interval. This has resulted in a significant amount of data on steam generator tubing performance collected for 18-24 month operating intervals but little for longer intervals.

8.4 IRIS Steam Generator Design and Inspection

The maintenance related barriers associated with currently used steam generators are applicable to the IRIS steam generator design but one fundamental operational difference exists. In a conventional steam generator, the higher reactor coolant pressure is on the inside of the steam generator tubing. Although an integral pressurized water reactor could be configured to have the pressure on the inside of the tubing also, the IRIS steam generators will have the reactor coolant pass over the tubing rather than inside of it. This configuration was selected because it reduces pressure losses in the reactor coolant loop and enhances natural circulation flow, which is a key characteristic in the IRIS accident mitigation scheme.

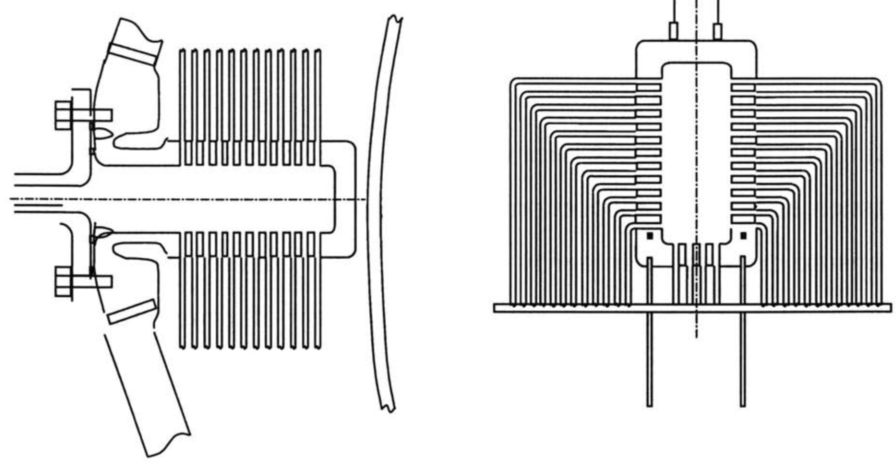
Regardless of whether the tubes are in tension (with the higher pressure on the inside of the tubing) or compression (as in the IRIS design), the requirement to inspect the steam generator tubing does not change. This reversal of differential pressure compared to current steam generators, however, makes it more difficult to design the steam generators to be accessible for inspection since the inspection method is not necessarily known. Efforts are currently underway by the IRIS team to identify the dominant failure mechanism for tubes in compression (vice tension, which are currently inspected using eddy current techniques) and the applicable inspection technique to detect this failure mechanism.

Rather than delay the steam generator development, it is assumed that whatever inspection method is to be utilized will use equipment and techniques similar to eddy current testing (i.e., an active element on a cable which is inserted into the tube). For the proposed solutions to the accessibility design problem, constraints are carried over to the inspection technique development problem. There is a risk in this approach that the equipment to perform the applicable inspection technique is not (as assumed) similar to eddy current testing. However it is possible that eddy current testing may turn out to be the applicable inspection technique for IRIS steam generators. And if it is not, advances in miniaturizing electronic components suggest that the equipment to perform the applicable technique is likely to be at least not larger than current eddy current inspection equipment.

The ultimate selection of steam generator tube configuration (modular helical-coil or u-tube) will have little impact on the accessibility design problem (although it will significantly impact the inspection method). All proposed configurations will have channel heads with the tubes penetrating the heads, similar to the conceptual c-tube design of Figure 8-2. In currently used steam generators, such as the Westinghouse Model F above, the tubes are hydraulically expanded into the tubesheet and a pressure tight seal is created when the tubes are internally pressurized. For IRIS, the tubes will tend to contract when externally pressurized which could lead to leakage between the tubes and channel head. Two methods will be utilized to mitigate this potential leakage: (1) a collar will be pressed into each tube, which will maintain the tube pressed against the channel head, and (2) the tubes will be seal welded to the channel head. The once-through configuration of the IRIS steam generators requires the flow to be balanced across the tube bundle within a steam generator, so the tube collar will also function as an orificing device. The implication of these orificing devices is that a large active element may be difficult (or impossible) to insert into the tube, necessitating partial steam generator disassembly or complete removal for inspection.

8.5 The Steam Generator Tube Inspection Maintenance Barrier

Failure of any given steam generator tube is not an inevitable occurance, nor is failure of a single tube a catastrophic event. However, in current design steam generators failure of a steam



Top View From R/V Inside View

Figure 8-2: Conceptual Drawing of IRIS C-Tube Steam Generator

generator tube does compromise the RCPB and can lead to a significant loss of primary coolant and contamination of the steam system if not immediately detected. Therefore, steam generator tube failure must be prevented for both safety and investment protection reasons. Currently, steam generator tube failure is prevented by detecting and eliminating conditions which are known to potentially lead to tube failure.

For IRIS, meeting the operating cycle length goal requires one of the following conditions:

- Eliminating the conditions which potentially lead to tube failure allowing the inspection interval to be extended to a periodicity consistant with the operating cycle length,
- Eliminating the use of the steam generator tubes as a reactor coolant pressure boundary eliminating the requirement to conduct tube inspections, or
- Making the tube bundle accessible for inspection at power.

At this early stage of IRIS development, it is not reasonable to assume that the conditions which potentially lead to tube failure can be eliminated since they have not yet been identified. The second option, moving the reactor coolant pressure boundary, is also not feasible since it would make portions of the steam and feed systems (from the steam generators to the main steam isolation valves and feedwater isolation valves) part of the RCPB and these components would need to be designed to the higher reactor coolant system pressure and tested as an active safety system. Designing a complex system to accomodate failure of a component is certainly prudent, but accomodating the failure simply because the designer chooses not to take adequate and reasonable steps (such as periodic inspections) to prevent the failure circumvents the intent of the regulations to ensure mechanical integrity of the RCPB. Therefore, resolving the steam generator tube inspection maintenance barrier requires development of a means to perform the required inspections at power.

8.6 Application of the Design Resolution Methodology

8.6.1 Currently Used Component

The currently used steam generator does not meet the functional requirement for a compact internal steam generator. Based on the significant design differences between currently operating PWRs and IRIS, no component modifications can be made which will make a design like the Westinghouse Model F suitable for use in IRIS. However, analysis of the Model F is not without benefit since much of the maintenance performed on the Model F will also be performed on the IRIS steam generators.

