Reducing Occupational Radiation Dose by Improving the Operational Flexibility of the Reactor Water Cleanup System at Pilgrim Nuclear Power Station

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Reducing Occupational Radiation Dose by Improving the Operational Flexibility of the Reactor Water Cleanup System at Pilgrim Nuclear Power Station
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ABSTRACT

The goal of this project was to develop a piping modification to allow the Reactor Water Cleanup system at Pilgrim Nuclear Power Station to function during a refueling outage. The process of shutting down the reactor before the refueling outage disturbs an oxide layer of radioactive contaminants that is settled on the primary piping of the reactor and releases it into the reactor coolant. The practice employed to remove the highly radioactive decontaminants before the refueling outage was a manual process that caused nuclear workers to obtain a higher occupational radiation dose. The normal reactor coolant decontamination system is shut down during a refueling outage due to maintenance; therefore, it was requested that a piping modification be developed to allow the operation of this system, the Reactor Water Cleanup System, during a refueling outage. The modification designed in this project connected the Reactor Water Cleanup System with the Fuel Pool Cooling system, allowing operation of the Reactor Water Cleanup system during a refueling outage and introducing a new mode of operation titled Alternate Injection Mode. This could eliminate the need for the original manual cleaning process and was estimated to reduce occupational dose by 2,000 mrem every refueling outage.
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1. INTRODUCTION

Nuclear power is a major emission-free electricity source in the United States and many other parts of the globe. Coal is the only fuel source that produces more electricity than nuclear power, but burning coal produces harmful greenhouse gases. Electricity from nuclear power units avoids the production of greenhouse gases (See Glossary). In 2011 alone, the Environmental Protection Agency [EPA] estimated that Pilgrim Nuclear Power Station avoided 4,000 tons of sulfur dioxide, 800 tons of nitrogen oxide, and two million tons of Carbon Dioxide (Entergy® , 2012). Nuclear power is also very sustainable meaning fuel supplies are not in short supply, especially with the prospect of breeding fuel and while the waste is hazardous, it has a very small volume comparative to other power sources.

Though producing electricity by means of a nuclear power plant is considered more environmentally friendly, there are a number of challenges involved with safely operating a nuclear power plant. It is no secret there is an occupational radiation dose involved with operating a nuclear power plant. It is of utmost importance to keep occupational radiation dose as low as reasonably achievable.

To gain unescorted access to Pilgrim Nuclear Power Station, each member of the team had to undergo 33 nuclear safety certifications and attend a 4-hour nuclear radiation worker practicum. As part of the training required for unescorted access to any nuclear power plant, nuclear workers learn in great detail about radiation and steps taken to reduce dose. The background reading also covers pertinent information regarding radiation.
The United States has 103 operational nuclear units (A. Andrews, 2011). According the US Census Bureau, Massachusetts has exactly 2,818,940 homes, and Pilgrim Nuclear Power Plant provides electricity to approximately 680,000 homes (J. Lindsey, 2012). Therefore, Pilgrim Nuclear Power Station provides electricity for approximately 24 percent of Massachusetts.

Pilgrim Nuclear Power Station, the only nuclear power station in Massachusetts, began operating commercially in 1972 (Entergy®, 2012). In 2012, Pilgrim’s license was renewed for twenty years by the United States Nuclear Regulatory Commission [USNRC] (Entergy®, 2012). Pilgrim Nuclear Power Station’s only reactor is a GE Mark I Type 3 boiling water reactor [BWR] (Entergy®, 2012). Additionally, the reactor has a 685 megawatt electric output (Nuclear Information and Resource Service, 2011).

During a refueling outage, Pilgrim Nuclear Power Station faces a difficult task of optimizing cost, worker dosage, and safety concerns in a short amount of time. The goal of an outage, in addition to refueling the core and correcting any mechanical and non-mechanical issues, is to minimize the total outage time and reducing worker dose. To start a refueling outage, the reactor needs to be shutdown. The traditional method of shutting down at Pilgrim Station was slow and costly. New procedures use a scramming process to minimize the time it takes to really begin the outage while also minimizing the total time in the outage. Despite this improved way of shutting down a reactor, an unintended consequence is release of debris in the water and is further compounded by the shutdown of the Reactor Water Cleanup (RWCU) system and its cleanup function during refueling outages. The RWCU is a non-safety related system that does not assist reactor shutdown but are vital in maintaining normal operation mode (M300, 2013).
The main functions of the RWCU are to purify reactor water during the running times and to remove small radioactive and non-radioactive particle debris from the reactor water. The inoperability of the RWCU during refueling outages is leading to a less optimal outage time, added complexity in purifying reactor coolant during outage, and increased radiation levels at the start of refueling outages.

The goal of the project was to develop a design and cost estimate for a piping modification to the Reactor Water Clean Up system that would address the above mentioned operational challenges during refueling outages, improve operational flexibility, reduce outage time, and reduce radiation levels during fuel outages, and filter out more Co-60 from the reactor.
2. BACKGROUND

In order to modify the design of the Reactor Water Clean Up system, the main functions of the system were evaluated. Technical documents were examined to establish the function of the RWCU in relation to the overall function of the reactor and nuclear plant. Both normal operation and refueling operation system function had to be understood in order to properly modify the operational flexibility of the RWCU. Pilgrim Nuclear Power Station’s reference and training text provided detailed information regarding even the smallest components of the RWCU. The background section covers the function of BWRs, the function of the RWCU with respect the reactor, the primary functions and components of the RWCU at Pilgrim Nuclear Power Station, the function of a BWR during a refueling outage, and sources radiation during a refueling outage. In addition, a previously proposed modification to the RWCU to allow function during refueling is presented.

2.1 Design of a Boiling Water Reactor

As previously mentioned, Pilgrim Nuclear Power Station only has one reactor. The sole reactor at Pilgrim is a General Electric [GE] Mark I Type three BWR (Andrews, 2011). GE created the BWR design in the mid-1950s. In fact, the design is employed by thirty-five BWR’s across the country (Andrews, 2011). Mark-I refers to the containment system of the reactor. Because Pilgrim Nuclear Power Station’s reactor began operating in 1972, it has the oldest containment system.

In order to understand the necessity of the RWCU system, the function of the BWR must be explained. The function of a BWR is surprisingly simple and can be explained in five steps. First, water acting as reactor coolant absorbs the heat created by
the fission of enriched uranium in the core of the reactor (USRNC, 2012). The heat from
the reactor converts water into a mixture of vapor and liquid. The steam is separated from
recirculation water, dried in the dryer located at the top of the vessel, and directed into the
steam line connected to the turbine (GE, 1980). The steam is only used once before
being condensed and returning to the reactor core to be reboiled. Finally, the turbine
powers the generator producing electricity (USNRC, 2012). The diagram below in
Figure 1: Schematic of a Boiling Water Reactor. (USNRC, 2005) is a visual
representation of the BWR process.

![Figure 1: Schematic of a Boiling Water Reactor. (USNRC, 2005)](image)

**Figure 1: Schematic of a Boiling Water Reactor. (USNRC, 2005)**

The heat generated in the reactor core is generated by a nuclear fission reaction
(See Glossary). The element generally used in the nuclear fuel for the fission process is
Uranium. Uranium exists in three different isotopes 234, 235 and 238 (Glasstone, S., &
Sesonske, A. 1994). However, only Uranium-235 can be used directly for the release of
fission energy (Glasstone, S., & Sesonske, A. 1994). After absorption of a neutron the nucleus divides in a process called fission. Fission releases enormous amounts of energy mainly in the form of kinetic energy (See Glossary) of the fission products.

In a BWR, as previously mentioned, water acts as the reactor coolant. The water flowing through the reactor must be kept extremely pure. The water primarily contracts impurities from the reactor core. Neutrons released in the fission process can activate the impurities via neutron capture and result in radioactivity within the coolant. The RWCU system removes impurities the reactor coolant and prevents fuel damage in the reactor core by reducing the potential for those impurities to corrode the fuel. It is very important to minimize corrosion, the possibility of fuel damage, and release of fission products in the coolant.
2.2 Design of the Reactor Water Cleanup System

The specifications of the RWCU system differ between nuclear power plants, but the main function is of the system is to maintain purity of reactor coolant. The RWCU is sized to process the entire volume of reactor system in approximately 4 hours (GE, 1980). According to the RWCU reference text at Pilgrim Nuclear Power Station, the RWCU has the following design functions:

Figure 2: BWR Detailed Diagram (GE, 1980)
Continuous removal of radioactive waterborne materials generated in the coolant from fission and corrosion processes.

Continuous removal of soluble inorganic impurities (e.g., chlorides) that enter with the reactor feedwater and could, if not controlled, subsequently concentrate to exceed the specified water quality limits (see PNPS 7.8.1 Water Quality Limits).

Maintain water quality requirements for water removed during startup and shutdown.

Limit heat and fluid losses from reactor system.

(Pilgrim Personnel, 2012b).

The RWCU system draws water from the reactor recirculation flow through the suction side of the reactors recirculation pumps. The inlet is a six-inch MO-1201-85 valve and is named the Clean Up suction valve (Pilgrim Personnel, 2012b). The suction side is on the left side of Figure 2: BWR Detailed Diagram, and it draws from the blue lines near the bottom of Figure 1: Schematic of a Boiling Water Reactor. (USNRC, 2005). The recirculation pumps in the reactor, shown in Figure 2: BWR Detailed Diagram, are different from the recirculation pumps in the RWCU. The reactors recirculation pumps control power in the reactor. The function of the RWCU recirculation pumps will be elaborated on later in this section.

After going through the inlet valve, the flow goes through an inboard isolation valve (See Glossary). The inboard isolation valve, labeled MO-5 in Figure 4, isolates the flow from the recirculation loop and the reactor drain line (Pilgrim Personnel, 2012b). The inboard isolation valve is a main valve separating the flow from primary
containment. Next, the outboard isolation valve follows the inboard isolation valve. Similarly, the outboard isolation valve isolates the flow from primary containment.

After going through the outboard isolation valve, the flow enters regenerative heat exchangers [HX]. The flow enters the regenerative heat exchangers first because the temperature of the flow must be lowered before the demineralizers. The flow enters the first regenerative heat exchanger at approximately 515°F and exits the last at 190°F (Pilgrim Personnel, 2012b). The regenerative heat exchangers are located at elevation 51’ in the RWCU HX Room (Pilgrim Personnel, 2012b). The regenerative heat exchangers act as the first phase of cooling. When the regenerative heat exchangers lower the temperature of the flow by removing heat, a portion of that heat is recovered (Pilgrim Personnel, 2012b). The heat recovered by the regenerative heat exchangers is important because it reduces overall heat loss of the system. Temperature and pressure elements are located upstream and downstream of the regenerative heat exchangers. These elements ensure the regenerative heat exchangers are lowering the temperature of the untreated flow and are functioning properly.

After flowing through the regenerative heat exchangers, the flow goes through non-regenerative heat exchangers. Non-regenerative heat exchangers do not recover the heat removed from the flow and simply reduce the temperature of the flow. The non-regenerative heat exchangers are also located at elevation 51’ in the RWCU HX room (Pilgrim Personnel, 2012b). The flow enters the regenerative heat exchangers at approximately 190°F and exits at 115°F (Pilgrim Personnel, 2012b). The non-regenerative heat exchangers still ensure the flow exits at a temperature less than 130°F (Pilgrim Personnel, 2012b) in the event that regenerative heat exchangers are not fully
functioning. After passing through the non-regenerative heat exchangers, the flow goes into the RWCU recirculation pumps.

The RWCU recirculation pumps are sized for a flow of 200 gpm (Pilgrim Personnel, 2012b). The RWCU recirculation pumps force the flow to the demineralizers for purification but will isolate the flow under any of the following conditions:

- Inboard isolation valve (MO-2) not full open
- Outboard isolation valve (MO-5) not full open
- Return isolation valve (MO-80) full closed
- RBCCW from pump high temp (140°F, TIS-1291-48A/B)

(Pilgrim Personnel, 2012b).

