HISTORICAL CIVILIAN NUCLEAR ACCIDENT BASED NUCLEAR REACTOR CONDITION ANALYZER

by

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STATEMENT OF THESIS APPROVAL

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ABSTRACT

There are significant challenges to successfully monitoring multiple processes within a nuclear reactor facility. The evidence for this observation can be seen in the historical civilian nuclear incidents that have occurred with similar initiating conditions and sequences of events. Because there is a current lack within the nuclear industry, with regards to the monitoring of internal sensors across multiple processes for patterns of failure, this study has developed a program that is directed at accomplishing that charge through an innovation that monitors these systems simultaneously.

The inclusion of digital sensor technology within the nuclear industry has appreciably increased many computer systems' capabilities to manipulate sensor signals, thus making the satisfaction of these monitoring challenges possible. One such manipulation to signal data has been explored in this study. The Nuclear Reactor Condition Analyzer (NRCA) program that has been developed for this research, with the assistance of the Nuclear Regulatory Commission's Graduate Fellowship, utilizes one-norm distance and kernel weighting equations to normalize all nuclear reactor parameters under the program's analysis. This normalization allows the program to set more consistent parameter value thresholds for a more simplified approach to analyzing the condition of the nuclear reactor under its scrutiny.

The product of this research provides a means for the nuclear industry to implement a safety and monitoring program that can oversee the system parameters of a nuclear power reactor facility, like that of a nuclear power plant.

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The foundation of this research was rooted in a layman's understanding of how the pathways of nuclear accidents arise and their repetition occurs. And although my understanding of the complexity that resides at the intersection of an accident's initiating conditions and the prior condition of that system has broadened, the layman roots of this thesis maintain its merit in the face of my technical enlightenment.

Any technical enlightenment that I have gained is partly due to the fellowship award graciously awarded to me in the name of the Nuclear Regulatory Commission (NRC). And for that, I would like to thank the NRC for supporting my graduate work at the University of Utah. This thesis is dedicated to the NRC in support of their mission to regulate and support the nuclear industry in the United States. This study has been directed towards assisting them to enhance the safety and effectiveness of our national nuclear interest.

I would like to express my deepest gratitude to my advisor, mentor, and the director of the University of Utah's Nuclear Engineering Program (UNEP), Dr. Tatjana Jevremovic. I would not be in this wonderful position in my life without Dr. Jevremovic's guidance. The UNEP facilities and staff are unparalleled in their knowledge and willingness to help others. And I will always be grateful for the opportunities that I was afforded while being part of that legacy.

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CHAPTER 1

INTRODUCTION

1.1 Objective

The core objective behind this research was based on the idea that the academic community, through innovative research, could provide assistance to the United States Nuclear Regulatory Commission (US NRC) with straightening national interest in the fields of nuclear science and technology. The product of this research provides a means for the nuclear industry to implement a safety and monitoring program that can oversee the system parameters of a nuclear power reactor facility, like that of a nuclear power plant (NPP).

The preliminary design of this program has targeted the primary and secondary processes of light water reactor (LWR) NPPs. More specifically, the parameters associated with the coolant and steam cycles of boiling water reactors (BWR) and pressurized water reactors (PWR) are the targets of this current research model. Because there is a current lack within the nuclear power generation industry, with regards to the monitoring of internal sensors across multiple processes for patterns of failure, this program is directed at accomplishing that charge through an innovation that monitors these systems simultaneously. This will provide an operational failsafe that is compatible with a variety of plant types. But its application within the nuclear power generation industry is particularly interesting. Due to the fact that the majority of nuclear power plants are still operating on analog, or analog-digital hybrid instrumentation as described by the 2010 IAEA report on "Integration of Analog and Digital Instrumentation and Control Systems in Hybrid Control Rooms." This type of individualized system reporting results in a plant's operators monitoring each internal system (coolant: temp, flow rate; steam temp, flow rate…etc.) individually; this lack of total system cohesion in NPP safety monitoring has resulted in a number of NPP incidents and accidents. This research intends to illustrate how the combination of a computation using modified kernel weighting (KW) and numerical representations of historical nuclear power plant parameter failure scenarios can provide an operational failsafe that has the ability to encompass an overall system awareness. This operational failsafe, named the Nuclear Reactor Condition Analyzer (NRCA), is aimed towards the prevention of operational failures within the various civilian nuclear process industries.

1.2 Motivation

The motivation behind this research was founded through the identification of key areas in our national nuclear interest and how nuclear engineering can effect change in those areas. There exists a veil of public uneasiness associated with the international expansion of nuclear energy, not unduly, because of some unfortunate events that have occurred throughout the world over the last 4 decades. But when the political climate demands that nuclear energy be brought to bear against the looming national energy crisis, nuclear technological expansion moves ahead in the national discussion of nuclear cost to benefit analysis.

Of course, every historical nuclear power generation industry accident was predated by the events that occurred in Hiroshima and Nagasaki during World War II

(WWII). Where the America public was provided a visual representation of the potential power and devastation that nuclear energy is capable of with the delivery of the Fatman and Little Boy bombs. This powerful display of nuclear energy's might, coupled with the average individual's lack of significant understanding about the various levels of nuclear potential, has left the public with the idea that nuclear power generation expansion is an undertaking that is full of risk. The events at Three Mile Island in the 1970s, Chernobyl in the 1980s, and Fukushima in 2010s have compounded this visual to form the current global idea of nuclear insecurity. And although nuclear power has remained one of the safest forms of power generation, especially in the United States of America, the following illustrates the key areas that frequently arise when the advancement of nuclear technology is discussed:

- **i)** Negative public opinion – Newsworthy nuclear incidents shift public opinion from a positive to a negative majority with respect to nuclear proliferation. Tragedy not only befalls the direct victims of the incident, the environment pays a heavy price, as well.
- **ii)** Isolated or incomplete databases – The amount of credible information on nuclear incidents is not readily accessible outside of select groups, nor is it often comprehensive. Therefore, the exploitation of vital information formed during the stages of an accident is limited to a select number of governmental or proprietary groups.
- **iii)** Misinterpreted or exaggerated information – As is often the case, when the amount of credible information is limited to the public, the public often misinterprets or exaggerates the information that is provided. And the public

sector, without thorough system analysis and parameter documentation, has a limited ability to assist the nuclear industry with incident research and prevention.

iv) Limited modeling and testing – Due to the security issues surrounding nuclear research, many nuclear research facilities have been downsized or eliminated. This has led to a national deficiency in the number of nuclear scientists and engineers that are qualified to advance the national nuclear interest.

These challenges illustrate how influential historical events can be on nuclear policy, therefore, it was felt that the research community must fully exhaust every possible type of data on civilian nuclear accidents in order to have those influences on nuclear policy be positive. By developing the Nuclear Power Process Condition Analyzer (NRCA), this research will provide the foundation for the continued exploration of the connections between individual system parameters and operational failures. And by doing so, the foundation will be placed to a more open avenue for academic research into nuclear industrial process safety and monitoring.

1.3 Thesis Overview

The outline to this thesis is as follows:

Chapter 1 provides the objective, motivation, and overview.

Chapter 2 provides the background information for this research by describing the conceptual foundation, information about nuclear industry incidents, and the basics of industrial incident pathways.

Chapter 3 describes nuclear power generation within the United States of America, USA nuclear reactor types, and the peripheral systems that support the main reactor within the USA's NPPs.

Chapter 4 explains the fundamentals of kernel weighting (KW), how kernel weights have been utilized within industrial process monitoring, and how KW has been modified and applied towards the objective for this research.

Chapter 5 describes the console program that was developed from this research that employs the modified KW algorithm.

Chapter 6 provides a summation of this study's objectives and outcomes.

Chapter 7 discusses the future direction of this research and desirable outcomes for further work on its program.

CHAPTER 2

BACKGROUND

2.1 Conceptual Foundation

The concept for a process analyzer that critiques the individual parameters of a system was constructed from the simplicity of cause and effect. The simplicity of cause and effect can be seen in the program's use of subsystem anomalies to identify patterns that may result in partial or total system failure. But the identification of patterns between multiple systems with differing parameters is difficult without the tools suited for this unique application. The tools for the realization of a process analyzer reside with the methods used in sensor validation. The idea being that a system of subsystems needs validation, just as a system of sensors requires validation. The validation of a system can be found by the comparison of real-time to nominal and real-time to incident system parameter values. This validation technique can be utilized for total system awareness because, typically, a system undergoes abnormal operations in one or more of its subsystems preceding a system wide failure.

A benign example of a subsystem anomaly preceding a system wide failure can be seen in the events that caused the incident at Davis-Besse Reactor One in 1977. Many of the preliminary events that occurred at Davis-Besse in 1977 were nearly identical to the preliminary events that occurred at Three Mile Island NPP on March 28th, 1979. The Davis-Besse 1977 events occurred as follows (Bertini, 1980):

(1) With the reactor operating at 10% power, there was a significant loss of feedwater to one of the steam generators in the reactor's secondary system.

(2) This loss of feed water flow to the steam generator resulted in a valve closure in the condensed water supply feeding the steam generator.

(3) Subsequently, the existing water in the steam generator boiled away leaving it incapable of removing heat from the primary system.

(4) Due to the loss of the reactor's heat sink, the temperature and pressure in the primary coolant system increase.

(5) Due to the elevated pressure in the primary coolant system, the overburden valve on the pressurizer opened but did so in an abnormal manner. The valve oscillated from open to closed numerous times before becoming stuck open.

(6) Subsequently, the steam pressure in the quench tank increased rapidly, and the rupture disk on the tank ruptured.

(7) Coolant and steam from the reactor was now in the containment building due to this excursion.

(8) The operator shutdown, or scrammed, the reactor by dropping the control rods fully into the core.

(9) Since the reactor was operating below its threshold for a step-wise shutdown, once the reactor was shut down, the primary system temperature and pressure began to decrease.

(10) Because of the malfunctioning pressurizer valve being stuck in the open position, the primary coolant system pressure decreased below its saturation threshold, causing a surge of steam in the primary coolant system.

(11) Due to the surge of steam in the primary coolant system, the primary coolant was forced into the pressurizer beyond its maximum value resulting in coolant escaping through the suck pressurizer overburden valve.

(12) At this point, the operators became aware of the open pressurizer valve and closed an isolation valve preventing further pressure losses in the primary coolant system.

(13) The pressure in the primary coolant system increased, the steam bubbles in the coolant collapsed, and the primary system stabilized to a nominal condition.

The series of events that are described above were predated by similar events that occurred on August 20th, 1974, at the Benzau NPP and postdated by similar preliminary events that occurred on March 28th, 1979, at the Three Mile Island NPP. The events that occurred at Three Mile Island ultimately resulted in that core's meltdown. This illustrates that similar, if not identical, preliminary events can predicate incidents that occur at nuclear power generation facilities utilizing similar processes. By exploiting these patterns with the NRCA, this research aims to provide the nuclear industry with a tool to eliminate the occurrence of incidents that are decodable. And the NRCA will help to eliminate operational anomalies by providing operators an early warning system that identifies patterns in the parameters of nuclear power generation processes.

2.2 Nuclear Industry Impacts

The prevention of future civilian nuclear industrial incidents is a core motivation for this research because the lasting and detrimental effects of civilian nuclear incidents are a multifold issue. The most obvious of which is the long-term impacts of radionuclides that are released into the biosphere. The nuclear power generation industry in the United States has a safety record that is exempt from a major release of radionuclides to the environment. But every significant industrial release of radioactive material internationally has national nuclear industry impacts. In fact according to Pew Research, in a March 21st article after the Fukushima NPP accident occurred in Japan, US public support for the idea of national nuclear energy expansion fell from 47% to 39%. And US public opposition to nuclear energy expansion increased 5% (Pew Research Center, 2011).

Outside of the environmental impacts, there exists a real potential for significant economic impacts associated with nuclear incidents. And with US public opinion often driven by worldwide political and economic climates, the economic feasibility of nuclear energy proliferation should not be sacrificed by any lack of industry safety efforts. Therefore, in an effort to satisfy the dual criteria of supplying a positive impact to national nuclear interest and the future prevention of civilian nuclear industry incidents, this program utilizes the incident trends of the past to increase the nuclear security of the future.

And with the foundations of this research rooted in National nuclear interest through preemptive process awareness, the NRCA has been founded through understanding civilian nuclear accident causes and pathways.

2.3 Incident Pathways

When incidents occur within industrial processes that share similar operational techniques and equipment, the incidents often share common pathways (Lee and McCormick, 2011). This can be seen in the initiating event and subsequent event tree analysis (ETA) of a Probability Risk Assessment (PRA). PRA's make use of an ETA to explore the probable outcomes of an incident following a specific initiating event. And the Nuclear Regulatory Commission (NRC) has committed a significant amount of resources since the late 1980s towards the areas of PRA and ETA with their Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) computer software.

PRA's support the idea that a specific operational trigger, by way of an initiating event, will precede a cascade of subsequent failure events with a certain probability. And because of the proven reliability with this approach with regards to accident progression, the NRCA shares this ideology. In its efforts to satisfy incident prevention, the NRCA utilizes a combination of incident pathway recognition and inference of failure probability. The NRCA recognizes incident pathways through the identification of process parameter patterns, and it heightens operator awareness through an inference of failure probabilities based on an incident pathway knowledge database. The NRCA's incident knowledge database has been formed through the primary and secondary process parameter values theorized from instances of civilian nuclear incidents and accidents. Ideally, the actual recorded process parameter data for these events would be verified data sets from nuclear power plants. But due to the proprietary and security obstacles that are attached to such data, theoretical parameter values have been drawn from accident reports for this program. The use of an authenticated data set from an actual NPP in operation today would serve to solidify the use of NRCA as an industry tool for corrective action.

2.4 Current Industry Monitoring Techniques

Current nuclear industry technologies used for monitoring within nuclear process plants are nonuniform and are not often used for short-term or even long-term incident analysis. According to the report “On-line Monitoring For Improving Performance of Nuclear Power Plants Part 2: Process And Component Condition Monitoring and Diagnostics” issued by the International Atomic Energy Agency (IAEA) in 2008, the initial efforts of NPP on-line monitoring (OLM) were focused on the calibration and response time of nuclear process instrumentation. The IAEA’s intention for their 2008 report was to outline the current and future industry efforts to make use of on-line monitoring for sensor and equipment validation/functionality monitoring. That is to say, the initial intention for OLM was to analyze a singular equipment sensor or a group of equipment sensors for a singular or group of processes. But OLM did not form with the intention of overall system monitoring and safety.

The majority of NPPs around the world still operate on systems that are a hybrid of analog and digital instrumentation/controls (Hashemian, 2011). This lack of digital autonomy in NPPs is a result of the many financial and technical challenges that arise during upgrades to a NPP’s critical systems. Due to the uniqueness of an operating environment that entails nuclear processes, the absolute need for a NPP's sustained operation, and its margin of capital to return of investment hinging on lifetime gigawatt-hours, NPP digital upgrades have lagged behind many comparable plant types. Part of the issues that arise with these types of hybrid operating systems is that analog sensor technology has been at the core of OLM's historical difficulty in monitoring complex patterns within complex systems.

OLM's historical difficulty with capturing patterns of disruption in the subsystems of a NPP lies at the ability of a digital interpretation of analog sensor data. Analog sensor data require a conversion step before being received by a digital monitoring system, such as a computer, and this conversion step can interfere with a monitoring system's ability to properly identify small operation anomalies. And with incident prediction, often, small operational anomalies are a prediction's sole foundation. Therefore, it is the responsibility of an incident monitoring system to operate in such a way that capturing these anomalies is possible without an abundance of false positives. This research strives to use this method of process and equipment conditional monitoring for total system awareness.

2.5 Industry Design Direction

While the industry uses probability risk assessments (PRA) for operator actions and facility planning for postfailure event actions, the current trends for process monitoring of facility equipment is typically directed towards preventative maintenance, sensor validation, and process disruption. This approach leaves a vacancy within the ability of the industry's operators to actively monitor the nuclear power generation facility as a singular system. Analyzing an entire system based on multiple parameter histories and values for the multiple subsystems can be seen in the condition monitoring of transformers as described in the 2012 Idaho National Laboratory Report (Lybeck et al., 2012). Obviously, capturing system wide trends within a transformer or other compact mechanical system is more easily accomplished than conducting a similar capture on a grand mechanical system, like that of a NPP. But this trend illustrates that scientists and engineers have seen the need for system wide analyses for the proper condition monitoring of mechanical systems.

An Event Tree Analysis (ETA) is one example of how sequential events within a collection of systems can result in a system wide failure. ETAs are an essential part of an operator's understanding of incident pathways. ETAs begin with a specific Initiating Event (IE). Once the specific IE has been established, a tree of possible events branches out from the IE. An ETA using the SAPHIRE software for the University of Utah's Nuclear Engineering Program's (UNEP) Training, Research, Isotopes, General Atomics (TRIGA) reactor is show in Figure 1. UNEP is leading the way towards research reactor safety and accountability by its implementation of software tools, such as DevonWay. These software tools coupled with the inherit safety built within the TRIGA reactors demonstrate the industry's awareness to move towards more computer based system monitoring.

2.6 Nonpower Plant Facilities

Nuclear reactors are not solely the tool of the power generation industry. Nuclear reactors are used within university type research facilities, as well. This includes national laboratories, hospitals, and universities of higher education. The TRIGA nuclear reactor is one of the more notable reactors in this class.

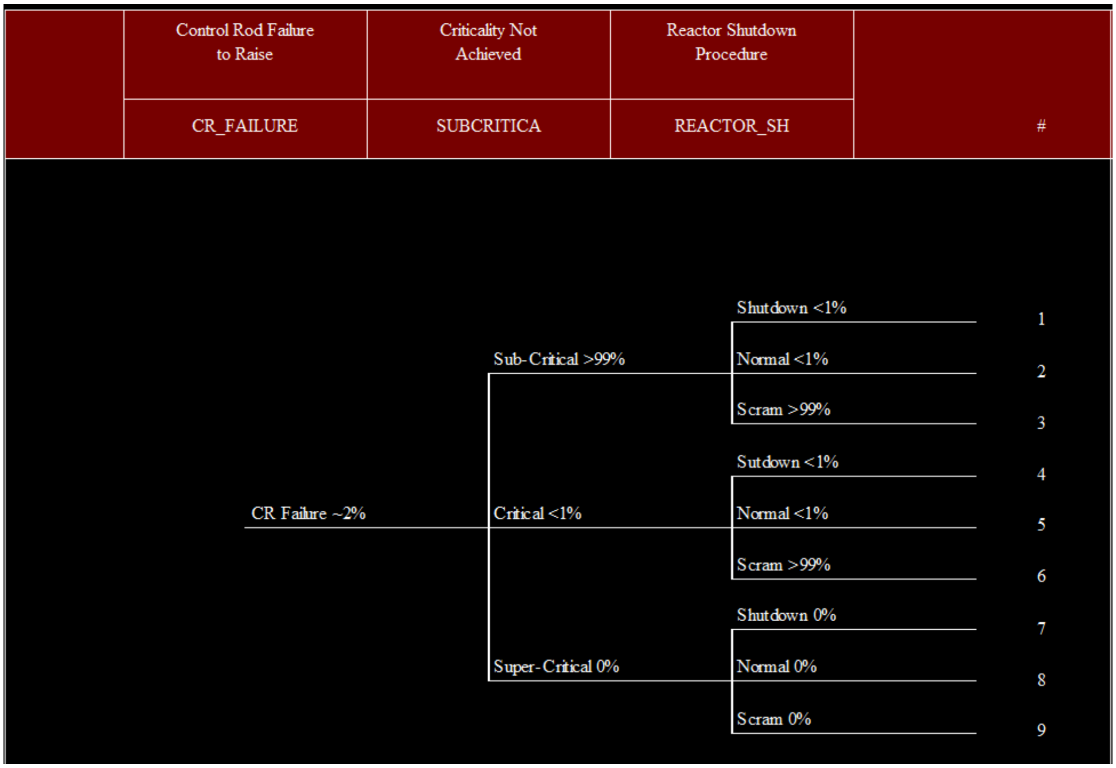

Figure 1. UNEP Control Rod (CR) ETA per SAPHIRE

Nuclear reactors, like those of the TRIGA, are of lower wattages, yet they are still subject to some failures that plague all nuclear facilities.

The NRC is the US regulating body that reviews the incidents that occur within US civilian nuclear facilities. And US Nuclear facilities issue incident reports to the NRC for internal review and release to the public. Research type reactors, like those of the TRIGA, are typically housed in facilities that were assembled during the 1950s and 1960s. This period in history is shared among most US nuclear reactor facilities in that the majority of research and power generation reactors were constructed, or constructed from reactor designs, during this time. The graph shown in Figure 2 illustrates the divergence of research reactor incidents that have occurred in the US from 1999–2012 as classified by this study (US.NRC, 2013).

