1.1 INTRODUCTION

1.1.1 GENERAL

This update of the FSAR for the Palisades Plant is pursuant to the requirements of 10 CFR 50.71(e). It represents the current safety evaluation of the Palisades Plant by incorporating information relating to Plant modifications, revised analysis, additional studies and results of NRC special programs which became available subsequent to the release of the original FSAR on November 1, 1968.

Inasmuch as possible, the original (existing as of 1980) FSAR chapter format has been retained except for the inclusion of new material or rearrangement of original material for editorial purposes. All appendices and amendments of the 1980 FSAR have been incorporated into the Update text. The Security Plan, Emergency Plan and the QA Program have been omitted from the Update and should be consulted independently. New material not easily incorporable into the text may appear as new appendices. Most pages of the FSAR Update are new and cannot be directly compared to the 1980 FSAR.

1.1.2 LICENSING HISTORY

Consumers Power filed, on Docket 50-255, a Construction Permit and Operating License (CPOL) Application (which included the PSAR) to the AEC on June 2, 1966 for the Palisades Plant to be located near South Haven, Michigan. The application was for development of a 2,650 MWt (design core power) commercial nuclear-powered electrical generating facility to be operated at 2,200 MWt or an equivalent electrical output of 700 MWe. On March 14, 1967, the AEC issued Consumers Power a Class 104 Construction Permit CPPR-25, pursuant to Section 104(b) of the Atomic Energy Act, to construct a Combustion Engineering pressurized water reactor (PWR) with a full-power design rating of 2,650 MWt. Subsequent to the original CPOL/PSAR application, eight additional PSAR amendments were filed addressing NRC concerns and addition of the Technical Specifications. On November 1, 1968, Consumers Power filed with the AEC an Operating License (OL) Application (which included the FSAR) as Amendment 9 to the CPOL Docket 50-255, to operate the Palisades Plant at 2,200 MWt core power. Following submission of the initial FSAR as Amendment 9 to the CPOL, 23 subsequent amendments to the FSAR were submitted to the NRC. They were identified as Amendments 10 through 32 to the CPOL Docket 50-255, the most extensive of which was the Full-Term Operating License (FTOL) Application, Amendment 28. In addition, nine minor revisions were subsequently submitted to the NRC.

On March 24, 1971, the NRC issued Interim Provisional Operating License IDPR-20 to be effective for 1-1/2 years to operate Palisades up to 1 MWt. Subsequent amendments to IDPR-20 were issued on November 20, 1971 to operate up to 440 MWt (20% power); March 10, 1972 to operate up to 1,320 MWt (60% power); September 1, 1972 continued operation at 1,320 MWt (60% power); October 16, 1972 authorized operations for 2,200 MWt (100% power - limited to 60% power); and March 23, 1973 authorized operations for 2,200 MWt (100% power - limited to 85% power). That Operating License has since been amended numerous times to keep the Plant current with NRC standards and to reflect Plant modifications.

On January 22, 1974, Consumers Power requested conversion of the Provisional Operating License DPR-20 to a Full-Term Operating License to operate at 2,638 MWt (845 MWe gross) for a period of 40 years from the date of the issuance of the Construction Permit. As part of the FOL Application, which was submitted as Amendment 28 to Docket 50-255, a complete amendment to the FSAR was provided. This FSAR amendment included major revisions based upon:

1. Incorporation of information related to the once-through circulating water conversion to a closed-cycle system with mechanical draft cooling towers and water treatment chemistry changes

2. Increase in operating power to 2,638 MWt core power

3. Incorporation of information related to Radwaste System modifications implemented to obtain conformance to Appendix I of 10 CFR 50

NRC action on the request for an authorization to increase operating power and a Full-Term Operating License was delayed. The Provisional Operating License remained in effect indefinitely beyond its expiration date, however, under 10CFR2.109.

On August 12, 1977, Consumers Power requested that the Provisional Operating License limit of 2,200 MWt be increased to 2,530 MWt based upon reanalysis of safety evaluations and the improvements made with steam generator repairs. On November 1, 1977, the NRC granted Amendment 31 to DPR-20, authorizing operation of the facility at 2,530 MWt core power.

On February 21, 1991 the NRC issued the Full Term Operating License. This license was based on an Environmental Assessment dated October 22, 1990 and an SER issued as NUREG 1424 on November 21, 1990. The license expiration date is specified as midnight on March 14, 2007.

On November 30, 1999, the NRC issued Amendment 189 to the Palisades Operating Licensee to approve the conversion of Appendix A, Technical Specifications, from the original plant-specific format to a format more consistent with "Standard Technical Specifications, Combustion Engineering Plants,"NUREG-1432. These were referred to as "Improved Technical Specification" until implementation, which occurred on October 24, 2000.

On December 14, 2000, the NRC issued Amendment 192 to the Palisades Operating License to extend the license expiration date from March 14, 2007 to March 24, 2011. This action recaptured the Palisades construction period and provided for 40 years of licensed operation.

On June 23, 2004, the NRC issued Amendment 216 to the Palisades Operating License to authorize operation of the facility at steady state reactor core power levels up to 2565.4 Megawatts thermal, which is the present plant operating limit.

On January 17, 2007, the Renewed Facility Operating License was issued by the NRC, extending the license expiration date to March 24, 2031.

On April 11, 2007, the NRC issued Amendment 224 to the Palisades Renewed Facility Operating License to reflect the transfer of ownership to Entergy Nuclear Palisades, LLC, and operating authority Entergy Nuclear Operations, Inc.

A chronological history of these license events is provided on Table 1-1.

1.2 GENERAL PLANT DESCRIPTION

1.2.1 PLANT SITE

The site for the Palisades Plant consists of approximately 432 acres on the eastern shore of Lake Michigan, in Covert Township, approximately four and one-half miles south of South Haven, Michigan. The area adjacent to the site is sparsely populated and is primarily farmland. The population along the lake increases during the summer months. See Subsection 2.1.2 for details on demography and Figure 2-2 for site layouts.

The exclusion area for Palisades is defined as the property boundary shown on Figure 2-2. The minimum exclusion distance for the site is approximately 2,300 feet (667 meters) and the nearest population center area of more than 24,000 residents is constituted by the cities of Benton Harbor and St Joseph which are approximately 16 miles south of the site.

1.2.2 PLANT ARRANGEMENT

Figure 1-1, Plant Site Plan and Plant Area Plan, displays the primary power block structures arrangement. The turbine building for the Palisades Plant is oriented parallel and adjacent to the shoreline of Lake Michigan, with the reactor containment building located on the east, or landward, side of the turbine building. The office and auxiliary facilities are situated east of the north end of the turbine building so that the entire complex is L-shaped. The reactor containment structure is located inside the corner of this "L." Equipment layouts are shown in Figures 1-2 through 1-18.

The containment building houses the NSSS, consisting of the reactor, steam generators, primary coolant pumps, pressurizer and some of the reactor auxiliaries which do not require access during power operation. The containment building is served by a circular bridge crane.

The turbine building houses the turbine generator, condenser, feedwater heaters, condensate and feed pumps, turbine auxiliaries and certain of the switchgear assemblies. The north end of the turbine building provides additional shop, laboratory and office space.

The auxiliary building and auxiliary building addition (radioactive waste building) houses the waste treatment facilities, engineered safeguards components, heating and ventilating system components, the emergency diesel generators, switchgear, laboratories, offices and the control room. The spent fuel pool and the new fuel storage facilities are located in a separate section of the auxiliary building (Chapter 9) which is under controlled ventilation whenever spent fuel is being moved or stored in that section. Fuel transfer to and from containment is through a fuel transfer tube.

The condensate and makeup demineralizer building (feedwater purity building) was constructed during the feedwater purity modification. It houses the raw water filtration system, the reverse osmosis pretreatment system, the makeup demineralizer system, various components of the condensate demineralizer system, regeneration chemicals handling system, feedwater purity service and instrument air, chemical storage and a boiler room. Because of continuing concern with resin leakage and sodium release, the condensate demineralizer system has been rendered inoperable and retired in place.

The intake structure houses the service water and fire protection pumps. Prior to converting the Plant from once-through cooling to closed-cycle cooling, this building contained the circulating water pumps.

The cooling tower pump house contains two vertical pumps with sufficient head capacity to circulate the tube side condenser cooling water up to the cooling tower inlet near the tower top. The cooling tower basins are elevated some 20 feet above the Plant.

The circulating water cooling towers are cross-flow mechanical draft, located approximately 500 and 1,000 feet from the Plant. One tower contains 18 cells and is designed for a 30°F range and the other tower contains 16 cells and is designed for a 32°F range.

1.2.3 CONTAINMENT

The containment building uses a prestressed concrete design. The building is a vertical right cylindrical structure with a dome and a flat base. The building interior is lined with carbon steel plate for leak tightness. Inside the structure, the reactor and other NSSS components are shielded with concrete. An unlined steel ventilation stack is attached to the outside of the containment building and extends to an elevation equal to the top of the containment dome. Access to portions of the containment building during power operation is permissible.

The containment building, in conjunction with engineered safeguards, is designed to withstand the internal pressure and coincident temperature resulting from the energy released in the event of a DBA. The original structure design conditions are an internal pressure of 55 psig, a coincident temperature of 283°F and a leak rate of 0.1% per day by weight at design temperature and pressure. Actual containment conditions calculated to occur following accidents are discussed in Chapter 14.

The containment is equipped with two independent, full-capacity systems for cooling by air recirculation or building sprays after the postulated DBA. The recirculation system is designed to provide maximum containment atmosphere mixing, however, fan operation is not credited in the analysis for mixing. The cooling coils and fans are sized to provide adequate containment cooling following a DBA with three of the four units in service on emergency power. The building sprays supply borated water to cool and simultaneously remove some of the released fission products from the containment atmosphere. The spray system is sized to provide adequate cooling with two of the three containment spray pumps in service and the two shutdown heat exchangers in operation. Actual system capabilities and operating requirements for fans, coolers and sprays are discussed in Chapters 6 and 14.

The pumps initially take suction from the safety injection and refueling water storage tank. When this supply is depleted, the suction is transferred automatically to the containment sump. By the onset of this recirculation phase, sodium tetraborate is dissolved in the sump solution to neutralize the boric acid.

1.2.4 NUCLEAR STEAM SUPPLY SYSTEM (NSSS)

The NSSS consists of a pressurized water reactor with two closed loops. The principal components and supporting systems of the NSSS are the reactor vessel, internals, control rods, control rod drives, slightly enriched fuel, two "U" tube steam generators, four primary coolant pumps, primary system piping, pressurizer, quench tank, Chemical and Volume Control System, Safety Injection System, nuclear and process instrumentation, and the Reactor Protective System.

The NSSS uses chemical shim and control rods for reactivity control and supplies steam to a four-flow, tandem-compound, hydrogen-cooled turbine generator.

The NSSS is expected to have adequate margin to obtain an ultimate output of 2,650 MWt. The steam and power conversion equipment is designed for a maximum expected gross capability of 865 MWe. See Table 1-2 for equipment design. The Primary Coolant System operates at a nominal pressure of 2,060 psia. The primary coolant enters the upper section of the reactor vessel, flows downward between the reactor vessel shell and the core barrel, and passes through the flow skirt and into the lower plenum where the flow distribution is equalized. The coolant then flows upward through the core removing heat from the fuel rods, exits from the reactor vessel, and passes through the tube side of the two vertical "U" tube steam generators where heat is transferred to the secondary system. Two primary coolant pumps per steam generator return the primary coolant to the reactor vessel.

1. Reactor Vessel and Internals

   The reactor vessel and its removable hemispherical closure head are fabricated from carbon steel and are lined with 308/309 stainless steel. In the areas of internal attachments, the interior is clad with Ni-Cr-Fe alloy. A fixed hemispherical head is attached to the lower end of the shell. The reactor vessel is supported on three pads welded to the underside of the coolant nozzles.

   The reactor core is supported from the reactor vessel flange and is fueled with uranium in the form of slightly enriched UO_2 pellets. Zircaloy-4 or M-5 tubing is used for the fuel cladding. The core contains 204 fuel bundles and 45 control rods.

   A three-to-four batch, mixed central zone fuel management plan is employed and a further reduction in nuclear peaking is obtained by local enrichment zoning within the bundles. Boric acid dissolved in the coolant is used as the neutron absorber to provide long-term reactivity control. In order to reduce the boric acid concentration required at the beginning of the fuel cycle, and thus to make the moderator coefficient of reactivity more negative, mechanically fixed, burnable poison rods are utilized.

2. Steam Generators

   The two steam generators are vertical shell and "U" tube units (see Table 4-4).

   The steam generated in the shell side of the steam generator flows upward through moisture separators which reduce its moisture content. All surfaces in contact with the primary coolant are either stainless steel or Inconel in order to maintain primary coolant purity.

3. Primary Coolant Pumps

   The coolant in the primary loop is circulated by four primary coolant pumps of the single suction centrifugal type. The pump shafts are sealed by mechanical seals. The seal performance is monitored by pressure and temperature sensing devices in the seal water circulation system.