Filter demineralizer units remove the corrosive agents from the untreated cooling water. The ion exchange resins can only remove contaminants if the flow is under 140°F (Pilgrim Personnel, 2012b). The filter demineralizer units, shown in Figure 3, consist of many elements.
The most vital components of the filter demineralizer system are as follows:

a. Cleanup Filter Demineralizers- These are used in the RWCU at Pilgrim Nuclear Power Station as pressure precoat filters. Pressure precoat filters have a removable flanged top section and a dished head pressure vessel (Pilgrim Personnel, 2012b). Tubes support the vessel flange. The tubes are fed through a steel screen supported by a tube sheet (Pilgrim Personnel, 2012b). These tubes collect the precoat material and powdered resin. The resins and precoat material are entirely responsible for removing contaminants from the flow. The flow enters the cleanup filter demineralizers through an inlet at the bottom of the vessel. During purification, the contaminated water flows radially through the tubes and
tube sheets mentioned previously (Pilgrim Personnel, 2012b). The filter
demineralizers are designed for a flow of approximately 111 gpm.

b. Holding Pumps- the pumps maintain the minimum flow required for the
demineralizers to function properly. The resins and precoat material, responsible
for removing contaminants, are held in place by the normal flow (Pilgrim
Personnel, 2012b). If the flow drops below 75 gpm, the holding pumps will
engage to increase the flow.

c. Precoat Resin Tanks- these tanks consist of agitators that prepare a slurry of both
precoat material and resin (Pilgrim Personnel, 2012b). The material is dispensed
by the following system.

d. Precoat Pumps- these pumps distribute the slurry created in the precoat resin
tanks.

e. Cleanup Backwash Receiving Tank- this tank receives no longer active resin from
the demineralizer. The single tank collects the spent media during demineralizer
backwash and drains to designated storage tanks in the Radwaste Department.

f. Screen Strainer- this unit prevents any resin from entering the reactor via purified
reactor coolant. This situation could occur if any of the resin containing elements
mentioned earlier were to fail. Resin entering the reactor could be indicated by a
pressure difference, so a pressure differential switch rated at 5psid is located in
the system (Pilgrim Personnel, 2012b).

g. Flow Control Valves- these valves are located after each filter demineralizer unit
(Pilgrim Personnel, 2012b). The valves manually control the flow and can
manually isolate the flow from the demineralizers. A pressure differential higher
than 10psid will cause the flow control valves to isolate the flow leaving the demineralizer.

After the reactor coolant has been treated by the filter demineralizers, it passes through the return isolation valve labeled MO-80 in Figure 3. The return isolation valve controls the flow of treated water back to the regenerative heat exchangers (Pilgrim Personnel, 2012b). The regenerative heat exchangers receive the purified flow at approximate 115°F and exits at 460°F (Pilgrim Personnel, 2012b). The regenerative heat exchangers return the purified flow back to the reactor via a feedwater line (Pilgrim Personnel, 2012b). A visual representation of the RWCU system is displayed below in Figure 4.

![Figure 4: RWCU System Schematic](image_url)
The RWCU system is a crucial part to the proper function of all operational modes of the system; however, due to the need to maintain the return path at the feedwater line, it is not operational during a refueling outage.

2.3 Refueling Outages

The Nuclear Energy Institute [NEI] states that “U.S. nuclear reactors shut down once every 18 to 24 months to refuel approximately one-third of the reactor” (NEI, 2013). The refueling outage process usually lasts about one month and is time sensitive because the plant is not generating electricity when the reactor is shut down. A refueling outage is a multi-step process that requires complete understanding of all systems and their functions during the outage.

System and RWCU operations during a refueling outage are important to understand when considering the need for the proposed RWCU modification. The design of the modification depends heavily upon which systems would be operational during a refueling outage.

2.3.1 Reactor Shutdown Procedure

Shutting down the reactor begins with a process called scramming. The process of scramming a Boiling Water Reactor is used to manually or automatically shut down a reactor. SCRAM stands for Safety Control Rod Axe Man and is the quickest way to kill any chain reaction in a reactor or to shut down a reactor rapidly. The name comes from the early prototypes when the control rods were lifted by ropes and pulleys. The lone job of the axe man was to cut the ropes in an emergency to shut down the reactor. It is generally a very simple process and can successfully shut down a reactor, and in
emergency conditions, protect workers, and the public. The process of scramming a reactor involves the complete insertion of control and safety rods into the reactor vessel to poison the reaction and shut it down promptly.

At Pilgrim Nuclear Power Station, scramming is used as a routine part of shutdown. Once the reactor has reached about 20% power, the control rods are inserted rapidly during the scramming process. This rapid insertion causes a ‘kick up’ of small particles that are hiding in pipes, joints, and valves. Once this happens, the water then becomes too murky to see through and the refueling process cannot be completed without this water being cleaned.

After shutdown, cooling or cool down is the gradual decrease in temperature and removal of decay heat and is referred to as reactor cool down (Moore, 1987). It is accomplished through operation of the residual heat removal system (RHR). Cool down is then followed by cold shutdown. During cold shutdown, the reactor coolant is at atmospheric pressure and below two hundred degrees Fahrenheit.

2.3.2 Water Clarity During Refueling Outage

Ideally, the RWCU system would be used to clean the water following the scramming process. The RWCU system at Pilgrim is not functional during outages because the primary return for the water in the Feed Water system is shut down and isolated for maintenance. This prevents the RWCU from performing its primary function of purifying the water. Portable filters and demineralizers are used for a brief period of time to purify the water, reduce radiation levels associated with the radioactive debris in the water, so work can officially start on refueling.
There are only two systems available to purify reactor coolant during a refueling outage. These modes are also not designed to purify the reactor water during a refueling outage. The augmented fuel pool cooling [AFPC] modes are designed to extend the range of heat removal within the system. The AFPC system can operate under two different modes. The first mode is only available during operation of RHR shutdown cooling when the reactor basin (See Glossary) is flooded and the fuel pool gate is open. The second mode is only available when the RHR shutdown cooling loop is not operating when the reactor basin is flooded and the fuel pool gate is open, or when the fuel pool has a high heat load and is isolated from the reactor basin.

To compensate for the unavailability of the RWCU the augmented fuel pool cooling with purification [AFPC & P] mode was established to allow operation of the fuel pool cooling [FPC] system filter and demineralizer with the residual heat removal pump and heat exchanger. The connection between the RHR and FPC systems is limited however, due to the difference in operating conditions of the two systems; therefore, the operating procedures contain instructions for appropriate system startup and operating procedures. The efficiency of the AFPC & P mode is also restricted by the units within both systems. The FPC system consists of one demineralizer that is designed to process 670 gal/min of water during normal operation. A schematic of the FPC system layout can be found below in Figure 5. The FPC demineralizer is highlighted in orange while both FPC heat exchangers are highlighted in blue.
Figure 5: Fuel Pool Cooling System

Operation of the AFPC & P system during a refueling outage does not provide the complete water purification capacity that is desired. Pilgrim Nuclear Power Station is currently using portable demineralizers to augment the capacity of the combined RHR/AFPC system. The filters are lowered into the flooded reactor vessel to purify the water before any refueling and maintenance can be performed. This requires extra man hours and radiation exposure that could otherwise be avoided. Thus the goal of the RWCU piping system modification is to allow operation during a refueling outage. With the AFCP & P and the RWCU in operation simultaneously during a refueling outage, the required water purity can be achieved.
2.3.3 Administrative Dose Limitations and Goals

Some basic information about the administrative dose limits set by Entergy® are available through the training modules on the NANTeL training site. Entergy® has set a variety of limits based on the portion of the body that is exposed and in cases of a declared pregnancy. The whole body, which includes the head, trunk, active blood-forming organs and gonads, is limited to two rem per year. This is generally the most restricting portion for dose limit. A declared pregnant worker has a limit of 0.4 rem for the full term of the pregnancy or 50 mrem per month (NANTeL, 2010). The maximum amount of dose that can be received annually is 40 rem; however, this value only applies to the internal organs, extremities and skin. Due to the severe negative effects of radiation exposure to a fetus, it is highly encouraged that all female radiation workers that become pregnant declare their pregnancy to the company and fill out the appropriate paperwork. Any female radiation worker who decides to refrain from declaring her pregnancy is treated as any other radiation worker and assigned the normal administrative dose limits.

Entergy® in congruence with the Radiation Protection department also sets yearly man-rem goals for refueling outage years and non-refueling outage years. Man-rem refers to the total dose accumulated by a large number of workers. For non-refueling outage years, the man-rem occupation radiation dose goal is <25 rem (Pilgrim Personnel, personal communication, January 9, 2013). For refueling outage years, the man-rem occupation radiation dose goal is <37 rem (Pilgrim Personnel, personal communication, March 13, 2013). These limits are set to ensure the safety of all the employees at Entergy® and are strict guidelines that provide insight into the goals and limitations of the project.
2.4 Water Clarity and Radiation

As previously mentioned the water clarity is of great concern during a refueling outage in terms of time and dose exposure. During the scramming process a large quantity of Cobalt-60 is released due to the mechanical disturbance within the system. In fact, at the beginning of a refueling outage the dose associated with cobalt-60 when the reactor basin is flooded is 12mrem/hr (Pilgrim Personnel, personal communication, March 25, 2013). This form of cobalt later becomes the most significant source of dose concern and is one of driving forces behind this project. The following paragraphs discuss in detail the creation and risk associated with cobalt in the reactor coolant.

The presence of Co-60 in the reactor coolant creates the greatest radiation exposure during a fuel outage and in order to remove more Cobalt-60, it is vital to understand the source of the Cobalt-60. Stellite is a hard and corrosion-resistant metal alloy composed mostly of Cobalt (Gooch, 2007). Stellite has a general composition of about 57% Cobalt (Co), 18% Chromium (Cr), 15% Tungsten (W) and 10% Nickel (Ni). It is used as a coating to extend the life of valve seats and ensure complete closure of valves. It is designed to protect valves from the harsh conditions including temperature and pressure. The RWCU system, as well as the majority of the systems at Pilgrim contains many of these valves throughout the system. The stellite coating is composed mostly of Cobalt, Chromium and Tungsten (Gooch, 2007). The coating contains Cobalt which is entirely in the form of the stable Cobalt-59 (Co-59) isotope. As the stellite comes into contact with the reactor water it corrodes and releases Co-59 into the reactor coolant water. After a cobalt atom gets leaches off of the cobalt chromium alloy stellite coating and into the reactor coolant water, it is bombarded with neutrons in the reactor
core. Because cobalt will accept neutrons into its nucleus, the cobalt-59 (Co-59 or $^{59}\text{Co}$) atom becomes radioactive cobalt-60 (Co-60 or $^{60}\text{Co}$) isotope:

$$^{59}\text{Co} + \frac{1}{0}n \rightarrow ^{60}\text{Co}$$

With the addition of Co-60 to the coolant, the dose of radiation that is encountered when working near reactor coolant water, especially when the reactor head is open during refueling outage is increased significantly (T. Setzer, Personal Communication, October 25, 2012).

The reason Co-60 is a radiation exposure issue during refueling outages is because of its long half-life and its decay mode. Co-60 has a half-life of 5.2714 years, and it decays via beta-particle decay, $\beta^\prime$. When an element goes through beta decay, the products are one daughter atom (Ni-60), a Beta particle, an antineutrino, and two gamma rays. The gamma ray energies are 1.17 and 1.33 MeV and pose the exposure risk since they are most penetrating.

### 2.5 Related RWCU Modifications

Before researching potential locations for the piping modification, other modifications involving the RWCU were researched. The most recent and related
modification design was a project proposed in 2000. A member of the Mechanical Structural Civil Engineering Department [MSCD], proposed a piping modification that would have improved the operational flexibility of the RWCU.

Our project and the project proposed in 2000 by the Pilgrim personnel involved a potential intertie between the RWCU and FPC, but the purpose behind each modification was very different. As stated in the Introduction, the goal of this project was to modify the RWCU, so that it can purify reactor water during a refueling outage. The goal of the project proposed in 2000 was to modify the RWCU, so that it would be able to support the RHR system and increase outage decay heat removal (Pilgrim Personnel, 2000a). Despite the differences between the project statements, the Pilgrim personnel’s comprehensive research, detailed scope, justification approval, and technical review provided detailed information pertinent to this project. In the technical review of the proposal, the Pilgrim personnel explained the limited availability of the RWCU during a refueling outage. As previously mentioned, the RWCU is primarily unavailable during a refueling outage because feedwater line A, responsible for returning the purified flow to the reactor, is isolated for maintenance. Disregarding the isolated feedwater line, if the RWCU were operational during a refueling outage it could increase thermal efficiency by maximizing decay heat removal.

During normal operation, the configuration of both the regenerative and non-regenerative heat exchangers optimizes thermal efficiency (Pilgrim Personnel, 2000a). Besides normal operation, the configuration of the RWCU Heat Exchanger System could also optimize thermal efficiency during cooling, cold shutdown, and reactor vessel class one pressure test. Lastly, reactor vessel class one pressure test is a test performed near the
end of a refueling outage. During these procedures, the RWCU could provide approximately three times more heat removal at a rate of 12MBtu/Hr at 210°F (Pilgrim Personnel, 2000a).