8.6.2 Isolating Steam Generators for Inspection

Section 11.3, 'IRIS Integrated Testing and Coordinated Maintenance,' describes a comprehensive inspection scheme to perform required maintenance with the steam generator isolated but the reactor still at power. For this steam generator inspection to be conducted, consideration must be given to the extremes of temperature and radiation present inside the steam generator (for equipment) and in the vicinity of the reactor vessel (for personnel). Therefore, the inspection equipment must remotely operated with the following characteristics:

- The inspection equipment must be flexible enough to make up to a 180-degree bend and be directed into the desired tube, yet rigid enough to be pushed through all turns (including the multiple turns of a helical-coil steam generator). To transit the entry path and through the tube may require an additional force (such as flow, which would result in the generation of steam). To find the desired tube implies an imaging capability, although physically marking the tubes at the entrance (such as unique etched bands) would allow tube identification after inserting the active element.

- Current eddy current inspection equipment requires small tolerances between the coil and the tube wall for sensitivity and accuracy. The inspection method must be such that active element (probe, coil, transducer, etc.) is small enough to fit into the flow orificing device yet still maintain directional detection accuracy.

- The inspection device must be able to be removed for inspection and calibration. This is most feasibly accomplished by inserting the inspection equipment into the line used to drain the steam generator after conducting the integrated safety test. However the inspection equipment penetration into the feedwater system must be pressure tight to provide personnel protection and prevent primary coolant loss in the unlikely event of tube failure during inspection.

8.6.3 Continuous On-Line Inspection

A potential use of the method described above for isolated steam generator inspection is as a continuous on-line method. Rather than guiding the active element into a particular tube, normal feedwater flow could be used to force the active element into a random tube (which will be positively identified by the active element) as the inspection equipment control cable is let out. If the channel head entrance is sufficiently turbulent, then every tube will have a non-zero probability that the active element will enter that specific tube. Therefore, the inspection rate must be high enough to ensure a high statistical likelihood of all tubes being inspected during a specified interval. If all tubes are not inspected during the interval, then an assessment must be made based on the number of inspections made and the number of tubes inspected whether a representative sample has been collected to ensure the reliability of all tubes in that generator.

8.6.4 'Intelligent' Inspection Methods

Early research is being conducted on 'intelligent' inspection methods which use microminiature electronics in a small probe which travels in the fluid stream. The probe is inserted into the system and gathers data as it traverses the system to the exit point. For steam generator tube inspection, the probe would need to be small enough to be entrained by the flowing steam and carried to an exit point in the main steam header. At this point, the probes under development cannot localize detected flaws and thus provides only an indication that a flaw exists somewhere in the (unknown) flow path.

8.7 Summary

This chapter has addressed the steam generator tube integrity inspection barrier, and a summary of design solution alternatives is presented in Table 8.1. Although more steam generator design definition is required to fully analyze this barrier, steam generator tube integrity inspection will be the greatest challenge to achieving the target operating cycle length. Fundamentally, a satisfactory solution will not be developed without a significant technology investment that identifies and develops a novel technique for performing the required inspections.

Table 8.1: Steam Generator Tube Inspection Alternatives Summary

Design Alternative	Report Section	Maintainability	Anticipated Cost	Technology Risk
Isolated Steam Generator Inspection	8.6.2	medium	medium	medium
Continuous On-line Inspection	8.6.3	medium	medium	medium
Traveling Probe Inspection	8.6.4	high	high	high

Chapter 9

Application of Resolution Methodology–Main Condenser

9.1 Introduction

The main condenser is the primary heat sink for the power plant, and the ability to effectively transfer heat to the main condenser is vital to the efficient performance of the entire plant. If the main condenser heat transfer capability is degraded, the plant must be either operated at lower power (to maintain condenser operating conditions) or risk overheating. At it's extreme, overheating can lead to potentially severe condenser shell damage such as overpressurization and rupture.

The primary contributors to main condenser heat transfer degradation are clogging and fouling of the inlet tube sheets and tubes from:

- biofouling (organic debris that adheres to the inside diameter of the tube surface or blocks the intake flow at the tubesheet),
- slime/algae (bacteria that adheres to the condenser tube surface and reduces the usable tube surface area and cooling water flow area while aggravating and accelerating corrosion, erosion and pitting of the condenser tubes),
- barnacles/mussels/clams (small marine creatures which block cooling water flow at the tubesheet and/or adhere to the inner diameter of the tube surface which increases flow

velocity and accelerates tube erosion),

- lodged foreign material (which causes flow deflection leading to localized pitting and erosion), and
- scale (a hard deposit that adheres to the condenser tube surface which reduces heat transfer, decreases plant performance, and causes pitting of the condenser tubes).

All tube fouling will increase flow velocity, reduce heat transfer, increase back pressure and decrease efficiency of the condenser.

9.2 Main Condenser Cleaning and Inspection Barrier

Current nuclear power plants were typically outfitted with shell and straight-tube (copper) condensers with two or three waterboxes. At that time, due to the relatively short fuel cycle, it was anticipated that although fouling would occur it would not result in significant degradation of the overall plant thermal efficiency. Operating experience revealed this assumption to be overly optimistic. Plants experienced significant reductions in thermal efficiency (especially those using silted brackish water) as well as accelerated corrosion leading to tube leaks.

As a result of the early condenser experiences many plants have changed (or plan to change) their condenser tubes to titanium, which is much less susceptible to corrosion. To control fouling a number of strategies have been employed to chemically treat and/or mechanically filter the inlet cooling water. The extent to which a given plant employs these methods depends strongly on the economic balance between capital investment in the systems and the ability of the systems to maintain (or slow the reduction of) plant overall thermal efficiency. In some cases, excessive fouling cannot be prevented for an entire operating cycle and a mid-cycle reduced power window is required to sequentially clean the waterboxes.

Even with the IRIS mid-cycle maintenance shutdown strategy (at the 48 month point), a means to clean the main condenser waterboxes during the cycle must be provided. Design strategies which enable main condenser waterbox cleaning are discussed below.

9.3 Application of the Design Resolution Methodology

Evaluation of the IRIS requirements and main condenser operating history using the methodology of Chapter 5 leads to the conclusion that advances in condenser materials will not allow for extended operating cycles without significant fouling, and so a means to clean the main condenser tubes at power must be developed. Design solution alternatives which meet the requirements are described below.