4. Primary System Piping

   Each of the two loops which make up the Primary Coolant System consists of one 42-inch ID pipe and two 30-inch ID pipes. The larger pipe carries the water from the reactor to the steam generator. The flow from the steam generators is pumped to the reactor through the 30-inch ID pipes.

5. Pressure Control System

   The pressure in the Primary Coolant System is controlled by regulating the temperature of the coolant in the pressurizer, where steam and water are held in thermal equilibrium. Steam is formed by the pressurizer heaters or condensed by the pressurizer spray to reduce pressure variations caused by expansion and contraction of the primary coolant due to primary system temperature changes.

   Overpressure protection is provided by spring-loaded safety valves connected to the pressurizer. The discharge from the pressurizer safety valves is released under water in the pressurizer quench tank, where it is condensed and cooled. In the event that the discharged volume of steam exceeds the capacity of the quench tank, the tank relieves via a rupture disc to containment.

6. Reactor Control

   The reactor is controlled by a combination of 45 control rods and dissolved boric acid in the primary coolant. Forty-one of the control rods are full length, and four partial-length rods are also provided. The part-length rods are maintained in the fully withdrawn position during reactor operation and do not insert following a reactor trip.

   Boric acid addition or removal is used for reactivity changes associated with major changes in water temperature during start-up and shutdown, fuel burnup and xenon variations. Additions of boric acid also provide an increased shutdown margin during initial fuel loading, refuelings and approaches to cold shutdown condition. The boric acid solution is prepared in a boric acid batching tank, stored in two storage tanks, and maintained at a temperature sufficient to prevent precipitation. The tanks are connected to the charging pumps through locked open manual and automatic valving.

Control rod movement provides changes in reactivity required for power changes or for shutdown to a hot condition. The control rods are made of a silver-indium-cadmium alloy clad with stainless steel welded into a cruciform configuration. They are actuated by control rod drive mechanisms mounted on the head of the reactor vessel. The control rod drive mechanisms, which are rack-and-pinion units, are designed to permit rapid insertion of the control rods into the reactor core by gravity.

7. Chemical and Volume Control System

The purity of primary coolant is controlled by continuous purification of a portion, "letdown," of the total primary coolant volume. Coolant is removed from the primary system and is initially cooled in the regenerative heat exchanger. The coolant letdown is then reduced in pressure by orifices and letdown back pressure valves and again in temperature as it passes through the letdown heat exchanger. The letdown then flows through one of three demineralizers where corrosion and fission products are removed through a filter which traps particulate matter in the effluent from the demineralizer. It is then sprayed into the volume control tank.

The volume control system automatically controls the rate and amount of coolant returned to the Primary Coolant System to maintain the pressurizer level within a control band and thereby compensates for changes in volume due to primary coolant temperature changes. The volume control tank is sized to accommodate primary coolant inventory changes resulting from load changes from hot standby to full power. This mode of operation, using the volume control tank as a surge tank, decreases the quantity of liquid and gaseous waste which otherwise would be generated.

8. Chemical Treatment

Primary system makeup water is taken from the demineralized water storage system and from the concentrated boric acid tanks. The makeup water is pumped through the regenerative heat exchanger into the primary loop by the charging pumps.

Bleed from the primary system during a boron concentration reduction is routed to the radwaste liquid receiver tanks for processing through the Radwaste System before reuse in the Plant or disposal to the lake.

Chemical injection equipment is provided for the addition of corrosion control chemicals to the primary loop water. Hydrogen is added to primary coolant for oxygen scavenging through the volume control tank.

Depleted zinc ions are added to primary coolant through the Zinc Addition System for the removal of radioactive cobalt ions from PCS piping (inner walls). Removal of the radioactive cobalt ions reduces dose to personnel from PCS piping.

9. Nuclear Control and Instrumentation

a. Nuclear Plant Control

The reactor control system provides for start-up and shutdown of the reactor and for adjustment of the reactor power in response to turbine load demand. The NSSS is capable of following a ramp change from 15% to 100% power at a rate of 5% per minute and at greater rates over smaller load change increments up to a step change of 10%. This control is accomplished by manual rod motion. A temperature computing station calculates the reactor average temperature and a reference temperature value corresponding to turbine power. The reactor average coolant temperature and the reference temperature values are displayed to operators who manually adjust coolant temperature by moving control rods. Regulation of the primary temperature in accordance with this program maintains the secondary steam pressure and matches reactor power to load demand.

b. Reactor Neutron Monitoring

The nuclear instrumentation consists of excore and incore flux monitoring chambers. Eight channels of excore instrumentation monitor the neutron flux and six of the eight channels provide reactor protection signals during start-up and power operation. Two of the channels follow the neutron flux through the start-up range.

The incore monitors consist of rhodium neutron detectors and a thermocouple. This system provides information on neutron flux and temperatures in the core.

c. Reactor Protective System

The reactor parameters are maintained within acceptable limits by the inherent characteristics of the reactor, by the control rod system, by boron control and by the operating procedures. Departures from these limits are indicated audibly and visually in the control room. A Reactor Protective System initiates reactor shutdown if selected values of parameters are exceeded. The protective system is divided into four channels. Each channel receives trip signals from sensors when the relevant parameter values are exceeded and a two-of-four coincident logic system sends a "deenergize" signal to the control rod drive mechanism clutch power supplies.

The control rods are released and the reactor is shut down when the clutch power supplies are deenergized. Redundant sensors are provided for the reactor shutdown functions so that failure of any one sensor does not prevent a reactor trip.

d. Other Safety-Related Protection and Control Systems

While the Reactor Protective System protects the reactor core and the engineered safeguards controls protect against a Loss of Coolant Accident (LOCA), other safety-related Class 1E control and instrumentation systems are provided to allow a safe shutdown of the Plant, assure decay heat removal and protection of fluid systems boundaries. Such systems are reactor shutdown controls, primary coolant and other liquid boundaries overpressure protection, automatic auxiliary feedwater initiation and containment hydrogen control.

e. Process Instruments

Critical primary system parameters are monitored by redundant channels. Additional temperature, pressure, flow and liquid level monitoring is provided as required to keep the operating personnel informed of Plant conditions and to provide information from which Plant processes can be evaluated or regulated. The plant gaseous and liquid effluents are monitored for radioactivity. The levels are displayed and recorded and high values are annunciated.

Area monitoring stations are provided to monitor radioactivity at selected locations around the Plant. High-pressure or high-radiation conditions within the containment building initiate control action to isolate the containment.

10. Safety Injection System

Four safety injection tanks are provided, each connected to one of the four reactor inlet lines. Each tank has a volume of approximately 2,000 cubic feet containing approximately 1,000 cubic feet of borated water at a concentration of 1,720-2,500 ppm and pressurized by approximately 1,000 cubic feet of nitrogen at approximately 200 psia. In the event of a large Loss of Coolant Accident, the borated water is forced into the Primary Coolant System by the expansion of the nitrogen. The water in three tanks will adequately refill and reflood the entire core. In addition, borated water will be injected into the reactor vessel to cool the core via the same nozzles used by the SI tanks by two low-pressure and two high-pressure injection pumps taking suction from the 285,000-gallon safety injection and refueling water storage tank (SIRW). For maximum reliability, the designed capacity from the combined operation of one high-pressure and one low-pressure pump provides adequate injection flow for any Loss of Coolant Accident. Upon depletion of the storage tank supply, the high-pressure pump suction automatically transfers to the containment sump and the low-pressure pumps are shut down. One high-pressure pump has sufficient capacity to maintain the core water level at the start of recirculation. In the event of a DBA, at least one high-pressure and one low-pressure pump would receive power from the emergency power sources. Both high- and low-pressure injection pumps are located outside the containment building to permit access for periodic testing during normal operation. The pumps discharge into separate headers which lead to the containment. Test lines are provided to permit running the pumps for test purposes during Plant operation.

11. Shutdown Cooling System

The Shutdown Cooling System consists of a forced circulation heat removal loop which includes the low-pressure safety injection pumps and the shutdown heat exchangers. The system is designed to transfer heat from the Primary Coolant System to a closed loop cooling system during normal shutdown, refueling and maintenance operations.

Emergency shutdown cooling during a loss of normal and standby electrical power is accomplished by allowing natural circulation of the primary coolant to transfer heat from the core to the steam generators. The steam that is generated is released to the atmosphere as required. One of two auxiliary electric-driven feedwater pumps operating from either emergency diesel generator, or an auxiliary turbine-driven feedwater pump, supplies feedwater to the steam generators during this period. A 100,000-gallon supply of demineralized water available to these pumps is sufficient for eight hours of decay heat removal. In addition, the Plant has the capacity for long-term cooling incorporating the ability to flush the reactor core and prevent post-LOCA boric acid precipitation.

12. Shielding

Shielding is provided so that radiation exposure of personnel will not exceed the recommended limits of 10 CFR, Part 20. The design of radiation shielding is dependent both on the extent of access required to a particular location and on the sources of radiation adjacent to that location.

The control room is shielded to permit continuous occupancy following any accidental release of radioactivity in the containment.

1.2.5 TURBINE GENERATOR

The turbine is an 1,800 r/min tandem-compound unit with external moisture separation and live steam reheating. The double-flow high-pressure element exhausts to two double-flow low-pressure elements through moisture separators and reheaters. The low-pressure elements discharge to the main condenser and the condensate is returned to the steam generators through six stages of feedwater heating. Steam is extracted for feedwater heating and for two auxiliary turbines which drive the two half-sized steam generator feed pumps.

The feedwater cycle is of the closed type with deaeration effected in the condenser. Feedwater heaters are arranged in two parallel trains, each with one high-pressure and five low-pressure heaters. Separate feedwater regulating valves control the flow to each of the two steam generators.

The 1,800 r/min, hydrogen inner-cooled generator is rated at 955,000 kVA at 75 psig hydrogen pressure, 0.85 power factor and 0.62 short circuit ratio. Field excitation is provided by a brushless exciter directly coupled to the generator shaft.

The turbine generator has a guaranteed capability of 811,776 kWe gross at 1.8 inches Hg absolute back pressure and 0.25% makeup with inlet steam conditions of 735 psia and 509°F. The maximum calculated capacity of the turbine generator is 865 MWe gross.

1.3 IDENTIFICATION OF CONTRACTORS

Consumers Power engaged Combustion Engineering, Inc (Combustion Engineering) to design and supply the nuclear fuel and the NSSS. The NSSS includes the primary system (eg, reactor vessel, steam generators, pressurizer, pumps), reactor auxiliary system components, nuclear and certain process instrumentation and the Reactor Protective System. Bechtel Corporation and its affiliate, Bechtel Company, were engaged to design and supply the balance of the Plant equipment, systems and structures. Bechtel Corporation performed the onsite construction of the original Plant, with technical advice and consultation provided by Combustion Engineering for installation of the NSSS. Subsequent to the initial Plant start-up and turnover to Consumers Power, several major modifications involving other contractors have been undertaken. Those contractors are identified in Section 1.5.

Under its contract with Consumers Power, Combustion Engineering furnished Bechtel with the design data for the NSSS. Bechtel and Consumers Power could request that Combustion Engineering make changes in the NSSS design, but Combustion Engineering did not need to accede to any such request if the proposed change, in Combustion Engineering's judgment, would be unsafe or technically unsound.

Because of the interdependence of the NSSS and certain balance-of-Plant equipment, systems and structures, Combustion Engineering furnished Bechtel with certain functional requirements for such balance-of-Plant items that affect the operability and maintainability of the NSSS or the nuclear safety of the Palisades Plant. As Bechtel's engineering work progressed, Combustion Engineering reviewed Bechtel drawings, specifications and data and Combustion Engineering was satisfied that Bechtel has understood and applied the functional requirements specified by Combustion Engineering and was satisfied that the balance-of-Plant items are compatible with the NSSS and with nuclear safety.

Palisades' original fuel vendor for cycle 1 was Combustion Engineering. Starting with cycle 2, Exxon Nuclear Corporation designed and manufactured all fuel for the reactor. Over the years, Exxon Nuclear has undergone the following company name changes: from Exxon Nuclear Corporation (ENC), to Advanced Nuclear Fuels (ANF) Corporation, to Siemens Nuclear Power (SNP), to Siemens Power Corporation (SPC), to Framatome-ANP (FANP) to the present name AREVA.

1.4 PRINCIPAL DESIGN CRITERIA

1.4.1 STATION DESIGN

Principal structures and equipment which are necessary either to prevent accidents or to mitigate their consequences were designed, fabricated and erected in accordance with applicable codes and to withstand the effects of the most severe earthquakes, flooding conditions, windstorms, ice conditions, temperature and other deleterious natural phenomena which could be expected at the site during the lifetime of this unit. Principal structures and equipment were sized for the maximum expected NSSS and turbine generator outputs.