In order to operate the RWCU as a backup to RHR, Pilgrim Personnel proposed a piping modification that would begin after the filter demineralizers in the RWCU and end at the discharge header of the FPC. As stated previously, during the normal operation the flow leaving the demineralizers is heated before being returned to the feedwater line. Because of the need to maximize heat removal of reactor coolant system during cool down, cold shutdown, and reactor vessel class one pressure test, the piping modification must bypass the regenerative heat exchangers. Running through the regenerative heat exchangers would reduce heat removal capacity. First, the lack of a bypass around the regenerative heat exchangers would limit the net heat removal to 4MBtu/Hr at 210°F (Pilgrim Personnel, 2000d). Secondly, the use of the regenerative heat exchangers could inhibit the ability to maintain a constant reactor coolant temperature during reactor vessel class one pressure test. Bypassing the regenerative heat exchangers would also alleviate the temperature and pressure differential maximizing the net heat removal by the RWCU back up the RHR.

The Pilgrim personnel’s piping modification would have involved the addition of a 4” intertie line. The 4” inch line would be manually operated metal seated ball valves, butterfly valves, and a spectacle blind flange (See Glossary) arrangement (Pilgrim Personnel, 2000d). Approximately 60ft in the length, the modification would require supports and a core bore (Pilgrim Personnel, 2000d). The core bore pass through a wall on the 74’ elevation floor. The 4” intertie would divert flow to the reactor basin, so the
cleanup portion of the RWCU could also be used during a refueling outage (Pilgrim Personnel, 2000a).

The second part of the Pilgrim personnel’s modification would have connected the 4” intertie line upstream of the inboard isolation valve. The 2” line would bypass the regenerative heat exchangers and return the flow to the feedwater line. The 2” line would allow maximum decay heat removal during a reactor vessel class one pressure test (Pilgrim Personnel, 2000a). Both portions of the Pilgrim personnel’s piping modification were designed according to the ASME/ ANSI B31.1 piping modification, material fabrication and inspection specifications (Pilgrim Personnel, 2000b). Although the second portion of the Pilgrim personnel’s modification would benefit the reactor vessel pressure test, it was not considered effective because it is not directly pertinent to the project.

The modification proposed in 2000 would have been constructed in two phases. Because locations for the intertie pipes are in high radiation areas, the areas would have needed to be chemically decontaminated in accordance with the As Low As Reasonably Achievable [ALARA] radiation plan. In the ALARA review of his modification, the Pilgrim personnel suggested the use of decontamination valves despite the potential modification being located on the clean side of the RWCU filter demineralizer. Adding decontamination valves would potentially allow short decontaminations to be performed on the modifications during short RWCU outages (Pilgrim Personnel, 2000). Even more, The Pilgrim personnel’s potential modification would only be used during refueling outages, so it would not necessarily become contaminated during normal operation. The ALARA review also contained a suggestion to lower installation dose. The Pilgrim
personnel suggested a portable shield cart system be installed by Radiation Protection [RP] personnel.

With respect to the adequacy of the Pilgrim personnel’s design, the specifications for the piping were based on ISI Safety Class one, two, or three, Q/Non-Q Safety, and PNPS Class I or II Designations.

Figure 6: Safety Class One and ISI Safety Class One portion of the RWCU
Figure 7: Safety Class One and ISI Safety Class One portion of the RWCU

Figures 6 and 7 illustrate the PNPS Class One and ISI Safety Class areas of the RWCU. Class one areas were not the most desirable places for a modification because the Safety Specifications are the most rigorous and limited the feasibility of most modifications. Also regarding safety, both phases of the Pilgrim personnel’s design were evaluated according to the High Energy Line Break [HELB] Analysis. High Energy Line Breaks include any system or portion of system that could exceed the designated maximum temperature and pressure threshold during normal operation (Berkovsky et al, 2007). The cost, safety related issues, location etc. involved with the HELB shown in Figure 7 prevented the Pilgrim personnel’s potential modification from being installed.

If phase one and two of the Pilgrim personnel’s modification had been installed, their modes of operation would have been referred to as Refueling Bypass and Shutdown Bypass, respectively. During the operation of Refueling Bypass mode, the following conditions could have been expected.
Table 1. Conditions expected during Refueling Bypass mode

<table>
<thead>
<tr>
<th>Flow Rate</th>
<th>Temperature</th>
<th>Pressure</th>
<th>Reject Flow Rate</th>
</tr>
</thead>
<tbody>
<tr>
<td>222 GPM</td>
<td>125 °F</td>
<td>49 PSIA</td>
<td>0 GPM</td>
</tr>
</tbody>
</table>

Refueling Bypass mode would have involved the operation 4” intertie between the RWCU and FPC. The intertie would have aided the RHR by removing decay heat and would have purified reactor water.

During the operation of Shutdown Bypass mode, the following conditions could have been expected.

Table 2. Conditions expected during Shutdown Bypass mode

<table>
<thead>
<tr>
<th>Flow Rate</th>
<th>Temperature</th>
<th>Pressure</th>
<th>Reject Flow Rate</th>
</tr>
</thead>
<tbody>
<tr>
<td>150-222 GPM</td>
<td>210 °F</td>
<td>1050 PSIA</td>
<td>0 GPM</td>
</tr>
</tbody>
</table>

Shutdown bypass mode would have removed decay heat and maintain pressure during reactor vessel pressure head. Though the Pilgrim personnel’s modification was not implemented due to the HELB, his detailed design documentation and process contained vital information for this project.
3. METHODOLOGY

3.1 Overview

The project was split into two different phases, the preliminary design phase and the detailed design phase. The preliminary design phase included identifying candidates for inlets and outlets of the modification from the RWCU to the reactor vessel. This phase included gathering and organizing basic information on critical design attributes and their requirements. The detailed design phase included development of detailed drawings, bill of materials, cost estimates, and dose savings estimates. The progression of the project, all necessary information necessary for the completion of the project, and the final project design were presented to Entergy®. Within these phases there was a review process completed between the PNPS engineers and the WPI team as part of the fulfillment of the requirements for a project proposal with Entergy®. The review process consisted of three meetings: 10%, 50%, and 90% scope meetings. The output of these review meetings, in addition a peer review, was typically a design decision. The 10% scope meeting was included as part of the preliminary design phase, while the 50% and 90% scope meetings were included in the detailed design phase. Each meeting discussed different aspects of the project as well as the progression of the project. The following table outlines the design attributes that were of relevance to the project and at which meetings they were discussed.

3.2 Design Attributes

There are two types of design factors that were utilized in the preliminary design phase to narrow down all possible inlet and outlet locations: Imperative design factors
and achievable design factors. The imperative design factors are attributes that could completely rule out a design candidate. These design factors are as follows:

1. Must be in a non-safety related area
2. Must NOT be in and around a High Energy Line Break
3. Inlet and Outlet must be on the same elevation

If a design met all of the imperative design factors, the achievable design factors were then used to determine the most ideal location. The achievable design factors are displayed in Table 3. The values associated with these attributes evolved over the design of the modification and are presented in the results section. Each of these design attributes played a significant role in the progression and development of the modification, some of which were discussed at multiple meetings.

**Table 3: Achievable Design Attributes Development Overview**

<table>
<thead>
<tr>
<th></th>
<th>Preliminary Design Phase</th>
<th>Detailed Design Phase</th>
</tr>
</thead>
<tbody>
<tr>
<td>Design Attributes</td>
<td>10% Scope meeting?</td>
<td>50% Scope Discussion</td>
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<tr>
<td>Flow Rate</td>
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<td></td>
</tr>
<tr>
<td>Temperature</td>
<td>✓</td>
<td>✓</td>
</tr>
<tr>
<td>Pressure</td>
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<td>✓</td>
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<tr>
<td>Elevation</td>
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<tr>
<td>Location</td>
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<td>✓</td>
</tr>
<tr>
<td>Operational Reliability</td>
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<tr>
<td>Radiation</td>
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<td>✓</td>
</tr>
<tr>
<td>Estimate</td>
<td></td>
<td>✓</td>
</tr>
</tbody>
</table>
3.3 Preliminary Design Phase

The project goal was to improve the operational flexibility of the RWCU by modifying the system so that it can run during a refueling outage. A variety of components of the RWCU are shutdown during outages. Examples of these components are the feedwater line from the RWCU into the reactor, and the recirculation pumps that pump the reactor coolant into the RWCU. The entire RWCU system must be shut down simply because the feedwater line is not in operation; therefore, leaving the system without an outlet, hence the need for a modification to use alternate path to return to the reactor vessel. A major consideration at the outlet of the project was determining if enough suction head was provided by the RWCU recirculation pumps to feed the reactor coolant into the RWCU. Fluid mechanics, specifically Bernoulli’s equation, was used to prove the reactor coolant could make it to the RWCU with the use of the recirculation pumps in the system.

\[ p_2 := p_1 + \rho \cdot \frac{(v_1^2 - v_2^2)}{2} + \rho \cdot g \cdot (z_1 - z_2) - (\rho \cdot g \cdot h_L) + (\rho \cdot g \cdot h_P) \]  

(1)

The variables for this equation are all defined below.

- \( P_1 \) – Atmospheric pressure
- \( \rho \) – Density of water
- \( v_1 \) – Velocity of water entering the system
- \( v_2 \) – Velocity of water entering the pressure header
- \( g \) – Gravity
- \( z_1 \) – Height of the entrance to the system
- \( z_2 \) – Height of the exit to the system
- \( h_p \) – Head created from the RWCU pump
- \( h_L \) – Head loss due to friction
- \( Q \) – Volumetric flow rate
- \( d_{pipe} \) – diameter of the pipe
- \( A_{pipe} \) – Cross sectional area of the pipe
The next step in the preliminary design phase was to examine all of the possible modification inlet and outlet locations. Mechanical drawings of the RWCU provided a detailed layout of the equipment that comprises the system. From the mechanical drawings, the team determined potential inlet and outlet locations of the modification. For each potential outlet and inlet located on the relevant Piping and Instrumentation Diagrams [P&ID] (See Glossary) the various inlet and outlet location options were determined. The name and class on the relevant pipes was then found on the corresponding Isometric Roadmap [IRM] (See Glossary). The pipe class elicited from the IRM was then researched in an M300 pipe class document to determine the specifications of each pipe. The M300 designated the ASME/ANSI B31.1 piping modification, material fabrication, and inspection requirements for the piping. Also from the IRM, a Piping Isometric Drawing was found. The Piping Isometric Drawing showed the elevation and exact location of the potential inlets and outlets. The conceptual design section shows the four types of drawings mentioned in this paragraph and following the drawings is a table with information gathered from the M300.

The most ideal inlet and outlet locations were determined based on the following criteria.

- The maximum flow rate during an outage
- Temperature differential across the RWCU during an outage
- Pressure differential across the RWCU during an outage
- Difference in elevation between the inlet and outlet piping modification
- Radiation in terms of location
It was important for the most cost beneficial, safe, and functional modification to be designed. Therefore, the need to look into each of these factors was of great importance for each possible design. Of course much of this information (i.e. cost estimate) was not available at the time of the preliminary design phase. Therefore, any decisions based on those particular factors were made using the limited information that had been provided at that time. For example, it was known that running the modification between multiple floors was more expensive than to run the pipe through walls on the same elevation, therefore, inlet and outlet locations on the same floor would be more ideal.

3.3.1 10% Scope Meeting

The design attributes listed above were used to eliminate potential inlet/outlet locations on an engineering judgment standpoint. The preliminary designs were presented before the design team in a 10% Scope Meeting. During a 10% Scope Meeting, the preliminary designs are discussed and evaluated according to the design attributes previously listed. Members on the design team gave relevant feedback regarding the pros and cons of each potential location.

3.4 Detailed Design Phase

The detailed design phase consisted of developing a complete scope of the best-fit design from the preliminary design phase. Detailed drawings, bill of materials, cost estimates, and dose savings estimates were outputs of this phase. The design attributes used to evaluate the preliminary design were quantified in the detailed design. The detailed design was constructed according to the Design Considerations and

The Design Considerations document encompasses the factors that must be taken into consideration when modifying any nuclear system. The Design Considerations document contains general considerations, mechanical considerations, electrical considerations and foreign material exclusion [FME] considerations. These were all considered along with the other design attributes.

When developing a detailed design for a piping modification to a non-safety related system in a nuclear power plant, the first design consideration involved project scoping and the problem definition. The problem definition did not change during this process, so the project scoping documents and problem statements were reviewed prior to beginning work on the detailed design. These documents provided an outline of the relevant information that could be pertinent to the design modification. The information from these documents that was pertinent to this project is listed as the design attributes.

After reviewing the basic functions and critical characteristics of the RWCU, design conditions such as pressure, temperature and flow rate were reviewed. This was necessary information to ensure that the connection between the two systems would be possible and compatible.