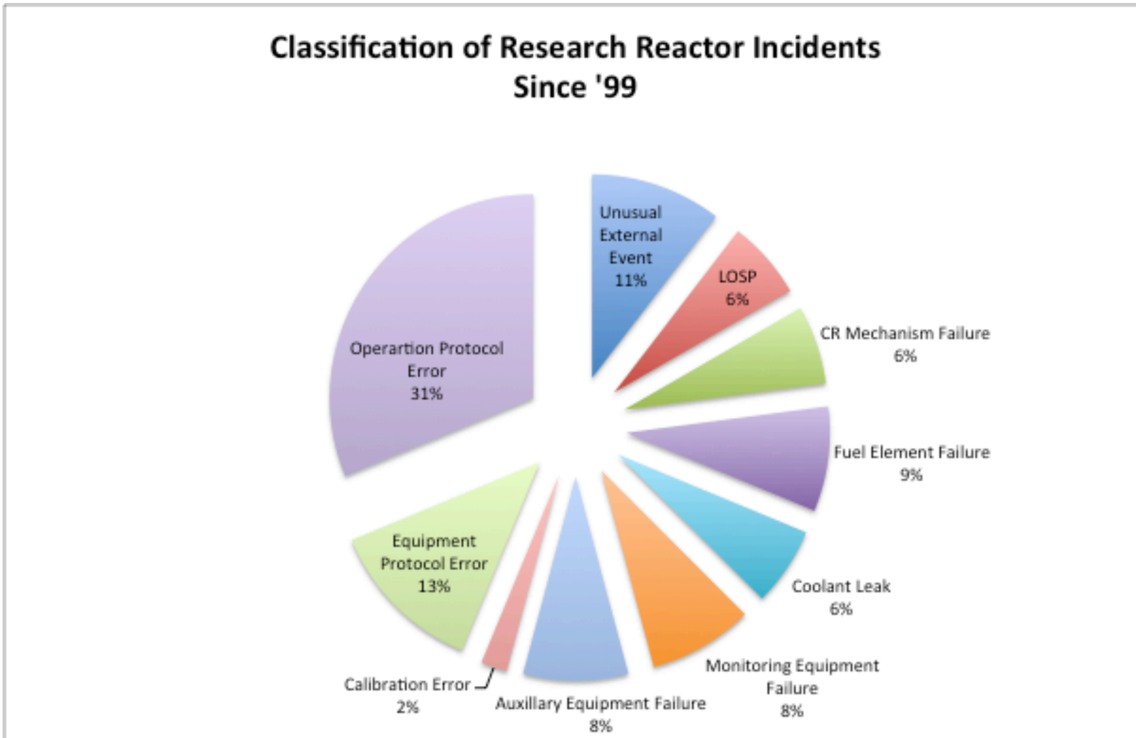

Figure 2. Research Reactor Incidents

Furthermore, due to high cost and technical challenges attached to upgrading nuclear reactor facilities, research reactors have lagged further behind their counterparts in the power generation industry with regards to sophisticated safety and monitoring systems.

Figure 2 illustrates that over 80% of the incidents that have occurred in research reactor facilities can be attributed to mechanical or protocol failures. By harnessing the information that can be found in the parameter values associated with mechanical failures, with the PRA information to prevent equipment and operational protocol errors, the NRCA program is poised to fill this current safety and monitoring gap within nonpower generation and power generation facilities alike.

CHAPTER 3

NUCLEAR POWER GENERATION

3.1 Nuclear Power Generation Facilities

There are many types of nuclear power generation facilities that are currently in use throughout the international nuclear industry. High power channel reactors (RMBK), Canadian Deuterium Uranium reactors (CANDU), and light water reactors (LWR) are the most commonly utilized types of nuclear reactor technologies. Countries of the former Soviet Union utilize RMBK technology. The uniqueness of the RMBK technology is its use of a graphite moderator, instead of a water moderator like most civilian nuclear reactor designs. CANDU technology originated and is the only type of civilian nuclear power reactor that is used by Canada. CANDU reactors are unique in their ability to make use of natural uranium without the addition of fissile isotopes in their fuel. These types of discrete individual differences between the nuclear reactors around the world are not of significance to this research because the NRCA program focuses on the system parameters, not the specific uniqueness of the parameters themselves. Furthermore, the nuclear power generation industry within the US is dedicated to the use of the LWR designs; LWR technology is the current focus of the NRCA. Boiling water reactors (BWR) and pressurized water reactors (PWR) are the two generic classifications and most utilized LWR designs. The following sections 3.1.1 and 3.1.2 discuss these two LWR designs in more detail.

Figure 3 illustrates a sampling of the most significant safety values and piping circuits within a nuclear power generation facility. These key safety devices and piping systems are of noteworthy importance to this study, because the monitoring system that is proposed here has considerable interaction with these components. Therefore, Figure 3 provides a valuable visual reference to the examples given in Chapters 5 and 6.

3.1.1 Boiling Water Reactors (BWRs)

Boiling water reactors (BWR) comprise 24 of the 62 NPPs that are currently operational within the US. Because the internal incident initiating events are nearly identical between these two types of LWRs, the major failure concerns to a BWR are very nearly identical to that of a PWR (Haskin et al., 2002).

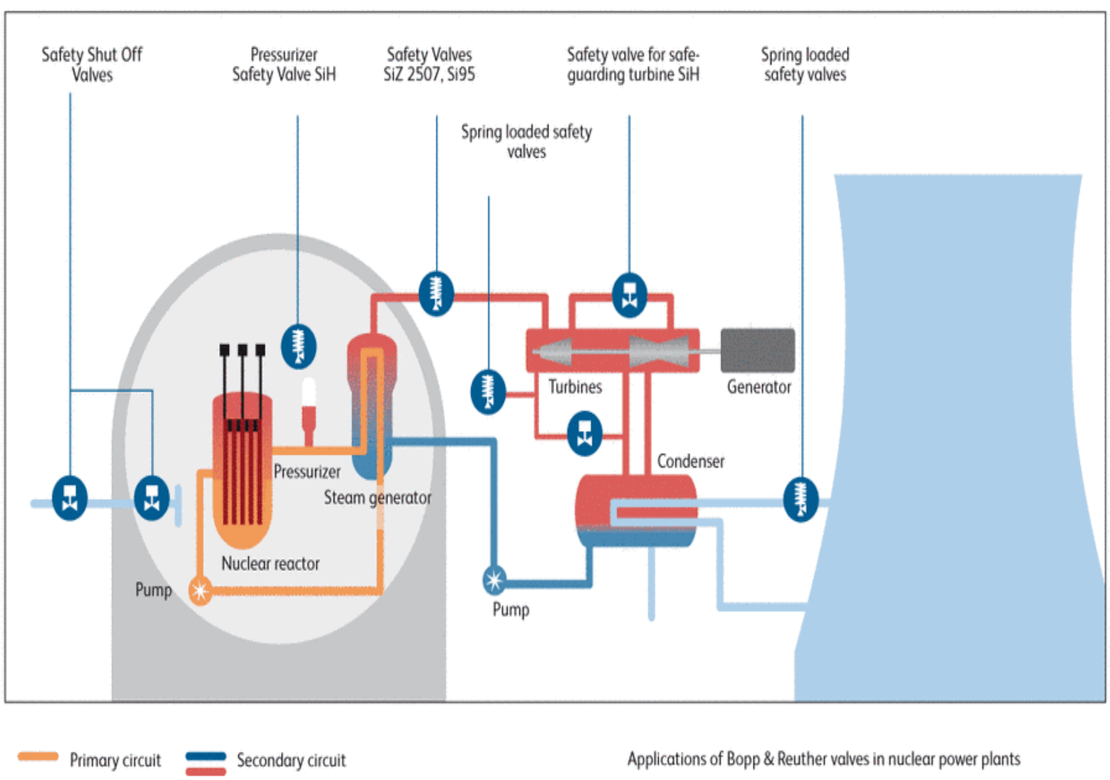

Figure 3. Nuclear Reactor Systems (Boop and Reuther, 2014)

The significance of the differences between a BWR and a PWR NPP as far as the Nuclear Reactor Condition Analyzer (NRCA) is concerned is in the parameters monitored by the NRCA. Since the primary concerns to any NPP are fluctuations or failures with its power conversion system (PCS), a NPP's coolant and steam systems' parameters are of most interest to a preventative monitoring system. Therefore, understanding the subtle variations in the operation of a BWR are crucial to the NRCA's ability to detect internal plant disruptions that may lead to a overall system failure.

One of the most significant system parameters unique to a BWR is the operational allowance to boil its coolant and therefore its moderator. The significance of boiling coolant in the reactor results in a reactivity control. This reactivity control comes by the way of steam bubbles that form in the BWR's coolant/moderator with increasing thermal energy. These steam bubbles in the moderator produces a negative void coefficient, and a negative void coefficient is constructed by the BWR by reducing the moderator's ability to thermalize neutrons. This reactivity control of thermal neutrons available in a BWR's moderator has a second unique effect of on the parameters of a BWR, as compared to a PWR.

3.1.2 Pressurized Water Reactors (PWRs)

Pressurized water reactors (PWR) comprise 38 of the 62 NPPs that are currently operational within the US. And as previously stated in section 3.1.1, the internal incident initiating events are nearly identical between these two types of LWRs; the major failure concerns to a PWR are very nearly identical to that of a BWR (Haskin et al., 2002). And the significance of the differences between a PWR and a BWR NPP as far as the NRCA is concerned is in the parameters monitored by the NRCA.

The internal reactor pressure that a PWR operates with separates it from a BWR. This more than two-fold increase in the pressure of a PWR creates challenges that are unique to a PWR. One such challenge is a loss of reactor system pressure. And although a loss of reactor pressure is not innocuous to a BWR, it is especially detrimental to the operations of a PWR. Due to the lack of coolant boiling during the normal operation of a PWR, when any substantial loss of pressure is suffered within the PWR's closed reactor system, the coolant being at such high temperature during full power conditions can flash boil creating a high loss to the heat sink or heat removing capacity of the system. Unlike the BWR design of a negative void coefficient as a result of the increased, albeit gradual increase, of steam bubbles within its boiling coolant, a PWR is not designed for steam bubbles within its coolant. Therefore, this condition creates unpredictability with the PWR system. The NRCA attempts to capture and interpret the small anomalies that precede this type of unpredictability.

3.1.3 LWR Coolant and Steam Systems

A LWR is comprised of many systems that achieve the coolant and steam cycles, and these systems require symphonic operation for the success of a LWR. This requirement is especially unique with NPP operations as compared with coal or natural gas fired power plants due to the strong, and during times of overexposure, volatile relationship between temperature and neutronics. Therefore, the synchronicity of the primary and secondary systems of a NPP predicates the harmonious operation of that NPP. All engineered systems strive for intersystem harmony. But one of the challenges that exist with ensuring that an engineered system remains within the bounds of harmonious operation is catching the small glitches that might disrupt this intersystem synchronicity.

Both US reactor types utilize light water as a neutron moderator and system coolant. However, some of the main systems that pertain to the coolant circuit of a LWR vary with the application of a BWR as compared to a PWR. Although this work is not charged with exploring the subtleties between US LWR types, the main differences between these two US LWRs is important to illustrate for the purposes of operational, and therefore NRCA, variances. The following describes the differences between the two LWRs that are worth mentioning to this study:

A BWR's coolant is allowed to boil.

A BWR's coolant is not within a closed circuit (the coolant supplies the steam cycle as well).

A BWR operates at a lower temperature and pressure than a PWR.

A PWR's coolant is superheated and not allowed to boil.

A PWR's coolant is contained within a closed circuit (the coolant does not supply the steam).

A PWR operates at a temperature and pressure that is two to three time greater than that of a BWR.

A PWR's greater coolant temperature, greater coolant pressure, and closed coolant circuits combines for a more complex operation as compared to that of a BWR. But due to the elimination of radioactive coolant entering the steam cycle, the risk of significant radioactive coolant escaping the reactor is less for the PWR than that of the

BWR. Though for the purposes of this research, the only implication that is felt by these differences is in the interpretation of operational margins and predictability.

3.1.4 LWR System Parameters

As with the operational uniqueness to a BWR, as compared to that of a PWR, the observed parameters of a BWR that would be observed by the NRCA will vary from a PWR to some degree. But the essence of the NRCA's capacity to observe the parameters of each US LWR remains, in general terms, unchanged. The LWR system parameters that are the current focus of this research pertain to a reactors cooling and steam generation capabilities. The justification for this focus resides in the strong relationship between a nuclear reactor's core temperature, its cooling capacity, and the heat sink available via its steam turbine/condenser system. This relationship results in an intersystem dependence that can be exploited by the NRCA to identify operational anomalies that might lead to failures in one or more systems of a reactor.

When it comes to the identifiable BWR and PWR anomalies that are of significance to a reactor's safe operation, Table 1 illustrates the general differences between these two LWR types. Since there is not a substantial benefit to this research in encompassing every aspect to each LWR's complexity, Table 1 is a simplified list of certain operational separations between BWRs and PWRs.

The information contained within Table 1 serves as a baseline for the selection of LWR parameters that are significant in this work. The resulting parameters that are of interest here were chosen to present a hypothetical model that displays the ability of the NRCA to harness a resident intersystem dependence.

Table 1. LWR Operating State

Operating State	Boiling Water Reactor (BWR)	Pressurized Water Reactor (PWR)
Overpressure	Depressurization Valve System (DPVS) relieves reactor overpressures directly into a secondary containment with a dry well and wet well or torus.	Pressure Relief Valve (PRV) relieves steam system overpressures from a pressurizer into a secondary containment quench tank.
Steam Generation	Steam is a product of the coolant being boiled directly by the core.	Steam is a product of the coolant being routed in a closed loop within a steam generation coil.
Power Transients	Increased core temperature result in pressure initiated transients due to the increase of steam bubbles in moderator.	Increased core temperature results in thermal initiated transients due to the increase of temperature in moderator.

That being said, the simplified list of PWR parameters given below will be used hence forth in this work for illustrating the benefits of the NRCA in nuclear power generation facilities:

(1) Main Feedwater Pump Valve (MFPV) – The MFPV serves as a throttle for the flow rate of condensed water entering the steam generator. This valve serves an important intersystem connection because it functions as one of the controls to the amount of heat sink that is available at the steam generator via the MFPV. Also, the steam flow rate entering the

turbines is partially dictated by the amount of condensed water entering the steam generators via the MFPV.

(2) Turbine Main Bypass Position (TMB) – The TMB serves as a throttle for the flow rate of steam entering the turbine. This valve serves as an important overall system condition indicator because it directs the energy produced by the reactor in the form of steam. If the TMB is in bypass mode, the energy produced by the reactor is not being directed towards the turbine; therefore, the reactors product is not being utilized.

(3) Reactor Vessel Pressure (RVP) – The RVP is a critical parameter to the safe and effective operation of nuclear power reactor. Given the direct relationship between the temperature and pressure of a fluid, the cooling and neutron moderating capacity of light water in BWR and PWR pressure vessels is dependent on the pressure within those vessels.

(4) Pressure Relief Valve (PRV) – The position of the PRV, whether open, venting pressure, or closed, holding pressure, is a direct indicator that can correlate the reported RVP via the control panel.

(5) Reactor Inlet Coolant Temperature (RICT) – The RICT is a critical parameter to the safe and effective operation of a nuclear power reactor. Due to the RICT's heavy dependence upon the RVP and the system's available heat sink, the intersystem health of a reactor can be directly interpreted by the RICT.

(6) Reactor Outlet Coolant Temperature (ROCT) – The ROCT, and the RICT/ROCT ratio, is a direct indicator that can correlate the reported

condition of the heat generated in the reactor core via a core thermocouple. The quality, or dryness, of the steam produced above the core in a BWR, or in the steam generators of a PWR, can be inferred form the ROCT, also.

(7) Reactor Coolant Recirculation Rate (RCRR) – The RCRR is a critical parameter, and its proper rate is imperative to the safe and effective operation of a nuclear power reactor. The directing of the excess heat from the reactor core is determined, in part, by the rate of coolant being passed through the cooling channels surrounding the nuclear fuel.

(8) Steam Flow Rate (SFR) – The SFR is a reactor output parameter that provides an operational view of the reactor's proficiency. The SFR can provide a correlation between the ROCT, PRV, MFPV, TMB, and turbine output

(9) Reactor Power (RP) – The RP is a static parameter for this study. But a nuclear reactor's level of power is a crucial element in the analysis of its operational state. Whether the reactor is at startup, full, standby, or shutdown power determines the nominal parameter values for all systems directly related to its operation. Therefore, to ensure the correct interpretation of this work's place in those reactor power states, the reactor power remained at what would be considered a steady state full power condition.

CHAPTER 4

KERNEL-WEIGHTING

4.1 Fundamentals

Kernel weighting algorithms have been thoroughly investigated for their use in process signal monitoring with good results (Henkel, 2011). This work expands on the use of kernel weighting into the field multiple process monitoring. By using modified Gaussian and Logistic KW to apply a process analysis, a program has been developed to compare historical nuclear reactor parameters values that resulted in operational failure and real-time nuclear power plant process parameter values for the safety monitoring of those systems. The inception for this program is described by the article titled “Learning from Accidents: A Conceptual Model for Automated Monitoring of Nuclear Power Plants under Corrective Management” for the International Journal of Monitoring and Surveillance Technologies Research (McCoy et al., 2013).

The ability of this program to apply an appropriate numerical scheme is accomplished by normalizing the nuclear reactor’s individual system parameter values with the KW. Afterwards, the appropriate amount of numerical weight can be calculated in the form of process condition values. The NRCA program, in order to provide an operational failure threshold, compares the summation of the real-time process parameter values to sets of nominal, or incident, process values. The sets of historical incident values are intended to be recorded from sets of operational parameter values for instances of system failure.

In order to confirm the validity of multiple sensor signals from the same process, auto-associative kernel regression (AAKR) algorithms are used in some depth for the monitoring of multiple sensor signals from the same process origin (Hines and Garvey, 2006). The Nuclear Regulatory Commission, in collaboration with the University of Tennessee, has published a number of technical reviews on the use of AAKR algorithms for the on-line monitoring of system performance (Hines et al., 2008). This research has broken down the individual components of AAKR type algorithms and modified their use for the application of a nuclear reactor condition analysis program.

The typical approach to detecting multiple sensor signals from a singular process utilizes the combination of distance and kernel weighting functions to detect the amount of variance among signals of a similar process origin. For example, an AAKR algorithm could be employed to measure the level of agreement between three parallel sensors residing within the flow of a nuclear reactor's coolant conduit. Signal identification and cohesion from the three sensors are the objectives of a process signal analysis. But this type of analysis is not compatible when the system under question contains multiple parameters with differing values of magnitude and differing units, like the systems of a nuclear power generation facility. In this case, a modified use of the distance and kernel weighting functions are used to normalize the inputs across a single scale.

4.2 Distance Functions

In order to apply a numerical scheme, KW must first compute the numerical gap between two values, or in other words, the displacement between two values. Because of the emphasis on operational variances, this numerical gap is crucial to this work's application of KW. The applications of numerical distance in KW algorithms, typically, utilize a two-norm distance function. Euclidean-norm distance functions have been utilized to normalize multiple data sets from the same process. Therefore, the gap for each numerical set is not as crucial as the Euclidean distance for all sets. An example of multiple signals for a single process that would result in Euclidean computed data sets can be seen in a series of 3 thermocouples in a conduit as shown in Figure 4.

The distances, (d_{TA}), (d_{TB}), and (d_{TC}), illustrate the signal variations between the three thermocouples "real-time", (R_{ij}), and exemplar, (E_{ij}), values. These distances are typically used in an auto-association kernel regression algorithm to predict the next series of thermocouple output values.

$$d_{TA} = \sqrt[2]{(E_{1A} - R_{1A})^2 + (E_{2A} - R_{2A})^2} \quad (1)$$

$$d_{TB} = \sqrt[2]{(E_{1B} - R_{1B})^2 + (E_{2B} - R_{2B})^2} \quad (2)$$

$$d_{TC} = \sqrt[2]{(E_{1C} - R_{1C})^2 + (E_{2C} - R_{2C})^2} \quad (3)$$

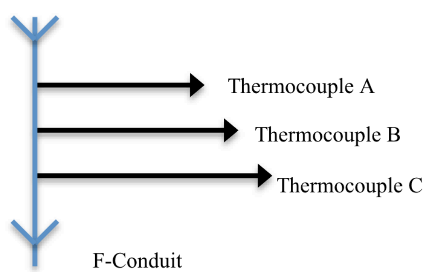

Figure 4. Multiple Signal Single Process

That being said, these distances are coupled with kernel weighting for sensor validation in multiple signal analysis algorithms.

When it comes to the NRCA, the application multiple signal analysis is multifaceted. The thermocouple example above provides an insight into a singular process with multiple signals, either for redundancy, validation, and/or calibration purposes. But with an analysis by the NRCA, there are multiple systems with multiple signals under analysis. And the analysis is not directed towards sensor validation. It is directed towards systems awareness and optimization. Therefore, the numerical distance comparison, as utilized by the NRCA, is an instantaneous comparison of one real-time to nominal, and/or to a historical incident, parameter per subsystem. Then each subsystem comparison product is used twice: once in a weighted comparison with that subsystem's set static parameter and once combined with system-wide parameters in a probability analysis. The typically used two-norm distance equation would negate the significance of each instantaneously reported signal. Therefore, the NRCA utilizes a one-norm distance equation as given below:

$$d_{ij} = x_{ij} - X_{ij} \tag{4}$$

where (d_{ij}) is the distance of each individual present, or real-time, system parameter value, (x_{ij}), from its corresponding stored exemplar, or static, parameter value, (X_{ij}). This distance value is important to the operations of the nuclear reactor, because it represents the displacement of the real-time process parameter value, (x_{ij}), from a nominal, or incident parameter value of the exemplar process parameter, (X_{ij}).

An example of this would be the (d_{ict1}) distance of the "real-time" reactor inlet coolant temperature value (x_{ict1}) from the nominal reactor inlet coolant temperature value (X_{ict1}). The nominal values (X_{ij}) for each of the individual system parameters are preset values that are properly suited for each nuclear power plant or other nuclear facility at a given time and condition for that specific plant. For example, the nominal reactor inlet coolant temperature value (X_{ict1}) for "NPP-A" when it is operating at 50% power would be a preset value different from that for "NPP-A" operating at 60% power, or "NPP-B" operating at 50% power. Boolean logic functions for each of the plant's operational conditions and stages could be added to this program to accommodate the implementation of nominal parameter sets at given operating power intervals.

4.3 Kernel Equations

The distance value (d_{ij}) is passed to either a modified version of the Gaussian kernel (G_{ij}) (5) or Logistic kernel (L_{ij}) (6) equations. Where (d_{ij}) is the distance of each individual current system parameter value (x_{ij}) from its corresponding stored exemplar parameter value (X_{ij}), and (h) is a bandwidth value that changes the scale of the response from each function with regards to the changes in magnitude of the (d_{ij}) distance value. The bandwidth value, (h), in the typical application of kernel weighting algorithms is a dynamic variable that changes with each a priori update of the algorithm. That is to say that the typical application of a bandwidth value adjusts with each new batch of historical data used by the kernel equations to make predictions. In this study, the bandwidth value, (h), is a predetermined parameter that is statically set according to each reactor's operational thresholds, thresholds that are determined by the historical operations specific to each reactor. This allows the users of the NRCA program to determine with what magnitude the kernel weighting equations represents the operational deviations in their system.

$$G_{ij} = \frac{1}{\sqrt{2\pi h^2}} e^{d_{ij}^2 / 2h^2} \quad (5)$$

$$L_{ij} = e^{(d_{ij})h} + 2 + \ e^{(-d_{ij})h} \quad (6)$$

Typically, the Gaussian kernel (G_{ij}) equation (5) utilizes a negative value for the distance (d_{ij}). But with the modified version for this program, the (G_{ij}) distance is a positive value. This modified version of the Gaussian kernel slope agrees with the modified version of the Logistic kernel (L_{ij}) equation (6) that provides a positive slope, as well. The reason that this program utilizes two separate kernel-weighting functions, and two separate bandwidth values, can be seen in their response to equal changes to their (d_{ij}) distance value inputs in Table 2. The over two-fold difference in the bandwidth value, (h), as shown in Table 2, is present to illustrate how this value is treated within the two different kernel types. The values 0.099 for the Gaussian, and 9.00 for the Logistic were chosen purely for this study's desired effect on the hypothetical parameter distances as given to the kernels.