9.3.1 On-Line Cleaning Enabled by Multiple Waterboxes

Access to the condenser tubes during condenser operation can be readily enabled by utilizing multiple (independently isolable) waterboxes. The strategy requires using *n* waterboxes, each with enough heat removal capability such that only *n-1* waterboxes are required to remove the maximum plant heat load at the least efficient condenser conditions. These conditions are calculated assuming maximum tube fouling for all on-service waterboxes, worst case cooling water conditions (maximum inlet temperature, minimum flowrate), all auxiliary steam loads secured, and maximum plant thermal power (including instrumentation uncertainties). The net effect of these assumed conditions is to have maximum condenser heat input under worst-case heat removal conditions.

Although this method will enable access for tube cleaning, it suffers from two significant drawbacks: cleaning is man-intensive and increasing the number of waterboxes increases condenser complexity (leading to increased capital cost). The condenser is a large component which is fabricated and assembled from a large number of metal parts, and is a small but significant portion (typically on the order of 2-3%) of the total capital investment. A detailed analysis is necessary to find the optimal economic point which balances the number of waterboxes against the amount of installed over-capacity per waterbox. General discussions conducted with a condenser manufacturer indicated that the optimal number of waterboxes is on the order of ten (each with 10% over-capacity).

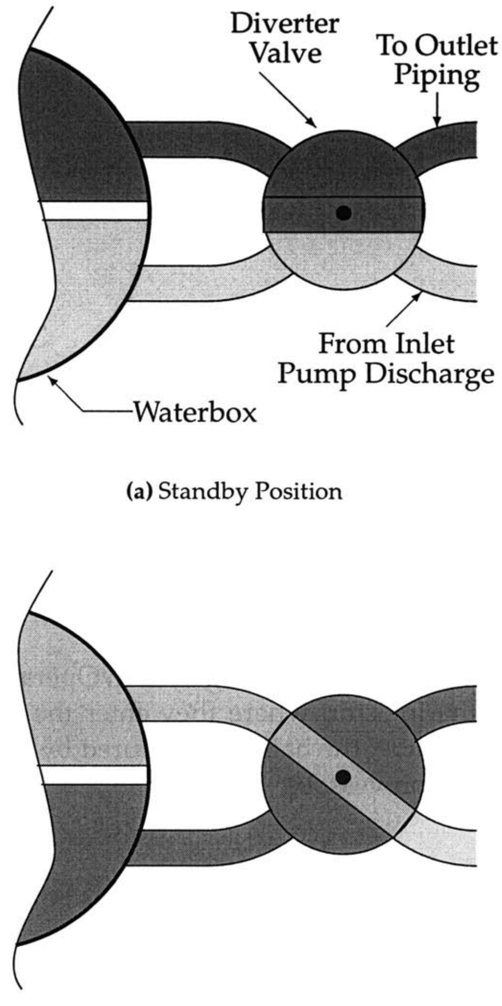
9.3.2 On-Line Cleaning

Improved on-line cleaning methods have emerged in recent years which are effective in reducing fouling to the point where a through off-line mechanical cleaning is required only infrequently (on the order of ten years). Two methods dominate the on-line cleaning market, brush-type and ball-type.[1] Both methods pass an abrasive device through the condenser tubes, using the differential pressure across the waterbox to move the device. However, although both methods are generally effective enough to prevent heat transfer degradation the entire tube circumference may not be throughly cleaned. This streaking inside the tube can result in conditions conducive to galvanic corrosion, although titanium tubes are much less susceptible.

9.3.2.1 Brush-Type On-Line Condenser Cleaning System

In the brush-type method (Figure 9-1) a brush is inserted into each condenser tube. A perforated basket is attached to the end of each condenser tube which prevents the brush from leaving the tube and entering the inlet or outlet waterbox head. A diverter valve is installed between the tubeside piping to and from the unit, and is used to reverse flow direction through waterbox causing the brushes to travel from one basket (through the tube) to the other basket. Figure 9-1(a) shows the flow diverter in the standby normal flow position. With flow in the normal direction, the brushes rest in their "home" baskets. The flow diverter is shown in the reverse flow position in Figure 9-1(b). The brushes are carried through the tubes, cleaning as they pass through the tubes. The brushes are caught by the "temporary" catch baskets at the opposite ends of the tubes and held there for a brief period. When the flow diverter is brought back to the normal flow position, the brushes are carried back through the tubes to their "home" positions where they wait until the next cleaning cycle is initiated. After sufficient time delay, the diverter valve reverses the flow direction through the waterbox causing the brush to again travel the length of the tube. This process is repeated until the waterbox thermal performance is restored.

[1]There are several manufacturers of both brush-type and ball-type condenser tube cleaning systems. The figures and descriptions here are for systems manufactured by WSA Engineered Systems, Milwaukee, WI.