Another relevant source of concern for the design process was the pipe class of the modification. When selecting a pipe class the normal operating conditions of both the RWCU and FPC systems had to be considered. It was easiest to base the modification off of the pipe class of the line that the modification was tying into and just ensure that
the pipe class in the FPC system of the outlet pipe would operate properly under those conditions.

### 3.4.1 50% Scope Meeting

At the 50% scope meeting, it was also determined what kinds of materials would be necessary for the implementation of the modification. Such materials included valves, pipe connections, and piping itself. Following the 50% scope there was a lot of progress still to be made in the design process and in completion of cost and dose estimates.

Subsequently, the layout arrangement requirements were evaluated. The layout arrangement requirements included: Fire Boundary Areas, Security considerations, piping configuration, plant walkdowns, and device location (Pilgrim Personnel, 2012a). The team conducted walkdowns to determine the piping configuration in both the RWCU and the FPC. The layout of the systems was partially evaluated using Panomap and live cameras outside the Locked High Radiation Areas [LHRA] and Very High Radiation Areas [VHRA].

Following the layout and arrangement requirements, the ALARA considerations of the detailed design were assessed. The ALARA considerations include creation of new field zones, crud traps, shielding, operations, potential for reducing cobalt, effects of cobalt bearing alloys, and all things that will reduce dose (Pilgrim Personnel, 2012a). Also, the ALARA review completed by Pilgrim personnel in 2000 provided detailed information.

Next the detailed design was analyzed according to the operational reliability design considerations. The operational reliability design considerations include plant
start-up, normal operation, plant shutdown, system maintenance, plant emergency
operation, special or infrequent operations, system transients, and system transient
operations (Pilgrim Personnel, 2012a).

After operational reliability was analyzed, the accessibility for maintenance was
assessed. In the design specifications manual, accessibility for maintenance cover a
variety of topics. The topics most related to the detailed design included: accessibility for
existing plant components and components.

A number of general design considerations followed layout and arrangements
requirement. These general design requirements that did not directly lend themselves to
the project were Environmental Impacts, Discharge Impacts, Seismic specifications,
Interface Requirements, Redundancy, Diversity, Transportation Requirements, Handling
Requirements, Material special processes.

Once the detailed design was complete, a cost estimate for the modification was
performed. The estimate was divided into two main savings categories: dose and
monetary. The goal was to keep both categories at a minimum. A cost-benefit analysis
compares the upfront costs of the implementation of the modification to the long-term
benefits associated with the operation of the modification. The monetary analysis
includes all aspects from man hours to material cost. Also, as safety is a primary concern
in any such project, a thorough dose analysis weighed the one-time exposure during
construction with the potential radiation exposure savings with the operational
modification in place. Essentially, it was successfully proved that the design that was
developed is not only beneficial in a monetary sense, but also in the pursuit of reducing worker dose.

### 3.4.2 90% Scope Meeting

The 90% scope meeting consisted of the team presenting the final design including dose and cost estimates, and all the necessary paperwork to the design engineering department, as well as some critical stakeholders from the chemistry and systems operations departments at PNPS. This meeting served as the final presentation for completion of the Major Qualifying Project requirements with WPI and also as a means of project approval through PNPS. With the help of Pilgrim personnel, the project was able to be approved.
4. RESULTS

The results portion of this project consists of three sections; preliminary design, detailed design, and estimates for cost and worker radiation exposure levels. These sections provide a cohesive explanation of the analysis process and a detailed final modification design that satisfies the requirements set forth by Pilgrim Nuclear Power Station.

4.1 Preliminary Design

The piping modification consisted of an inlet from the RWCU and an outlet to an appropriate system that is interconnected with the reactor vessel and is operational during refueling outages. The goal of the preliminary design phase consisted of developing multiple modification options and determining the best-fit option based on the criteria defined in the methodology.

Before the teams inlets and outlets were evaluated, it was vital to clearly state the definitions of the words inlet and outlet. The intent of the modification was to allow flow of lower purity water at the bottom of the vessel to enter the RWCU, run through purification steps in the RWCU, then re-enter the upper portion of the reactor vessel. It was realized that the inlet to the modification was an outlet from the RWCU, and the outlet of the modification was an inlet to another system. The flow from the outlet would then re-enter the upper portion of the reactor vessel. For this reason, the term inlet is used to describe the beginning of the modification. And the term outlet is used to describe the end of the modification.
The following sub-sections consist of potential inlets and outlets for the modification. There are three inlet and four outlet options listed, along with design specifications and a brief description of each. The drawings mentioned in the methodology that must be assembled when determining the general locations for a potential inlet or outlet are shown prior to each description.

There are three inlet and four outlet options listed, along with design specifications and a brief description of each, which were reviewed during the 10% Scope meeting. Pilgrim personnel on the design team were apprised of the assessment Nikole Stone, Scott Gallagher, and Katherine Goldberg completed regarding each potential inlet and outlet using P&ID, Isometric Roadmap, Piping Isometric, and Elevation drawings.

Factors that could rule a design completely were referred imperative design factors. The imperative design factors were as follows:

1. Must be in a non-safety related area
2. Must NOT be in and around a High Energy Line Break
3. Inlet and Outlet must be on the same elevation

Factors that the ideal design would accommodate were referred to as achievable design factors. The achievable design factors were as follows:

1. Flow Rate
2. Temperature
3. Pressure
4. Location
5. Radiation Level

The first imperative design factor required the areas selected for the modification be non-safety related areas. As previously stated, non-safety related areas have fewer
design requirements and increase the feasibility of accomplishing the modification. Materials required for safety related areas are also much more expensive than those required in non-safety related areas. The selected pipes were verified using a Q-list. A Q-list has every safety-related area in each numbered system of the power plant.

The second imperative design factor required the design stay outside of HELB areas. HELB areas impose strict limitations on modifications, require costly equipment, and require strict maintenance. Furthermore, a HELB struck down the modification proposed in 2000.

Lastly, the third imperative design factor, elevation, was evaluated in terms of whether the potential inlet or outlet would require drilling through walls or floors to complete the modification. Drilling through walls was considered less attractive from a cost standpoint.

After being assessed according to the imperative design factors, the candidates were analyzed according to maximum flow rate, temperature, pressure, location, radiation level, and elevation. Flow rate was assessed according the flow that would go through the selected pipe during normal operation. Normal operation refers to normal operation mode the most frequent mode of operation. During normal operation, demineralizer recirculation pumps have a flow rate of 111 gpm each, 222gpm total (Pilgrim Personnel, 2012b). Flow rate was assessed during normal operation because the new mode of operation would not deviate from the flow rate used in normal operation mode.

Temperature was assessed by comparing the temperature of the flow during shutdown mode to the pipe class of the pipe the modification would branch off of. As
previously stated, during a refueling outage or reactor shutdown mode the reactor coolant is approximately 140°F (Pilgrim Personnel, 2012b). Each pipe class has a designated design temperature. Two different pipe classes could be used in modification, but the design temperatures of the two pipes used would need to accommodate the temperature of the flow during shutdown mode.

Pressure was also assessed by evaluating the candidates according to the pressure of the flow during shutdown mode. During shutdown mode, the pressure of the flow leaving the demineralizers is 34.3psig. Stated in the previous paragraph, two different pipe classes could be used in the design of the modification, but both pressure classes would have to accommodate the pressure of the flow.

General location and radiation level were evaluated together because some locations in the plant had consistently high radiation levels and ruled out specific locations completely. The location and radiation level of the potential inlet were crucial design factors because some areas and rooms in the plant have higher levels of radiation at any given moment during any mode of operation. Though radiation and contamination can change, some rooms are consistently High Radiation Area [HRA] or Locked High Radiation Area. Given that the overall goal of the project was to reduce workers radiation dosage, it was most ideal to choose the location that would expose workers to the least amount of radiation during the time of implementation and maintenance.

After members Nikole Stone, Scott Gallagher and Katherine Goldberg verbally evaluated each inlet and outlet candidate, the other design team members gave input
regarding the different locations. The information gathered from the 10% Scope meeting regarding the inlet and outlet.
4.1.1 Candidate Inlet One

The first potential inlet was located shortly after the filter demineralizer of the RWCU, shown in Figure 8. Information regarding the specifications of potential inlet one is shown above in Table 4. Downstream of valve MO-1201-75, the demineralizer bypass line joins the flow leaving the demineralizers displayed in Figure 9. The three dimensional layout and elevation of the pipe are shown in Figure 10 and 11 respectively.

The temperature of the flow leaving the demineralizer was also well under the design temperature of the pipe class selected in the above table. The design pressure for that pipe was also at 600 psig; which will more than compensate for the 34.3 psig water pressure running through the system during an outage.

A known HRA was located in close proximity to Candidate Inlet One. The next concern was finding a portion of the length of pipe between the demineralizer and valve MO-1201-75 that is outside of any high radiation area.

Based on the information gathered in the preliminary design regarding the inlet location, the general area after the demineralizer was deemed an attractive location. The general location was considered ideal because the pipe class in that area would accommodate the temperature, pressure and capacity of the flow rate leaving the demineralizers. A walkdown of the general location would further justify whether candidate inlet could have a location outside a high radiation area. The more detailed location of potential inlet one was drawn from M-248 and IR-M248, and these drawing are displayed in the detailed design section.
Figure 8: M-247 P&ID for RWCU System

Figure 9: IR-M247 Isometric Roadmap for RWCU System
Figure 10: Piping Isometric Drawing for Pipe M100BC20-3 in the RWCU System

Figure 11: Elevation Map of Pipe M100BC-20-3
Table 4: Potential Inlet One Specifications

<table>
<thead>
<tr>
<th>Valve Name</th>
<th>Pipe Name</th>
<th>Pipe Class</th>
<th>Design Temperature</th>
<th>Design Pressure</th>
<th>Elevation/Room</th>
</tr>
</thead>
<tbody>
<tr>
<td>MO-1201-75</td>
<td>M-100BC-20-3</td>
<td>4&quot;-EA-12</td>
<td>850°F</td>
<td>600psig</td>
<td>CEL 61'-3&quot;, CEL 61'-3&quot;</td>
</tr>
</tbody>
</table>
4.1.2 Candidate Inlet Two

Potential inlet two was located in close proximity to potential inlet one. The schematic location of the pipe is shown in Figure 12. The name of the pipe was elicited from Figure 13. Figures 14 and 15 illustrate a three-dimensional version and an elevation view respectively.

Upon further examination during the detailed design phase, the potential inlets were found to be located on different P & ID drawings. They had the same pipe class and were both designed to handle the flow leaving the filter demineralizer.

The elevation of the second potential inlet made it a less attractive candidate. The elevation of potential inlet listed above in table four placed it very close to the elevation of the regenerative heat exchangers. This potentially put this inlet in a high radiation area, which could eliminate this option as a viable candidate. Given that potential inlets and outlets only designate general areas, the specific location could not be determined until a walkdown was performed.

Though potential inlet two was in the same general area as potential outlet one, it was located closer to the condenser. The excess length would cause a greater amount of head loss in the pipe. The head loss could require more work from the one operational RWCU recirculation pump, which was already over-compensating due to the shutdown of the reactor recirculation pumps. Therefore candidate inlet two was considered less attractive than Inlet one.
Figure 12: M-247 P&ID for RWCU System

Figure 13: IR-M247 Isometric Roadmap for RWCU System
Figure 14: Piping Isometric Drawing for Pipe M100BC20-3 in the RWCU System

Figure 15: Elevation Map of Pipe M100BC-20-3
Table 5: Potential Inlet Two Specifications

<table>
<thead>
<tr>
<th>Valve Name</th>
<th>Pipe Name</th>
<th>Pipe Class</th>
<th>Design Temperature</th>
<th>Design Pressure</th>
<th>Elevation/Room</th>
</tr>
</thead>
<tbody>
<tr>
<td>MO-1201-76</td>
<td>M-100BC-20-3</td>
<td>4&quot;-EA-12</td>
<td>850°F</td>
<td>600psig</td>
<td>EL. 52'-6&quot;</td>
</tr>
</tbody>
</table>
4.1.3 Candidate Inlet Three

The third potential inlet location was after the regenerative heat exchangers show in Figure 16. The pipe name in the area being evaluated was drawn from Figure 17. Figure 18 shows a three-dimensional view of the pipe. Figure 19 shows the general elevation in the area.

This Candidate does not meet two of the imperative design factors. The pipe is located in a safety related area and the portion of piping located after Regenerative Heat Exchanger E-208-A is a HELB. Because this Candidate did not meet two of the imperative design factors, it was ruled out completely.