The significance of the difference between the response of the modified Gaussian kernel (G_{ij}) equation and the modified Logistic kernel (L_{ij}) equation to magnitude changes in the value of the distance (d_{ij}) between the exemplar and present parameter value is utilized by the program in two ways: the modified Logistic kernel responds to greater changes in (d_{ij}) with less positive weight and both KW equations respond differently to changes in the bandwidth (*h*) value. This adds a level of flexibility to the program in its application of process analysis that strives for the elasticity required when dealing with the varied processes of an entire nuclear reactor system.

Table 2. Kernel Equation Responses

Distance (d_{ij})	**Gaussian (G_{ij}) h = 0.099**	**Logistic (L_{ij}) h = 9.00**
-0.1	6.7 e0	4.0 e0
-0.2	3.1 e1	4.9 e0
-0.3	4.0 e2	8.2 e0
-0.4	1.4 e4	1.7 e1
-0.5	1.4 e6	3.7 e1
0	4.0 e0	4.0 e0
.5	1.4 e6	9.2 e1
1	5.7 e22	8.1 e3
1.5	2.8 e50	7.2 e5
2	1.7 e89	6.6 e7
2.5	1.2 e139	5.9 e9

CHAPTER 5

NUCLEAR REACTOR CONDITION ANALYZER PROGRAM

5.1 Basics

The NRCA program operates with an algorithm that includes one-norm distances, Gaussian kernel weights, Logistic kernel weights, and comparison equations. This series of steps are performed using exemplar and flow process parameter data. The following is a simplified example of how the NRCA would perform a comparative analysis:

For "NPP-A" operating at 90% power, the nominal reactor inlet coolant temperature value (X_{ict90}) is 270°C, and its nominal reactor coolant flow-rate value (X_{ifr90}) is 13,700 Kg/s. For "NPP-A" at a given instance of time operating at 90% power, the "real-time" reactor inlet coolant temperature value (x_{ict1}) at 90% power is 276°C, and its "real-time" reactor coolant flow-rate value (x_{ifr1}) is 13,675 Kg/s.

The distances (d_{ict1}) and (d_{ifr1}) are 6°C and -25 Kg/s, respectively. These distances, (d_{ict1}) and (d_{ifr1}), are used in a first line NRCA analysis. Based on the operational status of the nuclear reactor, in this case "NPP-A", exemplar (X_{ictj}) and (X_{ifrj}) would be stored in a matrix for all "NPP-A" operational parameters at each state of normal reactor operation. The "real-time" (x_{ict1}) and (x_{ifr1}) values, along with all other "NPP-A" operational values, would be queried against every value in the exemplar parameter data matrix. This simple query by the NRCA is a validation of all systems that are required for the current state of reactor operation. Table 3 illustrates an abbreviated example of an exemplar parameter matrix for “NPP-A.”

Given the exemplar values in Table 3, the “real-time” reactor inlet coolant temperature value (x_{ict1}) at 90% power of 276°C and its “real-time” reactor coolant flow-rate value (x_{ifr1}) of 13,675 Kg/s correspond to 93% and less than 80% reactor power, respectively. This accounts for the distances (d_{ict1}) and (d_{ifr1}) of 6°C and -25 Kg/s, respectively. The NRCA program, through its first line of analysis, will recognize the mismatch between the “real-time” and exemplar parameter values of reactor power, inlet coolant temperature, and inlet coolant flow. The second line of analysis that the NRCA performs is a summation of the operational variations within the entire reactor system.

This summation acts as an operational penalty system for the reactor parameter values that are deviations from nominal and/or near incident conditions. The NRCA’s penalty system is driven by KW equations that supplement the distance equations from its first line analysis. The values of (d_{ict1}) and (d_{ifr1}) are passed to the modified Gaussian kernel and Logistic kernel weighting equations. In the above example for “NPP-A”, this would result in a (G_{ict1}) equal to a reactor inlet coolant temperature penalty value of 287.0 e10 and (G_{ifr1}) equal to a reactor inlet flow rate penalty value of 1.084 e271. If the historical penalty value database for “NPP-A” contains a value near 287.0 e10 x 1.084 e271 equal to 3.113 e281, the program will perform a predetermined output based on the known result of the before-mentioned conditions. The threshold value can be inferred through a variety of percentages of the historical penalty combinations or summations.

Table 3. "NPP-A" Exemplar Parameter Values

Reactor Power (%)	85	86	87	88	89	90	91	92	93
Inlet Coolant Temp. (°C)	260	262	264	266	268	270	272	274	276
Inlet Coolant Flow rate (Kg/s)	13690	13692	13694	13696	13698	13700	13702	13704	13706
Outlet Coolant Temp. (°C)	300	302	304	306	308	310	312	314	316
Outlet Coolant Flow rate (Kg/s)	13690	13692	13694	13696	13698	13700	13702	13704	13706
Reactor Vessel Pressure (Mpa)	15.0	15.1	15.2	15.3	15.4	15.5	15.6	15.7	15.8
Condensed Water Temp (°C)	70	72	74	76	78	80	82	84	86
Steam Flow rate (Kg/s)	1810	1812	1814	1816	1818	1820	1822	1824	1826

There would ideally be hundreds of historical scenarios within the historical parameter value database for the program to perform its queried comparative analysis. The comparative analysis portion of this program infers the condition of the entire system based on the summation of each real-time process parameter penalty and that resulting penalty value's comparison to a historical database of known operational penalty values at times of system failure or disruption.

5.2 Three Mile Island Example

The benefits to the implementation of the NRCA can be seen in its application to a historical civilian nuclear accident such as the Three Mile Island PWR Accident of 1979. The Three Mile Island Nuclear Power Plant Incident in 1979 was arguably the most influential nuclear accident in US history. An abbreviated sequence of the Three Mile Island 1979 events occurred as follows (Bertini, 1980):

On March 28, 1979, at 4:00am the following order of events began:

(1) The condensate pumps (CPs) that supplied secondary system water to the steam generator feedwater pumps (SGFPs) shut down without notice.

(2) Subsequently, the steam generator feedwater pumps (SGFPs) shut down.

(3) Due to the loss of feedwater to the steam generators (SGs), the steam turbines (STs) shut down because of the steam loss.

(4) Once the steam turbines (STs) shut down, the reactor automatically scrammed.

(5) Due to the lack of feedwater to cool the steam generators (SGs), the temperature and pressure of the steam generators (SGs) began to rise.

(6) The auxiliary steam generator feedwater pumps (ASGFPs) activated because of the lack of secondary coolant and elevated temperature in the steam generators (SGs).

(7) The auxiliary feedwater did not reach the steam generators (SGs) because the auxiliary steam generator feedwater valves (ASGFV) were closed.

(8) Subsequently, the pressure and temperature of the primary coolant system began to rise precipitously as a result of the loss of heat sink in the steam generators (SGs).

(9) The pressurizer relief valve (PRV) opened because of the significant pressure rise in the primary system.

(10) Due to the open pressurizer relief valve (PRV), the pressure in the primary coolant system lowered.

(11) The primary coolant's high-pressure injection pumps (HPIPs) were activated because of the loss of coolant resultant from the elevated temperature, pressure, and open pressure relief valve (PRV).

(12) The high-pressure injections pumps (HPIPs) increased the volume of coolant in the primary system as to completely fill the pressurizer.

(13) The reactor operator shut down the high pressure injection pumps (HPIPs).

(14) Since the pressurizer relief valve (PRV) was still open, the pressure in the primary system dropped below the saturation level of the primary coolant's temperature. This resulted in a flash boiling of the primary coolant and uncovering of the reactor core.

(15) The reactor operator manually opened the auxiliary steam generator feedwater valves (ASGFV).

(16) The temperature in the primary coolant system began to lower after the re-introduction of circulating secondary system feedwater, re-establishing the heat sink at the steam generators (SGs).

(17) The reactor operator manually re-activated the high-pressure injection pumps (HPIPs) to restore the necessary primary coolant level in the reactor.

(18) The pressurizer relief valve (PRV) drain tank ruptured due to the pressurizer relief valve remaining in the open position and the rise in coolant level into the drain tank.

(19) The primary system's main coolant pumps (MCPs) developed vibrations from their cavitation during the saturation event.

(20) The reactor operator shut down the main coolant pumps (MCPs) as a result of their vibrations.

(21) The temperature and pressure in the primary system began to rise because of the main coolant pumps (MCPs) being shut down.

(22) The reactor operator, noticing the pressurizer relief valve stuck in the open position, closed a pressurizer relief bypass valve (PRVBV).

(23) The pressure in the primary coolant system began to rise, collapsing the steam bubbles in the coolant system, thus eliminating the cavitation condition for the main coolant pumps (MCPs).

(24) The reactor operator manually relievied the high level of pressure that was building in the primary coolant system by opening the pressurizer relief valve bypass valve (PRVBV).

(25) The reactor operator restarted one of the main coolant pumps (MCPs) now that the pressure had been stabilized.

(26) The reactor operator continued to maintain the proper pressure in the primary coolant system through the manual operation of the pressurizer relief valve bypass valve (PRVBV) until the reactor was in a shutdown state.

Hypothetically, the NRCA can be applied to the Three Mile Island Nuclear Power Plant, and the events that occurred on March 28^{th}, 1979 can be re-examined. For a more direct view of the events and the hypothesized results that the NRCA determines during that time, Table 4 list a simplified list of the events and the NRCA results.

The hypothetical parameter values listed in Table 4 for the sequence of events that took place at the Three Mile Island NPP on March 28^{th}, 1979, were chosen based on a generic PWR. The exact parameter values for the event are not of significant importance to this study; the focus lies in the fluctuations and complexity of the values being hypothesized. Figures 5–8 illustrate how the operational variations during incident conditions can be graphically interpreted from nominal operations by one-norm distance equations and embellished by kernel weighting equations.

The x-axis of Figures 5–8 are labeled as iteration #. This iteration # is a generic representation of the data sampling intervals that are performed by a system to ascertain the condition of a process operating under that system's observation. Therefore, the iteration # as it pertains to the NRCA program is a representation of the process parameter sample data at any given instance of operational time, for instance, whether that time is set at millisecond or second reporting intervals. The advantages of this approach can be seen in the comparison of Figures 5 and 6 to Figures 7 and 8. Figures 5 and 6 represent multiple NPP parameters values on the same scale.

Table 4. NRCA Three Mile Island Example

Event #	Event Action	Hypothesized Incident Parameter Value (x_{ij})	Hypothesized Nominal Parameter Value (X_{ij})	One Norm Distance (d_{ij})	Gaussian Weight (G_{ij}) h = 0.099	Logistic Weight (L_{ij}) h = 9.000
1	CPs Failure[1]	0.0	1.0	-1.0	5.8e22	8.1e3
	Condensate Flow Loss[1]	100.0 (m^3/s)	400.0 (m^3/s)	-300 (m^3/s)	∞	∞
2	SGFPs Failure[1]	0.0	1.0	-1.0	5.8e22	8.1e3
3	STs Failure[1]	0.0	1.0	-1.0	5.8e22	8.1e3
	Steam Flow Loss	220.0 (m^3/s)	1820.0 (m^3/s)	-1200.0 (m^3/s)	∞	∞
4	Reactor Scrammed[1]	0.0	0.9	-0.9	3.6e18	3.3e3
5	SGs Temp Rise	295.0 (˚C)	285.0 (˚C)	10 (˚C)	∞	1.2e29
	SGs Pressure Rise	7.5 (MPa)	7.0 (MPa)	0.5 (MPa)	1.4e6	9.2e1
6	ASGFPs Activated[1]	1.0	0.0	1	5.8e22	8.1e3
7	ASGFV Closed[1]	0.0	1.0	-1.0	5.8e22	8.1e3
8	Primary Temp Rise	340.0 (˚C)	290.0 (˚C)	50.0 (˚C)	∞	2.7e195
	Primary Pressure Rise	16.0 (MPa)	15.0 (MPa)	1.0 (MPa)	5.8e22	8.1e3
9	PRV Opened[1]	1.0	0.0	1.0	5.8e22	8.1e3
10	Primary Pressure Drop	14.0 (MPa)	15.0 (MPa)	-1.0 (MPa)	5.8e22	8.1e3
11	HPIPs Activated[1]	1.0	0.0	1.0	5.8e22	8.1e3
12	Pressurizer Max Volume[2]	-	-	-	-	-
13	HPIPs Shutdown[1]	0.0	1.0	-1.0	5.8e22	8.1e3

Table 4. Continued

14	Primary Pressure Drop	9.0 (MPa)	15.0 (MPa)	-6.0 (MPa)	∞	2.8e23
15	ASGFV Opened[1]	1.0	0.0	1.0	5.8e22	8.1e3
16	Primary Temp Drop	290.0 (˚C)	290.0 (˚C)	0.0 (˚C)	4.0e0	4.0e0
17	HPIPs Activated[1]	1.0	0.0	1.0	5.8e22	8.1e3
18	Drain Tank Rupture[2]	-	-	-	-	-
19	MC Flow Loss (MCPs Vibrations)	11000 (m^3/s)	13700 (m^3/s)	-2700 (m^3/s)	∞	∞
20	MCPs Shutdown	0.0	1.0	-1.0	5.8e22	8.1e3
21	Primary Temp Rise	310.0 (˚C)	290.0 (˚C)	20.0 (˚C)	∞	1.5e78
	Primary Pressure Rise	10.0 (MPa)	15.0 (MPa)	-5.0 (MPa)	∞	3.5e9
22	PRVBV Closed[1]	0.0	1.0	-1.0	5.8e22	8.1e3
23	Primary Pressure Rise	13.0 (MPa)	15.0 (MPa)	-2.0 (MPa)	1.7e89	6.6e7
24	PRVBV Opened[1]	1.0	0.0	1.0	5.8e22	8.1e3
25	MCPs Restart[1]	1.0	1.0	0.0	4.0e0	4.0e0
26	Sequence Stabilized[2]	-	-	-	-	-

[1] Valves and pumps are hypothesized in the NRCA on a percentage scale.
[2] There are no parameter values assigned to this event number.