(a) Standby Position

(b) Reverse Flow Position

Figure 9-1: Brush-Type On-Line Condenser Cleaning System Flowpath

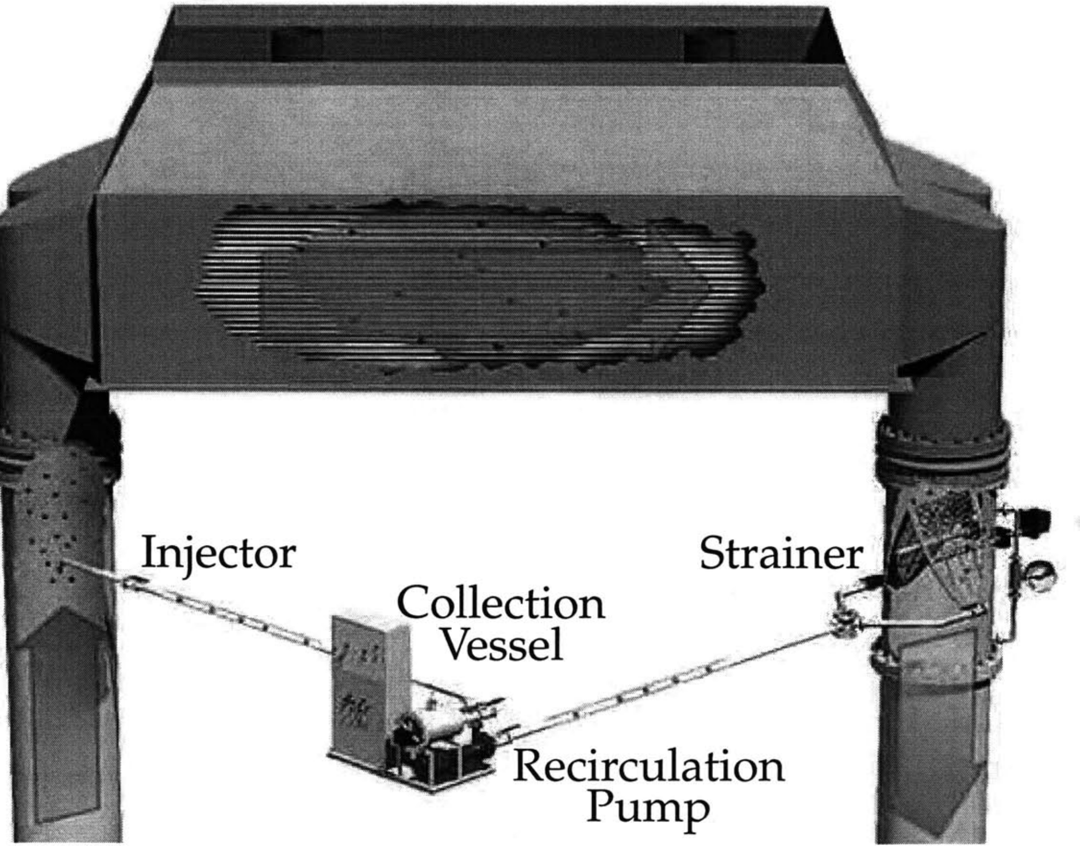


Figure 9-2: Ball-Type On-Line Condenser Cleaning System Operation. Abrasive balls are pumped from the collection vessel to the injector, where they enter the condenser waterbox inlet head. After passing through a random tube, the balls are captured by the strainer and directed (via the recirculation pump) to the collection vessel.

9.3.2.2 Ball-Type On-Line Condenser Cleaning System

The ball-type method (Figure 9-2) uses a large number of abrasive balls which are introduced into the inlet cooling water stream, pass through the condenser tubes, and are recovered from the outlet cooling water stream. Unlike the brush-type method (where one brush is uniquely associated with one tube) this method does not ensure that a cleaning ball will (even after several passes) travel through any individual tube.

9.4 Summary

This chapter has presented several alternatives for consideration in the IRIS design. The first alternative is simply an extension of current design practice to make waterboxes isolable and accessible. The two on-line methods described use an abrasive object (brush or ball) which cleans the condenser tube as it passes through.

Table 9.1 summarizes the alternatives qualitatively ranked by maintainability, economic viability, and technology risk. Within the alternatives generated, installation of *n-1* waterbox redundancy presents essentially no technology risk since it is a simple extension of today's technology. However, without an economic assessment of the cost impact of the additional waterboxes, valves, and associated piping a recommendation of the best alternative to pursue cannot be made.

Table 9.1: Main Condenser Cleaning Alternatives Summary

Design Alternative	Report Section	Maintain-ability	Anticipated Cost	Technology Risk
n-1 Waterbox Redundancy	9.3.1	medium	unknown	very low
Brush-Type On-Line Condenser Cleaning System	9.3.2.1	high	medium	medium
Ball-Type On-Line Condenser Cleaning System	9.3.2.2	medium	medium	medium

Chapter 10

Application of Resolution Methodology–Turbine Generator Throttle Control

10.1 Introduction

The main turbine generator is a multi-stage steam driven turbine coupled directly to a large three phase generator. Current turbine generators use an electro-hydraulic control (EHC) system to position hydraulically-actuated throttle valves which control steam flow to the turbine. Once synchronized to the grid, the turbine generator rotates at constant speed and steam flow controls the amount of power sent to the grid. Nuclear generation is typically used for baseline loading, and little throttle valve movement is necessary for long periods once the plant is at maximum power. When throttle valve movement is required, electric signals are sent to electrically-actuated control valves which reposition and allow the system hydraulic fluid to operate on the throttle valves.

10.2 Main Turbine Generator Maintenance Barrier

When the turbine generator is synchronized to the grid and producing constant (maximum) power, little hydraulic fluid flows through the EHC system. As a result, impurities and wear products (sludge) in the hydraulic fluid collect in low flow regions. These regions typically are in the immediate vicinity of the electrically-actuated control valves, and often lead to sluggish control valve actuation.

When this sluggish actuation occurs, the EHC system does not respond as expected and the turbine generator throttle valve tends to cycle about the desired operating point. This leads to a plant power output to the grid which oscillates about the mean (maximum) power point. If allowed to grow, the oscillation peak can lead to generator overheating and potential stator damage.

Operating experience with current EHC systems shows that the system operates reliably for the current operating cycle length (on the order of 24 months), but that sludge deposits do form. Extrapolating this performance data to longer operating cycles indicates that reliable operation cannot be ensured.

10.3 Application of the Design Resolution Methodology

Evaluation of the IRIS requirements and main turbine generator EHC system operating history using the methodology of Chapter 5 leads to the conclusion that advances in the throttle control system to prevent sludge buildup in low hydraulic fluid flow regions must be developed. Potential design solution alternatives which support development of an advanced EHC system are described below.

10.3.1 Prevention of Sludge Buildup

The root cause of the EHC system reliability issue is inadequate hydraulic fluid flow through the EHC system. In other low hydraulic flow applications, this problem is solved by sending dithering signals in a programmed sequence to the control valves causing them to stroke and disturb the low flow regions. If sufficient control valves exist in the system (which is the case for a modern turbine generator EHC system) then these dithering signals move only a small amount of hydraulic fluid from control valve to control valve and do not result in motion of the hydraulically-actuated component. Although 'as-built' turbine generators did not have this feature, dithering systems are finding increasing application in turbine generator control systems for both baseline and load following applications.

Another technique which addresses the low hydraulic fluid flow problem is the use of ultrasonic transducers to agitate the hydraulic fluid at high frequency to prevent sludge from settling. Unlike the dithering technique which surges hydraulic fluid from one control valve to the next, this method simply keeps the sludge suspended in the fluid allowing it to move through the system when the control valves are actuated.