Table 6: Potential Inlet Three Specifications

<table>
<thead>
<tr>
<th>Valve Name</th>
<th>Pipe Name</th>
<th>Pipe Class</th>
<th>Design Temperature</th>
<th>Design Pressure</th>
<th>Elevation/Room</th>
</tr>
</thead>
<tbody>
<tr>
<td>MO-1201-80</td>
<td>M-100BC-19-3</td>
<td>4&quot;-EA-12</td>
<td>850°F</td>
<td>600psig</td>
<td>58’4</td>
</tr>
</tbody>
</table>
Figure 16: M-247 P&ID for RWCU System

Figure 17: IR-M247 Isometric Roadmap for RWCU System
Figure 18: Piping Isometric Drawing for Pipe M100BC-19-3 in the RWCU System

Figure 19: Elevation Map of Pipe M100BC-20
4.1.4 Candidate Outlet One

During a refueling outage the Fuel Pool Cooling and demineralizer system is run with the Residual Heat Removal system as described previously in the AFPC mode. The potential outlet for the pipe modification would tie into the Fuel Pool Cooling Demineralizer bypass line shown in Figure 20. The pipe name in the general area was selected from Figure 21. Lastly, three-dimensional and elevation views are shown in Figures 22 and 23.

The location of this inlet does not accommodate the imperative design factor requiring the inlet and outlet to be on the same elevation. Because the specific pipe selected required drilling through the floor, a modification to this particular pipe was ruled out.

The Fuel Pool Cooling system would be the most attractive location for outlet. However, the general area selected and assessed in Candidate Outlet One was not attractive because the pipe into which the outlet would tie did not meet the pressure requirements and would require drilling through the floor.

Table 7: Potential Outlet One Specifications

<table>
<thead>
<tr>
<th>Valve Name</th>
<th>Pipe Name</th>
<th>Pipe Class</th>
<th>Design Temperature</th>
<th>Design Pressure</th>
<th>Elevation/Room</th>
</tr>
</thead>
<tbody>
<tr>
<td>8314M3</td>
<td>M100B-1027</td>
<td>8&quot;-HA-19</td>
<td>500°F</td>
<td>150 psig</td>
<td>74'9</td>
</tr>
</tbody>
</table>
Figure 20: M-231 P&ID for RWCU System

Figure 21: IR-M231 Isometric Roadmap for RWCU System
Figure 22: Piping Isometric Drawing for Pipe M100B-1027

Figure 23: Elevation Map of Pipe M100B-1027
4.1.5 Candidate Outlet Two

The second potential outlet would send the purified coolant to the Reactor Core Isolation Cooling system. A schematic of this location is shown in Figure 24. The specific pipe being evaluate was drawn from Figure 25. Figure 26 shows a three-dimensional view of this pipe.

Unfortunately this system returns water to the reactor via the same feedwater line as the RWCU. The feedwater is a safety related area, and therefore does meet imperative design factor one. Also, the elevation of the pipe would require drilling through the floor. This does not meet imperative design factor three. The issues surrounding an outlet in the RCIC deemed the system not ideal for an intertie with the RWCU.

Table 8: Potential Outlet Two Specifications

<table>
<thead>
<tr>
<th>Valve Name</th>
<th>Pipe Name</th>
<th>Pipe Class</th>
<th>Design Temperature</th>
<th>Design Pressure</th>
<th>Elevation/Room</th>
</tr>
</thead>
<tbody>
<tr>
<td>N/A</td>
<td>M-100-539-7</td>
<td>6&quot;-DL-6</td>
<td>850°F</td>
<td>900psig</td>
<td>31’</td>
</tr>
</tbody>
</table>
Figure 24: M-252 SH2 P&ID for RWCU System

Figure 25: IR-M252 SH2 Isometric Roadmap for RWCU System
Figure 26: Piping Isometric Drawing for Pip M-100-539-7
4.1.6 Candidate Outlet Three

The third potential outlet possessed a number of conceptual design issues that made it less attractive. This outlet tied directly into the feedwater line from the RWCU back into the reactor vessel shown in Figure 27. The pipe name was elicited from Figure 28, and a three-dimensional version of the pipe in show in Figure 29. As previously mentioned, during normal RWCU operation, the filtered reactor coolant is returned to the reactor vessel via the feedwater system; however, during a refueling outage feedwater line A is shut off and is a safety related area.

Because of the general location of potential outlet three with respect to the RWCU system, this outlet was only compatible with potential inlet candidate three. As mentioned previously, the pipeline after the regenerative heat exchanger 208-A is located in a safety-related area. Therefore, it was not desirable area for the modification.

Table 9: Potential Outlet Three Specifications

<table>
<thead>
<tr>
<th>Valve Name</th>
<th>Pipe Name</th>
<th>Pipe Class</th>
<th>Design Temperature</th>
<th>Design Pressure</th>
<th>Elevation/Room</th>
</tr>
</thead>
<tbody>
<tr>
<td>N/A</td>
<td>M-100-538-5</td>
<td>6&quot;-DL-6</td>
<td>850°F</td>
<td>900psig</td>
<td>N/A</td>
</tr>
</tbody>
</table>
Figure 27: M-245 P&ID for RWCU System

Figure 28: IR-M245 Isometric Roadmap for RWCU System
4.1.7 Candidate Outlet Four

Potential outlet four would have discharged purified reactor coolant into the FPC system, shown in Figure 30. The pipe name was elicited from Figure 31. Three-dimensional and elevation views are shown by Figures 32 and 33 respectively.

The FPC contains one demineralizer located downstream from this outlet. This posed a problem because this potential outlet would send the purified coolant through a second demineralizer. Despite the redundant purification step, the unnecessary flow through the FPC demineralizer would use more resin. Using more resin is not cost effective and is also expensive to dispose. It was more ideal to put the outlet at another the location in the FPC downstream of the demineralizer.

A design issue with potential outlet four is incompatibility with the pipe class. The pipe class mentioned above in the Table 7 only has a design pressure of 150 psig. Based on the design attributes, the pipe to which the modification ties into must be of great enough design pressure to handle the in-coming flow from the RWCU. Although the difference in diameter between the potential inlets and this outlet would alleviate some of the pressure differential, there would need to be a pressure orifice to keep the flow within the design pressure of the outlet pipe. Selection of a different pipe within the same system could easily solve this compatibility issue.

Table 10: Potential Outlet 4 Specifications

<table>
<thead>
<tr>
<th>Valve Name</th>
<th>Pipe Name</th>
<th>Pipe Class</th>
<th>Design Temperature</th>
<th>Design Pressure</th>
<th>Elevation/Room</th>
</tr>
</thead>
<tbody>
<tr>
<td>6&quot; 29M3</td>
<td>M100BC151-3</td>
<td>6&quot;-HA-19</td>
<td>500°F</td>
<td>150 psig</td>
<td>between 77'-7&quot; and 80'-6&quot;</td>
</tr>
</tbody>
</table>
Figure 30: M-231 P&ID for RWCU System

Figure 31: IR-M231 Isometric Roadmap for RWCU System
Figure 32: Piping Isometric Drawing for Pipe M100BC-151-3

Figure 33: Elevation Map of Pipe M100BC-151-3
**Preliminary Design Summary**

During the Preliminary Design Phase, the most attractive location for inlet from the RWCU and outlet from to the reactor vessel was determined according to maximum flow rate, temperature, pressure, location, radiation level, and elevation. From the information gathered during the Preliminary Design Phase, it was determined that the most attractive inlet was inlet 1. None of the outlets were selected as most ideal, but the Fuel Pool Cooling System was deemed the most ideal for the intertie. Table 11 summarizes the imperative design factors for each possible inlet and outlet.

**Table 11: Evaluation of Imperative Design Factors**

<table>
<thead>
<tr>
<th></th>
<th>Candidate Inlets</th>
<th>Candidate Outlets</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>1</td>
<td>2</td>
</tr>
<tr>
<td>Crucial Design Factor 1</td>
<td>✓</td>
<td>✓</td>
</tr>
<tr>
<td>Crucial Design Factor 2</td>
<td>✓</td>
<td>✓</td>
</tr>
<tr>
<td>Crucial Design Factor 3</td>
<td>✓</td>
<td>✓</td>
</tr>
</tbody>
</table>

The designs that were not ruled out by the crucial design factors were then assessed according to the achievable design factors, summarized in Table 12.
### Table 12: Evaluation of Achievable Design Factors

<table>
<thead>
<tr>
<th>Achievable Design Factor</th>
<th>Candidate Inlets</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>1</td>
</tr>
<tr>
<td>Flow Rate</td>
<td>✓ 222 GPM</td>
</tr>
<tr>
<td>Temperature</td>
<td>✓ 140 °F</td>
</tr>
<tr>
<td>Pressure</td>
<td>✓ 34.3 psig</td>
</tr>
<tr>
<td>Location</td>
<td>✓ elevation 51’</td>
</tr>
<tr>
<td>Radiation Level</td>
<td>✓ not HRA or VHRA</td>
</tr>
</tbody>
</table>

The FPC system deemed to be the most ideal outlet location, however, a pipe in the FPC that was on the same elevation, 51’, needed to be found to avoid drilling through the floor.

#### 4.2 Detailed Design

The goal of the detailed design phase was to produce a complete scope and estimate for the piping modification, as well as fill out all the necessary paperwork to have the project approved by PNPS. The inlet and outlet locations were narrowed down based on the information listed in the preliminary design summary. The detailed design phase consisted of developing a piping isometric drawing of the modification, completing
a cost estimate, filling out the Site Integrated Project Database [SIPD] paperwork, and presenting the final design to the engineers at PNPS.

4.2.1 50% Scope Design Phase

In the 50% Scope of the detailed design phase, several aspects of the design were addressed and analyzed and increased the maturity of the design. A name was given to the new mode of operation the modification would provide, Alternate Injection (AI) Mode. At the outset of 50% phase, it was recognized that AI mode of operation would enhance decay heat removal of the system during refueling, which was an initial goal of a modification proposed in 2000. Estimation of the pressure head to necessary to drive the flow from the RWCU through the FPC was performed. Finally, the following aspects of the design progressed:

- Pipe identification of inlet and outlet locations,
- specifications for operating conditions,
- selection of pipe class, valve, and weldolet.

4.2.1.1 Enhanced Decay Heat Removal

The outline of the modification proposed in 2000 illuminated new benefits to constructing an intertie between the RWCU and FPC. An intertie between the RWCU and FPC would maximize decay heat removal during a refueling outage. Phase one of the modification proposed in 2000 provided detailed information that directly applied to the selected detailed design of this modification. The purpose of the previous design was to optimize thermal efficiency by operating the RWCU as a back-up to RHR. This tangible benefit of the modification designed in 2000 became an additional benefit for this project.
As mentioned in the Background, phase one of the project developed in 2000 explained that an intertie between the FPC and RWCU bypassing the regenerative heat exchangers would remove heat at a rate of 12MBtu/Hr at 210°F (Pilgrim Personnel, 2000a). In order to exploit this modification for decay removal, the modification would need to operational during shutdown. Given that the scope of this project did not include operation procedures, the operation of the modification to maximize decay heat removal would need to be surveyed in a separate project.

4.2.1.2 Outlet Discharge Head Pressure

Given that RWCU is a high-pressure system and the FPC is a low-pressure system, the design specifications for the chosen pipes selected were specifically evaluated. The pipe class of the inlet to the modification was EA. The design pressure of an EA class pipe is 850°F at 600psig pressure. For any pipe class, the pressure it is able to withstand depends on the pressure class and the temperature of the flow. The pressure class information for pipe class EA is located in Table 8. The pipe in the FPC selected for outlet to the modification was M100B1026. The information regarding the outlet location pressure class is in Table 9.

The outlet pressure highlighted in yellow in the Bernoulli’s equation and designated P2 had to be higher than the discharge head pressure of pipe M100B1026 selected for the outlet. The discharge head pressure for the pipe M100B1026 during a refueling outage is 42.5psi (M538, 1996). For the intertie between the RWCU and FPC to be successful, the flow during a refueling outage had to be higher than 42.5 psi. The calculation performed using Bernoulli’s equation proved the pressure of the flow coming from the RWCU could overcome the discharge head pressure of pipe M100B1026.
The flow leaving the RWCU demineralizers would be a nominal 222 GPM (Pilgrim Personnel, 2012b). As mentioned previously, during a refueling outage AFPC is run. Because RHR is run in congruence with FPC, it had to be ensured that the flow added by the modification would not pose a design issue. To make sure this modification would be possible a calculation was completed to check the feasibility of the potential inlets and outlets.

\[
P_2 := P_1 + \rho \cdot \frac{v_1^2 - v_2^2}{2} + \rho \cdot g \cdot (z_1 - z_2) - (\rho \cdot g \cdot h_l) + (\rho \cdot g \cdot h_p)
\]

The variables for this equation are all defined below.