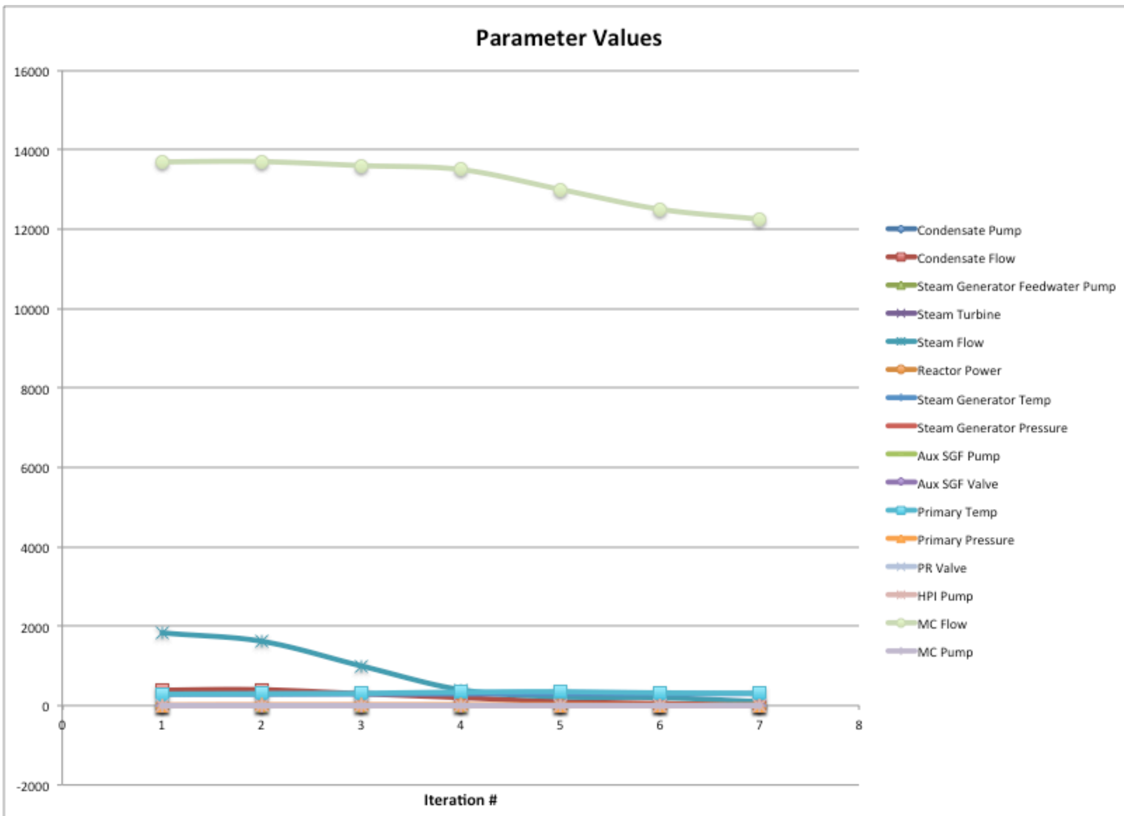

Figure 5. NRCA Parameter Values Example

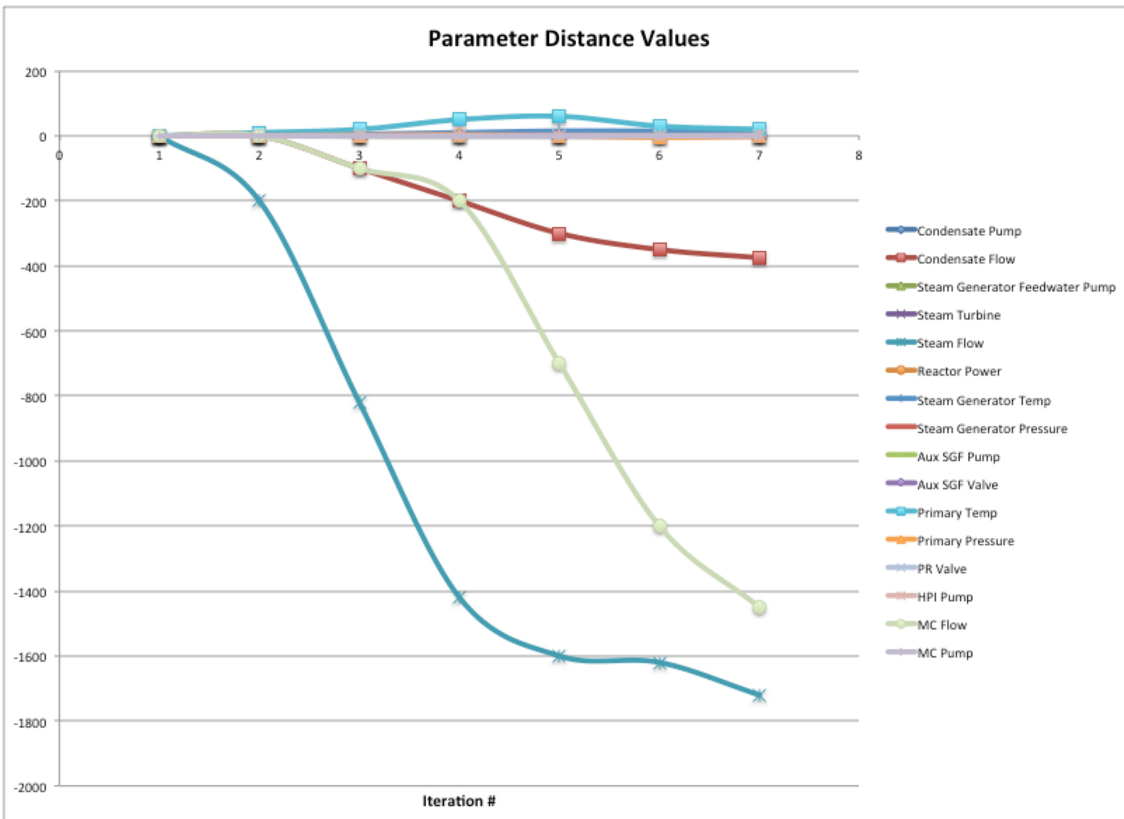

Figure 6. NRCA Distance Values Example

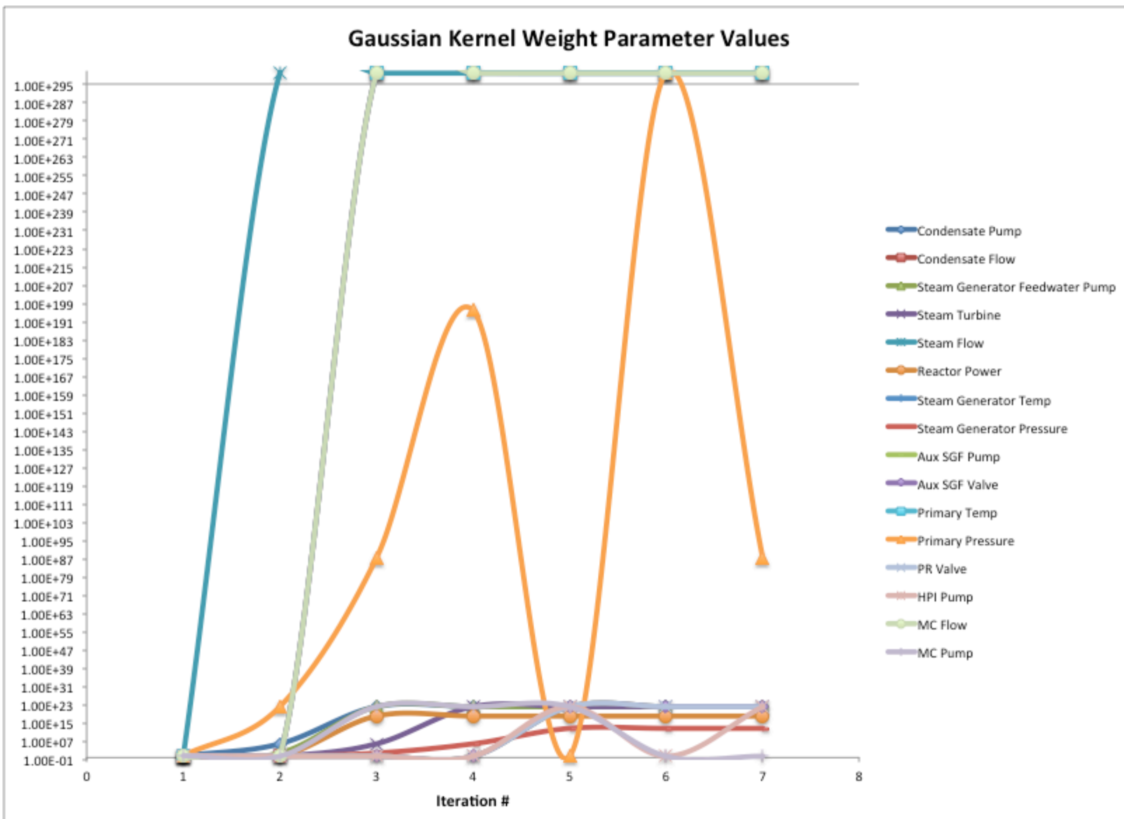

Figure 7. NRCA Gaussian KW Values Example

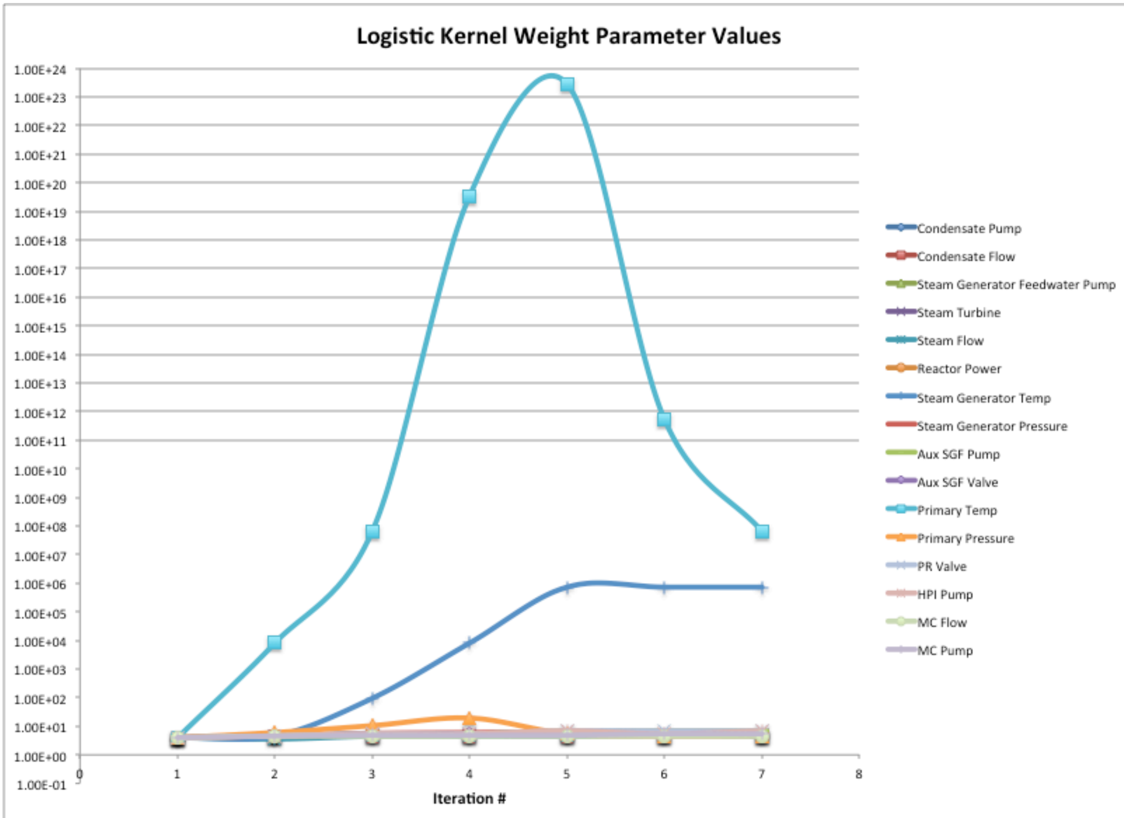

Figure 8. NRCA Logistic KW Values Example

This means that parameters like that of condensate pump power, which is reported here in decimal percent power, and that of coolant flow rate, which is reported here in cubic meters per second, are not fairly represented on the same graph due to the grand disparity between a value of 1.00 as compared to 13,700 m^3/s. Therefore, this type of biased parameter difference causes a hindrance with the monitoring of multiple processes, with multiple value scales in the same frame of reference. That is the purpose of the NRCA's normalization step that is accomplished by the KW equations, as can be seen in Figures 7 and 8. KW utilizes the one-norm distance values for each parameter analyzed and applies a numerical weight based on the value of that distance. The bandwidth variable within the KW equations allows for the varied application of weight for the equal values of distance.

CHAPTER 6

IMPLICATIONS

The results from the hypothetical Three Mile Island example in section 5.2 illustrate the embellishments that can be placed upon operational parameters through the use of one-norm and KW equations. These embellishments are useful in the safety and monitoring of multiple nuclear reactor processes when coupled with a normalization step and threshold values. The objective of the NRCA is to simplify this process as to ensure that a nuclear reactor is operating as efficiently and safely as possible. Towards this end, the NRCA utilizes KW equations as to ensure that all parameters related to the operation of a reactor are normalized to a related scale and analyzed together. Figures 9–12 are of the hypothetical condensate pump and flow parameter values from the Three Mile Island example in section 5.2. The dashed lines on Figures 9 and 10 represent a nominal threshold that might be assigned to the condensate pump and flow parameters for a nuclear reactor. This monitoring disparity increases considerably with each parameter addition, illustrating the challenges to monitoring multiple parameter values that are separated by significant differences in magnitude and units of measure in one frame of reference.

The dashed lines on Figures 11 and 12 represent a nominal threshold that can be applied to all parameter values that have been processed by the NRCA algorithm.

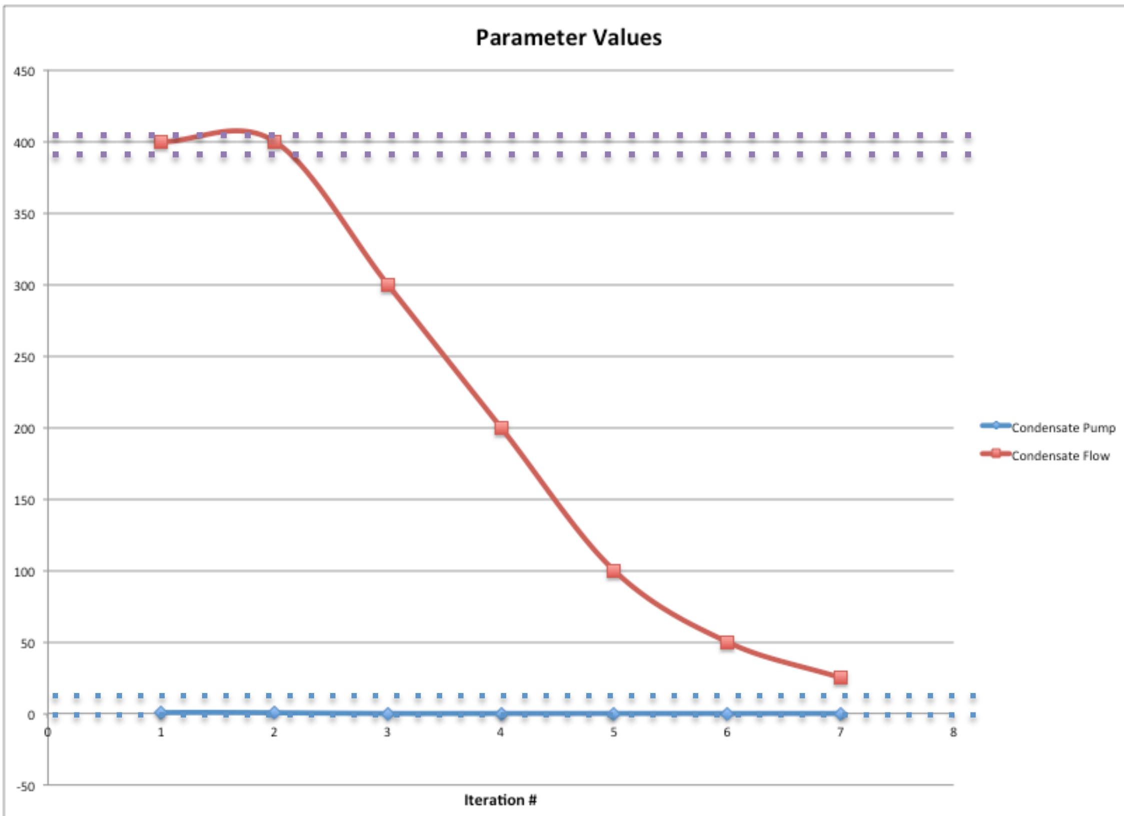

Figure 9. Parameter Value Comparison

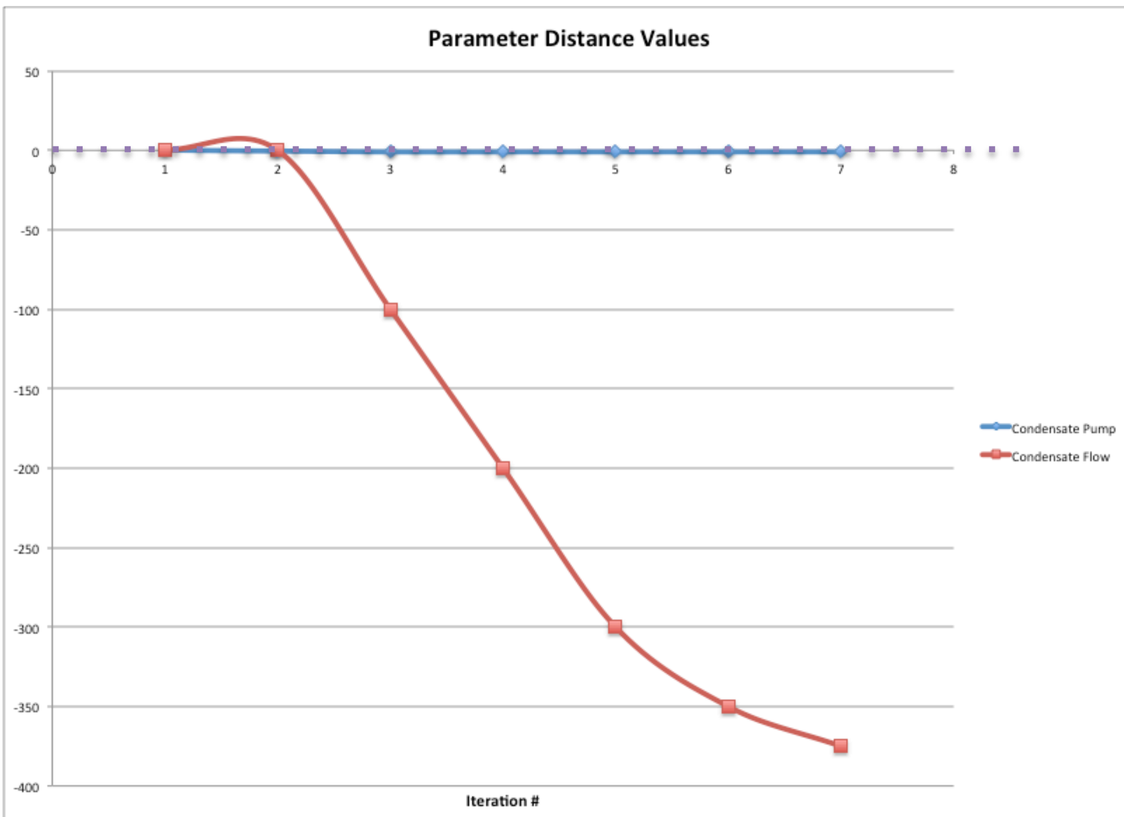

Figure 10. Distance Value Comparison

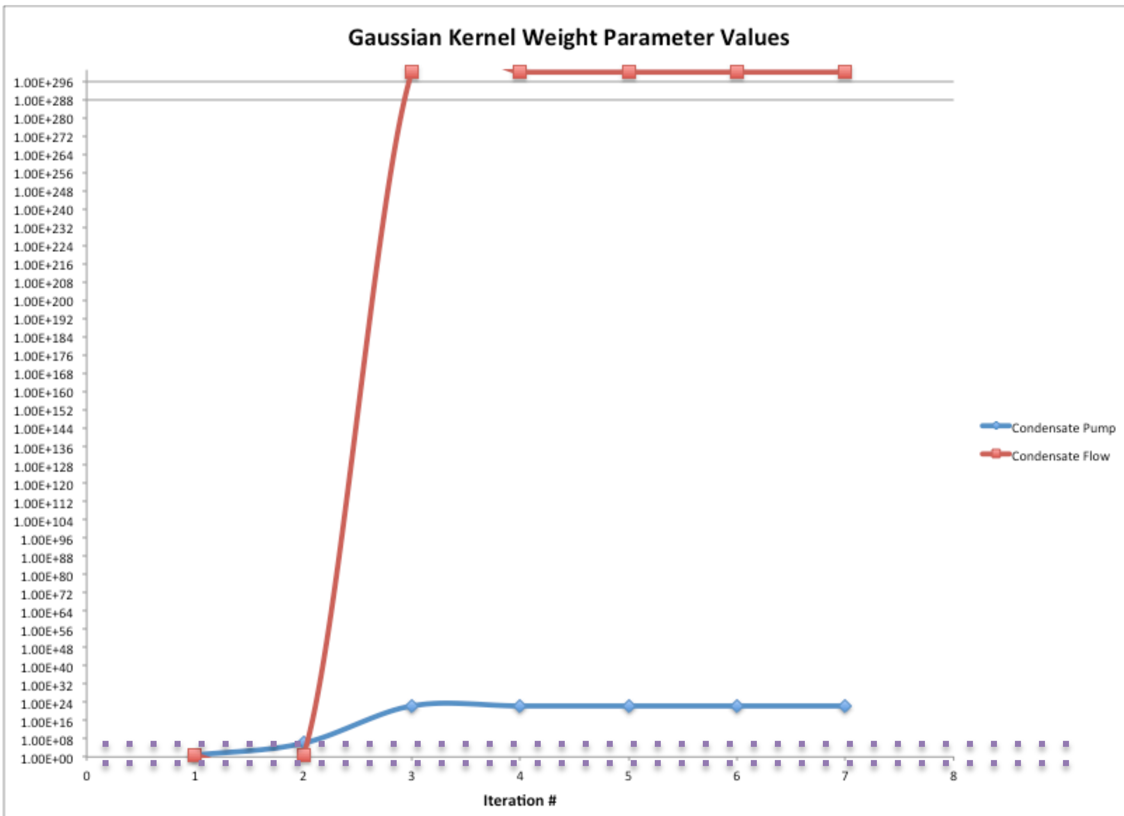

Figure 11. Gaussian Kernel Value Comparison

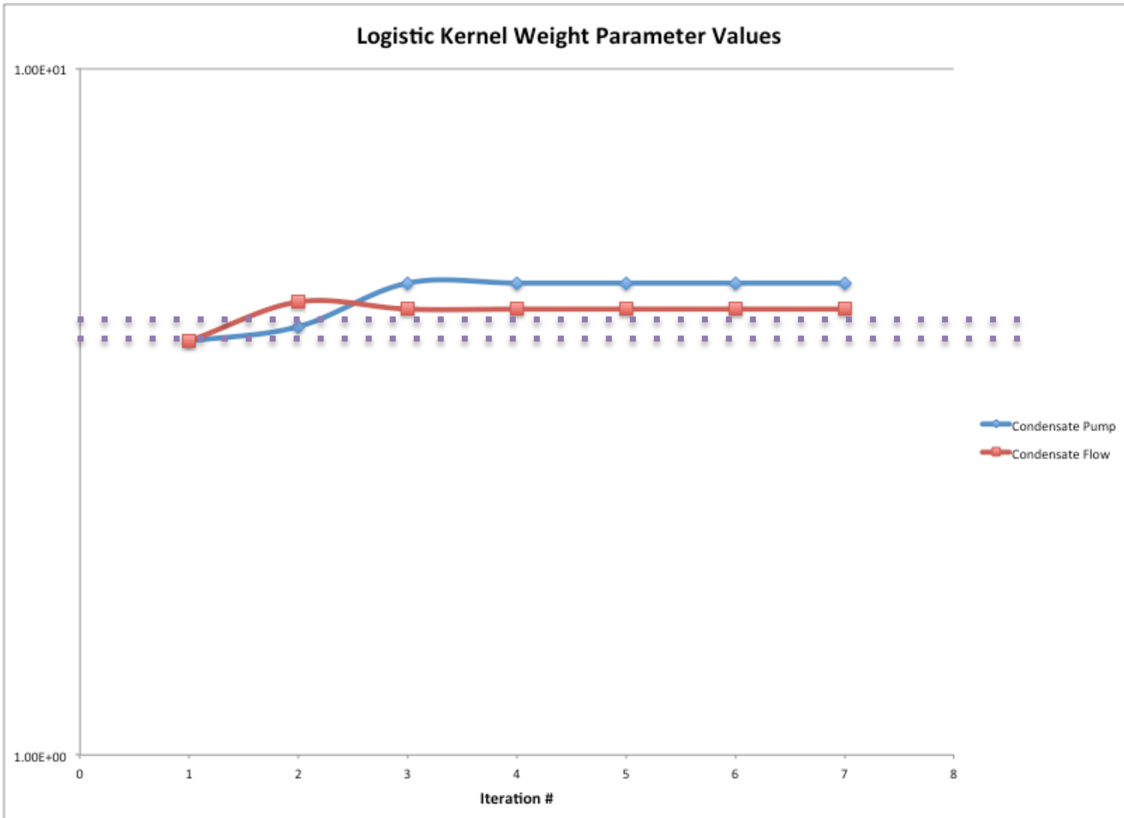

Figure 12. Logistic Kernel Value Comparison

Since every reactor parameter value that is passed through the NRCA algorithm is normalized to a single baseline value for nominal conditions, reactor operation thresholds can be set according to a collection of processes, in the place of individual process threshold, and each process can be made to achieve the threshold at differing rates with the adjustment of the NRCA's KW equations' bandwidth value.

6.1 Probability Risk Assessment Coupling

The NRCA's ability to combine multiple sensor signals in one frame of reference allows the program to exploit intersystem dependencies within nuclear reactor operations. This exploitation can be combined with a PRA for an effective incident prediction tool. Table 5 illustrates the type of condition combinations that would lead to a failure event within nuclear reactor. A similar sequence of events to that of Table 5 occurred during the Three Mile Island Accident as described in section 5.2.

PRAs outline event sequences that could lead to operational circumstances of failure, and the NRCA analyzes reactor parameters, normalized to a singular scale, for operational anomalies. Therefore, operating the NRCA with programmed knowledge of a nuclear facilities PRA results would be an effective safety tool. Figure 13 displays the Gaussian kernel values for the four events described in Table 5. The three dashed lines shown on Figure 9 are representative of operational thresholds that could be set. The dashed line threshold set near 1.0 on Figure 13 is below the Gaussian values at iteration number one for the steam flow rate and reactor vessel pressure. These two events, serving as initiating events for a PRA, could trigger an NRCA inference with the PRA information for the reported initiating events. The NRCA would provide further confirmation of the events taking place within the nuclear reactor when the coolant outlet temperature passes the second dashed line threshold on Figure 13. This would provide another valuable piece of inference information to the operators. Lastly, the NRCA program once the steam generator valve signal passed the first threshold would identify the root cause of the event series. Since each parameter has been normalized, the thresholds would be applicable across all four processes, thus allowing a single program to analyze multiple sets of parameters against historical and/or probability information embedded in its inferences.