All core physics and thermal hydraulics information contained in this report are based upon the reference core design of 2,650 MWt unless otherwise noted. The structures, systems and all postulated accidents are evaluated at either the 2,650 MWt design NSSS output or the licensed 2,565.4 MWt core output. Consult the specific FSAR chapters on systems or transient analyses for more detailed discussions. Section 5.1 details Palisades' conformance to General Design Criteria per 10 CFR 50, Appendix A. Section 5.2 specifies Design Codes, Structures/Systems/ Components Classification and establishes the basis for "CP Co Design Class" terminology.

1.4.2 REACTOR

1. The reactor is of the pressurized water type, designed to produce steam to drive a turbine generator. The reactor was initially operated at 2,200 thermal megawatts to produce steam at 770 psia and presently operates at 2,565.4 thermal megawatts, core power, producing steam at a nominal pressure of 765 psia.

2. The reactor is fueled with slightly enriched uranium dioxide contained in Zircaloy or M5 tubes.

3. The minimum departure from nucleate boiling ratio and maximum fuel center line temperature evaluated at the design overpower condition must be below values which could lead to fuel rod failures. The melting point of the UO_2 will not be reached during normal operation including expected transients.

4. Fuel rod clad thicknesses are designed to maintain cladding integrity throughout the anticipated fuel life. Fission gas release within the rods and other factors affecting design life must be considered for the maximum expected exposures.

5. The reactor and control system must be designed so that any xenon transients will be adequately damped.

6. The reactor must be designed to accommodate safely and without fuel damage tripping of the turbine generator, loss of power to the primary coolant pumps and station transients and maneuvers.

7. Power excursions which could result from any credible reactivity addition accident must not cause damage, either by motion or rupture, to the pressure vessel or impair operation of required safeguards.

8. Neutron absorption for reactivity control is provided by control rods and by dissolved boric acid in the coolant. The boron chemical shim system is completely independent of the control rod system.

9. For all operating conditions, the control rods are capable of providing an adequate shutdown margin at hot, zero power conditions following a trip, even with the most reactive rod stuck in the fully withdrawn position.

10. The boron chemical shim system is capable of adding boric acid to the primary coolant at a rate sufficient to maintain an adequate shutdown margin during primary system cooldown at the maximum design rate following a reactor trip.

11. The combined response of the fuel temperature coefficient, the moderator temperature coefficient, the moderator void coefficient and the moderator pressure coefficient to an increase in reactor thermal power is a decrease in reactivity. In addition, the reactor power transient remains bounded and damped in response to any expected changes in any operating variable.

12. The Primary Coolant Gas Vent System is designed to relieve steam or gas bubbles in the reactor vessel head and pressurizer areas of the Primary Coolant System. The system consists of a flow-limiting orifice on both the reactor vessel vent and pressurizer vent lines, solenoid valves, a pressure transmitter for pressure indication, and connecting piping. The primary vent path is directed into the open area of containment where adequate mixing with the containment atmosphere is assured.

Automatic and redundant reactor trips are provided to prevent anticipated plant transients from producing fuel or clad damage.

1.4.3 PRIMARY COOLANT AND AUXILIARY SYSTEMS

Heat removal systems are provided which can safely accommodate core heat output under all credible circumstances. Each of these heat removal systems has sufficient redundancy to provide reliable operation under all credible circumstances.

1.4.4 CONTAINMENT SYSTEM

The containment structure, including the associated access openings and penetrations, was designed to contain the pressures and temperatures resulting from a design basis accident (DBA) in which (a) the total energy contained in the Primary Coolant System water was assumed to be released into the containment through a double-ended break of one of the primary coolant pipes immediately adjacent to the reactor vessel outlet nozzle, (b) there was a simultaneous loss of external electric power, (c) heat was transferred from the reactor to containment by water supplied from the Safety Injection System but no credit was taken for cooling the fuel by water injected by the Safety Injection System, (d) the containment air recirculation and cooling system and the Containment Spray System function, and (e) the containment engineered safeguards do not operate until 30 seconds following the accident. For a discussion of the maximum hypothetical accident (MHA) refer to Section 14.22.

Means were provided for pressure and leak rate testing of the entire containment system including provisions for leak rate testing of individual piping and electrical penetrations that rely on gasketed seals, sealing compounds, or expansion bellows. Integrated leak rate testing is conducted according to Technical Specifications.

1.4.5 ENGINEERED SAFEGUARDS

Containment engineered safeguards systems with redundant features were incorporated in the Plant design which, in conjunction with the containment system and without relying upon the Emergency Core Cooling System, provide a high degree of assurance that the release of fission products to the environment following any credible Loss of Coolant Accident will not exceed the tolerances set forth in 10 CFR, Part 100.

An Emergency Core Cooling System was provided to prevent fuel and cladding damage that could interfere with adequate emergency core cooling and to limit the cladding water reaction to less than approximately 1% for all break sizes in the primary system piping up to the double-ended rupture of the largest primary coolant pipe, for any break location, and for the applicable break time. For discussion, refer to Chapter 6.

1.4.6 INSTRUMENTATION AND CONTROL

Interlocks and automatic protective systems were provided along with administrative controls to ensure safe operation of the Plant. A Reactor Protective System was provided which initiates reactor trip if reactor parameters exceed pre-established limits.

Sufficient redundancy was installed to permit periodic testing of the Reactor Protective Systems and so that failure or removal from service of any one protective system component or portion of the system will not preclude reactor trip or other safety action when required.

1.4.7 ELECTRICAL SYSTEMS

Offsite and emergency sources of auxiliary electrical power were provided to assure safe and orderly shutdown of the Plant and the ability to maintain a safe shutdown condition under all credible circumstances. "Redundancy and Separation" criteria were incorporated into the associated cabling from these sources for "Safety-Related" systems/components.

1.4.8 RADIOACTIVE WASTES AND RADIATION PROTECTION

The radioactive waste treatment system was designed so that discharge of radioactivity to the environment is in accordance with the requirements of 10 CFR, Part 20, and Appendix I to 10 CFR 50.

The Plant was provided with a centralized control room having adequate shielding to permit occupancy during all credible accident situations. The radiation shielding in the Plant, in combination with Plant radiation control procedures, ensures that operating personnel do not receive radiation exposures in excess of the applicable limits of 10 CFR, Part 20, during normal operation and maintenance.

1.4.9 FUEL HANDLING AND STORAGE

Fuel handling and storage facilities were provided for the safe handling, storage and shipment of fuel and will preclude accidental criticality.

1.4.10 FIRE PROTECTION

A "Fire Protection Program" (FPP) consisting of Plant design considerations, fire detection and suppression equipment, and Plant procedures assures that the Plant can safely shut down after a major fire. The FPP complies with 10 CFR 50.48, and with National Fire Protection Association (NFPA) 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," except where exemptions have been granted by the NRC.

1.4.11 CIRCULATING WATER SYSTEM

In order to minimize the environmental impact associated with "hot water" discharges, the circulating water system provides condenser cooling water supplied from two mechanical draft evaporative cooling towers. A further temperature dilution flow is provided before discharge to Lake Michigan. The discharges are within the Plant's NPDES Permit limitations.

1.4.12 SECURITY

Access and egress to all "protected" areas of the Plant are monitored/controlled through the utilization of card readers. Access to the Plant is controlled at the security entrance via explosive detectors, metal detectors, guards and card readers. A physical security force is always present. Details of conformance are identified in the commission-approved physical security, safeguards contingency, and guard training and qualification plans.

1.4.13 EMERGENCY PLANNING

In the unlikely event of a Plant accident resulting in, or potentially capable of allowing, offsite releases of radioactivity in excess of federal regulations, a system of emergency warning sirens is in place. Established "emergency implementing procedures" in conjunction with the Plant's "Emergency Plan" have been developed to assure minimum risk to the general public in compliance with 10 CFR 50.54(q) and 10 CFR 50, Appendix E.

1.4.14 PLANT OPERATION

The plant’s Facility Operating License requires operation to be in accordance with the Technical Specifications, which are contained in Appendix A to that license. Technical Specifications contain Safety Limits, Limiting Safety System Settings, Limiting Conditions for Operation, Surveillance Requirements, Design Features, and Administrative Controls, in accordance with the Code of Federal Regulations, Title 10, Part 50.36 (10CFR50). Operation of the Plant within Safety Limits and in accordance with the Limiting Safety System Settings and Limiting Conditions for Operation assures that plant operation will remain within the assumptions and initial conditions of the safety analyses. The Administrative Controls provide NRC requirements for plant staff Responsibilities, Organization, Qualifications, Procedures, Programs and Manuals, Reporting Requirements to the NRC, and High Radiation Area Control.

1.4.15 STRUCTURES

Plant structures were designed in accordance with the design criteria identified in Chapter 5. Structures were identified as CP Co Design Class 1, 2 or 3 according to Section 5.2. Specific design criteria for containment is discussed in Section 5.8, and other CP Co Design Class 1 structures are discussed in Section 5.9.

1.4.16 SINGLE FAILURE CRITERIA

1.4.16.1 Licensing Basis

Palisades submitted application for an operating license in 1968. At that time, the General Design Criteria (GDCs) were in draft form. The original FSAR contained Appendix I, which presented a comparison of plant design features with the 1967 draft GDCs. From the wording of Criterion 39, "Emergency Power," and Criterion 41, "Engineered Safety Features Performance," of the original FSAR, it is clear that design considerations for single failure concerns were limited to "failure of a single active component."

Palisades was not designed with system redundancy (electrical or fluid systems) comparable to newer plant designs. As such, and in general, only the failure of a single active component (and not a passive failure) was considered.

In 1977, the NRC initiated the Systematic Evaluation Program (SEP) to review the designs of older operating plants. The review provided: 1) an assessment of the significance of differences between the then-current technical positions on safety issues and positions that existed when a particular plant was licensed; 2) a basis for deciding how these differences would be resolved; and 3) documented evaluations of plant safety. Palisades was one plant selected for the SEP reviews. Based on the SEP reviews, topics were closed based on the adequacy of the existing system designs or, in some cases, after the licensees made procedural or design changes. Single failure criteria adequacy (electrical and fluid systems) was evaluated in several topics and no requirement to address passive failures on a plant-wide, system level basis was backfit by NRC or committed to by Palisades. Specific issues were addressed on a case-by-case basis.

Also of concern is the assumed timing of a failure. The NRC Safety Evaluation Report for Design Basis Events (Reference 6) shows that the types of failures considered as most limiting for design basis events were assumed to occur at the time of the demand for the components being called upon to function. NRC Information Notice 93-17 Revision 1 (Reference 7) addressed the issue by acknowledging that some plants' safety systems have been designed to respond properly to a failure upon demand but not for other possible sequences. The notice stated that no backfitting was intended or approved, and that the generic issue was dropped based on extremely low probability of occurrence.

In general, when performing design modifications or evaluating system performance, the licensing basis for Palisades for failures in electrical and fluid systems only considers single active failures, and the failure is only considered at the time the demand is placed on the component to function. Exceptions to this treatment of single failures have been addressed on a case-by-case basis as design requirements have evolved over time and significant safety concerns have been addressed. Where it has been determined to be applicable, specific criteria have been incorporated into the licensing basis. For example, the single failure criterion of IEEE 279-1971 has been applied to protection systems between the sensors and the actuation devices (Chapters 7 and 8). Future design modifications should consider current guidance per Subsection 1.4.16.3.

1.4.16.2 Active and Passive Failures

Active failures considered in Palisades design require a malfunction of a component that relies on a mechanical movement to complete its intended function upon demand. For example, the inadvertent opening of a normally closed breaker, absent a component fault, would be considered a passive failure of the breaker.

NRC Information Report, SECY-77-439, August 17, 1977, "Single Failure Criteria," (Reference 8) gives the following definition and example of a passive failure:

> "A passive failure in a fluid system means a breach in the fluid pressure boundary or a mechanical failure which adversely affects a flow path. Examples include the failure of a simple check valve to move to its correct position when required,…"

As shown by this statement, the NRC considered the failure of a check valve as a passive failure. Check valves were considered passive components when Palisades was originally designed and constructed.

1.4.16.3 Current Design Considerations

Standards and requirements have changed since the original licensing of Palisades. The plant has been reevaluated in light of these newer standards and requirements, as well as industry experience, and has at times been required to make changes on a case-by-case basis. Therefore, as new issues arise, the plant staff should use current guidance and review criteria in order to make appropriate decisions regarding the adequacy of systems design and performance.

While not part of the Palisades licensing basis, ANSI/ANS-58.9-1981, "American National Standard, Single Failure Criteria for Light Water Reactor Safety-Related Fluid Systems," provides current guidance. That standard provides the following definition of an active failure:

> "An active failure is a malfunction, excluding passive failures, of a component that relies on mechanical movement to complete its intended function upon demand."