- \( P_1 \) – Atmospheric pressure
- \( \rho \) – Density of water
- \( v_1 \) – Velocity of water entering the system
- \( v_2 \) – Velocity of water entering the pressure header
- \( g \) – Gravity
- \( z_1 \) – Height of the entrance to the system
- \( z_2 \) – Height of the exit to the system
- \( h_p \) – Head created from the RWCU pump
- \( h_l \) – Head loss due to friction
- \( Q \) – Volumetric flow rate
- \( d_{pipe} \) – Diameter of the pipe
- \( A_{pipe} \) – Cross sectional area of the pipe

First, the variables with known values had to be listed.

Variables

\[
P_1 = 14.7 \text{psi} = 1.014 \times 10^5 \text{ m}^{-1} \cdot \text{kg} \cdot \text{s}^{-2}
\]

\[
g = 9.807 \text{ m} \cdot \text{s}^{-2}
\]
Secondly, the velocity entering the system from the reactor vessel had to be determined.

\[ v_1 \]

Next, the actual velocity of the flow in the pipe was determined by dividing volumetric flow rate by the area of the opening of the orifice.

Variables

\[ Q_1 = 220 \text{gpm} = 0.014 \text{m}^3 \cdot \text{s}^{-1} \]

\[ d_{\text{pipe}} = 28 \text{in} \]

\[ d_{\text{pipe}} = 0.711 \text{m} \]

Equations

\[ \Lambda_{\text{pipe}} = \pi \left( \frac{d_{\text{pipe}}}{2} \right)^2 \]
Then, the velocity of the water exiting the piping modification was found by dividing volumetric flow rate by the area of the opening of the orifice.

Variables

\[ Q_1 = 220 \text{gpm} = 0.014 \text{m}^3 \cdot \text{s}^{-1} \]

\[ d_{\text{pipe}} = 4.0 \text{in} \]

\[ d_{\text{pipe}} = 0.102 \text{m} \]

Equations

\[ A_{\text{pipe}} = \pi \left( \frac{d_{\text{pipe}}}{2} \right)^2 \]

\[ v_2 = \frac{Q_1}{A_{\text{pipe}}} \]

Solutions

\[ A_{\text{pipe}} = 8.107 \times 10^{-3} \text{m}^2 \]
Finally, the pressure of the outlet based on the variables given was solved for.

Equation

\[ p_2 = p_1 + \rho \left( \frac{v_1^2 - v_2^2}{2} \right) + \rho \cdot g \cdot (z_1 - z_2) - (\rho \cdot g \cdot h_L) + (\rho \cdot g \cdot h_P) \]

\[ P_2 = 185.33 \text{ psi} \]

During refueling mode the pressure of the flow leaving the RWCU demineralizer was determined to be 34.3 psig (Pilgrim Personnel, 2000c). The pressure class information displayed in Tables 8 and 9 proved the pipes could withstand the temperature and pressure of the flow during refueling mode. Despite the low design pressure of the outlet into the FPC, the pipe was eight inches in diameter. As mentioned in the preliminary design, an increase in cross sectional area reduces the overall pressure applied to the pipe.

### 4.2.1.3 Pipe Identification of Inlet and Outlet Locations

Based on the information from the preliminary design phase, it was determined that the most ideal inlet location to the modification would be the general location following the RWCU demineralizers and downstream of valve 1201-75. The specific pipe selected was a 4”EA pipe named M100BC203-7SF.

The preliminary design phase also determined that the most ideal outlet location would be after the demineralizer in the FPC. The specific pipe selected was an 8” HA
pipe class and was named M100B1026. It was determined both pipes selected for the intertie would be able to process the feed leaving the RWCU.

4.2.1.4 Specification of Operating Conditions

It is important to understand the normal operating conditions of the RWCU and FPC systems so that it can be ensured that operation in AI mode will run smoothly. The following tables outline the normal operating conditions and design specifications of both systems.

Table 13: RWCU Conditions During Normal Operation (M300, 2013)

<table>
<thead>
<tr>
<th>Pipe Class</th>
<th>Service Description</th>
<th>Design Pressure (psig)</th>
<th>Design Temp. (°F)</th>
<th>Normal Operating Pressure (psig)</th>
<th>Normal Operating Temp. (°F)</th>
<th>Max Operating Pressure (psig)</th>
<th>Max Operating Temp. (°F)</th>
</tr>
</thead>
<tbody>
<tr>
<td>EA</td>
<td>E216B to E206C Inlet</td>
<td>1175</td>
<td>545</td>
<td>1175</td>
<td>120</td>
<td>1362</td>
<td>545</td>
</tr>
</tbody>
</table>

Table 14: FPC Conditions During Normal Operation (M300, 2013)

<table>
<thead>
<tr>
<th>Pipe Class</th>
<th>Service Description</th>
<th>Design Pressure (psig)</th>
<th>Design Temp. (°F)</th>
<th>Normal Operating Pressure (psig)</th>
<th>Normal Operating Temp. (°F)</th>
<th>Max Operating Pressure (psig)</th>
<th>Max Operating Temp. (°F)</th>
</tr>
</thead>
<tbody>
<tr>
<td>HA</td>
<td>T-204 to Fuel Pool</td>
<td>175</td>
<td>165</td>
<td>130</td>
<td>110</td>
<td>175</td>
<td>165</td>
</tr>
</tbody>
</table>

The large difference in operating pressure between the two systems was of greatest concern in this case, and was the driving force for the piping selection for the
modification. The piping selection is crucial for the safe and effective operation of the piping modification during AI mode.

The specifications of the modification for the development of AI mode are based on the criteria described in the previous table. AI mode has to be able to be designed to accommodate the normal operating conditions of both the RWCU and FPC systems. In this case the pipe class EA was selected for the modification because it is more durable than the HA tie in the FPC. Given that the flow leaving the RWCU demineralizers will always be less than 130°F, the inlet flow to the modification during AI mode was assumed to operate under the conditions listed in the following table.

**Table 15: Modification Inlet Conditions During AI Mode (Pilgrim Personnel, 2000c)**

<table>
<thead>
<tr>
<th>Pipe Class</th>
<th>Flow Rate (GPM)</th>
<th>Temperature (°F)</th>
<th>Pressure (psig)</th>
</tr>
</thead>
<tbody>
<tr>
<td>EA</td>
<td>222</td>
<td>125</td>
<td>34.3</td>
</tr>
</tbody>
</table>

**4.2.1.5 Pipe Class Selection**

The selected pipe specifications for the modification ensure that the pipe will be able to handle the volume, temperature and pressure of the process water passing through during AI mode. These specifications are listed in the following table.

**Table 16: AI Mode Modification Piping Specifications (M300, 2013)**

<table>
<thead>
<tr>
<th>Pipe Class</th>
<th>Material Description</th>
<th>Schedule</th>
<th>Material Specification</th>
</tr>
</thead>
<tbody>
<tr>
<td>4” EA</td>
<td>stainless steel</td>
<td>80</td>
<td>2.5” or larger</td>
</tr>
</tbody>
</table>
4.2.1.6 Valve Selection

The detailed design of the modification also includes an inboard isolation valve. This valve is used to open and close the flow into the modification during AI mode. The design of the valve must accommodate the flow, pressure and temperature of the process water passing through the system. The specifications for the necessary valve are listed in the table below.

Table 17: AI Mode Inboard Isolation Valve Design Specifications (M300, 2013)

<table>
<thead>
<tr>
<th>Pipe Class</th>
<th>Material Description</th>
<th>Pressure Class (lb)</th>
<th>Material Specification</th>
</tr>
</thead>
<tbody>
<tr>
<td>4” EA</td>
<td>stainless steel</td>
<td>600</td>
<td>2.5” to 10”</td>
</tr>
</tbody>
</table>

There are two different types of valves that are optional for the inboard isolation valve. A globe valve is capable of varying the flow into the modification pipe, whereas a gate valve would either completely shut off access to the pipe or allow maximum flow to the pipe. The pros and cons of each type of valve are outlined in the diagram below.
Figure 34: Globe Valves vs. Gate Valves

These pros and cons weigh heavily into the final modification designed are discussed in greater detail in the 90% Scope Meeting section. At the time of the 50% design phase, both valves were still under consideration. The following table outlines the material specification options for the inboard isolation valve.

Table 18: AI Mode Inboard Isolation Valve Material Specifications (M300, 2013)

<table>
<thead>
<tr>
<th>Valve Type</th>
<th>Valve Mark Numbers</th>
</tr>
</thead>
<tbody>
<tr>
<td>Globe</td>
<td>N136M3</td>
</tr>
<tr>
<td>Gate</td>
<td>N14M3</td>
</tr>
</tbody>
</table>

Due to the large difference in operating pressure between the RWCU and FPC systems during normal operation, a malfunction of the primary isolation valve during AI mode could damage the FPC system. For this reason, the installation of secondary
isolation valve was considered. The design and material specifications for the secondary isolation valve would be the same as those listed for the primary isolation valve. The decision for implementation of the secondary isolation valve would consider the cost versus the severity of the damage that could be caused pending the failure of the primary valve.

Both valves would have the option of being automated or operated manually. These manners of operation were reviewed for safety and cost related issues in order to ensure that the most realistic design was constructed. For the purposes of integrity and ensuring all possibilities are considered, the pros and cons of both manners of operations are outlined in the following figure.

![Figure 35: Manual vs. Automated Valve Operation](image)

**Figure 35: Manual vs. Automated Valve Operation**

Worker dose that can be accumulated during AI mode operation is part of the achievable design factors. The modification from 2000 outlined the specific worker dose
that would be accumulated during a refueling outage to manually operate these valves.

The details are outlined in the table below.

**Table 19: Dose Associated with Manual Valve Operation (Pilgrim Personnel, 2000b)**

<table>
<thead>
<tr>
<th>RWCU/FPC</th>
<th>Operation During Each RFO (man-hrs)</th>
<th>Effective Dose Rate (rem/hr)</th>
<th>Estimated Dose (man-rem)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Intertie / Bypass</td>
<td>2.0</td>
<td>0.02</td>
<td>0.04</td>
</tr>
</tbody>
</table>

The pros and cons of the different valve types and manners of operations along with a final valve choice were considered in greater detail in the 90% design phase.

**4.1.1.7 Weldolet Connection to FPC**

Due to the difference in pipe size between the modification inlet and outlet, a weldolet would have to be implemented on the dispersal end to connect the 4” modification pipe to the 8” return line in FPC. There was some discretion about whether the outlet should intersect the FPC line due to the presence of valve 19-HO-188. The intertie could be implemented either up-stream or down-stream of this valve depending on the feasibility and benefits of each location.
4.2.2 90% Scope Design Phase

The final, 90% phase of the detailed design addressed in detail all remaining aspects of the design and resulted in detailed drawings, bill of materials, cost estimates, and dose savings estimates for the modification.

The first design decision in this phase concerned a gate valve to be used to isolate the modification because there is no need to throttle (See Glossary) the valve.

4.2.2.1 Precise location of Outlet Upstream of 19-HO-188

The second matter considered in the 90% phase was the benefits of placing the outlet upstream or downstream of valve 19-HO-188. The image below is of a valve; specifically valve 19-HO-188, which was located in close proximity to the outlet of the modification into FPC. The placement of the outlet to the modification both upstream (See Glossary) and downstream (See Glossary) of valve 19-HO-188 posed specific benefits.
If the outlet to the modification was located downstream, the flow could have only been discharged into the spent fuel pool. Also, a walkdown confirmed downstream of the valve was extremely obstructed by previously installed piping. Furthermore, there is a hot spot (See Glossary) at the connection of the two pipes shown in Figure 36 downstream of valve 19-HO-188. Hot spots are generally contaminated or high radiation areas, so they are not ideal for a piping modification for dose reasons.

Installing the outlet upstream of valve 19-HO-188 posed less design constraints than upstream. If the outlet to the modification was located upstream, the flow could be discharged to different locations. If valve 19-H0-188 is open, the flow will be discharged into the spent fuel pool. If valve 19-HO-188 is closed, the flow can be discharged into the reactor basin. Given that the flow would be of high purity, the reactor basin was considered a more ideal location.
Installing the outlet upstream of valve 19-HO-188 was also considered more practical after the team completed a walkthrough of the Fuel Pool Corridor. There is a wall adjacent to the upstream location, and it could be used to run the length of pipe. The lack of obstruction and diverse discharge options made upstream of valve 19-HO-188 a more ideal location for the outlet.