Table 5 Multiple Parameter Anomalies' Data

Event	Iteration #	Distance (d_{ij})	Gaussian (G_{ij})	Logistic (L_{ij})
Steam Generator Valve Activated (Ratio)	1	-0.00	$4.0e^{00}$	$4.0e^{00}$
	2	-0.10	$6.7e^{00}$	$4.9e^{00}$
	3	-0.15	$1.3e^{01}$	$6.1e^{00}$
	4	-0.20	$3.1e^{01}$	$8.2e^{01}$
	5	-0.25	$9.8e^{01}$	$1.2e^{01}$
with Elevated Reactor Vessel Pressure (Mpa)	1	0.00	$4.0e^{00}$	$4.0e^{00}$
	2	0.50	$1.4e^{06}$	$9.2e^{01}$
	3	0.50	$1.4e^{06}$	$9.2e^{01}$
	4	1.00	$5.8e^{22}$	$8.1e^{03}$
	5	1.50	$2.9e^{50}$	$7.3e^{05}$
with Elevated Coolant Outlet Temp (˚C)	1	0.00	$4.0e^{00}$	$4.0e^{00}$
	2	0.50	$4.0e^{00}$	$9.2e^{01}$
	3	0.50	$1.4e^{06}$	$9.2e^{01}$
	4	1.00	$1.4e^{06}$	$8.1e^{03}$
	5	1.50	$5.8e^{22}$	$7.3e^{05}$
with Reduced Steam Flow Rate (m^3/s)	1	-0.00	$4.0e^{00}$	$4.0e^{00}$
	2	-0.50	$1.4e^{06}$	$9.2e^{01}$
	3	-1.00	$5.8e^{22}$	$8.1e^{03}$
	4	-1.50	$2.9e^{50}$	$7.3e^{05}$
	5	-2.00	$1.7e^{89}$	$6.6e^{07}$

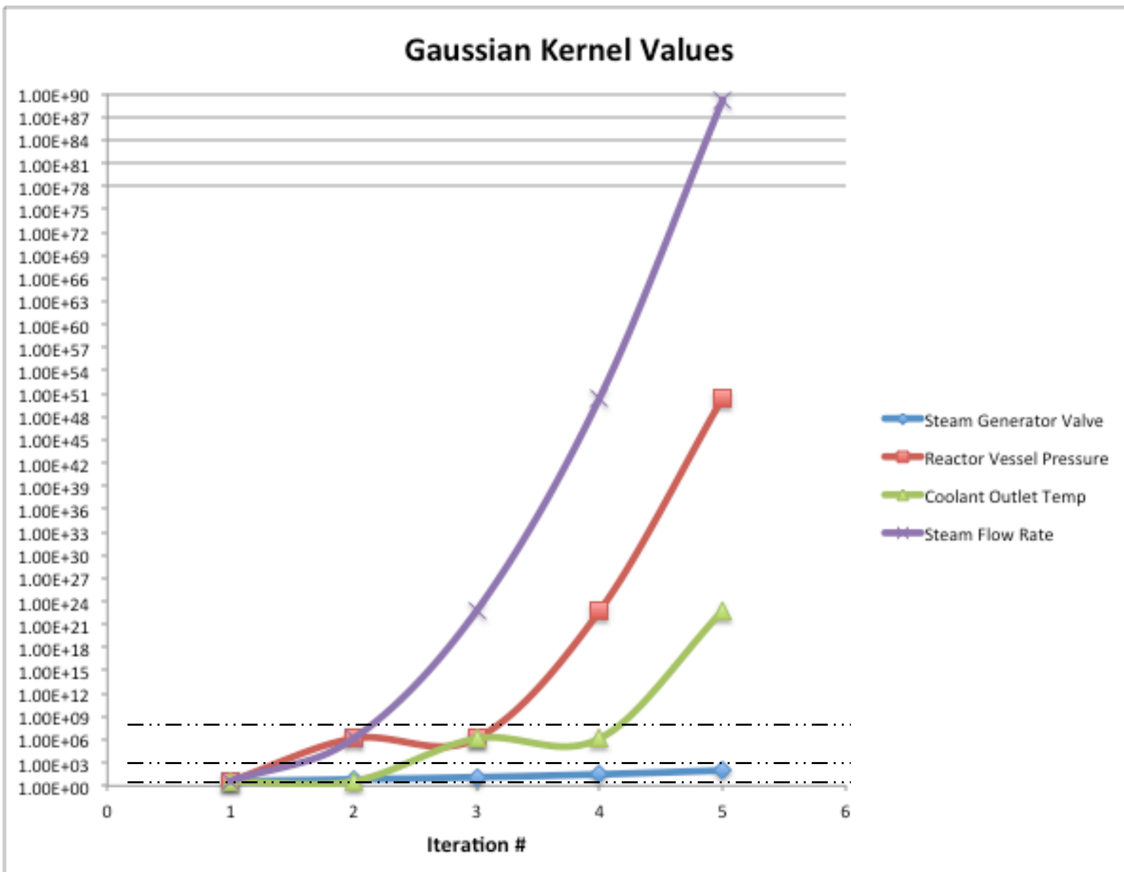

Figure 13. Multiple Parameter Anomalies with Thresholds

CHAPTER 7

CONCLUSION

This research was based on the idea that the academic community, through innovative research, could provide assistance to the United States Nuclear Regulatory Commission (US NRC) with straightening national interest in the fields of nuclear science and technology. The product of this research provides a means for the nuclear industry to implement a safety and monitoring program that can oversee the system parameters of a nuclear reactor facility, like that of a nuclear power plant.

The Nuclear Reactor Condition Analyzer (NRCA) program utilizes one-norm distance and kernel weighting equations to normalize onto one scale all nuclear reactor parameters under the program's analysis. This normalization allows the program to set more consistent parameter value thresholds for a more simplified approach to analyzing the condition of the nuclear reactor under its scrutiny. Although there has not been a thorough analysis of an authentic NPP data set, the preliminary results of this research display good results for an inherit flexibility within the KW algorithm to apply a wide variety of numerical weights to individual parameters with predictability. The inference step of this program, although not thoroughly concreted here, shows promise as a simple but effective early warning safety system for overall process disruptions within a nuclear process facility.

The program's early results show good KW agreement with variations in the query values across process parameters. And the inclusion of multiple parameter values in its analysis does not interfere with its application of KW to each parameter individually but provides a hastened response with the summation of multiple process parameter KW values through its inference step.

The inference portion of the NRCA program displays excellent agreement with the objectives of PRAs. Nuclear reactor facilities, with their use of PRAs, are poised to incorporate the inference information from their facility into the backend of the NRCA program. This coupling, of the NRCA's ability to preemptively interpret operational failures, and the intended prevention of incident progression that predicates PRA studies, illustrates the depth of integration that can be achieved by the inclusion of more extensive on-line monitoring systems within the nuclear industry.

CHAPTER 8

FUTURE WORK

Future research with this program will continue with operational, nominal, and incident parameter values that are time dependent from an authenticated NPP data set. The inclusion of operational and incident data from a US NPP will provide the benchmark data that this study requires to confirm the NRCA's ability to predict operational failures. This will be coupled with PRAs for the same facility, in order that the inference portion of the NRCA be calibrated to provide correct operator awareness.

The integration of a graphical user interface (GUI) for displaying the operational status and trends of the systems parameters under the NRCA's analysis is a future expansion for this work. An NRCA GUI will assist nuclear reactor operators in the monitoring of all systems under the NRCA umbrella by visualizing the operational trends of the reactor. This will allow the operators to more easily monitor the reactor systems for optimum operation as well as operational failures.

APPENDIX A

NRCA GENERIC PWR INPUT FILE

The following file contains input values that simulate the elevation in a reactor's coolant system.

0.0

1.0

1000.0

1.0

220.0

287.2

13702.0

1820.0

90.0

0.0

1.0

1000.0

1.0

220.1

287.2

13702.0

1820.0

90.0

0.0

1.0

1000.0

1.0

220.2

287.4

13702.0

1820.0

90.0

0.0

1.0

1000.0

1.0

220.4

287.8

13702.0

1820.0

90.0

0.0

1.0

1000.0

1.0

220.9

288.2

13702.0

1820.0

90.0

APPENDIX B

NRCA GENERIC PWR EXAMPLE CODE

The following C++ code is the console NRCA program:

```
//
//  main.cpp
//  PWRVectorInputFile
//
//  Created by Kaylyn Marie McCoy on 6/24/13.
//  Copyright (c) 2013 Kaylyn Marie McCoy. All rights reserved.
//

#include <iostream>
#include <fstream>
#include <stdio.h>
#include <vector>
#include <math.h>
#include <iostream>
#include <iterator>
#include <fstream>
#include <vector>
#include <algorithm> // for std::copy

int main()
{
    std::ifstream is("/Users/Kaylyn/PWR/ElevatedCoolantTemp.txt");
    std::istream_iterator<float> start(is), end;
    std::vector<float> a(start, end);
    std::cout << "Read " << a.size() << " vectors" << std::endl;

    // print the numbers to stdout
    std::cout << "numbers read in:\n";
    std::copy(a.begin(), a.end(),
              std::ostream_iterator<float>(std::cout, " "));
    std::cout << std::endl << std::endl;

    //std::vector <float> rbmk = {11.0, 12.0, 13.0, 14.0, 15.0, 0.75};

    //Exemplar vectors (0)SG-I/O, (1)TMB-I/O, (2)System P, (3)PRV-I/O,
(4)System Inlet T, (5)System Outlet T,(6)Coolant Flow, (7)Steam Flow
(8)Reactor Power
    std::vector <float> pelevation = {0.0, 0.0, 1100.0, 0.0, 242.0,
315.9, 13500, 1000.0, 90.0}; //pwr
    std::vector <float> ploss = {0.0, 0.0, 900.0, 1.0, 242.0,
315.9,15000, 1920, 90.0}; //pwr
    //counting vector for the iterations loop
    std::vector <int>::size_type j = 0;
```

```
    for (std::vector<float>::iterator i = a.begin(); i != a.end();
i++)
    {
        j++;
            std::cout << "  BELOW IS ITERATION #: " << j << std::endl
<< "  -----------------------------------------------------" <<
std::endl << std::endl;
        //v.push_back(*j);
        //Distance vector definitions
        std::vector <float> distancepwr;
        std::vector <float> distancepelevation;
        std::vector <float> distanceploss;
        //

        //Normal Operational vector values(0)SG-I/O, (1)TMB-I/O,
(2)System P, (3)PRV-I/O, (4)System Inlet T, (5)System Outlet
T,(6)Coolant Flow, (7)Steam Flow (8)Reactor Power
        std::vector <float> optimumpwr = {0.0, 1.0, 1000.0, 1.0,
220.0, 287.2, 13702, 1820.0, 90.0};

        //Realtime vector index input position from file and distance
comp
        std::cout << "RealTime SG Valve Position is: " << std::endl;
        std::cout << *i << "% (0 - 100% Open, 1 - 100% Closed)" <<
std::endl << std::endl;
        distancepwr.push_back(*i - optimumpwr[0]);
            std::cout << "RealTime SG Valve Deviation is: "  <<
distancepwr[0] << " %";
        std::cout << std::endl << std::endl;
        i++;

        std::cout << "RealTime TMB Valve Position is: " << std::endl;
        std::cout << *i << " (0 - 100% Open, 1 - 100% Closed)" <<
std::endl << std::endl;
        distancepwr.push_back(*i - optimumpwr[1]);
        std::cout << "RealTime TMB Valve Deviation is: "<<
distancepwr[1] << " %";
        std::cout << std::endl << std::endl;
        i++;

        std::cout << "RealTime System Pressure Value is: " <<
std::endl;
        std::cout << *i << " Psia" << std::endl << std::endl;
        distancepwr.push_back(*i - distancepwr[2]);
        std::cout << "RealTime System Pressure Deviation is: "<<
distancepwr[2] << " Psia ";
        std::cout << std::endl << std::endl;
        i++;

        std::cout << "RealTime PRV Position is: " << std::endl;
        std::cout << *i << " (0 - open, 1 - Closed)" << std::endl <<
std::endl;
        distancepwr.push_back(*i - distancepwr[3]);
        std::cout << "RealTime PRV Deviation is: "<< distancepwr[3] <<
" % ";
        std::cout << std::endl << std::endl;
        i++;
```

```
        std::cout << "RealTime Coolant Inlet Temperature Value is: "
<< std::endl;
        std::cout << *i << " Deg C" << std::endl << std::endl;
        distancepwr.push_back(*i - distancepwr[4]);
        std::cout << "RealTime Inlet Temperature Deviation is: "<<
distancepwr[4] << " Deg C ";
        std::cout << std::endl << std::endl;
        i++;

        std::cout << "RealTime Coolant Outlet Temperature Value is: "
<< std::endl;
        std::cout << *i << " Deg C" << std::endl << std::endl;
        distancepwr.push_back(*i - distancepwr[5]);
        std::cout << "RealTime Outlet Temperature Deviation is: "<<
distancepwr[5] << " Deg C";
        std::cout << std::endl << std::endl;
        i++;

        std::cout << "RealTime Coolant Recirculation Flow Value is: "
<< std::endl;
        std::cout << *i << " Kg/s" << std::endl << std::endl;
        distancepwr.push_back(*i - distancepwr[6]);
        std::cout << "RealTime Coolant Recirculation Flow Deviation
is: "<< distancepwr[6] << " Kg/s";
        std::cout << std::endl << std::endl;
        i++;

        std::cout << "RealTime Steam Flow Rate is: " << std::endl;
        std::cout << *i << " Kg/s" << std::endl << std::endl;
        distancepwr.push_back(*i - distancepwr[7]);
        std::cout << "RealTime Steam Flow Rate Deviation is: " <<
distancepwr[7] << " Kg/s";
        std::cout << std::endl << std::endl;
        i++;

        std::cout << "RealTime Reactor Power Value is: " << std::endl;
        std::cout << *i << " %" << std::endl << std::endl;
        distancepwr.push_back(*i - distancepwr[8]);
        std::cout << "RealTime Reactor Power Deviation: "<<
distancepwr[8] << " %";
        std::cout << std::endl << std::endl;

        //Distance analysis b/w input and exemplar vectors
        //for (std::vector<float>::size_type i = 0; i < 9; i++)
        //    distancebwr.push_back(optimumbwr[i]-a[i]);

        for (std::vector<float>::size_type i = 0; i < 9; i++)
            distancepelevation.push_back(distancepwr[i]-
pelevation[i]);

        for (std::vector<float>::size_type i = 0; i < 9; i++)
            distanceploss.push_back(distancepwr[i]-ploss[i]);

        std::cout << "PARAMETER DISTANCE VALUES:" << std::endl <<
std::endl;
        //Distance screen outputs
        //for (std::vector<float>::size_type i = 0; i < 9; i++)
            //std::cout << distancebwr[i] << " ";
```

```
        //std::cout << std::endl << std::endl;

        for (std::vector<float>::size_type i = 0; i < 9; i++)
            std::cout << distancepelevation[i] << " ";
        std::cout << std::endl << std::endl;

        for (std::vector<float>::size_type i = 0; i < 9; i++)
            std::cout << distanceploss[i] << " ";
        std::cout << std::endl << std::endl;

        //Bandwidth (h) and (Pi) definitions for the Kernel Functions
        double h1 = 0.099;      //0.399;
        double h2 = 0.01;
        double pi = 3.14159265359;

        //Gaussian Weight vector passing variables
        std::vector <double> goptimumpwr;
        std::vector <double> gnearpelevation;
        std::vector <double> gnearploss;

        //Weight application with Gassiun Kernel function
        for (std::vector<float>::size_type i = 0; i < 9; i++)

goptimumpwr.push_back(1/(sqrt((2*pi*(pow(h1,2)))))*(exp((pow(distancep
wr[i],2))/(2*pow(h1,2)))));

        for (std::vector<float>::size_type i = 0; i < 9; i++)

gnearpelevation.push_back(1/(sqrt((2*pi*pow(h1,2))))*exp((pow(distance
pelevation[i],2)/(2*pow(h1,2)))));

        for (std::vector<float>::size_type i = 0; i < 9; i++)

gnearploss.push_back(1/(sqrt((2*pi*pow(h1,2))))*exp((pow(distanceploss
[i],2))/(2*pow(h1,2))));


        //Logistic vector definitions
        std::vector <double> toptimumpwr;
        std::vector <double> tnearpelevation;
        std::vector <double> tnearploss;

        //Weight application with Logistic Kernel function
        for (std::vector<float>::size_type i = 0; i < 9; i++)
            toptimumpwr.push_back(exp(distancepwr[i]*h2)+2+(-
exp(distancepwr[i]*h2)));

        for (std::vector<float>::size_type i = 0; i < 9; i++)

tnearpelevation.push_back(exp(distancepelevation[i]*h2)+2+(-
exp(distancepelevation[i]*h2)));

        for (std::vector<float>::size_type i = 0; i < 9; i++)
            tnearploss.push_back(exp(distanceploss[i]*h2)+2+(-
exp(distanceploss[i]*h2)));

        std::cout << " INDIVIDUAL PARAMETER PENALTY VALUES: " <<
std::endl << std::endl;
```

```
        //Gaussian Weighted vector indices outputs
        std::cout << "Gaussian: " << std::endl;

        // First Bandwidth set Gaussian w/ Vector 0
        std::cout<< goptimumpwr[0] << " ";
        if (goptimumpwr[0] <= 0.42)
        {
            std::cout <<  "<-The SG Valve is 100% Open! " << " "<<
std::endl;
        }
        else if (goptimumpwr[0] > 1)
        {
            std::cout <<  "<-The SG Valve is NOT 100% Open!! " << "
"<< std::endl;
        }

        // First Bandwidth set Gaussian w/ Vector 1
        std::cout<< goptimumpwr[1] << " ";
        if (goptimumpwr[1] == 0.75)
        {
            std::cout <<  "<-The TMB Valve is 100% Closed! " << " "<<
std::endl;
        }
        else if (goptimumpwr[1] > 0.42)
        {
            std::cout <<  "<-The TMB Valve is Not 100% Closed!! " << "
"<< std::endl;
        }

        // First Bandwidth set Gaussian w/ Vector 2
        std::cout<< goptimumpwr[2] << " ";
        if (goptimumpwr[2] <= 1)
        {
            std::cout <<  "<-The System Pressure is Near Optimum! " <<
" "<< std::endl;
        }
        else if (goptimumpwr[2] > 1)
        {
            std::cout <<  "<-The System Pressure is Not Normal!! " <<
" "<< std::endl;
        }

        // First Bandwidth set Gaussian w/ Vector 3
        std::cout<< goptimumpwr[3] << " ";
        if (goptimumpwr[3] <= 0.42)
        {
            std::cout <<  "<-The PRV is 100% Closed! " << " " <<
std::endl;
        }
        else if (goptimumpwr[3] > 0.42)
        {
            std::cout <<  "<-The PRV is Open!!  " << " "<< std::endl;
        }

        // First Bandwidth set Gaussian w/ Vector 4
        std::cout<< goptimumpwr[4] << " ";
        if (goptimumpwr[4] <= 2)
        {
```

```
            std::cout <<  "<-The Coolant Inlet T is Near Optimum! " <<
" "<< std::endl;
        }
        else if (goptimumpwr[4] > 2)
        {
            std::cout <<  " <-The Coolant Inlet T is NOT Normal!! " <<
" " << std::endl;
        }

        // First Bandwidth set Gaussian w/ Vector 5
        std::cout<< goptimumpwr[5] << " ";
        if (goptimumpwr[5] <= 2)
        {
            std::cout <<  "<-The Coolant Outlet T is Near Optimum! "
<< " "<< std::endl;
        }
        else if (goptimumpwr[5] > 2)
        {
            std::cout <<  " <-The Coolant Outlet T is NOT Normal!! "
<< " " << std::endl;
        }

        std::cout<< goptimumpwr[6] << " ";
        if (goptimumpwr[6] <= 2)
        {
            std::cout <<  "<-The Coolant Recirculation Rate is Near
Optimum! " << " "<< std::endl;
        }
        else if (goptimumpwr[6] > 2)
        {
            std::cout <<  " <-The Coolant Recirculation Rate is NOT
Normal!! " << " " << std::endl;
        }

        std::cout<< goptimumpwr[7] << " ";
        if (goptimumpwr[7] <= 2)
        {
            std::cout <<  "<-The Steam Flow Rate is Near Optimum! " <<
" "<< std::endl;
        }
        else if (goptimumpwr[7] > 2)
        {
            std::cout <<  " <-The Steam Flow Rate is NOT Normal!! " <<
" " << std::endl;
        }


        // Reactor Power Input a[8]
        std::cout <<  "The Reactor Power is at: " << a[8] << " % " <<
std::endl << std::endl;


        //Logistic vector indices outputs
        std::cout << "Logistic: " << std::endl;
        // First Bandwidth set Triweight w/ Vector 0
        std::cout<< toptimumpwr[0] << " ";
        if (toptimumpwr[0] <= 1)
        {
```

```
            std::cout <<  "<-The SG Valve is Open! " << " "<<
std::endl;
        }
        else if (toptimumpwr[0] > 1)
        {
            std::cout <<  "<-The SG Valve is NOT Open!! " << " "<<
std::endl;
        }

        // First Bandwidth set Logistic w/ Vector 1
        std::cout<< toptimumpwr[1] << " ";
        if (toptimumpwr[1] <= 1)
        {
            std::cout <<  "<-The TMB Valve is Closed! " << " "<<
std::endl;
        }
        else if (toptimumpwr[1] > 1)
        {
            std::cout <<  "<-The TMB Valve is NOT Closed!! " << " "<<
std::endl;
        }

        // First Bandwidth set Logistic w/ Vector 2
        std::cout<< toptimumpwr[2] << " ";
        if (toptimumpwr[2] <= 1)
        {
            std::cout <<  "<-The System Pressure is Near Optimum! " <<
" "<< std::endl;
        }
        else if (toptimumpwr[2] > 1)
        {
            std::cout <<  "<-The System Pressure is Not Normal!! " <<
" "<< std::endl;
        }

        // First Bandwidth set Logistic w/ Vector 3
        std::cout<< toptimumpwr[3] << " ";
        if (toptimumpwr[3] <= 1)
        {
            std::cout <<  "<-The PRV is Closed! " << " " << std::endl;
        }
        else if (toptimumpwr[3] > 1)
        {
            std::cout <<  "<-The PRV is Open!!  " << " "<< std::endl;
        }

        // First Bandwidth set Logistic w/ Vector 4
        std::cout<< toptimumpwr[4] << " ";
        if (toptimumpwr[4] <= 1)
        {
            std::cout <<  "<-The Coolant Inlet T is Near Optimum! " <<
" "<< std::endl;
        }
        else if (toptimumpwr[4] > 1)
        {
            std::cout <<  " <-The Coolant Inlet T is NOT Normal!! " <<
" " << std::endl;
        }
```

```
        // First Bandwidth set Logistic w/ Vector 5
        std::cout<< toptimumpwr[5] << " ";
        if (toptimumpwr[5] <= 1)
        {
            std::cout <<  "<-The Coolant Outlet T is Near Optimum! "
<< " "<< std::endl;
        }
        else if (toptimumpwr[5] > 1)
        {
            std::cout <<  " <-The Coolant Outlet T is NOT Normal!! "
<< " " << std::endl;
        }

        std::cout<< toptimumpwr[6] << " ";
        if (toptimumpwr[6] <= 1)
        {
            std::cout <<  "<-The Coolant Recirculation Rate is Near
Optimum! " << " "<< std::endl;
        }
        else if (toptimumpwr[6] > 1)
        {
            std::cout <<  " <-The Coolant Recirculation Rate is NOT
Normal!! " << " " << std::endl;
        }

        std::cout<< toptimumpwr[7] << " ";
        if (toptimumpwr[7] <= 2)
        {
            std::cout <<  "<-The Steam Flow Rate is Near Optimum! " <<
" "<< std::endl;
        }
        else if (toptimumpwr[7] > 2)
        {
            std::cout <<  " <-The Steam Flow Rate is NOT Normal!! " <<
" " << std::endl;
        }

        // Reactor Power Input a[7]
        std::cout <<  "The Reactor Power is at: " << a[8] << " % " <<
std::endl << std::endl;

        std::cout << " SYSTEM PARAMETER SUMMATION PENALTY VALUES: " <<
std::endl << std::endl;
        //Gaussian Weighted vector indices outputs

        //Gaussian Summation Values

        float goptimumsum;
        goptimumsum = goptimumpwr[0] + goptimumpwr[1] + goptimumpwr[2]
+ goptimumpwr[3] + goptimumpwr[4] + goptimumpwr[5] + goptimumpwr[6] +
goptimumpwr[7] + goptimumpwr[8];
        std::cout << "The Gaussian PWR Realtime Penalty Value is: " <<
goptimumsum<< std::endl << std::endl;

        float gpelevationsum;
        gpelevationsum = gnearpelevation[0] + gnearpelevation[1] +
gnearpelevation[2] + gnearpelevation[3] + gnearpelevation[4] +
```

```
gnearpelevation[5] + gnearpelevation[6] + gnearpelevation[7] +
gnearpelevation[8];
        std::cout << "The Gaussian Pressure Elevation Accident Penalty
Value is: " << gpelevationsum<< std::endl << std::endl;

        float gplosssum;
        gplosssum = gnearploss[0] + gnearploss[1] + gnearploss[2] +
gnearploss[3] + gnearploss[4] + gnearploss[5] + gnearploss[6] +
gnearploss[7] + gnearploss[8];
        std::cout << "The Gaussian Pressure Loss Accident Penalty
Value is: " << gplosssum<< std::endl << std::endl;

        //Logistic Summation Values

        float toptimumsum;
        toptimumsum = toptimumpwr[0] + toptimumpwr[1] + toptimumpwr[2]
+ toptimumpwr[3] + toptimumpwr[4] + toptimumpwr[5] + toptimumpwr[6] +
toptimumpwr[7] + toptimumpwr[8];
        std::cout << "The Logistic PWR Realtime Penalty Value is: " <<
toptimumsum<< std::endl << std::endl;

        float tpelevationsum;
        tpelevationsum = tnearpelevation[0] + tnearpelevation[1] +
tnearpelevation[2] + tnearpelevation[3] + tnearpelevation[4] +
tnearpelevation[5] + tnearpelevation[6] + tnearpelevation[7] +
tnearpelevation[8];
        std::cout << "The Logistic Pressure Elevation Accident Penalty
Value is: " << tpelevationsum<< std::endl << std::endl;

        float tplosssum;
        tplosssum = tnearploss[0] + tnearploss[1] + tnearploss[2] +
tnearploss[3] + tnearploss[4] + tnearploss[5] + tnearploss[6] +
tnearploss[7] + tnearploss[8];
        std::cout << "The Logistic Pressure Loss Accident Penalty
Value is: " << tplosssum<< std::endl << std::endl;

        std::cout << " SYSTEM INFERENCE: " << std::endl << std::endl;
        //Distance vector indices outputs

        if (a[2] > optimumpwr [2])
        {
            std::cout <<  "!!The System Pressure is Above Optimum!! "
<< " "<< std::endl;
        }

        if (a[2] > optimumpwr [2] && a[0] >= 0.2)
        {
            std::cout <<  "!!The System Pressure is Over-Burdened Due
to SG Valve Closure!! " << " "<< std::endl;
        }

        if (a[2] > (1.05 *optimumpwr [2]) && a[3] == 1)
        {
            std::cout <<  "!!The System Pressure is Over-Burdened Due
to PRV Closure!! " << " "<< std::endl;
        }
```

```
        if (a[2] <= (0.995 *optimumpwr [2]) && a[3] <= 0.95 *
optimumpwr [3])
        {
            std::cout <<  "!!The System Pressure is Falling Due to
Open PRV!! " << " "<< std::endl;
        }
        //Penalty Summation Comparisons

        if (goptimumsum >=  0.85 * gpelevationsum)
        {
            std::cout <<  "!!THE SYSTEM IS NEAR A SATURATION CONDITION
FAILURE!! " << " "<< std::endl;
        }

        if (goptimumsum >=  0.85 * gplosssum)
        {
            std::cout <<  "!!THE SYSTEM IS NEAR A SATURATION CONDITION
FAILURE!! " << " "<< std::endl;
        }

        if (toptimumsum >=  0.85 * tpelevationsum)
        {
            std::cout <<  "!!THE SYSTEM IS NEAR A SATURATION CONDITION
FAILURE!! " << " "<< std::endl;
        }

        if (goptimumsum >=  0.85 * tplosssum)
        {
            std::cout <<  "!!THE SYSTEM IS NEAR A SATURATION CONDITION
FAILURE!! " << " "<< std::endl << std::endl;

        }
        std::cout << std::endl << std::endl;
    }

    return 0;
}
```