Finally, synthetic oils typically contain much less impurities than do petroleum fluids and have (or can be formulated to have) similar properties. Synthetic oil types include polyalphaolefins, diesters, polyol esters, alkylbenzenes, polyalkylene glycols, phosphate esters, silicones, and halogenated hydrocarbons. Synthetic oils are generally organic compounds and cost much more than petroleum oils. However, each type has one (or more) specific properties that are better than petroleum oils, and the limitations can generally be corrected by chemical additives. For IRIS, a detailed analysis of the EHC system characteristics must be made to match the synthetic oil properties to the application.

10.3.2 Electric Control System

A control system which uses electric linear motors to position the throttle valves would not be subject to the stability problems experienced by an EHC system. However, there is no industry impetus or manufacturer initiatives to improve the current technology EHC system (since it is generally reliable throughout the current operating cycle). Therefore, a significant research and development expenditure would be necessary to make an electric control system commercially viable.

10.4 Summary

In procuring a main turbine generator, performance specifications are typically given to the manufacturer and then the manufacturer uses it's own technologies to meet those specifications. Reactor plant designers rarely are involved directly in the main turbine generator design. The IRIS design team intends to be indirectly involved in the main turbine generator design process, funding research initiatives where necessary and applicable. Directly designing a main turbine generator system (including throttle control system) does not adequately leverage the design experience of the manufacturer and involves significant technology and economic risk.

It is likely that integrating available current technologies into the EHC system is more cost effective than development of an advanced technology control system. Only operating experience with this improved system will determine if this solution meets the long-term IRIS target cycle length goal of eight years, but it is anticipated that this improved EHC system will operate reliably for at least four (and possible to eight) years. This 'operate and assess' strategy has been adopted by the IRIS design team to maximize the potential for success while minimizing risk.

Chapter 11

Application of Resolution Methodology–Reduced Power Window Surveillances

11.1 Introduction

There are a large number of investment protection based surveillances (67 total) which are currently performed off-line but could be performed on-line at reduced power. Most of these surveillances have performance intervals much less than 48 months (typically 18 months) and therefore can also be considered to be barriers to a 48 month (full power) operating cycle. These surveillances have been generalized into six broad categories, and their resolution in IRIS is described below.

11.2 Resolution of Reduced Power Window Surveillances

11.2.1 Circulating Water/Service Water Pump and Traveling Screen Inspections

To conduct the required pump and traveling screen inspections requires one traveling screen-pump-heat exchanger train to be secured and drained. At the candidate PWR there are three identical parallel trains, each capable of removing approximately 46% of the maximum heat load.

Therefore, when one train is secured for inspection the total plant power (which is proportional to heat load) is limited to approximately 92%. In IRIS, additional train heat removal capacity or additional redundancy will be utilized to allow these surveillances to be conducted on-line with no power restrictions.

11.2.2 Generator Stator Cooling

Current main turbine generator sets use two identical cooling loops to cool the generator stator, both of which must be on-line for full power operation. As above, this limitation can be readily solved in IRIS using by adding redundancy to allow one cooling loop to be removed from service with no power restrictions.

11.2.3 Main Turbine Lube Oil System Pressure Switch Calibrations

These surveillances can be performed on-line with adequate installed redundancy. Installation of one additional pressure switch would maintain the original number of required on-service switches while allowing one to be removed from service for testing or repair. To avoid an inadvertent turbine generator trip, a digital trip control system will need to be developed but the technology required is founded in current practices.

11.2.4 Nuclear Instrument Calibration

Calibration of the power range nuclear instruments cannot be conducted without a change in power level since a single data point cannot establish the required instrument gain setting (i.e., the slope of the calibration curve). The current calibration method requires steady power to be maintained at a low level (approximately 20% reactor power) for data collection and then at high (near maximum) power. New techniques are being developed[1] which use automatic data collection from in-core flux monitors and require only a small reduction in reactor power (without a change in steam flow). This technology is anticipated to be available in 1-3 years.

[1] "An In-Core Power Deposition and Fuel Thermal Environmental Monitor for Long-Lived Reactor Cores," U.S. Department of Energy Nuclear Energy Research Initiative, Proposal No.: 2000-069, awarded to Ohio State University

11.2.5 Main Steam Isolation Valve Maintenance

Main steam isolation valve stroke check and actuating system surveillances will be conducted as part of the integrated passive safety systems operability testing described in Section 11.3 below.

11.2.6 Feedwater System Inspections and Calibrations

These surveillances will be conducted as part of the integrated passive safety systems operability testing described in Section 11.3 below.

11.3 IRIS Integrated Testing and Coordinated Maintenance

Based on the AP600 proof of operability burden, it is anticipated that demonstrating operability of the IRIS passive cooling scheme will require initiation of cooling and measurement of both cooling loop flowrate and heat transferred to the heat sink. Although the flowrate and heat transfer can be determined a priori for the (at power) primary coolant circuit conditions, proper flow conditions in the passive cooling loop will not be established without steam and feed flow being secured. Therefore, this operability test could be conducted with the reactor at power but with the steam and feed headers for the passive cooling loop secured. This corresponds to 75% total steam flow (three-fourths of all installed steam generators operating) for testing.

The integrated test begins with isolation of steam and feed flow for the header being tested, followed by initiation of passive cooling flow, and ends with measurement of the parameters necessary to demonstrate operability. After conducting the operability test, the passive cooling system is then drained to allow for in-situ steam generator tube inspections (if these inspections cannot be deferred to a maintenance outage). Finally, necessary maintenance on the main steam isolation and feedwater isolation valves is performed. After completion of maintenance and before restoring normal operation, feedwater regulating valve maintenance and feed pump maintenance is conducted as well as the containment safety features response time testing of Chapter 6, Section 6.2.1.2. It is estimated that this entire maintenance block can be completed, for one passive cooling circuit, in one week.

Also to be considered is reactivity control testing which may need to be conducted. Operation within the reactor safety analysis assumptions regarding rod control system performance cannot be assured for eight years of continuous operation without demonstration of the normal and emergency reactor shutdown mechanism. The IRIS team is currently examining different reactivity control and reactor shutdown methods, so it is not possible at this point to identify likely testing requirements. What can be assumed, however, is the core design must allow for at-power testing of the reactor shutdown mechanisms if a mid-cycle reactor shutdown is to be avoided.

From the above testing profile, it is estimated that IRIS will enter a two week 75% power window every two years (testing two passive cooling loops sequentially during each reduced power outage). Therefore, IRIS will operate 75% power for two weeks every two years or six weeks over the entire eight-year fuel cycle. The fourth testing and maintenance period would be scheduled to coincide with the refueling outage. Including a one-month refueling outage over the 96 week fuel cycle, the conservatively estimated theoretical unit capability factor is 98.4%.