4.2.2.1 Piping Isometric Drawing

A piping isometric drawing is essential for understanding the exact dimensions and location of the modification, as well as physically visualizing the modification. The use of the piping isometrics for the inlet, M100BC-203-7SF, and outlet, M100B1026, locations, as well as drawing M19, all of which can be found in Appendix C, allowed for the development of a new piping isometric for the modification. Using the same coordinate system as all other available drawings, the following piping isometric was developed.
Figure 37: Piping Isometric Drawing of Modification

This is a basic piping isometric drawing that would be used for the implementation of the modification. It includes all necessary dimensions, elevations, reference pipes, valves and support locations for a complete understanding of the piping. For visualization purposes, a second copy of the piping isometric has been included in Figure 38. This drawing includes a 3-D representation of the wall through which the modification travels, the relative location of the pipe to the back wall of the FPC corridor and RWCU holding pump room, as well as a few highlighted reference locations.
Figure 38: Piping Isometric Drawing for Modification with Extended Features

This allows for a better understand of the piping layout and modification function. The entire pipe is highlighted in blue for easy reference. Highlighted in green is the tee that connects the mod to the outlet from the RWCU holding pump room. The isolation valve that would be implemented to control flow during AI mode is pictured in red. When this valve is open the flow will pass from the RWCU, through the valve, up and through the wall of the RWCU holding pump room and into the FPC corridor. The pipe travels along the back wall of the corridor and finally turns off to connect to the FPC line. Lastly, the weldolet that connects the modification to the FPC line is highlighted in orange. The piping isometric shows the location of the piping relative to gridlines K and 13. This gives a reference point for the modification so that other drawings that may or
may not include the modification can be put into perspective with the new piping. All “EL” values give the piping elevation at that particular location relative to the ground. The floor level of the drawing is show in the bottom left corner of the drawing at a value of 74’0”.

Following the pipe design, a list of necessary materials had to be developed to begin the estimate portion of the design.

4.2.2.2 Materials

The materials lists followed the discussion regarding the piping isometric in the 90% Scope Meeting. The following is a list of all the materials necessary for implementation of the modification. The column QA designates whether the part needed was safety related or not. As previously stated, for a number of reasons the team chose all non-q or non-safety related materials. The Catalog Identification [ID] (See Glossary) and IAS description (See Glossary) column were drawn from a program called Asset Suite©. Direct access to Asset Suite was originally unavailable but was gained via the design team member. The program contains information for all parts ordered by the plant. For any listed material Catalog ID, the corresponding price and potential warehouse location was elicited. It was imperative to use Asset Suite© because many material necessary for the modification were already in the warehouse ready for implementation. Though not every material has Catalog ID number, the prices elicited from the Catalog ID allowed the materials estimate for the AI modification to be calculated.
### Table 20: EA Pipes and Fittings

<table>
<thead>
<tr>
<th>QA</th>
<th>Catalog ID.</th>
<th>IAS Description</th>
<th>Quantity Required</th>
</tr>
</thead>
<tbody>
<tr>
<td>Non-Q</td>
<td>00PIL33999</td>
<td>valve, gate, *,033907, 4x600#, SST, a-351, 316, SCH80, CF8M, ANC</td>
<td>1</td>
</tr>
<tr>
<td>Non-Q</td>
<td>00PIL90491</td>
<td>PIPE, S.S, ASTM A-312 SEAMLESS, TP-304 GR 4&quot;, SCH 80, 10</td>
<td>50ft</td>
</tr>
<tr>
<td>Non-Q</td>
<td>00PIL82861</td>
<td>ELBOW, PIPE, *, 009842, 4, 45DEG,LR, SCH 80, A-403, GRWP316</td>
<td>4</td>
</tr>
<tr>
<td>Non-Q</td>
<td>00PIL92528</td>
<td>ELBOW, PIPE, *, 009927, 4, 90DEG, LR, SCH80, A-403, GRWO304, BW</td>
<td>6</td>
</tr>
<tr>
<td>Non-Q</td>
<td>00PIL82862</td>
<td>TEE PIPE, *, 031840, 4x4x4, SCH80, A-403, GRWP 316</td>
<td>1</td>
</tr>
</tbody>
</table>

### Table 21: HA2 Pipes and Fittings

<table>
<thead>
<tr>
<th>QA</th>
<th>Catalog ID.</th>
<th>IAS Description</th>
<th>Quantity</th>
</tr>
</thead>
<tbody>
<tr>
<td>Non-Q</td>
<td>NOT FOUND IN IAS</td>
<td>NOT FOUND IN IAS</td>
<td>1</td>
</tr>
</tbody>
</table>
Table 22: Structural Materials

<table>
<thead>
<tr>
<th>QA</th>
<th>Catalog ID.</th>
<th>IAS Description</th>
<th>Quantity Required</th>
</tr>
</thead>
<tbody>
<tr>
<td>Non-Q</td>
<td>32074301</td>
<td>PLATE, CS, A36, 1/2&quot;THK, 12&quot; WD, 12&quot; LG</td>
<td>3</td>
</tr>
<tr>
<td>Non-Q</td>
<td>00PIL29628</td>
<td>ANGLE, *, 000549, 3x3x1/4 X20' CS HOT DIPPED</td>
<td>3</td>
</tr>
<tr>
<td>Non-Q</td>
<td>00PIL83275</td>
<td>BOLT, ANCHOR, *, 003746, 3/4X8, CS, KWIK, II, HILTI 00045377</td>
<td>12</td>
</tr>
<tr>
<td>Non-Q</td>
<td>NOT FOUND IN IAS</td>
<td>NOT FOUND IN IAS</td>
<td>3</td>
</tr>
</tbody>
</table>

Based on the list of materials necessary for the construction of the modification and the corresponding prices found in Asset Suite®, a materials estimate was developed.

4.2.2.3 Cost Estimate

The overall project estimate consists of two parts: a bill of materials, and man-hours. These are the two components of the estimate that are of relevance to this particular project and are all that was required by PNPS for the completion of the project estimate.
The bill of materials below in Table 23s, Table 24, and Table 25, outlines the individual cost of all the necessary materials as well as a total material cost.

**Table 23: EA pipes and Fittings**

<table>
<thead>
<tr>
<th>Description of Material</th>
<th>Technical Requirements</th>
<th>Cost per unit</th>
<th>Cost for material</th>
</tr>
</thead>
<tbody>
<tr>
<td>4” Gate Valve</td>
<td>Maximum Carbon Content 0.035% For All Grades ASME SA Materials May Be Used For M300 Pipe Class EA</td>
<td>15,772.00</td>
<td>15,772.00</td>
</tr>
<tr>
<td>4” Schedule 80</td>
<td>Maximum Carbon Content 0.035% For All Grades ASME SA Materials May Be Used For M300 Pipe Class EA</td>
<td>95.88</td>
<td>4794.00</td>
</tr>
<tr>
<td>Stainless Steel Pipe</td>
<td></td>
<td>94.86</td>
<td>379.44</td>
</tr>
</tbody>
</table>

93
<table>
<thead>
<tr>
<th>WP316LN</th>
<th>4” 90 Deg Long Radius Elbow Buttwelding Ends Schedule 80 Stainless Steel ASTM A-403 Gr WP304 ASTM A-403 Gr WP304-W ASTM A-403 Gr WP316L ASTM A-403 Gr WP316LN</th>
<th>Maximum Carbon Content 0.035% For All Grades ASME SA Materials May Be Used For M300 Pipe Class EA</th>
<th>195.58</th>
<th>1173.48</th>
</tr>
</thead>
<tbody>
<tr>
<td>4” Straight Tee Buttwelding Ends Schedule 80 Stainless Steel ASTM A-403 Gr WP304 ASTM A-403 Gr WP304-W ASTM A-403 Gr WP316L ASTM A-403 Gr WP316LN</td>
<td>Maximum Carbon Content 0.035% For All Grades ASME SA Materials May Be Used For M300 Pipe Class EA</td>
<td>173.41</td>
<td>173.41</td>
<td></td>
</tr>
</tbody>
</table>
### Table 24: HA2 Pipes and Fittings

<table>
<thead>
<tr>
<th>Description of Material</th>
<th>Technical Requirements</th>
<th>Cost per unit</th>
<th>Cost for material</th>
</tr>
</thead>
<tbody>
<tr>
<td>4&quot; Butt Welding Outlet for 8&quot; Run Weldolet or Trans-O-Con</td>
<td>Maximum Carbon Content 0.035% For All Grades ASME SA Materials May Be Used For M300 Pipe Class HA2</td>
<td>130.00</td>
<td>130.00</td>
</tr>
<tr>
<td>Schedule 40 Stainless Steel</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>ASTM A-403 Gr WP316L</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>ASTM A-182 Gr F316L</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

### Table 25: Structural Materials

<table>
<thead>
<tr>
<th>Description of Material</th>
<th>Technical Requirements</th>
<th>Cost per unit</th>
<th>Cost for material</th>
</tr>
</thead>
<tbody>
<tr>
<td>Carbon Steel Plate A36 12&quot;x 12&quot;x 1/2&quot; Thick</td>
<td>A36 Carbon Steel</td>
<td>100</td>
<td>300</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Carbon Steel Angle A36 3”x 3”x 1/4” w/ insulating pad</td>
<td>A36 Carbon Steel</td>
<td>341.24</td>
<td>1023.72</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>HILTI QUICK BOLTS</td>
<td>n/a</td>
<td>5.39</td>
<td>64.68</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Stainless 4” U-bolt Double Nut</td>
<td>SST</td>
<td>50</td>
<td>150</td>
</tr>
</tbody>
</table>

Based on the bill of materials above the total cost of all necessary materials for construction of this modification is $23,960.73. However, this does not cover man-hour costs for design and implementation of the modification. This is further explained in the following sections.

4.2.2.4 Man Hours Cost of Implementation

To calculate the man-hours cost of implementation, all the types of work needing to be done were identified and hourly rates were researched. After deciding what work needed to be done, the amount of time needed for each task was researched. Once the hourly rates were found, the amount of hours needed for each task was entered into the Man Hours Cost of Implementation Excel sheet and the total cost was calculated.

<table>
<thead>
<tr>
<th>Position (type over text below with craft description)</th>
<th>Rates</th>
</tr>
</thead>
<tbody>
<tr>
<td>FME Monitor (Apprentice Decon)</td>
<td>$25.00</td>
</tr>
<tr>
<td>Scaffolding</td>
<td>$75.00</td>
</tr>
<tr>
<td>Manufacturing Prep</td>
<td>$100.00</td>
</tr>
<tr>
<td>Welding and Construction</td>
<td>$100.00</td>
</tr>
<tr>
<td>Henry Stanley</td>
<td>$100.00</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Rates</th>
</tr>
</thead>
<tbody>
<tr>
<td>ST</td>
</tr>
<tr>
<td>OT</td>
</tr>
<tr>
<td>DT</td>
</tr>
<tr>
<td>$25.00</td>
</tr>
<tr>
<td>$37.50</td>
</tr>
<tr>
<td>$75.00</td>
</tr>
<tr>
<td>$112.50</td>
</tr>
<tr>
<td>$100.00</td>
</tr>
<tr>
<td>$150.00</td>
</tr>
<tr>
<td>$100.00</td>
</tr>
<tr>
<td>$150.00</td>
</tr>
</tbody>
</table>

Figure 39: Man hours Template Portion

| Labor Est  | $153,450 |
| Mob/Demob  | $0       |
| Other Costs: | $0     |
| Total Est:  | $153,450 |

Figure 40: Man hours Total Estimate
TOTAL IMPLEMENTATION + TOTAL MATERIALS
153,450.00 + 23,960.73 = 177,410.73

4.2.2.5 Dose Calculation

The calculation of dose savings was obtained shortly after the 90% Scope Meeting from the ALARA Supervisor. The ALARA supervisor works in the Radiation Protection Department at Pilgrim Nuclear Power Station. An ALARA is one of the individuals who determine the maximum dose for nuclear plant each year. The maximum occupational dose is the sum every individual's occupational dose for that year. For example, in 2012 Pilgrim Nuclear Power Station set a goal to keep the maximum occupational radiation dose less than 25 Rem for approximately 650 workers (Pilgrim Personnel, personal communication, January 9, 2013). While conducting research onsite at Pilgrim Nuclear Power Station, the team witnessed Pilgrim accomplish their goal of less than 25 Rem in 2012.

For realistic and achievable maximum occupational dose limits to be set, the ALARA Supervisor must have a working knowledge of dose accumulation for modification in different parts of the nuclear plant. For this reason, he was charged with the task of determining the amount of occupational dose the modification would reduce per refueling outage.

When making dose calculations, the ALARA Supervisor determines the dose calculations by references dose accumulated during similar modifications. However, the modification the team designed had only been successfully designed and installed in one
other nuclear power plant in the United States (Pilgrim Personnel, personal communication, March 29, 2013).