APPENDIX C

THREE MILE ISLAND EXAMPLE INPUT FILE

```
100.0
400.0
100.0
100.0
1820.0
285.0
7.0
0.0
100.0
290.0
15.0
0.0
0.0
100.0
13700.00
90.0

0.0
400.0
100.0
100.0
1820.0
285.0
7.0
0.0
100.0
290.0
15.0
0.0
0.0
100.0
13700.00
90.0

0.0
100.0
0.0
100.0
1820.0
285.0
7.0
0.0
100.0
290.0
```

```
15.0
0.0
0.0
100.0
13700.00
90.0

0.0
100.0
0.0
20.0
220.0
285.0
7.0
0.0
100.0
290.0
15.0
0.0
0.0
100.0
13700.00
10.0

0.0
100.0
0.0
10.0
220.0
295.0
7.5
0.0
100.0
290.0
15.0
0.0
0.0
100.0
13700.00
10.0

0.0
100.0
0.0
10.0
220.0
295.0
7.5
100.0
100.0
290.0
15.0
0.0
0.0
100.0
13700.00
10.0
```

```
0.0
100.0
0.0
10.0
220.0
295.0
7.5
100.0
100.0
340.0
16.0
100.0
0.0
100.0
13700.00
10.0

0.0
100.0
0.0
10.0
220.0
295.0
7.5
100.0
100.0
340.0
15.0
100.0
100.0
100.0
13700.00
10.0

0.0
100.0
0.0
10.0
220.0
295.0
7.5
100.0
100.0
340.0
14.0
100.0
100.0
80.0
11700.00
10.0
```

APPENDIX D

NRCA THREE MILE ISLAND EXAMPLE CODE

```
//
//  main.cpp
//  ThreeMileIsland
//
//  Created by Kaylyn Marie McCoy on 3/13/14.
//  Copyright (c) 2014 Kaylyn Marie McCoy. All rights reserved.
//

#include <iostream>
#include <fstream>
#include <stdio.h>
#include <vector>
#include <math.h>
#include <iostream>
#include <iterator>
#include <fstream>
#include <vector>
#include <algorithm> // for std::copy

int main()
{
    std::ifstream is("/Users/Kaylyn/PWR/ThreeMileIsland.txt");
    std::istream_iterator<float> start(is), end;
    std::vector<float> a(start, end);
    std::cout << "Read " << a.size() << " vectors" << std::endl;

    // print the numbers to stdout
    std::cout << "numbers read in:\n";
    std::copy(a.begin(), a.end(),
              std::ostream_iterator<float>(std::cout, " "));
    std::cout << std::endl << std::endl;


    //counting vector for the iterations loop
    std::vector <int>::size_type j = 0;

    for (std::vector<float>::iterator i = a.begin(); i != a.end();
      i++)
    {
        j++;
        std::cout << "  BELOW IS ITERATION #: " << j << std::endl << "
      ------------------------------------------------------" <<
      std::endl << std::endl;
        //v.push_back(*j);
```

```
  //Distance vector definitions
  std::vector <float> distancepwr;

  ////Exemplar vectors (0)Condensate-I/O, (1)Condensate Flow,
(2)Steam Generator FP-I/O, (3)Turbine-I/O, (4)Steam Flow,
(5)Secondary T,(6)Secondary P, (7) Aux Steam Generator FP-I/O,
(8) Aux Steam Generator FP Valve-I/O, (9)Primary T, (10) Primary
P, (11) PRV-I/O, (12) HPIP-I/O, (13) MCP-I/O, (14) Primary Flow,
(15) Reactor Power
  std::vector <float> optimumpwr = {100.0, 400.0, 100.0, 100.0,
1820.0, 285.0, 7.0, 0.0, 100.0, 290.0, 15.0, 0.0, 0.0, 100.0,
13700.00, 90.0};

  //Realtime vector index input position from file and distance
comp
  std::cout << "Secondary Circuit Condensate Pump is Operating
at: " << std::endl;
  std::cout << *i << "% " << std::endl << std::endl;
  distancepwr.push_back(*i - optimumpwr[0]);
  std::cout << "Secondary Circuit Condensate Pump Deviation from
Nominal Operation is: "  << distancepwr[0] << " %";
  std::cout << std::endl << std::endl;
  i++;

  std::cout << "Secondary Circuit Condensate Flowrate is : " <<
std::endl;
  std::cout << *i << " l/s" << std::endl << std::endl;
  distancepwr.push_back(*i - optimumpwr[1]);
  std::cout << "Secondary Circuit Flowrate Deviation from
Nominal Operation is: "<< distancepwr[1] << " l/s";
  std::cout << std::endl << std::endl;
  i++;

  std::cout << "Steam Generator Feedwater Pump is Operation at:
" << std::endl;
  std::cout << *i << " % " << std::endl << std::endl;
  distancepwr.push_back(*i - optimumpwr[2]);
  std::cout << "Steam Generator Feedwater Pump Deviation from
Nominal Operation is: "<< distancepwr[2] << " % ";
  std::cout << std::endl << std::endl;
  i++;

  std::cout << "Steam Turbine is Operating at: " << std::endl;
  std::cout << *i << " % " << std::endl << std::endl;
  distancepwr.push_back(*i - optimumpwr[3]);
  std::cout << "Steam Turbine Deviation from Nominal Operation
is: "<< distancepwr[3] << " % ";
  std::cout << std::endl << std::endl;
  i++;

  std::cout << "Secondary Circuit Steam Flowrate is: " <<
std::endl;
  std::cout << *i << " l/s" << std::endl << std::endl;
  distancepwr.push_back(*i - optimumpwr[4]);
  std::cout << "Secondary Circuit Steam Flowrate Deviation from
Nominal Operation is: "<< distancepwr[4] << " l/s";
  std::cout << std::endl << std::endl;
  i++;
```

```
  std::cout << "Secondary Circuit Temperature is: " <<
std::endl;
  std::cout << *i << " Deg C" << std::endl << std::endl;
  distancepwr.push_back(*i - optimumpwr[5]);
  std::cout << "Secondary Circuit Temperature Deviation from
Nominal Operation is: "<< distancepwr[5] << " Deg C";
  std::cout << std::endl << std::endl;
  i++;

  std::cout << "Secondary Circuit Pressure is: " << std::endl;
  std::cout << *i << " MPa" << std::endl << std::endl;
  distancepwr.push_back(*i - optimumpwr[6]);
  std::cout << "Secondary Circuit Pressure Deviation from
Nominal Operation is: "<< distancepwr[6] << " MPa";
  std::cout << std::endl << std::endl;
  i++;

  std::cout << "Auxiallary Steam Generator Feedwater Pump is
Operating at: " << std::endl;
  std::cout << *i << " % " << std::endl << std::endl;
  distancepwr.push_back(*i - optimumpwr[7]);
  std::cout << "Auxiallary Steam Generator Feedwater Pump
Deviation from Nominal Operation is: " << distancepwr[7] << "
%";
  std::cout << std::endl << std::endl;
  i++;

  std::cout << "Auxiallary Steam Generator Feedwater Valve
Closure Position is: " << std::endl;
  std::cout << *i << " %" << std::endl << std::endl;
  distancepwr.push_back(*i - optimumpwr[8]);
  std::cout << "Auxiallary Steam Generator Feedwater Valve
Closure Deviation from Nominal Operation is: " << distancepwr[8]
<< " %";
  std::cout << std::endl << std::endl;
  i++;

  std::cout << "Primary Circuit Temperature is: " << std::endl;
  std::cout << *i << " Deg C" << std::endl << std::endl;
  distancepwr.push_back(*i - optimumpwr[9]);
  std::cout << "Primary Circuit Temperature Deviation from
Nominal Operation is: "<< distancepwr[9] << " Deg C";
  std::cout << std::endl << std::endl;
  i++;

  std::cout << "Primary Circuit Pressure is: " << std::endl;
  std::cout << *i << " MPa" << std::endl << std::endl;
  distancepwr.push_back(*i - optimumpwr[10]);
  std::cout << "Primary Circuit Pressure Deviation from Nominal
Operation is: "<< distancepwr[10] << " MPa";
  std::cout << std::endl << std::endl;
  i++;

  std::cout << "Steam Generator Pressure Relief Valve Closure
Position is: " << std::endl;
  std::cout << *i << " % " << std::endl << std::endl;
  distancepwr.push_back(*i - optimumpwr[11]);
```

```
  std::cout << "Steam Generator Pressure Relief Valve Closure
Deviation from Nominal Operation is: " << distancepwr[11] << "
%";
  std::cout << std::endl << std::endl;
  i++;

  std::cout << "Primary Circuit High Pressure Injection Pump is
Operating at: " << std::endl;
  std::cout << *i << " % " << std::endl << std::endl;
  distancepwr.push_back(*i - optimumpwr[12]);
  std::cout << "Primary Circuit High Pressure Injection Pump
Deviation from Nominal Operation is: " << distancepwr[12] << "
%";
  std::cout << std::endl << std::endl;
  i++;

  std::cout << "Primary Circuit Main Circulation Pump is
Operating at: " << std::endl;
  std::cout << *i << " % " << std::endl << std::endl;
  distancepwr.push_back(*i - optimumpwr[13]);
  std::cout << "Primary Circuit Main Circulation Pump Deviation
from Nominal Operation is: " << distancepwr[13] << " %";
  std::cout << std::endl << std::endl;
  i++;

  std::cout << "Primary Circuit Coolant Flowrate is: " <<
std::endl;
  std::cout << *i << " l/s" << std::endl << std::endl;
  distancepwr.push_back(*i - optimumpwr[14]);
  std::cout << "Primary Circuit Coolant Flowrate Deviation from
Nominal Operation is: "<< distancepwr[14] << " l/s";
  std::cout << std::endl << std::endl;
  i++;

  std::cout << "Reactor Power is: " << std::endl;
  std::cout << *i << " %" << std::endl << std::endl;
  distancepwr.push_back(*i - optimumpwr[15]);
  std::cout << "Reactor Power Deviation from Nominal Operation
is: "<< distancepwr[15] << " %";
  std::cout << std::endl << std::endl;



  //Bandwidth (h) and (Pi) definitions for the Kernel Functions
  double h1 = 0.099;       //0.399;
  double h2 = 9.00;
  double pi = 3.14159265359;

  //Gaussian Weight vector passing variables
  std::vector <float> goptimumpwr;


  //Weight application with Gassiun Kernel function
  for (std::vector<float>::size_type i = 0; i < 16; i++)

goptimumpwr.push_back(1/(sqrt((2*pi*(pow(h1,2)))))*(exp((pow(dis
tancepwr[i],2))/(2*pow(h1,2)))));
  //Logistic vector definitions
```

```
  std::vector <double> toptimumpwr;

  //Weight application with Logistic Kernel function
  for (std::vector<float>::size_type i = 0; i < 16; i++)
      toptimumpwr.push_back(exp(distancepwr[i]*h2)+2+(-
exp(distancepwr[i]*h2)));

  std::cout << " INDIVIDUAL PARAMETER PENALTY VALUES: " <<
std::endl << std::endl;
  //Gaussian Weighted vector indices outputs
  std::cout << "Gaussian: " << std::endl;

  // First Bandwidth set Gaussian w/ Vector 0
  std::cout<< goptimumpwr[0] << " ";
  if (goptimumpwr[0] > 4.02972)
  {
      std::cout <<  "<-The Secondary Circuit Condensate Pump is
Not Operating Nominally! " << " "<< std::endl;
  }
  if (goptimumpwr[0] > 40.02972)
  {
      std::cout <<  "<-The Secondary Circuit Condensate Pump is
Operating in a Dangerous State! " << " "<< std::endl;
  }
  // First Bandwidth set Gaussian w/ Vector 1
  std::cout<< goptimumpwr[1] << " ";
  if (goptimumpwr[1] > 4.02972)
  {
      std::cout <<  "<-The Secondary Circuit Condensate Flow is
Not  Nominal! " << " "<< std::endl;
  }
  if (goptimumpwr[1] > 40.02972)
  {
      std::cout <<  "<-The Secondary Circuit Condensate Flowrate
is in a Dangerous State! " << " "<< std::endl;
  }

  // First Bandwidth set Gaussian w/ Vector 2
  std::cout<< goptimumpwr[2] << " ";
  if (goptimumpwr[2] > 4.02972)
  {
      std::cout <<  "<-The Steam Generator is Not Operating
Nominally! " << " "<< std::endl;
  }
  if (goptimumpwr[2] > 40.02972)
  {
      std::cout <<  "<-The Steam Generator is in a Dangerous
State! " << " "<< std::endl;
  }

  // First Bandwidth set Gaussian w/ Vector 3
  std::cout<< goptimumpwr[3] << " ";
  if (goptimumpwr[3] > 4.02972)
  {
      std::cout <<  "<-The Steam Generator Feedwater Pump is Not
Operating Nominally! " << " "<< std::endl;
  }
  if (goptimumpwr[3] > 40.02972)
```

```
  {
      std::cout <<  "<-The Steam Generator Feedwater Pump is
Operating in a Dangerous State! " << " "<< std::endl;
  }


  // First Bandwidth set Gaussian w/ Vector 4
  std::cout<< goptimumpwr[4] << " ";
  if (goptimumpwr[4] > 4.02972)
  {
      std::cout <<  "<-The Steam Turbine is Not Operating
Nominally! " << " "<< std::endl;
  }
  if (goptimumpwr[4] > 40.02972)
  {
      std::cout <<  "<-The Steam Turbine is Operating in a
Dangerous State! " << " "<< std::endl;
  }

  // First Bandwidth set Gaussian w/ Vector 5
  std::cout<< goptimumpwr[5] << " ";
  if (goptimumpwr[5] > 4.02972)
  {
      std::cout <<  "<-The Steam Flow Rate is Not Nominal! " <<
" "<< std::endl;
  }
  if (goptimumpwr[5] > 40.02972)
  {
      std::cout <<  "<-The Steam Flow Rate is in a Dangerous
State! " << " "<< std::endl;
  }

  // First Bandwidth set Gaussian w/ Vector 6
  std::cout<< goptimumpwr[6] << " ";
  if (goptimumpwr[6] > 4.02972)
  {
      std::cout <<  "<-The Secondary Circuit Temperature is Not
Nominal! " << " "<< std::endl;
  }
  if (goptimumpwr[6] > 40.02972)
  {
      std::cout <<  "<-The Secondary Circuit Temperature is in a
Dangerous State! " << " "<< std::endl;
  }

  // First Bandwidth set Gaussian w/ Vector 7
  std::cout<< goptimumpwr[7] << " ";
  if (goptimumpwr[7] > 4.02972)
  {
      std::cout <<  "<-The Secondary Circuit Pressure is Not
Nominal! " << " "<< std::endl;
  }
  if (goptimumpwr[7] > 40.02972)
  {
      std::cout <<  "<-The Secondary Pressure is in a Dangerous
State! " << " "<< std::endl;
  }
```

```
  // First Bandwidth set Gaussian w/ Vector 8
  std::cout<< goptimumpwr[8] << " ";
  if (goptimumpwr[8] > 4.02972)
  {
      std::cout <<  "<-The Auxillary Steam Generator Feedwater
Pump is Not Operating Nominally! " << " "<< std::endl;
  }
  if (goptimumpwr[8] > 40.02972)
  {
      std::cout <<  "<-The Auxillary Steam Generator Feedwater
Pump is Operating in a Dangerous State! " << " "<< std::endl;
  }

  // First Bandwidth set Gaussian w/ Vector 9
  std::cout<< goptimumpwr[9] << " ";
  if (goptimumpwr[9] > 4.02972)
  {
      std::cout <<  "<-The Primary Temperature is Not Nominal! "
<< " "<< std::endl;
  }
  if (goptimumpwr[9] > 40.02972)
  {
      std::cout <<  "<-The Primary Temperature is in a Dangerous
State! " << " "<< std::endl;
  }

  // First Bandwidth set Gaussian w/ Vector 10
  std::cout<< goptimumpwr[10] << " ";
  if (goptimumpwr[10] > 4.02972)
  {
      std::cout <<  "<-The Primary Pressure is Not Nominal! " <<
" "<< std::endl;
  }
  if (goptimumpwr[10] > 40.02972)
  {
      std::cout <<  "<-The Primary Pressure is in a Dangerous
State! " << " "<< std::endl;
  }

  // First Bandwidth set Gaussian w/ Vector 11
  std::cout<< goptimumpwr[11] << " ";
  if (goptimumpwr[11] > 4.02972)
  {
      std::cout <<  "<-The Steam Generator Pressure Relief Valve
is Not Operating Nominally! " << " "<< std::endl;
  }
  if (goptimumpwr[11] > 40.02972)
  {
      std::cout <<  "<-The Steam Generator Pressure Relief Valve
is Operating in a Dangerous State! " << " "<< std::endl;
  }

  // First Bandwidth set Gaussian w/ Vector 12
  std::cout<< goptimumpwr[12] << " ";
  if (goptimumpwr[12] > 4.02972)
  {
      std::cout <<  "<-The High Pressure Injection Pump is Not
Operating Nominally! " << " "<< std::endl;
```

```
  }
  if (goptimumpwr[12] > 40.02972)
  {
      std::cout <<  "<-The High Pressure Injection Pump is
Operating in a Dangerous State! " << " "<< std::endl;
  }

  // First Bandwidth set Gaussian w/ Vector 13
  std::cout<< goptimumpwr[13] << " ";
  if (goptimumpwr[13] > 4.02972)
  {
      std::cout <<  "<-The Primary Circuit Main Circulation Pump
is Not Operating Nominally! " << " "<< std::endl;
  }
  if (goptimumpwr[13] > 40.02972)
  {
      std::cout <<  "<-The Primary Circuit Main Circulation Pump
is Operating in a Dangerous State! " << " "<< std::endl;
  }

  // First Bandwidth set Gaussian w/ Vector 14
  std::cout<< goptimumpwr[14] << " ";
  if (goptimumpwr[14] > 4.02972)
  {
      std::cout <<  "<-The Primary Circuit Flowrate is Not
Nominal! " << " "<< std::endl;
  }
  if (goptimumpwr[14] > 40.02972)
  {
      std::cout <<  "<-The Primary Circuit Flowrate is in a
Dangerous State! " << " "<< std::endl;
  }

  // Reactor Power Input a[15]
  std::cout <<  "The Reactor Power is at: " << a[15] << " % " <<
std::endl << std::endl;


  //Logistic vector indices outputs
  std::cout << "Logistic: " << std::endl;
  // First Bandwidth set Logistic w/ Vector 0
  std::cout<< toptimumpwr[0] << " ";
  if (toptimumpwr[0] > 2)
  {
      std::cout <<  "<-The Secondary Circuit Condensate Pump is
Not Operating Nominally! " << " "<< std::endl;
  }
  if (toptimumpwr[0] > 20.02972)
  {
      std::cout <<  "<-The Secondary Circuit Condensate Pump is
Operating in a Dangerous State! " << " "<< std::endl;
  }
  // First Bandwidth set Logistic w/ Vector 1
  std::cout<< toptimumpwr[1] << " ";
  if (toptimumpwr[1] > 2)
  {
      std::cout <<  "<-The Secondary Circuit Condensate Flow is
Not  Nominal! " << " "<< std::endl;
```

```
  }
  if (toptimumpwr[1] > 20.02972)
  {
      std::cout <<  "<-The Secondary Circuit Condensate Flowrate
is in a Dangerous State! " << " "<< std::endl;
  }

  // First Bandwidth set Logistic w/ Vector 2
  std::cout<< toptimumpwr[2] << " ";
  if (toptimumpwr[2] > 2)
  {
      std::cout <<  "<-The Steam Generator is Not Operating
Nominally! " << " "<< std::endl;
  }
  if (toptimumpwr[2] > 20.02972)
  {
      std::cout <<  "<-The Steam Generator is in a Dangerous
State! " << " "<< std::endl;
  }

  // First Bandwidth set Logistic w/ Vector 3
  std::cout<< toptimumpwr[3] << " ";
  if (toptimumpwr[3] > 2)
  {
      std::cout <<  "<-The Steam Generator Feedwater Pump is Not
Operating Nominally! " << " "<< std::endl;
  }
  if (toptimumpwr[3] > 20.02972)
  {
      std::cout <<  "<-The Steam Generator Feedwater Pump is
Operating in a Dangerous State! " << " "<< std::endl;
  }


  // First Bandwidth set Logistic w/ Vector 4
  std::cout<< toptimumpwr[4] << " ";
  if (toptimumpwr[4] > 2)
  {
      std::cout <<  "<-The Steam Turbine is Not Operating
Nominally! " << " "<< std::endl;
  }
  if (toptimumpwr[4] > 20.02972)
  {
      std::cout <<  "<-The Steam Turbine is Operating in a
Dangerous State! " << " "<< std::endl;
  }

  // First Bandwidth set Logistic w/ Vector 5
  std::cout<< toptimumpwr[5] << " ";
  if (toptimumpwr[5] > 2)
  {
      std::cout <<  "<-The Steam Flow Rate is Not Nominal! " <<
" "<< std::endl;
  }
  if (toptimumpwr[5] > 20.02972)
  {
      std::cout <<  "<-The Steam Flow Rate is in a Dangerous
State! " << " "<< std::endl;
```

```
  }

  // First Bandwidth set Logistic w/ Vector 6
  std::cout<< toptimumpwr[6] << " ";
  if (toptimumpwr[6] > 2)
  {
      std::cout <<  "<-The Secondary Circuit Temperature is Not
Nominal! " << " "<< std::endl;
  }
  if (toptimumpwr[6] > 20.02972)
  {
      std::cout <<  "<-The Secondary Circuit Temperature is in a
Dangerous State! " << " "<< std::endl;
  }

  // First Bandwidth set Logistic w/ Vector 7
  std::cout<< toptimumpwr[7] << " ";
  if (toptimumpwr[7] > 2)
  {
      std::cout <<  "<-The Secondary Circuit Pressure is Not
Nominal! " << " "<< std::endl;
  }
  if (toptimumpwr[7] > 20.02972)
  {
      std::cout <<  "<-The Secondary Pressure is in a Dangerous
State! " << " "<< std::endl;
  }

  // First Bandwidth set Logistic w/ Vector 8
  std::cout<< toptimumpwr[8] << " ";
  if (toptimumpwr[8] > 2)
  {
      std::cout <<  "<-The Auxillary Steam Generator Feedwater
Pump is Not Operating Nominally! " << " "<< std::endl;
  }
  if (toptimumpwr[8] > 20.02972)
  {
      std::cout <<  "<-The Auxillary Steam Generator Feedwater
Pump is Operating in a Dangerous State! " << " "<< std::endl;
  }

  // First Bandwidth set Logistic w/ Vector 9
  std::cout<< toptimumpwr[9] << " ";
  if (toptimumpwr[9] > 2)
  {
      std::cout <<  "<-The Primary Temperature is Not Nominal! "
<< " "<< std::endl;
  }
  if (toptimumpwr[9] > 20.02972)
  {
      std::cout <<  "<-The Primary Temperature is in a Dangerous
State! " << " "<< std::endl;
  }

  // First Bandwidth set Logistic w/ Vector 10
  std::cout<< toptimumpwr[10] << " ";
  if (toptimumpwr[10] > 2)
  {
```

```
      std::cout <<  "<-The Primary Pressure is Not Nominal! " <<
" "<< std::endl;
  }
  if (toptimumpwr[10] > 20.02972)
  {
      std::cout <<  "<-The Primary Pressure is in a Dangerous
State! " << " "<< std::endl;
  }

  // First Bandwidth set Logistic w/ Vector 11
  std::cout<< toptimumpwr[11] << " ";
  if (toptimumpwr[11] > 2)
  {
      std::cout <<  "<-The Steam Generator Pressure Relief Valve
is Not Operating Nominally! " << " "<< std::endl;
  }
  if (toptimumpwr[11] > 20.02972)
  {
      std::cout <<  "<-The Steam Generator Pressure Relief Valve
is Operating in a Dangerous State! " << " "<< std::endl;
  }

  // First Bandwidth set Logistic w/ Vector 12
  std::cout<< toptimumpwr[12] << " ";
  if (toptimumpwr[12] > 2)
  {
      std::cout <<  "<-The High Pressure Injection Pump is Not
Operating Nominally! " << " "<< std::endl;
  }
  if (toptimumpwr[12] > 20.02972)
  {
      std::cout <<  "<-The High Pressure Injection Pump is
Operating in a Dangerous State! " << " "<< std::endl;
  }

  // First Bandwidth set logistic w/ Vector 13
  std::cout<< toptimumpwr[13] << " ";
  if (toptimumpwr[13] > 2)
  {
      std::cout <<  "<-The Primary Circuit Main Circulation Pump
is Not Operating Nominally! " << " "<< std::endl;
  }
  if (toptimumpwr[13] > 20.02972)
  {
      std::cout <<  "<-The Primary Circuit Main Circulation Pump
is Operating in a Dangerous State! " << " "<< std::endl;
  }

  // First Bandwidth set Logistic w/ Vector 14
  std::cout<< toptimumpwr[14] << " ";
  if (toptimumpwr[14] > 2)
  {
      std::cout <<  "<-The Primary Circuit Flowrate is Not
Nominal! " << " "<< std::endl;
  }
  if (toptimumpwr[14] > 10)
  {
```

```
        std::cout <<  "<-The Primary Circuit Flowrate is in a
    Dangerous State! " << " "<< std::endl;
      }

      // Reactor Power Input a[15]
      std::cout <<  "The Reactor Power is at: " << a[15] << " % " <<
    std::endl << std::endl;

      std::cout << " SYSTEM PARAMETER SUMMATION PENALTY VALUES: " <<
    std::endl << std::endl;
      //Gaussian Weighted vector indices outputs

      //Gaussian Summation Values

      float goptimumsum;
      goptimumsum = goptimumpwr[0] + goptimumpwr[1] + goptimumpwr[2]
    + goptimumpwr[3] + goptimumpwr[4] + goptimumpwr[5] +
    goptimumpwr[6] + goptimumpwr[7] + goptimumpwr[8];
      std::cout << "The Gaussian PWR Realtime Penalty Value is: " <<
    goptimumsum<< std::endl << std::endl;


      //Logistic Summation Values

      float toptimumsum;
      toptimumsum = toptimumpwr[0] + toptimumpwr[1] + toptimumpwr[2]
    + toptimumpwr[3] + toptimumpwr[4] + toptimumpwr[5] +
    toptimumpwr[6] + toptimumpwr[7] + toptimumpwr[8];
      std::cout << "The Logistic PWR Realtime Penalty Value is: " <<
    toptimumsum<< std::endl << std::endl;

}

    return 0;
}
```