Per that standard, failure of a check valve to move to its correct position is an example of an active failure, which is different from the Palisades original licensing basis. The standard also provides a method for taking exceptions by stating:

> "Where the proper active function of a component can be demonstrated despite any credible condition, then that component may be considered exempt from active failure. Examples of such component functions may include opening of code safety valves and certain swing check valves. Where such exemption is taken, the basis for the exemption shall be documented in the single failure analysis."

Though an issue may not be part of the Palisades licensing/design basis, if current standards would require a different treatment of a particular type of failure, it should be evaluated on a case-by-case basis to assess the safety significance and determine if it would be prudent to adopt the current standard. While failure at the time of demand is the licensing basis for consideration of single active failures, other credible single active failures should not automatically be eliminated.

1.5 MAJOR PLANT MODIFICATIONS (DESIGN/CONSTRUCTION)

Following initial completion of the Palisades Plant in 1971, several major facility modifications have been made to improve the safety and operability of the Plant. These modifications are briefly outlined below, with references to the appropriate FSAR Update section and identification of the designer/constructor.

CONDENSER RETUBING - BECHTEL/J A JONES

In 1974, due to condenser tube leakage problems, the Admiralty tube section of the main condenser was replaced with 90-10 copper-nickel tubes.

CONDENSER REPLACEMENT - YUBA/TOWNSEND & BOTTUM

In 1990 the main condenser was replaced to eliminate copper-related corrosion concerns with the new steam generators. The replacement condenser is a Yuba design that contains Type 439 stainless steel tubing. See Subsection 10.2.3.1.

STEAM GENERATOR TUBE PLUGGING - CP CO/CP CO

A total of 2,044 tubes in original Steam Generator A and 2,442 tubes in original Steam Generator B were plugged due to tube wall degradation resulting from the following: (1) secondary side standard phosphate water treatment, (2) intergrannular corrosion and (3) tube denting. An all-volatile secondary system water chemistry with boric acid has been implemented to reduce further tube degradation. See Subsections 4.3.4.1 and 4.3.4.2.

STEAM GENERATOR REPLACEMENT - BECHTEL/BECHTEL

Switch over to an all-volatile secondary water chemistry decreased the rate of tube degradation but over time examination revealed further intergrannular attack (IGA) and other growing problems related to denting at tube support plates. With excess outage times and plant operation nearing the point of power limitation due to plugged tubes, replacement of both steam generators was undertaken in late 1990. The replacement steam generators are designed to match the essential parameters of the old steam generators and to be compatible with operation at 2,565.4 MWt. Consistent with other PCS equipment, the replacement steam generators are designed for operation at 2650 MWt should an increase in the licensed power level be pursued in the future. See Subsection 4.3.4.2.

FEEDWATER PURITY BUILDING ADDITION - BECHTEL/J A JONES

A completely new secondary side feedwater (condensate) purity system was installed to provide full flow condensate demineralization system utilizing powdered ion exchange resins and on-line resin body feed capability. This new system is housed in the feedwater purity building addition. See Subsection 10.2.3.2. This system has since been deactivated.

COOLING TOWERS ADDITION - ECODYNE/SPX MARLEY

Initially, the Plant was designed for a once-through Circulating Water System for providing cooling water to the condenser. For environmental reasons, the system was converted in 1974 to a closed-cycle system using two mechanical draft cooling towers and blowdown dilution. A cooling tower pump house was constructed to enclose the cooling tower pumps. See Subsections 10.2.4 and 10.2.4.1.

In 2012, the "A" cooling tower was replaced with an SPX Marley cooling tower. The replacement tower has 16 cells as apposed to the 18 cells of the original Ecodyne design. The new tower is a pultruded fiberglass design.

RADWASTE SYSTEM MODIFICATIONS/AUXILIARY BUILDING ADDITION - BECHTEL/BECHTEL

During 1971-1973 the liquid waste management system was modified to reduce liquid discharges to "Near Zero" and meet the requirements of 10 CFR 50, Appendix I. The auxiliary building was expanded to enclose much of the new equipment required. The 1972-1973 service building addition, in conjunction with the aforementioned changes, was made to accommodate solid radwaste system changes designed by Protective Packaging Inc (PPI). This system was subsequently replaced by a molten bitumen immobilizing system for waste concentrates. In 1996, the molten bitumen immobilizing system was replaced by a concentrated waste drying system. See Chapter 11 for details.

SPENT FUEL POOL STORAGE MODIFICATIONS - NUS/J A JONES 1977 - WESTINGHOUSE/WESTINGHOUSE 1987

In 1977, the spent fuel pool storage capacity was increased from a capacity of 272 assemblies to 798. In 1987, Amendment 105, dated July 24, 1987, authorized replacing existing racks with six high-density spent fuel racks that increased the storage capacity from 798 to 892 fuel assemblies. See Section 9.4 and Subsection 9.11.3.

HIGH-PRESSURE AIR ADDITION - BECHTEL/J A JONES

In 1977, the compressed air system was augmented by the addition of a high-pressure air system (325 psig) for supply to safety-related air-operated valves and components. See Section 9.5.

FIRE PROTECTION SYSTEM MODIFICATIONS - NUCLEAR SERVICES CORP/QUADREX/J A JONES

Following the 1975 Browns Ferry fire and subsequent NRC revised guidelines, Consumers Power undertook a series of studies and resultant Plant fire protection features modifications. This has included the addition of fire-fighting equipment, separation of cables, addition of fire stops, preparation of procedures, etc. See Section 9.6 and Chapters 7 and 8.

AUXILIARY FEEDWATER MODIFICATION - BECHTEL/BECHTEL

As a result of lessons learned at TMI, the Auxiliary Feedwater System has been upgraded to a safety-related system. See Sections 9.7 and 7.4.

METEOROLOGICAL PROGRAM IMPROVEMENTS - EG&G/CP CO

Several modifications were made to the onsite meteorological towers including the addition and relocation of new towers. A final meteorological program was attained in 1977 following a 1975 study by EG&G Environmental Consultants. See Subsection 2.5.2.3 for a description of the new tower and meteorological program.

AUXILIARY BUILDING TSC/EER/HVAC ADDITION - BECHTEL/BECHTEL

During 1983 an addition was added to the north side of the auxiliary building to house a Technical Support Center (TSC), an Electrical Equipment Room (EER) and a Heating, Ventilating and Air Conditioning (HVAC) area. The TSC was required to fulfill the guidelines of NUREG-0696, the HVAC area as a result of the control room habitability requirements of NUREG-0737, and the EER area as a result of loads placed on the electrical system by the addition of the TSC and HVAC areas. See Section 9.8 for discussion of the HVAC system, Chapter 8 for discussion of the electrical equipment and the Site Emergency Plan for the functional discussion of the TSC.

INTERIM OLD STEAM GENERATOR STORAGE FACILITY - BECHTEL/BECHTEL

In 1990, a reinforced concrete building was constructed for interim storage of two old steam generators. This facility is located in the controlled area of the site approximately 2,200 feet northeast of the containment building. The storage facility design provides sufficient radiation shielding such that the onsite and offsite dose rate will not exceed the limits defined in 10 CFR 20 and 40 CFR 190, respectively. The facility is designated as a secondary restricted area. The old steam generators will remain in this facility until an ultimate disposition method is selected.

INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) - PACIFIC SIERRA NUCLEAR

In 1993 Palisades constructed a suitable facility and began dry storage of spent nuclear fuel in casks under the General License provisions of 10CFR72. This facility is located north of the support building and is enclosed within the plant security fence. The ISFSI was planned to hold 25 Ventilated Storage Casks (VSCs) designed by Pacific Sierra Nuclear (later Sierra Nuclear) Corporation although other NRC-certified cask designs could also be utilized if the associated fuel handling equipment were procured.

In 2003, Palisades constructed an additional ISFSI pad for storage of NUHOMS casks under the General License provisions of 10CFR72. This facility is located east of the plant and is enclosed by a security fence. Other NRC-certified cask designs could also be stored at the pad.

REPLACEMENT OF REGION 1 CARBORUNDUM RACKS – HOLTEC INTERNATIONAL

In 2013, Palisades replaced six of seven Carborundum®-equipped Region 1 racks in the Spent Fuel Pool with six new Metamic™-equipped Region 1 racks under the provisions of 10CFR50. The new racks have the same storage capacity (the same number of fuel assemblies stored) as the replaced racks. The new racks were designed and supplied by Holtec International.

1.6 INSERVICE INSPECTION

1.6.1 HISTORICAL BACKGROUND

The Palisades Nuclear Power Plant was built in the late 1960s and was placed in commercial service on December 31, 1971. During the first 40-month life of the Plant, in order to comply with Paragraphs 4.3 and 4.12 of the Technical Specifications (dated September 1, 1972) of the Provisional Operating License DPR-20 for the Palisades Nuclear Plant, which discusses ISI requirements of ASME Class 1 components and systems, the nondestructive examinations were performed to satisfy the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, 1971 Edition, including the Winter 1972 Addenda (ASME B&PV Code, Section XI, 71W72a). In February 1976, the NRC amended Paragraph 55a (g) of 10 CFR 50 to require nuclear plants to upgrade their Technical Specifications in the areas of the ISI requirements and the functional testing of pumps and valves. By amending Paragraph 55a (g) and by invoking Regulatory Guide 1.26, the NRC required nuclear plants to upgrade their ISI program to include not only ASME Class 1 systems, but also ASME Class 2 and ASME Class 3 systems.

1.6.2 GENERAL

The Inservice Inspection Plan for the initial 10-year inservice intervals was developed by Southwest Research Institute and Consumers Power Company, and reviewed and approved by Consumers Power Company for use at the Palisades Nuclear Power Plant. Subsequent updating to remain responsive to industry requirements is anticipated.

The start of the first 10-year interval coincides with the date of first commercial operation, December 31, 1971. The length of the first 3-1/3-year period was extended to October 30, 1976 by adding 18 months cumulative shutdown time between August 1973 and April 1975 in accordance with ASME B&PV Code, Section XI, IS-241, 71W72a. The second period ran to June 1, 1980 due to the 1979/1980 extended refueling outage. The third period extended to November 9, 1983 per ASME B&PV Code, Section XI, IWA-2400(c), 77S78a.

The second 10-year interval began November 10, 1983, and lasted until May, 1995.

The third 10-year interval began May 12, 1995, and lasted until December 2006.

The fourth 10-year interval began December 13, 2006, and lasted until December 12, 2015.

The fifth 10-year interval began December 13, 2015.

See Section 6.9 for details of the Inservice Inspection Program.

1.7 RESEARCH AND DEVELOPMENT REQUIREMENTS

The design of the Palisades Plant is based upon concepts which have been successfully incorporated in pressurized water reactor systems. However, certain Palisades specific Combustion Engineering development tests have been performed and are listed below. These tests were completed prior to initial Plant start-up.

1.7.1 FLOW MIXING AND FLOW DISTRIBUTION

Tests have been run to measure the flow mixing factor. Dye dispersion rate was measured in a series of full-scale and larger than full-scale mock-ups of various fuel bundle flow channel configurations. These tests were run in a cold-water test loop.

A larger than full-scale model of the Palisades bundle inlet region has been tested in a cold-water flow loop to determine the effect of minor flow maldistributions due to inlet structure nonuniformities, and to verify the effectiveness of certain steps taken to improve inlet flow.

1.7.2 CONTROL ROD TESTS

A series of tests were run to demonstrate the adequacy of the control rod and its guidance system. Cold-water flow testing of a slightly underscale model of four bundles and a cruciform control rod was performed. These tests were conducted with mechanical misalignments exceeding design values.

In addition to the cold-water tests, a test program has been performed using a full-scale model, including a prototype mechanism under reactor conditions of flow, temperature and pressure in the CE Utility Reactor Components Test Facility at Windsor. The purpose of this program was to assess the effects of mechanical misalignments, of thermal distortions which have been measured on model fuel bundles and on a prototype control rod, of cross flow and of upper limit conditions of axial flow and system pressure.

A series of mechanical tests have been run on structural components of the Ag-In-Cd control rod blade.

1.7.3 CONTROL ROD DRIVE MECHANISMS

An extensive development program has been completed on the control rod drive mechanisms. This program has included up to 130,000 feet of travel on various components and 350 full-height drops. The production mechanism design incorporates improvements derived from experience gained on this program.

1.7.4 FUEL BUNDLE DESIGN

Combustion Design

Cold-water flow-induced vibration tests on fuel rods and subassemblies and mechanically induced vibration tests in air on model bundles have been completed. An extensive hot flow test program, including the effects of forced cross flow, has been completed using essentially full-length, though not full cross-section, bundles. Total time at reactor conditions (or conditions believed to be more severe) exceeded 13,000 hours. These tests have been supplemented by a basic program on the mechanism of grid to fuel rod wear conducted in a static autoclave with mechanically induced relative motion.

These tests substantiate the adequacy of the fuel bundle design for its expected service.

In addition to this program, four full-scale model fuel bundles were tested by Combustion Engineering at reactor conditions (or more severe conditions) in the Utility Reactor Components Test Facility. Beginning with the second fuel cycle, Combustion Engineering fuel has not been used.