11.4 Reduced Power Surveillance Strategy

It is unlikely that design solutions for all barriers with periodicities less than eight years can be found to allow maintenance to be performed on-line without power restriction. Although this thesis has attempted to assist in developing design solutions which enable this condition, surveillances have been identified (specifically, safety system operability demonstration and steam generator tube integrity inspections) which cannot feasibly be performed on-line without a very significant technology development effort. Therefore, given that a reduced power window will be required a strategy that considers capital investment, investment protection requirements, and availability should be developed.

For example, the design solution which allows performing circulating and service water system maintenance (Subsection 11.2.1, above) is installation of redundancy. However, if this maintenance were scheduled for completion during a reduced power window then the capital investment for an additional cooling train can be avoided without impacting overall plant availability. This analysis is beyond the scope of this thesis, but is described here since it contributes to an overall sound design strategy.

11.5 Summary

Design solutions to the most significant of the identified maintenance barriers requiring reduced power have been proposed, with the notable exception of safety system operability testing. Therefore, given that a reduced power window is required in IRIS, a strategy which minimizes the duration of this window has been proposed. This strategy meets the current regulatory requirements for operability demonstration without significant regulatory changes or technological advances.

It must be recognized that a balanced economic strategy does not always justify the required investment to eliminate reduced power windows. Development of materials and technologies to eliminate currently unresolvable reduced power surveillances potentially requires a large research and development investment and delay in fielding such solutions. Also, the capital cost of solutions which require the installation of additional capacity or redundancy may not be justified relative the the reduction in availability from performing maintenance of lower capacity systems during a reduced power window.

Chapter 12

Summary and Future Work

12.1 Summary

A renewed interest in new nuclear power generation in the United States has spurred interest in developing advanced reactors with features which will address the public's concerns regarding nuclear generation. However, it is economic performance which will dictate whether any new orders for these plants will materialize in the next decade. Economic performance is, to a great extent, improved by maximizing the time that the plant is on-line generating electricity relative to the time spent off-line conducting maintenance and refueling. Indeed, the strategy for the advanced light water reactor plant IRIS (International Reactor, Innovative & Secure) is to utilize an eight year operating cycle.

A formalized strategy to address, during the design phase, the maintenance-related barriers to an extended operating cycle does not exist. Therefore, the top-level objective of this thesis was to develop a methodology for injecting component and system maintainability issues into the reactor plant design process to overcome these barriers.

A primary goal was to demonstrate the applicability and utility of the methodology in the context of the IRIS design. The methodology developed has been demonstrated to narrow the design space to feasible design solutions which enable a desired operating cycle length, yet is general enough to have broad applicability. Feedback from the IRIS design team indicates that the proposed solutions to the investigated operating cycle length barriers are both feasible and consistent with sound design practice.

12.1.1 Methodology Development

The first step in meeting the top-level objective was to determine the types of operating cycle length barriers that the IRIS team is likely to face. An investigation into the regulatory and investment protection surveillance program barriers preventing a candidate operating PWR from achieving an extended (48 month) cycle has been recently completed.[1] This presented a logical starting point, and the results of the operating PWR investigation were examined in the context of the IRIS design. Relative to IRIS, the surveillances were generalized and placed into one of the following categories:

Category 1: On-line surveillances which will be performed on-line in IRIS;

Category 2: Candidate surveillances for design resolution to create an on-line performance mode in IRIS;

Category 3: Surveillances requiring further analysis to determine performance mode in IRIS; and,

Category 4: Off-line surveillances likely to have performance interval extended to at least eight years in IRIS.

The design methodology was developed to address those surveillances in Category 2.

The operating PWR investigation addressed a 48 month operating cycle, but the IRIS operating cycle length goal is eight years. Therefore, the 54 surveillances resolved to Category 2 represent a minimum number of potential IRIS operating cycle length barriers. It is likely that additional unidentified barriers exist which were already compatible with the 48 month operating PWR cycle length (i.e., performance periodicities already greater than 48 months) but may be a barrier to an eight year operating cycle. However, this thesis did not investigate surveillances with a periodicity greater than 48 months. But, since the 54 known barriers cover a broad spectrum of systems and components they were considered representative of the design challenges likely to be presented by the unidentified barriers.

[1] Moore Jr., Thomas Joseph, "A Surveillance Strategy for a Four Year Operating Cycle in Commercial Pressurized Water Reactors," Massachusetts Institute of Technology Department of Nuclear Engineering, Nuclear Engineer's Thesis, May 1996.

The design methodology developed is a four-step process. The first step is the synthesis of the general requirements that the component must satisfy. The second step is the synthesis of the design objectives with the design requirements. The third step is to bound the solution space by application of suitable and relevant constraints. The final step is to develop design alternatives which meet the specifications of the synthesized design requirements, objectives, and constraints. Like any design process, the methodology flowchart is iterative in nature.

12.1.2 Methodology Application

The methodology was applied to the identified (Category 2) operating cycle length barriers. Many of the barriers were considered (based on discussions with the IRIS design team) to be readily solved by design, and so a detailed investigation into these barriers was not conducted. However, several IRIS operating cycle length barriers emerged which required further investigation:

- Primary relief valve testing,
- Steam generator eddy current inspection,
- Condenser waterbox maintenance,
- Main turbine throttle control maintenance,
- Safety system testing,
- Reduce power window items, and
- Reactivity control system testing.

Detailed design of the IRIS core has not been completed, and so reactivity control system testing could not be addressed in this thesis. The resolution methodology was applied to the remaining barriers and feasible (as assessed by the IRIS design team) design alternatives were proposed which enable achievement of the eight year IRIS operating cycle length goal.

12.1.3 Resolution Methodology Limitations

The resolution methodology developed and applied in this thesis is not intended to eliminate the need for creative thought in the design process. This point cannot be emphasized enough, since it is the creative element that allows any design to be a significant improvement over the current standard. Although the results presented in this thesis for overcoming operating cycle length barriers are the produce of a structured methodology, they cannot be reproduced without including the creative design element. What can be reproduced, however, is the synthesis of relevant factors into a limited set of governing constraints which guide that creative process toward feasible solutions which meet the specified requirements.