APPENDIX E

NRCA THREE MILE ISLAND EXAMPLE OUTPUT

```
Read 144 vectors
numbers read in:
100 400 100 100 1820 285 7 0 100 290 15 0 0 100 13700 90 0 400 100 100
      1820 285 7 0 100 290 15 0 0 100 13700 90 0 100 0 100 1820 285 7
      0 100 290 15 0 0 100 13700 90 0 100 0 20 220 285 7 0 100 290 15
      0 0 100 13700 10 0 100 0 10 220 295 7.5 0 100 290 15 0 0 100
      13700 10 0 100 0 10 220 295 7.5 100 100 290 15 0 0 100 13700 10
      0 100 0 10 220 295 7.5 100 100 340 16 100 0 100 13700 10 0 100 0
      10 220 295 7.5 100 100 340 15 100 100 100 13700 10 0 100 0 10
      220 295 7.5 100 100 340 14 100 100 80 11700 10

  BELOW IS ITERATION #: 1
  ------------------------------------------------------

Secondary Circuit Condensate Pump is Operating at:
100%

Secondary Circuit Condensate Pump Deviation from Nominal Operation is:
      0 %

Secondary Circuit Condensate Flowrate is :
400 l/s

Secondary Circuit Flowrate Deviation from Nominal Operation is: 0 l/s

Steam Generator Feedwater Pump is Operation at:
100 %

Steam Generator Feedwater Pump Deviation from Nominal Operation is: 0
      %

Steam Turbine is Operating at:
100 %

Steam Turbine Deviation from Nominal Operation is: 0 %

Secondary Circuit Steam Flowrate is:
1820 l/s

Secondary Circuit Steam Flowrate Deviation from Nominal Operation is:
      0 l/s

Secondary Circuit Temperature is:
285 Deg C
```

Secondary Circuit Temperature Deviation from Nominal Operation is: 0 Deg C

Secondary Circuit Pressure is:
7 MPa

Secondary Circuit Pressure Deviation from Nominal Operation is: 0 MPa

Auxiallary Steam Generator Feedwater Pump is Operating at:
0 %

Auxiallary Steam Generator Feedwater Pump Deviation from Nominal Operation is: 0 %

Auxiallary Steam Generator Feedwater Valve Closure Position is:
100 %

Auxiallary Steam Generator Feedwater Valve Closure Deviation from Nominal Operation is: 0 %

Primary Circuit Temperature is:
290 Deg C

Primary Circuit Temperature Deviation from Nominal Operation is: 0 Deg C

Primary Circuit Pressure is:
15 MPa

Primary Circuit Pressure Deviation from Nominal Operation is: 0 MPa

Steam Generator Pressure Relief Valve Closure Position is:
0 %

Steam Generator Pressure Relief Valve Closure Deviation from Nominal Operation is: 0 %

Primary Circuit High Pressure Injection Pump is Operating at:
0 %

Primary Circuit High Pressure Injection Pump Deviation from Nominal Operation is: 0 %

Primary Circuit Main Circulation Pump is Operating at:
100 %

Primary Circuit Main Circulation Pump Deviation from Nominal Operation is: 0 %

Primary Circuit Coolant Flowrate is:
13700 l/s

Primary Circuit Coolant Flowrate Deviation from Nominal Operation is: 0 l/s

Reactor Power is:
90 %

Reactor Power Deviation from Nominal Operation is: 0 %

INDIVIDUAL PARAMETER PENALTY VALUES:

Gaussian:
4.02972 4.02972 4.02972 4.02972 4.02972 4.02972 4.02972 4.02972 4.02972 4.02972 4.02972 4.02972 4.02972 4.02972 4.02972 The Reactor Power is at: 90 %

Logistic:
2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 The Reactor Power is at: 90 %

SYSTEM PARAMETER SUMMATION PENALTY VALUES:

The Gaussian PWR Realtime Penalty Value is: 36.2675

The Logistic PWR Realtime Penalty Value is: 18

BELOW IS ITERATION #: 2

Secondary Circuit Condensate Pump is Operating at:
0%

Secondary Circuit Condensate Pump Deviation from Nominal Operation is: -100 %

Secondary Circuit Condensate Flowrate is :
400 l/s

Secondary Circuit Flowrate Deviation from Nominal Operation is: 0 l/s

Steam Generator Feedwater Pump is Operation at:
100 %

Steam Generator Feedwater Pump Deviation from Nominal Operation is: 0 %

Steam Turbine is Operating at:
100 %

Steam Turbine Deviation from Nominal Operation is: 0 %

Secondary Circuit Steam Flowrate is:
1820 l/s

Secondary Circuit Steam Flowrate Deviation from Nominal Operation is: 0 l/s

Secondary Circuit Temperature is:
285 Deg C

Secondary Circuit Temperature Deviation from Nominal Operation is: 0 Deg C
Secondary Circuit Pressure is:
7 MPa

Secondary Circuit Pressure Deviation from Nominal Operation is: 0 MPa

Auxiallary Steam Generator Feedwater Pump is Operating at:
0 %

Auxiallary Steam Generator Feedwater Pump Deviation from Nominal Operation is: 0 %

Auxiallary Steam Generator Feedwater Valve Closure Position is:
100 %

Auxiallary Steam Generator Feedwater Valve Closure Deviation from Nominal Operation is: 0 %

Primary Circuit Temperature is:
290 Deg C

Primary Circuit Temperature Deviation from Nominal Operation is: 0 Deg C

Primary Circuit Pressure is:
15 MPa

Primary Circuit Pressure Deviation from Nominal Operation is: 0 MPa

Steam Generator Pressure Relief Valve Closure Position is:
0 %

Steam Generator Pressure Relief Valve Closure Deviation from Nominal Operation is: 0 %

Primary Circuit High Pressure Injection Pump is Operating at:
0 %

Primary Circuit High Pressure Injection Pump Deviation from Nominal Operation is: 0 %

Primary Circuit Main Circulation Pump is Operating at:
100 %

Primary Circuit Main Circulation Pump Deviation from Nominal Operation is: 0 %

Primary Circuit Coolant Flowrate is:
13700 l/s

Primary Circuit Coolant Flowrate Deviation from Nominal Operation is: 0 l/s

Reactor Power is:
90 %

Reactor Power Deviation from Nominal Operation is: 0 %

INDIVIDUAL PARAMETER PENALTY VALUES:

Gaussian:

inf <-The Secondary Circuit Condensate Pump is Not Operating Nominally!
<-The Secondary Circuit Condensate Pump is Operating in a Dangerous State!
4.02972 4.02972 4.02972 4.02972 4.02972 4.02972 4.02972 4.02972 4.02972 4.02972 4.02972 4.02972 4.02972 4.02972 The Reactor Power is at: 90 %

Logistic:
2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 The Reactor Power is at: 90 %

SYSTEM PARAMETER SUMMATION PENALTY VALUES:

The Gaussian PWR Realtime Penalty Value is: inf

The Logistic PWR Realtime Penalty Value is: 18

BELOW IS ITERATION #: 3

Secondary Circuit Condensate Pump is Operating at:
0%

Secondary Circuit Condensate Pump Deviation from Nominal Operation is: -100 %

Secondary Circuit Condensate Flowrate is :
100 l/s

Secondary Circuit Flowrate Deviation from Nominal Operation is: -300 l/s

Steam Generator Feedwater Pump is Operation at:
0 %

Steam Generator Feedwater Pump Deviation from Nominal Operation is: -100 %

Steam Turbine is Operating at:
100 %

Steam Turbine Deviation from Nominal Operation is: 0 %

Secondary Circuit Steam Flowrate is:
1820 l/s

Secondary Circuit Steam Flowrate Deviation from Nominal Operation is: 0 l/s

Secondary Circuit Temperature is:
285 Deg C

Secondary Circuit Temperature Deviation from Nominal Operation is: 0 Deg C

Secondary Circuit Pressure is:

7 MPa

Secondary Circuit Pressure Deviation from Nominal Operation is: 0 MPa

Auxiallary Steam Generator Feedwater Pump is Operating at:
0 %

Auxiallary Steam Generator Feedwater Pump Deviation from Nominal Operation is: 0 %

Auxiallary Steam Generator Feedwater Valve Closure Position is:
100 %

Auxiallary Steam Generator Feedwater Valve Closure Deviation from Nominal Operation is: 0 %

Primary Circuit Temperature is:
290 Deg C

Primary Circuit Temperature Deviation from Nominal Operation is: 0 Deg C

Primary Circuit Pressure is:
15 MPa

Primary Circuit Pressure Deviation from Nominal Operation is: 0 MPa

Steam Generator Pressure Relief Valve Closure Position is:
0 %

Steam Generator Pressure Relief Valve Closure Deviation from Nominal Operation is: 0 %

Primary Circuit High Pressure Injection Pump is Operating at:
0 %

Primary Circuit High Pressure Injection Pump Deviation from Nominal Operation is: 0 %

Primary Circuit Main Circulation Pump is Operating at:
100 %

Primary Circuit Main Circulation Pump Deviation from Nominal Operation is: 0 %

Primary Circuit Coolant Flowrate is:
13700 l/s

Primary Circuit Coolant Flowrate Deviation from Nominal Operation is: 0 l/s

Reactor Power is:
90 %

Reactor Power Deviation from Nominal Operation is: 0 %

INDIVIDUAL PARAMETER PENALTY VALUES:

```
Gaussian:
inf <-The Secondary Circuit Condensate Pump is Not Operating
      Nominally!
<-The Secondary Circuit Condensate Pump is Operating in a Dangerous
      State!
inf <-The Secondary Circuit Condensate Flow is Not  Nominal!
<-The Secondary Circuit Condensate Flowrate is in a Dangerous State!
inf <-The Steam Generator is Not Operating Nominally!
<-The Steam Generator is in a Dangerous State!
4.02972 4.02972 4.02972 4.02972 4.02972 4.02972 4.02972 4.02972
      4.02972 4.02972 4.02972 4.02972 The Reactor Power is at: 90 %

Logistic:
2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 The Reactor Power is at: 90 %

 SYSTEM PARAMETER SUMMATION PENALTY VALUES:

The Gaussian PWR Realtime Penalty Value is: inf

The Logistic PWR Realtime Penalty Value is: 18



  BELOW IS ITERATION #: 4
  ------------------------------------------------------

Secondary Circuit Condensate Pump is Operating at:
0%

Secondary Circuit Condensate Pump Deviation from Nominal Operation is:
      -100 %

Secondary Circuit Condensate Flowrate is :
100 l/s

Secondary Circuit Flowrate Deviation from Nominal Operation is: -300
      l/s

Steam Generator Feedwater Pump is Operation at:
0 %

Steam Generator Feedwater Pump Deviation from Nominal Operation is: -
      100 %

Steam Turbine is Operating at:
20 %

Steam Turbine Deviation from Nominal Operation is: -80 %

Secondary Circuit Steam Flowrate is:
220 l/s

Secondary Circuit Steam Flowrate Deviation from Nominal Operation is:
      -1600 l/s

Secondary Circuit Temperature is:
285 Deg C
```

Secondary Circuit Temperature Deviation from Nominal Operation is: 0 Deg C

Secondary Circuit Pressure is:
7 MPa

Secondary Circuit Pressure Deviation from Nominal Operation is: 0 MPa

Auxiallary Steam Generator Feedwater Pump is Operating at:
0 %

Auxiallary Steam Generator Feedwater Pump Deviation from Nominal Operation is: 0 %

Auxiallary Steam Generator Feedwater Valve Closure Position is:
100 %

Auxiallary Steam Generator Feedwater Valve Closure Deviation from Nominal Operation is: 0 %

Primary Circuit Temperature is:
290 Deg C

Primary Circuit Temperature Deviation from Nominal Operation is: 0 Deg C

Primary Circuit Pressure is:
15 MPa

Primary Circuit Pressure Deviation from Nominal Operation is: 0 MPa

Steam Generator Pressure Relief Valve Closure Position is:
0 %

Steam Generator Pressure Relief Valve Closure Deviation from Nominal Operation is: 0 %

Primary Circuit High Pressure Injection Pump is Operating at:
0 %

Primary Circuit High Pressure Injection Pump Deviation from Nominal Operation is: 0 %