Current Fuel Design

Comparable developmental testings, as described previously, have also been performed by the current fuel vendor. Refer to Subsection 3.3.4.3 for details.

1.7.5 REACTOR VESSEL FLOW TESTS

A one-fifth scale model of the reactor vessel and its internals has been constructed and subjected to airflow testing at the Battelle Memorial Institute Laboratories at Columbus, Ohio. These tests have investigated flow distribution, pressure drop and the tracing of flow paths within the vessel for all four pumps running and various part-loop configurations.

1.8 SPECIAL MAJOR PROGRAMS

As a result of continued NRC concern with the "health and safety of the public" and its relationship to the safe operation of all nuclear facilities, the Palisades Plant and operations have been subjected to considerable NRC review.

The Inspection and Enforcement Branch of the NRC routinely provides IE Bulletins (for utilities to review and respond) regarding the identification of generic problems that could have a safety impact. In addition, the NRC on occasion establishes technical review programs as a result of legal mandates following court actions or NRC initiated programs resulting from unusual events in the industry.

Programs of special interest to Palisades resulting from these circumstances are discussed in the remainder of this section.

1.8.1 SYSTEMATIC EVALUATION PROGRAM

1.8.1.1 Description of Program

Between 1977 and the early 1980s, Consumers Power Company (CPCo) participated in the NRC’s Systematic Evaluation Program (SEP). The purpose of this program was to confirm the adequacy of certain as-licensed design features of eleven plants, including Palisades, whose construction permits were issued before the final General Design Criteria (GDC) (10 CFR 50 Appendix A), the associated Standard Review Plans (NUREG 75/087 and 0800), and other guidance documents. The program was also intended to provide a basis for converting the Provisional Operating Licenses (POL) held by some of the plants to Full Term Operating Licenses (FTOL). The program was designed to be conducted by the NRC with limited, voluntary support of licensees. The overall approach was to document major differences between each plant’s as-licensed design and then-current design criteria in written topic evaluation reports; to evaluate the safety significance of differences that were judged to be potentially significant; and, finally, to make decisions about whether any of those differences should be resolved through backfits, using an integrated safety assessment process with the assistance of probabilistic risk assessment (PRA) tools.

Topic evaluation reports were prepared for a number of subject areas, and these reports, in turn, provided inputs to the integrated safety assessment process. A separate integrated safety assessment report was issued to document the final NRC conclusions for selected issues raised in the topic evaluation reports which were judged to warrant discussion or action. The individual topic evaluation reports did not contain the final NRC conclusions about each topic, although they often did include the author’s opinions or recommendations about perceived safety significance and the desirability of backfits for certain issues.

The integrated safety assessment process was an internal NRC activity. Accordingly, the public record does not always contain detailed discussions of NRC bases for selecting certain design differences for additional study, of the NRC decisions process for selecting specific differences to be assessed as candidates for backfitting, nor of the internal NRC process for judging adequacy of actions proposed by CPCo. The outputs of the integrated safety assessment process, which documented the final agency conclusions for the evaluated topics, were published in NUREG 0820 (Integrated Plant Safety Assessment Report) (Reference 3), NUREG 0820 Supplement 1, NUREG 1424 {Safety Evaluation Report (SER) supporting conversion of license from POL to FTOL} (Reference 4), and in several later letters on topics that were not closed until after the Integrated Plant Safety Assessment Report was published.

Table 1-3 provides a listing of the final 90 topics reviewed in the SEP for Palisades, and summarizes the results of each review. The SEP review began by comparing the as-built plant design with the then current review criteria in 137 different areas defined as "topics."

During the review, 47 of the topics were deleted from consideration by SEP, based on one of the three following reasons: (1) topic was part of the Unresolved Safety Issue Program (USI), (2) topic was part of Three Mile Island Action Plan Tasks, or (3) the topic was not applicable to the Plant. The remaining 90 topics were reviewed for Palisades and are those listed in Table 1-3. Fifty-nine of the 90 topics met current criteria or were acceptable on another defined basis. These topics are identified as Status Code S on Table 1-3.

Thirty-one topics received further review and evaluation during the Integrated Assessment Program. A major part of the integrated assessment was the probabilistic risk assessment (PRA). PRA was used to determine which system failures would create an unacceptable risk because either a redundant system was not available or available systems were inadequate for the job required. During the integrated assessment, several of these topics were found to be acceptable and required no further work. These items are identified as Status Code 4 in Table 1-3.

The remaining topics evaluated using PRA, were each found to require one or more of the following modifications:

1. Plant changes

2. Technical Specifications changes

3. Refined engineering analysis required

These items are identified as Status Code 1, 2 or 3 in Table 1-3. The disposition column of Table 1-3 shows where specific topics are addressed.

1.8.1.2 SEP Reviews Confirmed Safety of Palisades Design

The SEP topic evaluations and integrated plant safety assessment documents confirmed that the level of safety provided by the Palisades design was adequate even though the design differed from later design requirements embodied in the General Design Criteria and other documents. The NRC letter of October 29, 1982, which transmitted the final report of the SEP review (NUREG 0820) to CPCo, specifically states, “The review has provided for … a documented evaluation of plant safety when all supplements to the IPSAR and the Safety Evaluation report for converting the license from a provisional to a full-term license have been issued.” The fact that Integrated Plant Safety Assessment Report did not identify backfits or other actions for many of the identified design differences, and the fact that the SEP results later provided a part of the basis to convert the Palisades POL to a FTOL, provide de facto evidence that the licensed plant design was found to be adequate.

While the GDC were used as reference standards for the reviews, the SEP did not backfit a requirement to comply with the GDC. The NRC generic position on applicability of the GDC to plants of Palisades’ age was later summarized as follows: “The General Design Criteria are not applicable to plants with construction permits issued prior to May 21, 1971. At the time of the promulgation of Appendix A, the Commission stressed that the GDC were not new requirements and were promulgated to more clearly articulate the licensing requirements and practice in effect at that time. While compliance with the intent of the GDC is important, each plant licensed before the GDC were formally adopted was evaluated on a plant specific bases, determined to be safe, and licensed by the Commission. Furthermore, current regulatory processes are sufficient to ensure that plants continue to be safe and comply with the intent of the GDC.” (Reference 5) The FSAR should be used to determine the extent of Palisades’ commitments, if any, to any particular revision or criterion of the GDC.

1.8.1.3 Continuing Applicability and Interpretation of SEP Information

The SEP topic evaluations and supporting information have continuing relevance in that they help to clarify the plant's original licensing and design bases, and they provide documentation of NRC review. The topic evaluations sometimes include summaries of the design and regulatory philosophy which existed when Palisades was designed and licensed. At times the topic evaluations discuss explicit bases for the reviewers' judgments about adequacy and safety significance of certain plant design features. However, the SEP topic evaluation should not be used independently from the final Integrated Plant Safety Assessment Report or other closure documents. The information in individual topic evaluations concerning perceived design weaknesses, or additional actions recommended by a topic evaluation's author, were not formal NRC conclusions, and the recommendations contained therein did not create licensee commitments. Some of the specific issues raised in topic evaluations have no documented closure. Closure sometimes has to be inferred from the fact that NRC did not include a discussion of a specific issue in NUREG 0820, its supplement, subsequent docketed letters on selected topics, or in NUREG 1424. If an issue was recommended for additional action within a topic evaluation, but that issue or action was not documented in NUREG 0820 or successor documents, it can be assumed that the issue was screened out during NRC's integrated safety assessment process as not having sufficient safety significance to warrant further action or discussion.

Most of the actions taken to resolve issues raised during the SEP were voluntarily proposed by CPCo and do not represent permanent obligations. When accepted by NRC, the CPCo-proposed actions were implemented as NRC commitments, and the actions were summarized by NRC in NUREG 0820 and its Supplement 1. The specific actions taken by CPCo as a result of the SEP, and the specific plant design and operating features which were reviewed during the SEP, can be changed with appropriate justification in accordance with 10 CFR 50.59, plant administrative requirements, and/or NRC-endorsed industry guidelines for NRC commitment management.

1.8.2 TMI ACTION ITEMS (NUREG-0737)

As a result of the incident at Three Mile Island Nuclear Power Plant, the NRC developed a list of requirements for other nuclear-powered generating stations. The list consists of 37 items, which are broken down into a total of 94 subitems as identified in NUREG-0737.

The list of items, compliance status and general description of how the item was to be resolved, are shown in Table 1-4. All of the 94 subitems have been closed out and are identified in the table as Status Code 1. The NRC SER for issuance of the Full Term Operating License, NUREG 1424 dated November 1990, confirms that the NRC views all TMI Action Items as being resolved for Palisades.

1.8.3 PIPE SUPPORT BASEPLATE DESIGNS USING CONCRETE EXPANSION ANCHOR BOLTS (IE BULLETIN 79-02)

Nuclear Regulatory Commission IE Bulletin 79-02, addressed Seismic Category I pipe supports using concrete expansion anchor bolts (CEBs) for loadings obtained from analysis of Seismic Category I piping systems.

All baseplates or structural steel members using CEBs for large piping identified in the course of responding to IE 79-02 were evaluated. The evaluation was performed in accordance to the load combinations specified in Section 5.10. Acceptance criteria were as specified in the Bulletin for CEBs and Chapter 5 for baseplates or structural steel members. Those baseplates or structural steel members and CEBs which did not satisfy acceptance criteria were modified (or will be modified).

Approximately 3,000 accessible CEBs for large bore piping were inspected and load tested. More than 96% of this population satisfied the load testing.

Approximately 4% of the CEBs for large bore piping were inaccessible for full testing and inspection. These CEBs and their baseplates or structural steel members were evaluated. If these CEBs and baseplates or structural steel members did not satisfy acceptance criteria, they were either modified or the piping support system was revised to yield acceptable results.
Small bore piping supports were designed using a conservative chart method. A sample of CEBs used for support of small piping was inspected and tested. This sample consisted of more than 1,000 CEBs (more than 2/3 of the population). This testing and inspection program used in conjunction with the conservative chart method, yields an acceptable confidence level for small piping.

Thus, the inspection, testing and evaluation performed for baseplates or structural steel members and CEBs for Seismic Category I piping, satisfy the requirements of IE 79-02, and the modifications have been completed.

1.8.4 SEISMIC ANALYSIS FOR AS-BUILT SAFETY-RELATED PIPING SYSTEMS (IE BULLETIN 79-14)

The bulletin required an inspection of approximately 18,100 feet of large diameter safety-related piping, 1,550 pipe supports and piping components at the Palisades Plant. Small piping systems (2 inches or less in diameter) were also inspected, noted items evaluated and a sample of the small piping systems was evaluated.

The Palisades Plant systems were reviewed and it was determined that 23 systems had safety-related piping. Data on and sketches of the safety-related systems were completed, potential nonconformance items were listed and the as-built data were evaluated. Approximately 320 piping support changes have been completed: (1) 45 new supports were added, (2) 23 supports were removed, and (3) 252 supports were modified.

There were approximately 3,250 listed conditions that were either questionable or constituted a discrepancy. These items were evaluated and resolved. About 75% of the items related to lack of or nonconformance with existing drawings. The remaining 750 items related to hardware conditions, such as nuts/bolts loose or missing, spring cans bottomed out or without load and bent/broken or missing pipe support components. Two Licensee Event Reports (LERs) were issued as a result of the program (LER 79-033 and LER 80-001). Corrective action has been completed on both items.

In 1989, discrepancies were noted in the original 79-14 bulletin analyses. It was determined that design/evaluation criteria, documentation and work quality of the original effort were not adequate. LER 89-23 and LER 89-23 Rev 1 were issued. They outlined a work scope to correct the deficiencies and established interim operability criteria to facilitate implementation of the effort. A major program was established "Safety Related Piping Reverification Program" for re-evaluating large bore piping. Subsequently, a small bore piping program was also established. These programs were developed to consist of walkdowns to establish as built inputs for piping analysis and support evaluation. When analysis or evaluation shows it is necessary, plant modifications are made to satisfy acceptance criteria.

1.8.5 UNRESOLVED SAFETY ISSUES (NUREG-0410)

Of the unresolved safety issues (also called generic safety issues) identified by the NRC and discussed in NUREG-0410, -0510, -0649, and -0705, 19 were considered by the NRC to require investigation for their potential impact on the Palisades Nuclear Plant. The 19 unresolved safety issues considered are listed in Table 1-5 with a cross-reference to the appropriate FSAR chapter wherein they are discussed. All 19 of the unresolved safety issues have been assessed by Consumers Power Company and are considered to have no undue risk to the health and safety of the public while longer term generic review of these issues is being conducted.

USI A-46, Seismic Qualification of Equipment in Operating Plants, has been resolved by the Seismic Qualification Utility Group (SQUG). Equipment in the Safe-Shutdown paths required for plant shutdown was evaluated using the SQUG Generic Implementation Procedure (GIP) (Reference 10). The "Report of SQUG Assessment at Palisades Nuclear Plant for the Resolution of USI A-46" was submitted to the NRC on May 19, 1995 (Reference 11). The NRC Safety Evaluation Report for the resolution of USI A-46 was issued on September 25, 1998 (Reference 12).