Because the modification concept itself was relatively new, the ALARA Supervisor drew his dose savings calculation from other nuclear power plants developing similar designs and dose calculations from prior refueling outages. When the refuel cavity is flooded, the dose rates associated with the cavity are 12mrem/hr (Pilgrim Personnel, personal communication, March 29, 2013). As previously stated, the radioactive contaminants swept into the reactor coolant during the shutdown process largely contribute to the high dose rate around the refuel cavity. Approximately 36 hours after the refuel cavity is flooded, AFPC mode is initiated (Pilgrim Personnel, personal communication, March 29, 2013). AFPC mode lowers the dose rates around the refuel cavity but does nothing to lower the initially high 12mrem/hr dose rate. After the implementation of the modification, RWCU Alternate Injection mode would significantly reduce the dose in the area around the refuel cavity prior to the initiation of AFPC mode (Pilgrim Personnel, personal communication, March 29, 2013). Based on the information provided, the conservative estimate for the occupational dose the modification would save every refueling outage was 2,000mrem (Pilgrim Personnel, personal communication, March 29, 2013).

4.3 Detailed Design Summary

During the Detailed Design Phase, it was determined the modification would connect pipe M100BC203-7SF in the RWCU to pipe M100B1026 in the FPC. Because pipe M100BC203-7SF had a higher durability than pipe M100B1026, the pipe class EA of M100BC203-7 was selected for the material specification. The material selected for
the pipes and fittings was determined to accommodate the maximum rate, temperature, and pressure of the flow leaving the demineralizer in the RWCU.

It was determined the modification would begin in the holding pump room on the 51’ elevation with manually operated gate valve. The holding pump room is generally classified as a HRA, but it only required the modification to go through a wall rather than a floor. The pipe would run through the holding pump wall and connect with pipe M100B1026 in the Fuel Pool Corridor via a weldolet. A walkdown of the areas was performed, and it was concluded building the modification in the selected areas was feasible.

The proper design materials were listed and then researched in IAS. From the Catalog ID numbers found, an estimate for the cost of materials was determined. The Design team used a man hours template to develop an estimate for the personnel cost for the modification. The total cost for man hours and materials was 177,410.73. ALARA supervisor at Pilgrim calculated the occupational dose that the implementation of the modification could save. He determined the occupational dose saved by the modification every refueling outage would conservatively be 2000mrem (Pilgrim Personnel, personal communication, March 29, 2013).
5. CONCLUSION

The goal of the project was to improve the operational flexibility of the RWCU system and reduce personnel exposure to Cobalt-60. Non-radioactive cobalt is a component of the valve hard facing alloy Stellite. Over time Stellite is eroded by reactor coolant causing non-radioactive cobalt to seep into the reactor coolant. When the non-radioactive cobalt is in the reactor coolant, it can pick up a neutron and become radioactive. The radioactive material settles on the primary piping, but when the reactor is shutdown for a refueling outage the radioactive layer is disturbed and mixes with reactor coolant (Pilgrim Personnel, personal communication, March 29, 2013). The presence of the radioactive cobalt in the reactor coolant causes high dose rates during a refueling outage (Pilgrim Personnel, personal communication, March 29, 2013). In addition, the removal of the Cobalt-60 is costly and prolongs the refueling outage. If the RWCU could be utilized, the radioactive layer could be filtered reducing personnel exposure considerably.

In 2000, a member of the MCSDE department developed a design for an intertie between the RWCU and FPC. Despite the tangible benefits of the modification, a HELB in the RWCU struck down the design. After meticulous work with members of the MCSDE department, a feasible intertie between the RWCU and FPC was developed. The intertie designed would improve the operational flexibility of the RWCU by allowing it to operate during a refueling outage. At the end of the 90% Scope Meeting on March 13, 2013, the design approved and sent to upper management for financial approval.

Given that Stellite is used in valves in almost every nuclear power plant, modifications involving RWCU operational flexibility are still being developed in
nuclear power plants all over the country. However, only one nuclear power plant has successfully implemented a design. Unfortunately, information regarding the successful modification design was not documented or available to the team. For this reason, the design team documented every phase and aspect of the modification. The meticulous documentation of the modification can now serve as guideline to other BWR power plants trying to reduce personnel exposure to Cobalt-60.

As stated above, the approved design for the modification will lessen the amount of time needed for a refueling outage. At the present time, personnel must remove the radioactive material from the reactor coolant prior to refueling. After the modification is implemented, the RWCU can filter out the Cobalt-60 directly after the scramming process.

The most important aspect of the modification is the occupational dose it will prevent after implementation. While completing research on site at Pilgrim Nuclear Power Station, the team witnessed the profound importance on reducing occupational radiation dose. The conservative estimate for personnel exposure saved by the modification every refueling outage was 2000mrem (Pilgrim Personnel, personal communication, March 29, 2013). For a refueling outage year, the maximum occupational radiation dose is <37 Rem (Pilgrim Personnel, personal communication, March 13, 2013). The implementation of the modification would lower the maximum occupational radiation dose each refueling outage following the implementation.
In conclusion, the implementation of the modification will serve as model for other nuclear plants attempting the modification, reduce outage time, and reduce occupational radiation exposure for outage personnel. The implementation of the modification will benefit Pilgrim Nuclear Power Station for the duration of its existence.
REFERENCES


NANTEL. (2010). Entergy® Radiation Worker Training Course. (p. 7-9). Retrieved February 6, 2013 from

https://nantel.org/Saba/Web/Main.


APPENDICES

Appendix A: Abbreviations

AFPC – Augmented Fuel Pool Cooling
AFPC & P – Augmented Fuel Pool Cooling and Purification
ALARA – As Low As Reasonably Achievable
BWR – Boiling Water Reactor
EPA – Environmental Protection Agency
FME – Foreign Material Exclusion
FPC – Fuel Pool Cooling
GE – General Electric
HELB – High Energy Line Break
HRA – High Radiation Area
HX – Heat Exchanger
IRM – Isometric Roadmap
LHRA – Locked High Radiation Area
MSCD – Mechanical Structural Civil Engineering Department
NEI – New England Energy Institute
PC – Protective Clothing

PNPS – Pilgrim Nuclear Power Station

PWR – Pressurized Water Reactor

P&ID – Piping and Instrumentation Diagram

RA – Radiation Area

RCIC – Reactor Core Isolation Cooling

RFO – Refueling Outage

RHR – Residual Heat Removal System

RP – Radiological Personnel

RWCU – Reactor Water Cleanup

SIPD – Site Integrated Project Database

USNRC – United States Nuclear Regulatory Commission

VHRA – Very High Radiation Area
Appendix B: Glossary

Activated Fission Products (noun) – nuclides formed during the transformation of stable reactor components

Catalog ID (noun) – reference number assigned for the specific material

Contamination (noun) – radioactive material where it should not be

Downstream (adjective) – refers to the location of an object that the flow has not gone through yet.

Flange (noun) – a projecting flat rim or collar on an object

Fuel Outage (noun) – the replacement of empty fuel canisters in the core of the reactor every 18-24 months

Gamma Rays (noun) – electromagnetic radiation capable of penetration, they are short in length than x-rays

Gate Valve (noun) – a valve that can only be set completely closed or completely open

Globe Valve (noun) – A valve that can be used to adjust flow

Green House Gases (noun) – gases in the atmosphere that absorb and emit radiation within the thermal infrared range

Hot Spot (noun) – an area with very high radiation dose relative to its size
IAS Description (noun) – the description entered manually into the IAS system for a given material. Because IAS descriptions are entered manually, they can vary even though the specific material being searched for is the same

Isolation Valve (noun) – valve used to separate flow in case of system failure

Isometric Roadmap (noun) – a mechanical drawing containing the pipe name and pipe class of the pipes that comprise the system, as well as all instrumentation labels

Kinetic Energy (noun) – energy characterized by a body in motion

Nuclear Fission Reaction (noun) – a reaction in which a heavy nucleus splits spontaneously releasing energy

Piping and Instrumentation Diagram (noun) – a mechanical drawing containing the relevant piping and instrumentation information that compromises a system

Piping Isometric (noun) – a 3-D mechanical drawing of an individual pipe within a system that includes all relevant elevations, valves, and connections to other pipes

Precoat material (noun) – substance used with the resin in the filter demineralizer to remove contaminants from reactor coolant

Radiation (noun) – Energy released in the form of radioactive waves

Reactor Basin (noun) – Large concrete tank where various secondary processes occur

Resin (noun) – substance used in the filter demineralizer for ion-exchange of contaminants
Spent Fuel Pool (noun) – storage area for empty fuel canisters

Throttle (verb) – to partially open or close a valve to limit or augment the flow rate

Upstream (adjective) – refers to an object that the flow has gone through already

Weldolet (noun) – a specialized type of reducer designed to connect large pipes to small pipes
Appendix C: Scope Meeting Checklists

Detailed Design Checklist for 50% Scope Meeting

1. Review Design Process
   a. Modification Inlet
      i. M-247
      ii. Detail A – M-247
      iii. Piping Isometric M100BC203-7SF
   b. Modification Outlet
      i. M-231
      ii. Detail B – M-231
      iii. Piping Isometric M100B1026

2. Alternate Injection Mode Design Specification
   a. Normal Operating Conditions in RWCU and FPC
   b. Alternate Injection Assumed Inlet Conditions
      i. Piping Specifications with Respect to Inlet Conditions
      ii. Inlet Valve Specifications
         1. Operational Secondary Isolation Valve
         2. Valve Specifications from M300 with Respect to Inlet Conditions
         3. Gate vs. Globe Valves

3. Outlet Specifications
   a. Weldolet
Detailed Design Checklist for 90% Scope Meeting

Alternate Injection Mode: Detailed Design Packed

Meeting: 90% Review / Final MQP Presentation

Date / Time: Wednesday, March 13, 2013 / 3:00pm

Location: PNPS Conference Room

Attendees:

<table>
<thead>
<tr>
<th>Name</th>
<th>Department</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pilgrim Personnel</td>
<td>PNPS</td>
</tr>
<tr>
<td>Robert Thompson</td>
<td>WPI Faculty</td>
</tr>
<tr>
<td>Peter Miraglia</td>
<td>WPI Faculty</td>
</tr>
<tr>
<td>Katherine Goldberg</td>
<td>MCSDE Intern / WPI Student</td>
</tr>
<tr>
<td>Scott Gallagher</td>
<td>MCSDE Intern / WPI Student</td>
</tr>
<tr>
<td>Nikole Stone</td>
<td>MCSDE Intern / WPI Student</td>
</tr>
</tbody>
</table>

Checklist:

1. 50% Review Meeting Minutes
   a. Review minutes
   b. Previous concerns
2. Piping Isometric Drawing
   a. M100B1026, M100BC-203-7SF and M19
   b. New piping isometric drawing explanation
3. Materials
   a. Necessary materials for implementation
   b. Spare materials
4. Estimate
   a. Bill of materials
   b. Man hour costs of implementation
5. SIPD
6. Concerns
50% Review Meeting Minutes

**Date / Time:** Thursday, February 21, 2013 / 3:30pm

**Location:** PNPS Conference Room 3F

1. Review Design Process
   a. Modification Inlet
      i. M-247
      ii. Detail A – M-247
      iii. Piping Isometric M100BC203-7SF

2. Alternate Injection Mode Design Specification
   a. Normal Operating Conditions in RWCU and FPC
   b. Alternate Injection Assumed Inlet Conditions
      i. Piping Specifications with Respect to Inlet Conditions
      ii. Inlet Valve Specifications
         1. Operational Secondary Isolation Valve
         2. Calve Specifications from M300 with Respect to Inlet Conditions
         3. Gate vs. Globe Valves

**Will the modification outlet be located upstream or downstream of valve 19-HO-188?**

The inlet will be located two feet upstream of valve 19-HO-188. This was chosen because of the apparent benefits of having the outlet at this location. These benefits include the following:

- Space availability
- Redirection to reactor basic
- Operation of RWCU, FPC and RHR simultaneously

   b. Modification Outlet
      i. M-231
      ii. Detail B – M-231
      iii. Piping Isometric M100B1026

2. Alternate Injection Mode Design Specification
   a. Normal Operating Conditions in RWCU and FPC
   b. Alternate Injection Assumed Inlet Conditions
      i. Piping Specifications with Respect to Inlet Conditions
      ii. Inlet Valve Specifications
         1. Operational Secondary Isolation Valve
         2. Calve Specifications from M300 with Respect to Inlet Conditions
         3. Gate vs. Globe Valves

**Will a globe valve or a gate valve be used?**

A gate valve will be used to isolate the modification.


**Will the valve be automated or manually operated?**

The valve will be manually operated to prevent malfunctions and eliminate the need for backup components in cases where an automated valve should fail.

3. Outlet Specifications
   a. Weldolet