12.2 Future Work

This thesis is the first attempt at developing a structured methodology to address maintenance-related barriers to an extended operating cycle. As with any methodology, refinement and improvement to the methodology can be made by identifying limitations in it's applicability. Although feasible solutions were generated which will assist the IRIS design team in achieving the target operating cycle length goal of eight years, there is a significant amount of future work that must be completed to improve the utility of this methodology as discussed below.

Chapter 2, 'Design Methodology Framework,' discussed the methodology inputs necessary to develop feasible design solution alternatives. These inputs were developed with the prior knowledge of the barriers requiring design resolution. Additional investigation into the creative design process needs to be conducted to ensure that all relevant factors have been adequately captured. If additional factors exist, then the methodology of Chapter 5, 'Eliminating the Maintenance–Related Barriers,' needs to be updated to include them.

Chapter 3, 'Operating Pressurized Water Reactor Surveillance Program,' presented the results of an investigation which considered only those surveillances which have periodicities less than 48 months. The identified barriers from the candidate operating PWR investigation were considered to be representative of the types of barriers likely to emerge from the unexamined (greater than or equal to 48 month timeframe) surveillances. Further examination of this timeframe is required to validate the assumption that the spectrum of potential barriers has been bounded.

Chapters 6 through 11 presented design solution alternatives to identified IRIS operating cycle length barriers. As the IRIS design matures, these solutions need to be continuously evaluated against the IRIS design requirements, objectives, and constraints to ensure that the proposed solutions remain feasible.

The resolution methodology is structured to generate design solution alternatives which enable performing all maintenance on-line at full power. Some barriers have been identified which either require a significant research and development expenditure or must be performed during a reduced power window. It would be useful to examine the economic dependence between lost revenue in reduced power windows and capital expenditure to eliminate those windows to develop a more balanced (economically driven) design strategy.

Finally, the operating cycle length barriers considered in this thesis were well defined and a large amount of supporting data was available (since these barriers were not IRIS unique) to assist in resolution. As the IRIS design matures, IRIS unique operating cycle length barriers not faced by currently operating plants will certainly emerge. Application of the resolution methodology to these barriers will challenge the fundamental structure of the methodology more thoroughly than has been done in this thesis.

Appendix A

Identified IRIS Maintenance Barriers

This appendix summarizes the maintenance-related barriers described in Chapter 3, 'Operating Pressurized Water Reactor Surveillance Program.' Table A.1 addresses the barriers to a four year operating cycle, Table A.2 addresses the reduced power window items, and Table A.3 addresses the barriers to an eight year operating cycle.

Table A.1: Identified Maintenance Barriers to Four-Year Operating Cycle

Description	Discussion	Basis
Installation of ASME Class 1 relief valves (including pressurizer relief valves and steam generator/main steam safety valves).	The ASME Boiler & Pressure Vessel Code primarily deals with relief valve construction, with limited discussion of installation requirements. Within the Code, installation of an inlet isolation valve is permitted under certain (inadequately specified) conditions. IRIS requires a relief valve inlet isolation valve to conduct in situ (but off-service) valve testing.	ASME Boiler & Pressure Vessel Code, Section III, 'Nuclear Power Plant Components,' Article NB-7000, 'Overpressure Protection.'
Periodic testing of ASME Class 1 relief valves (including pressurizer relief valves and steam generator/main steam safety valves).	All valves of each type and manufacture shall be tested within each subsequent[1] 5 year period with a minimum of 20% of the valves tested within any 24 months. This 20% shall be previously untested valves, if they exist.	'The Operation and Maintenance of Nuclear Power Plants,' ASME/ANSI OM-1989[2], Part 1, 'Requirements for Inservice Performance Testing of Nuclear Power Plant Pressure Relief Devices,' §1.3.3, 'Test Frequency, Class 1 Pressure Relief Devices.'

Table A.1 continued on next page

[1]During the initial 5 year period, no testing is required during the first 12 months. Testing shall be performed on a minimum 25% of the valves of each type and manufacture during each following 12 month interval such that at the end of 24 months of operation 25% have been tested, 50% in 36 months, 75% in 48 months, and 100% in 60 months. Additionally, during any running 24 month period a minimum of 20% of the valves (previously untested, if they exist) shall be tested.

[2]Subsequent updates to 'The Operations and Maintenance of Nuclear Power Plants' have occured but the 1989 Edition is referenced in 10 CFR §50.55a(b).

Table A.1 continued from previous page

Description	Discussion	Basis
In-Service testing of safety related pumps and valves.	All safety related pumps and valves are required to be tested for operability on a quarterly basis. Under certain circumstances, tests which cannot be conducted at power can be designated as either *cold shutdown tests* or *refueling tests* and be deferred to the next outage. Neither require prior NRC approval but they must be justified, augmented by risk-based arguments, and are auditable.	ASME Boiler & Pressure Vessel Code, Section XI, 'Rules for In-Service Inspection of Nuclear Power Plant Components.'
Periodic testing of motor operated valves (MOVs) in safety related systems.	NRC requires all MOVs in safety related systems to be diagnostically tested and, where practical, tested to their design basis condition. Testing interval is determined by combining the risk significance and failure rate.	ASME Boiler & Pressure Vessel Code, Section XI, 'Rules for In-Service Inspection of Nuclear Power Plant Components' supplemented by NRC Generic Letter 96-05, 'Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves,' issued September 18, 1996.
Engineered safety feature actuation tests (integrated tests which involve the complete safety signal path from sensor to system actuation).	These tests are currently performed each outage and are typically segregated into three tests: engineered safeguards actuation, containment isolation, and core cooling.	Numerous NRC and ASME regulations apply.

Table A.1 continued on next page

Table A.1 continued from previous page		
Description	**Discussion**	**Basis**
Steam generator eddy current testing.	Steam generator eddy current testing is currently performed shutdown at 18-24 month intervals. NRC inspection guidelines for steam generators allows for periods between steam generator eddy current testing of up to 40 months, after two previous successful inspections at shorter intervals.	Numerous NRC, ASME, and industry regulations and recommendations apply. NRC letter SECY-00-0078, 'Status and Plans for Revising the Steam Generator Tube Integrety Regulatory Framework,' dtd 30 March, 2000, indicates the intent for the NRC to accept the recommendations and inspection procedures contained in the Nuclear Energy Institute (NEI) initiative, NEI 97-06, 'Steam Generator Program Guidelines.'
Periodic testing of ASME Class 2 relief valves used in non-primary pressure boundary applications.	All valves of each type and manufacture shall be tested within each subsequent[3] 10 year period with a minimum of 20% of the valves tested within any 48 months. This 20% shall be previously untested valves, if they exist.	Investment protection based. Testing is governed by 'The Operation and Maintenance of Nuclear Power Plants,' ASME/ANSI OM-1989[4], Part 1, 'Requirements for Inservice Performance Testing of Nuclear Power Plant Pressure Relief Devices,' §1.3.4, 'Test Frequency, Classes 2 and 3 Pressure Relief Devices.'
		Table A.1 continued on next page

[3]During the initial 10 year period, no testing is required during the first 24 months. Testing shall be performed on a minimum 25% of the valves of each type and manufacture during each following 24 month interval such that at the end of 48 months of operation 25% have been tested, 50% in 72 months, 75% in 96 months, and 100% in 120 months. Additionally, during any running 48 month period a minimum of 20% of the valves (previously untested, if they exist) shall be tested.