There were several outliers identified (ie, equipment items that did not meet the SQUG GIP initial screening criteria). Essentially all of the outliers were resolved over the next five years by testing, analysis, modification or some combination of these approaches. It was concluded that the Safety Injection Refueling Water (SIRW) Tank (T-58) could not be resolved by any of the methods employed for the other outliers.

A multidisciplinary effort was pursued to develop a source and flow path of borated water that could be used in lieu of the SIRW Tank and its associated piping to provide suction to the charging system to ensure the maintenance of inventory and boron concentration in the primary coolant system while the plant is shutdown following a seismic event. This alternate path, which utilized the spent fuel pool as the source of borated water, was evaluated using the identical SQUG GIP approach as used to evaluate other Safe Shutdown Equipment. SQUG GIP assessments were performed on the additional equipment and plant modifications were performed as appropriate to designate the alternate path as operational.

The letter entitled “Final Closeout of Unresolved Safety Issue A-46 Outliers,” from Douglas E. Cooper to the USNRC Document Control Desk was submitted to the NRC on June 26, 2003 (Reference 13). This letter provided the final resolution of USI A-46 for the Palisades Nuclear Plant including the resolution of all outliers, including the SIRW tank outlier which was resolved by implementing an alternate path.

1.8.6 ENVIRONMENTAL QUALIFICATION OF "SAFETY-RELATED" ELECTRICAL EQUIPMENT (EEQ) (NUREG-0588) (USI A-24)

In order to assure the reliable functions of certain electrical equipment subjected to harsh environments following accident conditions, each licensee was requested to reevaluate all previously installed equipment. Pursuant to an NRC order issued August 29, 1980, Consumers Power Company engaged in the preparation of EEQ documentation. That information was submitted to the NRC on October 7, 1980 in a report entitled, "Environmental Qualification of Safety Related Electrical Equipment - Palisades Plant," September 1980. Revisions to the report were submitted to the NRC from October 29, 1980 to May 20, 1983. Industry resolution of this issue was embodied in new rule 10CFR50.49. For Palisades the NRC issued an SER on January 31, 1985 which contained the finding that the plant's environmental qualification program was in compliance with 10CFR50.49.

1.8.7 CONTROL ROOM HABITABILITY (NUREG-0696)

In 1983 Consumers Power Company completed several modifications to the control room HVAC system to satisfy the control room habitability requirements of NUREG-0737. These included extending the control room air intake from the then existing configuration, increasing the intake air duct to allow 100% makeup air, installing redundant charcoal filters, extending the control room habitability zone and replacing air intake and discharge dampers. The safety evaluation concluded that the systems will provide safe, habitable conditions within the control room under both normal and accident radiation and toxic gas conditions, including Loss of Coolant Accidents. See discussion in Section 6.10.

1.8.8 EFFECTS OF PIPE RUPTURE (SEP TOPICS III.5.A AND B)

In December 1972, the NRC initially raised the concern for the dynamic effects of pipe ruptures. In response, Consumers Power Company had an analysis done for postulated high-energy line breaks outside of containment. This report is entitled "Special Report No 6 - Analysis of Postulated High-Energy Line Breaks Outside of Containment" (SR-6), Revision 3A.

The concerns of high-energy line breaks and moderate-energy line breaks were further reviewed for both situations inside and outside containment in SEP Topics III.5.A and III.5.B, respectively. The conclusions were that the criteria used to assess pipe breaks at the Palisades Nuclear Plant was in accordance with present-day criteria. For discussion, see Section 5.6.

1.8.9 STATION BLACKOUT (10 CFR 50.63) (USI A-44)

In 1988 the NRC issued 10 CFR 50.63 to define a loss of all onsite AC power sources (Station Blackout) as an event with which all plants must be able to cope. By letters dated April 17, 1989, December 11, 1989, March 27, 1990, and July 3, 1990, Consumers Power Company certified that the Palisades evaluation of the issue was completed in accordance with the specified guidance, NUMARC 87-00, and that Palisades possessed the required coping ability. In a letter dated May 20, 1991, the NRC issued its SER, which stated "...we find that the Palisades Plant conforms to the SBO rule, and the guidance of Regulatory Guide (RG) 1.155, Nuclear Management and Resources Council (NUMARC) 87-00, and NUMARC 87-00 Supplemental Questions/Answers and Major Assumptions." NRC acceptance was contingent upon satisfactory resolution of several included recommendations. CPCo letter dated August 1, 1991 provided responses/commitments to resolve those recommendations. Final NRC closure was provided in an SER dated June 25, 1992.

1.8.10 SAFE SHUTDOWN

For Palisades' original design, "Safe Shutdown" was "Hot Shutdown". The original design did not require the ability to achieve cold shutdown conditions; therefore, the design of the systems used to get to cold shutdown was determined by the Architect Engineer or the Nuclear Steam Supply System vendor and was not based on any regulatory safety concern.

From the NRC requirements applied since the original design, being a "Hot Shutdown" plant does not mean the only consideration is getting to and maintaining hot shutdown. The need to get to Cold Shutdown for all plants has been made clear by the NRC. Specific references include Branch Technical Position RSB 5-1 "Design Requirements for Residual Heat Removal System" (Reference 1) and the NRC's SEP review position for Palisades (Reference 2).

Equipment and components needed to get to cold shutdown may be nonsafety-related if:

a. An alternate means exists to allow cooldown to cold shutdown assuming failure of the nonsafety-related equipment/component under consideration or,

b. Manual action or repairs are identified to correct a single failure within some reasonable time period and are found to be acceptable to allow cooldown to cold shutdown.

When alternate means or manual actions/repairs are used to justify equipment and components as nonsafety-related, such actions should be covered by procedures when the actions are required within the first several hours following the initiating event. If reasonable reaction time exists, or the issue is of negligible probability, then inclusion in a procedure may not be warranted. The complexity associated with either recognizing the need for or the performance of the alternate means, manual action, or repair is considered when determining the appropriateness of a procedure.

1.8.11 HEAVY LOADS

An NRC Generic Letter dated December 22, 1980, requested licensees to prepare responses to indicate their degree of compliance with certain guidelines for NUREG-0612, “Control of Heavy Loads.” Phase I of this effort involved identifying the load handling equipment within the scope of NUREG-0612 and describing general heavy load handling program activities, such as safe load path identification, load handling procedures, operator training, the use of special and general purpose lift devices, the maintenance, testing and repair of the cranes, and crane design (Reference 89). These program activities are implemented and controlled by plant procedures. Phase II of this effort involved further actions in response to additional NUREG-0612 guidance (Reference 90). The NRC accepted the response to Phase I for Palisades in an SER dated November 9, 1983 (Reference 15).

Subsequently, the NRC issued Generic Letter 85-11 (Reference 16), which stated that, based on improvements in the handling of heavy loads obtained from Phase I, further action to reduce the risks associated with the handling of heavy loads was not required, and the Phase II was considered to be completed.

The NRC issued Bulletin 96-02, “Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment” (Reference 91) which requested licensees to review their capabilities to handle heavy loads while the plant is operating and to remind licensees of their responsibilities for ensuring that heavy load activities are performed safety and within requirements. Palisades responded to the Bulletin (Reference 92) and the response was accepted by the NRC (Reference 93).

1.9 RENEWED FACILITY OPERATING LICENSE

On January 17, 2007, the Renewed Facility Operating License was issued by the NRC, extending the expiration date to March 24, 2031. License Condition 2.H of the renewed license required that the FSAR be supplemented in accordance with 10 CFR 54.21(d) to incorporate summary descriptions of the programs and activities credited for managing the effects of aging (Aging Management Programs), and of the evaluation of Time-Limited Aging Analyses (TLAAs), for the period of extended operation. The Application for Renewed Operating License (LRA) had been submitted on March 22, 2005. This section includes that supplement.

During the NRC review of the LRA, several commitments for future actions were made by the licensee. Table 1-9 provides the listing of final commitments as submitted by the licensee and confirmed in Appendix A of the Safety Evaluation Report Related to the License Renewal of Palisades Nuclear Plant (NUREG 1871). Some changes to the commitments were made through the commitment change process. All commitments that were required to be implemented prior to entering the period of extended operation were completed prior to entering the period.

Section 1.9.1 contains summary descriptions of the programs used to manage the effects of aging during the period of extended operation, and Section 1.9.2 contains summaries of TLAAs applicable to the period of extended operation.

Under the renewed operating license, Aging Management Programs are applied to certain non-safety related Systems, Structures, and Components (SSCs). To ensure appropriate controls are provided for non-safety related aging management activities, a commitment was made to apply certain Quality Assurance Program provisions to Aging Management Programs. These provisions apply to the program elements of corrective action, confirmation process, and administrative controls. FSAR Section 15.1.2 describes the generic quality assurance requirements to be applied to the Aging Management Programs.

1.9.1 SUMMARY DESCRIPTIONS OF AGING MANAGEMENT PROGRAMS

This section provides summaries of programs and activities credited in the License Renewal Application for managing the effects of aging during the period of extended operation.

The activities implemented to manage aging at the Palisades Plant may be performed under discrete programs as defined herein, or they may be incorporated into other plant programs. The program summaries should be interpreted as summaries of activities to be performed to manage aging, and not as specific commitments to maintain unique programs with the specific titles and content listed.

It should also be noted that these summaries do not specifically invoke or reference the Generic Aging Lessons Learned (GALL) Report, NUREG-1801. The activities credited for managing aging at Palisades were developed, to a large extent, to be responsive to the revision of the GALL that existed at the time that the License Renewal Application was developed. It is expected that changes will be made to these programs in the future as a result of advances in the state of knowledge in the industry, plant modifications, and operating experience. However, no commitment is made to update any aging management program in response to changes in the GALL. Future changes that may occur to aging management programs or activities will be managed under 10 CFR 50.59, "Changes, Tests and Experiments," and/or other regulatory and administrative requirements appropriate to the changes being made.

1.9.1.1 Nickel Alloy Program

The Nickel Alloy Program manages aging due to Primary Water Stress Corrosion Cracking (PWSCC) of the Primary Coolant System (PCS) pressure boundary Alloy 600 components, including Inconel 82/182 weld joints, reactor vessel head penetrations, etc. The program includes:

a. PWSCC susceptibility assessment using industry models to identify susceptible components

b. Monitoring and control of primary coolant chemistry to mitigate PWSCC

c. In-Service Inspections (ISI) of pressurizer penetrations, reactor vessel head penetrations and Alloy 82/182 PCS pressure boundary welds in accordance with American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (BPV Code) Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” Subsection IWB, Table IWB-2500-1

d. Augmented inspections or preemptive repair/replacement of susceptible components or welds

The License Renewal Application included a commitment to submit for NRC review and approval a revised nickel alloy (ie, Alloy 600) aging management program that updates the PWSCC corrosion rate assessments and inspection program consistent with the latest NRC requirements and industry commitments (Reference 57). The revised nickel alloy aging management program was submitted on March 13, 2008 (Reference 101). The NRC determined that the revised program was acceptable in a safety evaluation dated August 15, 2012 (Reference 102).

1.9.1.2 ASME Section XI, Subsections IWB, IWC, IWD, IWF Inservice Inspection Program

The applicable ASME BPV Code for the fifth ten-year interval of the inservice inspection program at the Palisades Plant is ASME Section XI, 2007 edition, including 2008 addenda.

ASME Section XI IWB, IWC, IWD, and IWF Inservice Inspection Program facilitates inspections to identify and correct degradation in Class 1, 2, and 3 piping, components, and their supports and integral attachments. The program includes periodic visual, surface, and/or volumetric examinations and leakage tests of all Class 1, 2, and 3 pressure-retaining components and their supports and integral attachments, including welds, pump casings, valve bodies, pressure-retaining bolting, piping/component supports, and reactor head closure studs. These are identified in ASME Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” or commitments requiring augmented inservice inspections, and are within the scope of license renewal. This program is in accordance with 10 CFR 50.55a, "Codes and Standards."

1.9.1.3 Bolting Integrity Program

Palisades Bolting Integrity Program relies on the guidelines delineated in NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants," EPRI NP-5769, "Degradation and Failure of Bolting in Nuclear Power Plants" (with the exceptions noted in NUREG-1339), for safety related bolting, and EPRI TR-104213, "Bolted Joint Maintenance & Applications Guide" (for non-safety related bolting. The program also includes repair/replacement controls for ASME Section XI related bolting and generic guidance regarding material selection, thread lubrication, and assembly of bolted joints. The program considers the guidelines delineated in NUREG-1339 for a bolting integrity program, EPRI NP-5769 (with the exceptions noted in NUREG-1339) for safety related bolting, and EPRI TR-104213 for non-safety related bolting.