Primary Circuit Main Circulation Pump is Operating at:
100 %

Primary Circuit Main Circulation Pump Deviation from Nominal Operation is: 0 %

Primary Circuit Coolant Flowrate is:
13700 l/s

Primary Circuit Coolant Flowrate Deviation from Nominal Operation is: 0 l/s

Reactor Power is:
10 %

```
Reactor Power Deviation from Nominal Operation is: -80 %

 INDIVIDUAL PARAMETER PENALTY VALUES:

Gaussian:
inf <-The Secondary Circuit Condensate Pump is Not Operating
      Nominally!
<-The Secondary Circuit Condensate Pump is Operating in a Dangerous
      State!
inf <-The Secondary Circuit Condensate Flow is Not  Nominal!
<-The Secondary Circuit Condensate Flowrate is in a Dangerous State!
inf <-The Steam Generator is Not Operating Nominally!
<-The Steam Generator is in a Dangerous State!
inf <-The Steam Generator Feedwater Pump is Not Operating Nominally!
<-The Steam Generator Feedwater Pump is Operating in a Dangerous
      State!
inf <-The Steam Turbine is Not Operating Nominally!
<-The Steam Turbine is Operating in a Dangerous State!
4.02972 4.02972 4.02972 4.02972 4.02972 4.02972 4.02972 4.02972
      4.02972 4.02972 The Reactor Power is at: 90 %

Logistic:
2 2 2 2 2 2 2 2 2 2 2 2 2 2 2 The Reactor Power is at: 90 %

 SYSTEM PARAMETER SUMMATION PENALTY VALUES:

The Gaussian PWR Realtime Penalty Value is: inf

The Logistic PWR Realtime Penalty Value is: 18



  BELOW IS ITERATION #: 5
  ------------------------------------------------------

Secondary Circuit Condensate Pump is Operating at:
0%

Secondary Circuit Condensate Pump Deviation from Nominal Operation is:
      -100 %

Secondary Circuit Condensate Flowrate is :
100 l/s

Secondary Circuit Flowrate Deviation from Nominal Operation is: -300
      l/s

Steam Generator Feedwater Pump is Operation at:
0 %

Steam Generator Feedwater Pump Deviation from Nominal Operation is: -
      100 %

Steam Turbine is Operating at:
10 %

Steam Turbine Deviation from Nominal Operation is: -90 %
```

Secondary Circuit Steam Flowrate is:
220 l/s

Secondary Circuit Steam Flowrate Deviation from Nominal Operation is: -1600 l/s

Secondary Circuit Temperature is:
295 Deg C

Secondary Circuit Temperature Deviation from Nominal Operation is: 10 Deg C

Secondary Circuit Pressure is:
7.5 MPa

Secondary Circuit Pressure Deviation from Nominal Operation is: 0.5 MPa

Auxiallary Steam Generator Feedwater Pump is Operating at:
0 %

Auxiallary Steam Generator Feedwater Pump Deviation from Nominal Operation is: 0 %

Auxiallary Steam Generator Feedwater Valve Closure Position is:
100 %

Auxiallary Steam Generator Feedwater Valve Closure Deviation from Nominal Operation is: 0 %

Primary Circuit Temperature is:
290 Deg C

Primary Circuit Temperature Deviation from Nominal Operation is: 0 Deg C

Primary Circuit Pressure is:
15 MPa

Primary Circuit Pressure Deviation from Nominal Operation is: 0 MPa

Steam Generator Pressure Relief Valve Closure Position is:
0 %

Steam Generator Pressure Relief Valve Closure Deviation from Nominal Operation is: 0 %

Primary Circuit High Pressure Injection Pump is Operating at:
0 %

Primary Circuit High Pressure Injection Pump Deviation from Nominal Operation is: 0 %

Primary Circuit Main Circulation Pump is Operating at:
100 %

Primary Circuit Main Circulation Pump Deviation from Nominal Operation is: 0 %

Primary Circuit Coolant Flowrate is:
13700 l/s

Primary Circuit Coolant Flowrate Deviation from Nominal Operation is:
0 l/s

Reactor Power is:
10 %

Reactor Power Deviation from Nominal Operation is: -80 %

INDIVIDUAL PARAMETER PENALTY VALUES:

Gaussian:
inf <-The Secondary Circuit Condensate Pump is Not Operating Nominally!
<-The Secondary Circuit Condensate Pump is Operating in a Dangerous State!
inf <-The Secondary Circuit Condensate Flow is Not Nominal!
<-The Secondary Circuit Condensate Flowrate is in a Dangerous State!
inf <-The Steam Generator is Not Operating Nominally!
<-The Steam Generator is in a Dangerous State!
inf <-The Steam Generator Feedwater Pump is Not Operating Nominally!
<-The Steam Generator Feedwater Pump is Operating in a Dangerous State!
inf <-The Steam Turbine is Not Operating Nominally!
<-The Steam Turbine is Operating in a Dangerous State!
inf <-The Steam Flow Rate is Not Nominal!
<-The Steam Flow Rate is in a Dangerous State!
1.39373e+06 <-The Secondary Circuit Temperature is Not Nominal!
<-The Secondary Circuit Temperature is in a Dangerous State!
4.02972 4.02972 4.02972 4.02972 4.02972 4.02972 4.02972 4.02972 The Reactor Power is at: 90 %

Logistic:
2 2 2 2 2 0 2 2 2 2 2 2 2 2 2 The Reactor Power is at: 90 %

SYSTEM PARAMETER SUMMATION PENALTY VALUES:

The Gaussian PWR Realtime Penalty Value is: inf

The Logistic PWR Realtime Penalty Value is: 16

BELOW IS ITERATION #: 6

Secondary Circuit Condensate Pump is Operating at:
0%

Secondary Circuit Condensate Pump Deviation from Nominal Operation is:
-100 %

Secondary Circuit Condensate Flowrate is :
100 l/s

Secondary Circuit Flowrate Deviation from Nominal Operation is: -300 l/s

Steam Generator Feedwater Pump is Operation at:
0 %

Steam Generator Feedwater Pump Deviation from Nominal Operation is: -100 %
Steam Turbine is Operating at:
10 %

Steam Turbine Deviation from Nominal Operation is: -90 %

Secondary Circuit Steam Flowrate is:
220 l/s

Secondary Circuit Steam Flowrate Deviation from Nominal Operation is: -1600 l/s

Secondary Circuit Temperature is:
295 Deg C

Secondary Circuit Temperature Deviation from Nominal Operation is: 10 Deg C

Secondary Circuit Pressure is:
7.5 MPa

Secondary Circuit Pressure Deviation from Nominal Operation is: 0.5 MPa

Auxiallary Steam Generator Feedwater Pump is Operating at:
100 %

Auxiallary Steam Generator Feedwater Pump Deviation from Nominal Operation is: 100 %

Auxiallary Steam Generator Feedwater Valve Closure Position is:
100 %

Auxiallary Steam Generator Feedwater Valve Closure Deviation from Nominal Operation is: 0 %

Primary Circuit Temperature is:
290 Deg C

Primary Circuit Temperature Deviation from Nominal Operation is: 0 Deg C

Primary Circuit Pressure is:
15 MPa

Primary Circuit Pressure Deviation from Nominal Operation is: 0 MPa

Steam Generator Pressure Relief Valve Closure Position is:
0 %

Steam Generator Pressure Relief Valve Closure Deviation from Nominal Operation is: 0 %

Primary Circuit High Pressure Injection Pump is Operating at:
0 %

Primary Circuit High Pressure Injection Pump Deviation from Nominal Operation is: 0 %

Primary Circuit Main Circulation Pump is Operating at:
100 %

Primary Circuit Main Circulation Pump Deviation from Nominal Operation is: 0 %

Primary Circuit Coolant Flowrate is:
13700 l/s

Primary Circuit Coolant Flowrate Deviation from Nominal Operation is: 0 l/s

Reactor Power is:
10 %

Reactor Power Deviation from Nominal Operation is: -80 %

INDIVIDUAL PARAMETER PENALTY VALUES:

Gaussian:
inf <-The Secondary Circuit Condensate Pump is Not Operating Nominally!
<-The Secondary Circuit Condensate Pump is Operating in a Dangerous State!
inf <-The Secondary Circuit Condensate Flow is Not Nominal!
<-The Secondary Circuit Condensate Flowrate is in a Dangerous State!
inf <-The Steam Generator is Not Operating Nominally!
<-The Steam Generator is in a Dangerous State!
inf <-The Steam Generator Feedwater Pump is Not Operating Nominally!
<-The Steam Generator Feedwater Pump is Operating in a Dangerous State!
inf <-The Steam Turbine is Not Operating Nominally!
<-The Steam Turbine is Operating in a Dangerous State!
inf <-The Steam Flow Rate is Not Nominal!
<-The Steam Flow Rate is in a Dangerous State!
1.39373e+06 <-The Secondary Circuit Temperature is Not Nominal!
<-The Secondary Circuit Temperature is in a Dangerous State!
inf <-The Secondary Circuit Pressure is Not Nominal!
<-The Secondary Pressure is in a Dangerous State!
4.02972 4.02972 4.02972 4.02972 4.02972 4.02972 4.02972 The Reactor Power is at: 90 %

Logistic:
2 2 2 2 2 0 2 nan 2 2 2 2 2 2 2 The Reactor Power is at: 90 %

SYSTEM PARAMETER SUMMATION PENALTY VALUES:

The Gaussian PWR Realtime Penalty Value is: inf

The Logistic PWR Realtime Penalty Value is: nan

BELOW IS ITERATION #: 7

Secondary Circuit Condensate Pump is Operating at:
0%

Secondary Circuit Condensate Pump Deviation from Nominal Operation is:
-100 %

Secondary Circuit Condensate Flowrate is :
100 l/s

Secondary Circuit Flowrate Deviation from Nominal Operation is: -300
l/s

Steam Generator Feedwater Pump is Operation at:
0 %

Steam Generator Feedwater Pump Deviation from Nominal Operation is: -
100 %

Steam Turbine is Operating at:
10 %

Steam Turbine Deviation from Nominal Operation is: -90 %

Secondary Circuit Steam Flowrate is:
220 l/s

Secondary Circuit Steam Flowrate Deviation from Nominal Operation is:
-1600 l/s

Secondary Circuit Temperature is:
295 Deg C

Secondary Circuit Temperature Deviation from Nominal Operation is: 10
Deg C

Secondary Circuit Pressure is:
7.5 MPa

Secondary Circuit Pressure Deviation from Nominal Operation is: 0.5
MPa

Auxiallary Steam Generator Feedwater Pump is Operating at:
100 %

Auxiallary Steam Generator Feedwater Pump Deviation from Nominal
Operation is: 100 %

Auxiallary Steam Generator Feedwater Valve Closure Position is:
100 %

```
Auxiliary Steam Generator Feedwater Valve Closure Deviation from
      Nominal Operation is: 0 %

Primary Circuit Temperature is:
340 Deg C

Primary Circuit Temperature Deviation from Nominal Operation is: 50
      Deg C

Primary Circuit Pressure is:
16 MPa

Primary Circuit Pressure Deviation from Nominal Operation is: 1 MPa

Steam Generator Pressure Relief Valve Closure Position is:
100 %

Steam Generator Pressure Relief Valve Closure Deviation from Nominal
      Operation is: 100 %

Primary Circuit High Pressure Injection Pump is Operating at:
0 %

Primary Circuit High Pressure Injection Pump Deviation from Nominal
      Operation is: 0 %

Primary Circuit Main Circulation Pump is Operating at:
100 %

Primary Circuit Main Circulation Pump Deviation from Nominal Operation
      is: 0 %

Primary Circuit Coolant Flowrate is:
13700 l/s

Primary Circuit Coolant Flowrate Deviation from Nominal Operation is:
      0 l/s

Reactor Power is:
10 %

Reactor Power Deviation from Nominal Operation is: -80 %

 INDIVIDUAL PARAMETER PENALTY VALUES:

Gaussian:
inf <-The Secondary Circuit Condensate Pump is Not Operating
      Nominally!
<-The Secondary Circuit Condensate Pump is Operating in a Dangerous
      State!
inf <-The Secondary Circuit Condensate Flow is Not  Nominal!
<-The Secondary Circuit Condensate Flowrate is in a Dangerous State!
inf <-The Steam Generator is Not Operating Nominally!
<-The Steam Generator is in a Dangerous State!
inf <-The Steam Generator Feedwater Pump is Not Operating Nominally!
<-The Steam Generator Feedwater Pump is Operating in a Dangerous
      State!
inf <-The Steam Turbine is Not Operating Nominally!
```

```
<-The Steam Turbine is Operating in a Dangerous State!
inf <-The Steam Flow Rate is Not Nominal!
<-The Steam Flow Rate is in a Dangerous State!
1.39373e+06 <-The Secondary Circuit Temperature is Not Nominal!
<-The Secondary Circuit Temperature is in a Dangerous State!
inf <-The Secondary Circuit Pressure is Not Nominal!
<-The Secondary Pressure is in a Dangerous State!
4.02972 inf <-The Primary Temperature is Not Nominal!
<-The Primary Temperature is in a Dangerous State!
5.76628e+22 <-The Primary Pressure is Not Nominal!
<-The Primary Pressure is in a Dangerous State!
inf <-The Steam Generator Pressure Relief Valve is Not Operating
      Nominally!
<-The Steam Generator Pressure Relief Valve is Operating in a
      Dangerous State!
4.02972 4.02972 4.02972 The Reactor Power is at: 90 %

Logistic:
2 2 2 2 2 0 2 nan 2 0 2 nan 2 2 2 The Reactor Power is at: 90 %

 SYSTEM PARAMETER SUMMATION PENALTY VALUES:

The Gaussian PWR Realtime Penalty Value is: inf

The Logistic PWR Realtime Penalty Value is: nan



  BELOW IS ITERATION #: 8
  -------------------------------------------------------

Secondary Circuit Condensate Pump is Operating at:
0%

Secondary Circuit Condensate Pump Deviation from Nominal Operation is:
      -100 %

Secondary Circuit Condensate Flowrate is :
100 l/s

Secondary Circuit Flowrate Deviation from Nominal Operation is: -300
      l/s

Steam Generator Feedwater Pump is Operation at:
0 %

Steam Generator Feedwater Pump Deviation from Nominal Operation is: -
      100 %

Steam Turbine is Operating at:
10 %

Steam Turbine Deviation from Nominal Operation is: -90 %

Secondary Circuit Steam Flowrate is:
220 l/s
```

Secondary Circuit Steam Flowrate Deviation from Nominal Operation is: -1600 l/s

Secondary Circuit Temperature is:
295 Deg C

Secondary Circuit Temperature Deviation from Nominal Operation is: 10 Deg C

Secondary Circuit Pressure is:
7.5 MPa

Secondary Circuit Pressure Deviation from Nominal Operation is: 0.5 MPa

Auxiallary Steam Generator Feedwater Pump is Operating at:
100 %

Auxiallary Steam Generator Feedwater Pump Deviation from Nominal Operation is: 100 %

Auxiallary Steam Generator Feedwater Valve Closure Position is:
100 %

Auxiallary Steam Generator Feedwater Valve Closure Deviation from Nominal Operation is: 0 %

Primary Circuit Temperature is:
340 Deg C

Primary Circuit Temperature Deviation from Nominal Operation is: 50 Deg C

Primary Circuit Pressure is:
15 MPa

Primary Circuit Pressure Deviation from Nominal Operation is: 0 MPa

Steam Generator Pressure Relief Valve Closure Position is:
100 %

Steam Generator Pressure Relief Valve Closure Deviation from Nominal Operation is: 100 %

Primary Circuit High Pressure Injection Pump is Operating at:
100 %

Primary Circuit High Pressure Injection Pump Deviation from Nominal Operation is: 100 %

Primary Circuit Main Circulation Pump is Operating at:
100 %

Primary Circuit Main Circulation Pump Deviation from Nominal Operation is: 0 %

Primary Circuit Coolant Flowrate is:
13700 l/s

```
Primary Circuit Coolant Flowrate Deviation from Nominal Operation is:
      0 l/s

Reactor Power is:
10 %

Reactor Power Deviation from Nominal Operation is: -80 %

 INDIVIDUAL PARAMETER PENALTY VALUES:

Gaussian:
inf <-The Secondary Circuit Condensate Pump is Not Operating
      Nominally!
<-The Secondary Circuit Condensate Pump is Operating in a Dangerous
      State!
inf <-The Secondary Circuit Condensate Flow is Not  Nominal!
<-The Secondary Circuit Condensate Flowrate is in a Dangerous State!
inf <-The Steam Generator is Not Operating Nominally!
<-The Steam Generator is in a Dangerous State!
inf <-The Steam Generator Feedwater Pump is Not Operating Nominally!
<-The Steam Generator Feedwater Pump is Operating in a Dangerous
      State!
inf <-The Steam Turbine is Not Operating Nominally!
<-The Steam Turbine is Operating in a Dangerous State!
inf <-The Steam Flow Rate is Not Nominal!
<-The Steam Flow Rate is in a Dangerous State!
1.39373e+06 <-The Secondary Circuit Temperature is Not Nominal!
<-The Secondary Circuit Temperature is in a Dangerous State!
inf <-The Secondary Circuit Pressure is Not Nominal!
<-The Secondary Pressure is in a Dangerous State!
4.02972 inf <-The Primary Temperature is Not Nominal!
<-The Primary Temperature is in a Dangerous State!
4.02972 inf <-The Steam Generator Pressure Relief Valve is Not
      Operating Nominally!
<-The Steam Generator Pressure Relief Valve is Operating in a
      Dangerous State!
inf <-The High Pressure Injection Pump is Not Operating Nominally!
<-The High Pressure Injection Pump is Operating in a Dangerous State!
4.02972 4.02972 The Reactor Power is at: 90 %

Logistic:
2 2 2 2 2 0 2 nan 2 0 2 nan nan 2 2 The Reactor Power is at: 90 %

 SYSTEM PARAMETER SUMMATION PENALTY VALUES:

The Gaussian PWR Realtime Penalty Value is: inf

The Logistic PWR Realtime Penalty Value is: nan



  BELOW IS ITERATION #: 9
  ------------------------------------------------------

Secondary Circuit Condensate Pump is Operating at:
0%
```

Secondary Circuit Condensate Pump Deviation from Nominal Operation is: -100 %

Secondary Circuit Condensate Flowrate is :
100 l/s

Secondary Circuit Flowrate Deviation from Nominal Operation is: -300 l/s

Steam Generator Feedwater Pump is Operation at:
0 %

Steam Generator Feedwater Pump Deviation from Nominal Operation is: -100 %

Steam Turbine is Operating at:
10 %

Steam Turbine Deviation from Nominal Operation is: -90 %

Secondary Circuit Steam Flowrate is:
220 l/s

Secondary Circuit Steam Flowrate Deviation from Nominal Operation is: -1600 l/s

Secondary Circuit Temperature is:
295 Deg C

Secondary Circuit Temperature Deviation from Nominal Operation is: 10 Deg C

Secondary Circuit Pressure is:
7.5 MPa

Secondary Circuit Pressure Deviation from Nominal Operation is: 0.5 MPa

Auxiallary Steam Generator Feedwater Pump is Operating at:
100 %

Auxiallary Steam Generator Feedwater Pump Deviation from Nominal Operation is: 100 %

Auxiallary Steam Generator Feedwater Valve Closure Position is:
100 %

Auxiallary Steam Generator Feedwater Valve Closure Deviation from Nominal Operation is: 0 %

Primary Circuit Temperature is:
340 Deg C

Primary Circuit Temperature Deviation from Nominal Operation is: 50 Deg C

Primary Circuit Pressure is:
14 MPa

```
Primary Circuit Pressure Deviation from Nominal Operation is: -1 MPa

Steam Generator Pressure Relief Valve Closure Position is:
100 %

Steam Generator Pressure Relief Valve Closure Deviation from Nominal
      Operation is: 100 %

Primary Circuit High Pressure Injection Pump is Operating at:
100 %

Primary Circuit High Pressure Injection Pump Deviation from Nominal
      Operation is: 100 %

Primary Circuit Main Circulation Pump is Operating at:
80 %

Primary Circuit Main Circulation Pump Deviation from Nominal Operation
      is: -20 %

Primary Circuit Coolant Flowrate is:
11700 l/s

Primary Circuit Coolant Flowrate Deviation from Nominal Operation is:
      -2000 l/s

Reactor Power is:
10 %

Reactor Power Deviation from Nominal Operation is: -80 %

 INDIVIDUAL PARAMETER PENALTY VALUES:

Gaussian:
inf <-The Secondary Circuit Condensate Pump is Not Operating
      Nominally!
<-The Secondary Circuit Condensate Pump is Operating in a Dangerous
      State!
inf <-The Secondary Circuit Condensate Flow is Not  Nominal!
<-The Secondary Circuit Condensate Flowrate is in a Dangerous State!
inf <-The Steam Generator is Not Operating Nominally!
<-The Steam Generator is in a Dangerous State!
inf <-The Steam Generator Feedwater Pump is Not Operating Nominally!
<-The Steam Generator Feedwater Pump is Operating in a Dangerous
      State!
inf <-The Steam Turbine is Not Operating Nominally!
<-The Steam Turbine is Operating in a Dangerous State!
inf <-The Steam Flow Rate is Not Nominal!
<-The Steam Flow Rate is in a Dangerous State!
1.39373e+06 <-The Secondary Circuit Temperature is Not Nominal!
<-The Secondary Circuit Temperature is in a Dangerous State!
inf <-The Secondary Circuit Pressure is Not Nominal!
<-The Secondary Pressure is in a Dangerous State!
4.02972 inf <-The Primary Temperature is Not Nominal!
<-The Primary Temperature is in a Dangerous State!
5.76628e+22 <-The Primary Pressure is Not Nominal!
<-The Primary Pressure is in a Dangerous State!
```

```
inf <-The Steam Generator Pressure Relief Valve is Not Operating
      Nominally!
<-The Steam Generator Pressure Relief Valve is Operating in a
      Dangerous State!
inf <-The High Pressure Injection Pump is Not Operating Nominally!
<-The High Pressure Injection Pump is Operating in a Dangerous State!
inf <-The Primary Circuit Main Circulation Pump is Not Operating
      Nominally!
<-The Primary Circuit Main Circulation Pump is Operating in a
      Dangerous State!
inf <-The Primary Circuit Flowrate is Not Nominal!
<-The Primary Circuit Flowrate is in a Dangerous State!
The Reactor Power is at: 90 %

Logistic:
2 2 2 2 2 0 2 nan 2 0 2 nan nan 2 2 The Reactor Power is at: 90 %

 SYSTEM PARAMETER SUMMATION PENALTY VALUES:

The Gaussian PWR Realtime Penalty Value is: inf

The Logistic PWR Realtime Penalty Value is: nan
```

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