[4]Subsequent updates to 'The Operations and Maintenance of Nuclear Power Plants' have occured but the 1989 Edition is referenced in 10 CFR §50.55a(b).

Table A.1 continued from previous page

Description	Discussion	Basis
Condenser waterbox inspection and cleaning.	The primary degradation mechanism of the heat trasfer capability of the condenser is the fouling and clogging of the condenser inlet tube sheets and tube surfaces from either debris or marine growth.	Investment protection based to maintain plant thermodynamic efficiency. Numerous organizations (Occupational Safety and Health Administration (OSHA), Employee's Union, etc.) address personnel safety issues.
Control rod drop and position indication testing	Control and shutdown rod drop testing is currently performed following refueling to guarantee that the control rods have an unimpeded path to the bottom of the core and that maximum drop times are consistent with the assumed drop times used in the plant safety analysis. No upper limit to testing frequency is specified by the NRC, primarily since no current plant operates on a long fuel cycle where fuel swelling (leading to control rod binding) is of concern. Position indication testing is currently deferred since chemical reactivity control is used and the control rods are fully withdrawn for the duration of the operating cycle (no position uncertainty).	Numerous NRC regulations including 'NRC Improved Standard Technical Specifications.'

Table A.1 continued on next page

Table A.1 continued from previous page		
Description	**Discussion**	**Basis**
Electrically operated safety component circuit breaker overcurrent relay checks	Overcurrent relay checks of electrically operated safety component circuit breakers (primarily pumps) is performed periodically to ensure that a bound safety component will trip off-line not be sensed by the safeguards systems as being in operation.	
Main turbine lubricating oil system low-pressure trip switch calibration.	Calibration of the main turbine low shaft pump discharge pressure and low bearing oil pressure switches require the main turbine to be shutdown.	Vendor provided investment protection.
Main turbine electrohydraulic control (EHC) system clean and inspect.	A complete EHC system clean and inspect, including software and filter replacement, is required at less than four-year intervals (typically 24 months). This maintenance requires the main turbine to be shutdown.	Vendor provided investment protection. Can be performed on-line if a non-zero steam demand (i.e., from a steam dump or large auxiliary loading) can be provided.

Table A.2: Identified Maintenance Barriers Requiring Reduced Power

Description	Discussion	Basis
Steam and feedwater flow meter calibrations.	Requires the applicable steam generator to be secured.	Investment protection.
Feedwater control valve indication and stroke check.	Requires the applicable steam generator to be secured.	Investment protection.
Feedwater isolation valve stroke check.	Requires the applicable steam generator to be secured.	Investment protection. Regulatory based if feedwater isolation valve closure is part of safety system actuation.
Hydraulically operated valve fluid change. (Main steam isolation valves and feedwater isolation valves are typically hydraulically operated.)	Requires the applicable steam generator to be secured.	Investment protection.
Main feed pump governor calibration.	Requires steam demand to be reduced to the capacity of remaining main feed pump(s).	Investment protection.
Main steam isolation valve stroke check.	Does not require complete stroke, and causes only a small reduction in steam flow. Steam demand typically reduced sufficiently to prevent oversteaming remaining generators if valve inadvertently closes fully.	Investment protection.

Table A.2 continued on next page

Table A.2 continued from previous page		
Description	**Discussion**	**Basis**
Component cooling water system pump maintenance.	Requires total system heat load to be reduced to the capacity of remaining component cooling water system pump(s).	Investment protection.
Heat exchanger inlet traveling screen clean and inspect.	Requires heat exchanger to be secured. Total system heat load is required to be reduced to the capacity of the remaining heat exchanger(s).	Investment protection.

Table A.3: Identified Maintenance Barriers to Eight-Year Operating Cycle[5]

Description	Discussion	Basis
Electrical switchgear clean and inspect including motor-operated valve inspections and cable meggers, starter checks, breaker inspections, unique features testing, and visual inspections.	Encompasses a large number (over 450) of unspecified surveillances with periodicities between 48 and 96 months. Shutdown may be required to permit accessibility or when entire (vital) switchboards must be secured.	Investment protection; fire safety.
Periodic maintenance of non-safety system manually operated valves. Scope is typically disassemble-inspect-reassemble followed by functional testing.	Periodicity established by vendor, typically between 48 and 72 months. May require shutdown if valve cannot be isolated from system.	Investment protection.
Reactor coolant system and containment integrated leak rate testing.	Extent and frequency based on material history.	ASME Boiler & Pressure Vessel Code, Section XI, 'Rules for In-Service Inspection of Nuclear Power Plant Components.'
Main turbine trip and throttle valve inspection.	Major valves in the main turbine steam supply path are typically inspected at 48-60 month intervals.	Vendor provided investment protection.
Fire station and snubber inspections.	Typically performed at 48-60 month intervals. Some stations inside containment are not accessible at power.	Investment protection; fire safety.

Table A.3 continued on next page

[5]Includes all known barriers identified in Table A.1.

Table A.3 continued from previous page		
Description	**Discussion**	**Basis**
Control rod drive mechanism (CRDM) motor-generator set mechanical and electrical maintenance.	Requires motor-generator set to be secured. Typically performed at 54 month intervals.	Investment protection.
Charging pump mechanical and electrical maintenance.	Performed at 48-72 month intervals. Requires shutdown if insufficient charging capacity is available.	Investment protection and regulatory based.
Main turbine generator (generator end) electrical inspections. Includes stator visual inspection and megger, and exciter inspection and megger.	Typically performed at 54 month intervals.	Vendor provided investment protection.