1.9.1.4 Boric Acid Corrosion Program

The Palisades Boric Acid Corrosion Program monitors component degradation due to boric acid leakage through the performance of periodic inspections. It implements the recommendations of NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR plants." The program requires periodic visual inspection of all systems within the scope of license renewal that contain borated water for evidence of leakage, accumulations of dried boric acid, or boric acid wastage. The program also provides for visual inspections and early discovery of borated water leaks such that structures and electrical and mechanical components that may be contacted by leaking borated water will not be adversely affected such that their intended functions are impaired.

1.9.1.5 Buried Services Corrosion Monitoring Program

The Buried Services Corrosion Monitoring Program manages aging effects on the external surfaces of carbon steel, low-alloy steel, and stainless steel components that are buried in soil or sand. This program includes (a) visual inspections of external surfaces of buried components for evidence of coating damage and substrate degradation to manage the effects of aging, (b) visual inspection of the external surfaces of buried stainless steel components for evidence of crevice corrosion, pitting, and Microbiologically Influenced Corrosion (MIC). The periodicity of these inspections for carbon, low-alloy, and stainless steel will be based on opportunities for inspection such as scheduled maintenance work.

1.9.1.6 Closed Cycle Cooling Water Program

The Closed Cycle Cooling Water Program manages aging effects in closed cycle cooling water systems that are not subject to significant sources of contamination, in which water chemistry is controlled and heat is not directly rejected to the ultimate heat sink. The program includes (a) maintenance of system corrosion inhibitor concentrations to minimize degradation, and (b) periodic or one-time testing and inspections to assess component aging. This program is based on the guidelines in EPRI TR-107396, "Closed Cooling Water Chemistry Guideline." The program scope includes activities to manage aging in the Component Cooling Water (CCS) System, Emergency Diesel Generator (EDG) Jacket Cooling Water (Emergency Power System), and Shield Cooling System (SCS).

1.9.1.7 Containment Inservice Inspection Program

The Containment Inservice Inspection (ISI) Program is designed to ensure that containment shell concrete, the post-tensioning system, and steel pressure retaining elements continue to provide an acceptable level of structural integrity. In addition, it is designed to ensure that the liner (with associated moisture barriers), other leakage limiting steel barriers, and pressure retaining bolted connections have not degraded.

1.9.1.8 Containment Leakage Testing Program

The Containment Leakage Testing Program ensures that containment leakage is maintained below the upper acceptance limit of La = 0.1% / day. This testing program, in conjunction with the Containment Inservice Inspection Program, provides assurance that age related (and other) deterioration of the containment leakage limiting boundary is appropriately managed to ensure that postulated post-accident releases are limited to an acceptable level. The program is implemented through the following testing and examination activities:

- Overall containment leakage (integrated leakage rate or Type A) test to assess the leak tight integrity of the entire pressure boundary
- Visual examinations of the containment exterior and interior
- Local (Type B & C) tests to assess the leak tight integrity of individual penetrations

1.9.1.9 Diesel Fuel Monitoring and Storage Program

The Diesel Fuel Monitoring and Storage Program assures the continued availability and quality of fuel oil to be used in diesel generators and diesel fire pumps. The program includes:

a. Monitoring and trending of fuel oil chemistry to maintain fuel oil quality and mitigate corrosion

b. Periodic draining, cleaning, and internal inspection of fuel oil storage tanks

c. Periodic ultrasonic measurement of thickness of the bottom of fuel oil storage tanks

Fuel oil quality is maintained by monitoring and controlling fuel oil contamination in accordance with the guidelines of the American Society for Testing Materials (ASTM) Standards D 1796, D 2276, D 2709, and D 4057, and by verifying the quality of new oil before its introduction into the storage tanks.

1.9.1.10 Fire Protection Program

The Fire Protection Program includes:

a. Fire barrier inspections

b. Electric and diesel-driven fire pump tests

c. Periodic maintenance, testing, and inspection of water-based fire protection systems

Periodic visual inspections of fire barrier penetration seals, fire dampers, fire barrier walls, and ceilings and floors and periodic visual inspections and functional tests of fire-rated doors are performed to ensure that functionality and operability is maintained. Periodic testing of the fire pumps ensures that an adequate flow of firewater is supplied and that there is no degradation of diesel fuel supply lines. Periodic maintenance, testing, and inspection activities of water-based fire protection systems provide reasonable assurance that fire water systems are capable of performing their intended function. Inspection and testing include periodic hydrant inspections, fire main flushing, sprinkler inspections, pipe wall thickness testing, and flow tests.

1.9.1.11 Flow Accelerated Corrosion Program

The Flow Accelerated Corrosion Program manages aging effects due to Flow-Accelerated Corrosion (FAC) on the internal surfaces of carbon or low alloy steel piping, elbows, reducers, expanders, and valve bodies which contain high energy fluids (both single phase and two phase). The program implements the Electric Power Research Institute (EPRI) guidelines in NSAC-202L-R3, "Recommendations for an Effective Flow-Accelerated Corrosion Program," for an effective FAC program and includes:

a. An analysis using a predictive code such as CHECWORKS™ to determine critical locations

b. Baseline inspections to determine the extent of thinning at these locations

c. Follow-up inspections to confirm the predictions

d. Repairing or replacing components, as necessary

1.9.1.12 Non-EQ Electrical Commodities Condition Monitoring Program

The Non-EQ Electrical Commodities Condition Monitoring Program manages aging in selected non-EQ commodity groups within the scope of 10 CFR 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants." Features of the program include periodic inspection and/or testing of the following commodity groups:

a. Accessible insulated cables and connections in scope of license renewal installed in adverse localized environments.

b. Sensitive instrumentation cables and connections in scope of license renewal.

c. Inaccessible medium voltage cables in scope of license renewal (not designed for submergence), subject to long periods of high moisture conditions and voltage stress.

d. Underground manholes for the accumulation of water over medium voltage cables in scope of license renewal.

e. Non-segregated bus and connections in scope of license renewal for insulation degradation, bus enclosure for degradation, and bus supports for structural integrity.

f. Representative sample of bolted electrical connections in scope of license renewal.

1.9.1.13 One-Time Inspection Program

The One-Time Inspection Program addressed potentially long incubation periods for certain aging effects, including various corrosion mechanisms, cracking, and selective leaching, and provided a means of verifying that an aging effect is either not occurring or progressing so slowly as to have negligible effect on the intended function of the structure or component. Hence, the One-Time Inspection Program provided measures for verifying an aging management program is not needed, verifying the effectiveness of an existing program, or determining that degradation is occurring which required evaluation and corrective action.

The program included:

a. Determination of appropriate inspection sample size

b. Identification of inspection locations

c. Selection of examination technique, with acceptance criteria

d. Evaluation of results to determine the need for additional inspections or other corrective actions

The inspection sample included locations where the most severe aging effect(s) would be expected to occur. Inspection methods included visual (or remote visual), surface or volumetric examinations, or other established Non-Destructive Examination (NDE) techniques.

This program was used for a variety of purposes, including the following:

- To verify the effectiveness of water chemistry control for managing the effects of aging in stagnant or low-flow portions of piping or components, exposed to a treated water environment

- To manage the aging effects of loss of material due to aging mechanisms such as general, crevice, pitting, and galvanic corrosion; selective leaching; and MIC

- To verify that cracking due to stress corrosion cracking or cyclic loading, in small bore (< 4" NPS) ASME class 1 piping, is not occurring

- To verify, for components in the Compressed Air System, that there are no aging effects requiring management in the dry air environment [This aspect of the program was superseded by the creation of a Compressed Air Monitoring Program as described in licensee letter dated October 31, 2005. This program is further described in Section 1.9.1.23.]

- To verify, for carbon steel storage tanks supported on earthen or concrete foundations, that excessive corrosion is not occurring on the bottom surfaces of the tanks

The One Time Inspection Program was completed on March 14, 2011. This completion date conforms with the program implementation schedule date of prior to the period of extended operation as described in NUREG-1871, "Safety Evaluation Report Related to the License Renewal of Palisades Nuclear Plant," (Reference 69) and guidance in NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," (Reference 68) that the population of components be inspected before the end of the current operating term. Palisades entered the period of extended operation on March 24, 2011.

1.9.1.14 Open Cycle Cooling Water Program

The Open Cycle Cooling Water Program manages aging effects such as loss of material due to general, pitting, and crevice corrosion, erosion, MIC, and loss of heat transfer due to biological/corrosion product fouling (e.g., sedimentation, silting) caused by exposure of internal surfaces of metallic components to raw, untreated (e.g., service) water. The program scope includes activities to manage aging in the Service Water System (SWS) and Circulating Water system (CWS).

The aging effects are managed through:

a. Monitoring and control of biofouling

b. Flow balancing and flushing

c. Heat exchanger testing

d. Routine inspection and maintenance program activities

e. System walkdowns

f. Review of maintenance, operating, and training practices and procedures to ensure that aging effects do not impair component intended function

Inspection methods include visual (VT), ultrasonic (UT), radiographic (RT), and eddy current (ECT). This program is responsive to NRC GL 89-13, "Service Water System Problems Affecting Safety-Related Equipment."

1.9.1.15 Overhead Load Handling Systems Inspection Program

The Overhead Load Handling Systems Inspection Program provides for inspections of the structural components and rails of cranes and fuel handling machines associated with heavy load handling that are subject to the requirements of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," and are within the scope of license renewal requiring aging management. For Palisades these are the Containment Building Polar Crane, the Spent Fuel Pool Overhead Crane, the Containment Building jib and boom cranes, and the reactor and spent fuel pool fuel handling machines. These cranes comply with the Maintenance Rule requirements provided in 10 CFR 50.65, "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." The Overhead Load Handling Systems Inspections Program is primarily focused on structural components that make up the bridge and trolley of the overhead cranes that are within the scope of NUREG-0612.

1.9.1.16 Reactor Vessel Integrity Surveillance Program

The Reactor Vessel Integrity Surveillance Program manages the aging effect reduction of fracture toughness due to neutron embrittlement of the low alloy steel reactor vessel. Monitoring methods will be in accordance with 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements."

This program includes:

a. Capsule insertion, withdrawal, and materials testing/evaluation (including upper shelf energy and RT_{NDT} determinations)

b. Fluence and uncertainty calculations

c. Monitoring of Effective Full Power Years (EFPY)

d. Development of pressure temperature limitations

e. Determination of Low Temperature Overpressure Protection (LTOP) set points

The program ensures the reactor vessel materials (a) meet the fracture toughness requirements of 10 CFR 50, Appendix G, "Fracture Toughness Requirements," and (b) have adequate margins against brittle fracture caused by Pressurized Thermal Shock (PTS) in accordance with 10 CFR 50.61a, "Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events."

1.9.1.17 Reactor Vessel Internals Inspection Program

The Reactor Vessel Internals (RVI) Inspection Program manages aging effects for reactor vessel internals. In response to the publication of Electric Power Research Institute (EPRI) MRP-227, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0)," December 2008, Palisades revised the Reactor Vessel Internals Inspection Program. This program was published as WCAP-17133-NP, "PWR Vessel Internals Program Plan for Aging Management of Reactor Internals at Palisades Nuclear Plant," Revision 0, November 2009, and was submitted to the NRC in a letter dated March 3, 2010. Submittal of this program satisfied License Renewal Commitment No. 33 listed in FSAR Table 1-9.

On September 13, 2012, Entergy Nuclear Operations, Inc. submitted a revised aging management program plan "Palisades Reactor Vessel Internals Aging Management Program," for NRC Review. The Palisades aging management plan (AMP) was developed based on the NRC staff approval topical report MRP-227-1, "Material Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines." The submittal of the

AMP fulfilled a regulatory commitment that originated from the license renewal activities as documented in NUREG-1871, "Safety Evaluation Report Related to the License Renewal of PNP Nuclear Plant." The NRC approved the revised program plan on December 11, 2014 (Reference 107).

The Reactor Vessel Internals Inspection Program provides for:

a. Inservice Inspection (ISI) in accordance with ASME Section XI requirements, including examinations performed during 10-year ISI examinations (See ASME Section XI IWB, IWC, IWD, IWF Inservice Inspection Program)

b. Monitoring and control of reactor coolant water chemistry in accordance with EPRI guidelines to mitigate SCC or IASCC (See Water Chemistry Program)

c. Augmented inspections as directed by MRP-227-A

In addition, the nuclear industry, through both EPRI/MRP and the PWR Owners Group, continue to sponsor activities related to RVI aging management. Entergy will maintain cognizance of industry activities related to PWR internals inspection and aging management and will address/implement industry guidance stemming from those activities, as appropriate, under the practices of Nuclear Energy Institute NEI 03-08, "Guidelines for the Management of Materials Issues." This includes the implementation of applicable changes in MRP-227-A.

1.9.1.18 Steam Generator Tube Integrity Program

The Steam Generator Tube Integrity Program manages the aging effects of steam generator tubes and tube repairs. The Program also manages the aging effects of accessible steam generator secondary side internal components and incorporates the guidance of NEI 97-06, "Steam Generator Program Guidelines." The program manages aging effects through a balance of mitigation, inspection, evaluation, repair, and leakage monitoring measures. Component degradation is mitigated by controlling primary and secondary water chemistry. Eddy current testing is used to detect steam generator tube flaws and degradation. Visual examinations are performed to identify degradation of accessible steam generator secondary side internal components. Primary to secondary leakage is monitored during plant operation.

1.9.1.19 Structural Monitoring Program

The Structural Monitoring Program is designed to ensure that age related (as well as other) deterioration of plant structures (including masonry walls) and components within its scope is appropriately managed to ensure that each such structure or component retains the ability to perform its intended function. The program is implemented through visual examination of these structures, components, and other specified items. In addition, the program provides for inspections of opportunity of normally inaccessible below grade concrete when excavation work uncovers a significant depth (several feet or more) to provide access for inspection. Damage or degradation found during visual examination may be further evaluated by measurements and testing techniques as appropriate. As part of the Structural Monitoring Program, groundwater sampling for pH, chlorides, and sulfates will be performed, with a periodicity not to exceed every 5 years, to ensure the below grade environment remains non-aggressive.

This program also implements provisions of the Maintenance Rule, 10 CFR 50.65, that relate to masonry walls and water-control structures. It conforms to the guidance contained in RG 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," and NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," as well as Nuclear Energy Institute publication NEI 96-03, "Guideline for Monitoring the Condition of Structures at Nuclear Power Plants." This NEI document, which supplements NUMARC 93-01, contains additional guidance specific to the monitoring of structures. In addition, the program specifies that inspections for unreinforced block walls that are not contained by bracing will be performed on a more frequent basis than the normal frequency of once each 10-year interval specified for reinforced or braced block walls.

1.9.1.20 System Monitoring Program

The System Monitoring Program manages aging effects for normally accessible, external surfaces of piping, tanks, and other components and equipment within the scope of License Renewal. These aging effects are managed through visual inspection and monitoring of external surfaces for leakage and evidence of material degradation. The program relies upon periodic system walkdowns to monitor degradation of the protective paint or coating, and/or the exterior steel surface area (if no paint or coatings exist, or if the existing protective paint and coatings are degraded to a point whereby the exterior steel surface is exposed). Palisades does not take credit for any above ground coating or paint for mitigating corrosion even though the tanks may be painted or coated. However, inspections of the above ground coating or paint will provide an indication of the condition of the material underneath the coating or paint.

1.9.1.21 Water Chemistry Program

The Water Chemistry Program manages aging effects such as loss-of-material due to general, pitting, and crevice corrosion; cracking due to SCC; and steam generator tube degradation caused by denting, Intergranular Attack (IGA), and Outer Diameter Stress Corrosion Cracking (ODSCC), by controlling the environment to which internal surfaces of systems and components are exposed. The aging effects are minimized by controlling the chemical species that cause the underlying mechanisms that result in these aging effects. The program provides assurance that an elevated level of contaminants and, where applicable, oxygen does not exist in the systems and components covered by the program, thus minimizing the occurrences of aging effects, and maintaining each component's ability to perform the intended functions. The program is based on the guidelines in EPRI Topical Report, "PWR Primary Water Chemistry Guidelines," and EPRI Topical Report, "PWR Secondary Water Chemistry Guidelines."

1.9.1.22 Inspections of Opportunity for Internal Surfaces of Selected Components and Corrosion Under Insulation

Internal surfaces of selected systems and components which are exposed during periodic system and component surveillances, or during the performance of maintenance activities, are subject to visual inspections of opportunity. These inspections are applicable to components in-scope for license renewal that have an internal environment of water, are constructed of materials that are potentially susceptible to internal aging degradation in a wetted environment, but are not subject to an Aging Management Program (e.g., Water Chemistry) that would manage the internal environment such that aging degradation of the internal surfaces would not be expected. Visual inspections are performed to assure that existing environmental conditions are not causing material degradation that could result in a loss of a component intended function. Inspection activities are performed by qualified personnel looking for corrosion (General, Pitting, Crevice, MIC) and fouling. Degraded conditions are documented in the Corrective Action Program and evaluated for acceptability, repair, or replacement.

External surfaces of selected insulated piping and components, which are exposed when insulation is removed for maintenance or surveillance, are subject to visual inspections of opportunity. The piping and components of interest are those within the scope of the System Monitoring Program, constructed of carbon or low alloy steel, with low normal operating temperatures in an indoor or outdoor environment such that the piping could be wetted under its insulation (e.g., from condensation or rain water) for extended periods without being detected. Degraded conditions are documented in the Corrective Action Program and evaluated for acceptability, repair, or replacement.

1.9.1.23 Compressed Air Monitoring Program

The Compressed Air Monitoring Program manages aging affects on the internal surfaces of carbon steel, low-alloy steel, copper alloys, and stainless steel components within the scope of License Renewal that are exposed to a compressed air environment. These include components such as piping, traps, heat exchangers, filter housings, dryer housings, accumulators, and valve bodies made of materials such as carbon steel, low alloy steel, copper alloys, and stainless steel. The program manages the aging effects of General, Crevice, and Pitting Corrosion and Stress Corrosion Cracking. The program includes maintenance of the compressors, dryers, and filters associated with the plant Instrument Air System, High Pressure Air System, Feedwater Purity Air System, and associated back-up systems.

1.9.1.24 Oil Sampling and Analysis

For selected components, in-scope for License Renewal, that have an internal environment of oil, and are constructed of materials that are potentially susceptible to internal aging degradation in that environment, the oil shall be subject to periodic sampling and analysis. The purpose of these activities is to ensure that oil system contaminants (primarily water and particulates) are maintained within acceptable limits, thereby preserving an environment that is not conducive to loss of material or reduction of heat transfer. Associated activities include:

a. Determination of appropriate analysis to be performed
b. Frequency of analysis
c. Acceptance criteria
d. Trending of results
e. Corrective actions, if required

These activities ensure that the lubricating oil environment of these components is maintained such that water and contaminants are minimized.

1.9.1.25 Electrical Equipment Qualification Program

The Electrical Equipment Qualification Program is an existing program that implements the requirements of 10 CFR 50.49, “Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants,” at Palisades. 10 CFR 50.49 defines the scope of components to be included, requires the preparation and maintenance of a list of in-scope components, and requires the preparation and maintenance of a qualification file that includes component performance specifications, electrical characteristics, the environmental conditions to which the components could be subjected, and the basis for qualification. 10 CFR 50.49(e)(5) contains provisions for aging that require, in part, consideration of all significant types of aging degradation that can affect component functional capability. 10 CFR 50.49(e)(5) also requires replacement or refurbishment of qualified components prior to the end of its designated life, unless additional life is established through ongoing qualification. EQ programs manage component thermal, radiation, and cyclical aging through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods.

1.9.1.26 Fatigue Monitoring Program

The Fatigue Monitoring Program is a new program that ensures that limits on fatigue usage are not exceeded during the renewal term. The program monitors and tracks selected cyclic loading transients (cycle counting) and their effects on susceptible components. Palisades has selected this option under 10 CFR 54.21, "Contents of Application--Technical Information," to manage cracking due to metal fatigue of the reactor coolant pressure boundary during the extended period of operation.

The Fatigue Monitoring Program provides cycle counting activities for confirming analytically derived cumulative usage values for applicable locations. Specific locations that may be subject to cyclic loading that could cause fatigue cracking are monitored using a computer-based monitoring program provided by EPRI, called FatiguePro. If warranted, other monitoring methods in addition to cycle counting may also be employed under this program to monitor specific locations.

1.9.2 SUMMARY DESCRIPTIONS OF TIME-LIMITED AGING ANALYSES

As part of a License Renewal Application, 10 CFR 54.21(c) requires that an evaluation of Time-Limited Aging Analyses (TLAAs) for the period of extended operation be provided. The following TLAAs were identified and evaluated during the license renewal process to meet this requirement.

1.9.2.1 Reactor Vessel Neutron Embrittlement

a. Upper Shelf Energy

The Charpy upper shelf energy is associated with the determination of acceptable reactor vessel toughness during the license renewal period. 10 CFR Part 50 Appendix G, "Fracture Toughness Requirements," Paragraph IV.A.1, requires that the reactor vessel beltline materials must have Charpy upper shelf energy of no less than 68 J (50 ft-lb) throughout the life of the reactor vessel, unless otherwise approved by the NRC. In the event that the 50 ft-lb requirement cannot be satisfied as stated in 10 CFR 50 Appendix G, or by alternative procedures acceptable to the NRC, the reactor vessel may continue to operate provided requirement 1 of Appendix G is satisfied. This requirement states that an analysis must conservatively demonstrate, the existence of equivalent margins of safety for continued operation.

Analysis

As shown in FSAR Table 1-6, "Estimated USE on March 24, 2031," the upper shelf energy for reactor vessel beltline materials at the end of the extended period of operation is expected to decrease to less than 50 ft-lbs based on predictions using Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials." A low upper-shelf fracture mechanics analysis has been performed to evaluate the plate and weld material for ASME Levels A, B, C, and D Service Loadings, based on the acceptance criteria of the ASME Code, Section XI, Appendix K. Combustion Engineering Report NPSD-993 has determined that Combustion Engineering reactor vessels maintain equivalent margins of safety, if plate material and circumferential weld material maintains at least 30 ft-lb upper shelf energy, and if longitudinal weld material maintains at least 34 ft-lb upper shelf energy. This analysis has not been submitted to the NRC.

<u>Disposition</u>: Aging Management, 10 CFR 54.21(c)(1)(iii)

10 CFR 50, Appendix G requires licensees to submit an analysis at least 3 years prior to the time that the upper-shelf energy of any of the reactor vessel material is predicted to drop below 50 ft-lb., as measured by Charpy V-notch specimen testing. Entergy has complied with this requirement.

Entergy submitted an equivalent margins analysis, completed in accordance with 10 CFR 50 Appendix G Section IV.A.1, for NRC approval, before any reactor vessel beltline material upper shelf energy decreased to less than 50 ft-lb (Reference 104). The analysis documented that the materials with upper shelf energy values that drop below the USE lower limit of 50 ft-lb throughout the remaining life of the reactor vessel have margins of safety against fracture equivalent to those required by Appendix G of Section XI of ASME Code. The analysis was subsequently approved by the NRC (Reference 105).

This issue will be dispositioned using the method of 10 CFR 54.21(c)(1)(iii), the effects of aging on the intended function will be adequately managed for the period of extended operation.

b. Pressurized Thermal Shock

The pressurized thermal shock (PTS) rule, 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," established screening criteria that are a measure of a limiting level of reactor vessel material embrittlement beyond which operation cannot continue without further plant-specific evaluation. The screening criteria are given in terms of reference temperature RT_{PTS}. The screening criteria are 270°F for plates and axial welds, and 300°F for circumferential welds.

The alternate pressurized thermal shock (PTS) rule, 10 CFR 50.61a, "Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," established screening criteria that are a measure of a limiting level of reactor vessel material embrittlement beyond which operation cannot continue without further plant-specific evaluation. The screening criteria are given in terms of an embrittlement reference temperature RT_{MAX-X}, and are provided in Table 1 of 10 CFR 50.61a.

Analysis

The results of the PTS analysis for the limiting material were reviewed for compliance with 10 CFR 50.61. The methodology used in the PTS analysis was based on the projected neutron fluence at the end of the period of extended operation. The RT_{PTS} values for the intermediate and lower shell plates remained below the NRC screening criterion of 270ºF, and the RT_{PTS} value for the circumferential weld also remained below the NRC screening criterion of 300°F. The RT_{PTS} values for the axial welds were projected to exceed the NRC screening criterion of 270°F. The vessel was projected to reach the PTS screening criterion of 270ºF on the beltline axial welds fabricated with weld wire heat W5214 in August 2017.

Subsequently, an alternate PTS analysis was performed, and the results of the analysis for the limiting material was reviewed for compliance with 10 CFR 50.61a. The methodology used in the alternate PTS analysis was based on the projected neutron fluence at the end of the period of extended operation. As shown in FSAR Table 1-7, "RT_{MAX-X} Values at 42.1 Effective Full Power Years," the RT_{MAX-X} values for the axial welds, the shell plates, and the circumferential welds remained below the 50.61a NRC screening criteria through the period of extended operation.

Disposition: Aging Management, 10 CFR 54.21(c)(1)(iii)

10 CFR 50.61 requires the licensee to implement a flux reduction program that is reasonably practicable to avoid exceeding the screening criteria. If the flux reduction program does not prevent the reactor vessel from exceeding the PTS screening criterion at the end of life, 10 CFR 50.61 allows two options. The licensee can submit a safety analysis pursuant to 10 CFR 50.61(b)(4) to determine what, if any, modifications to equipment, systems, and plant operation are necessary to prevent failure of the reactor vessel from a postulated PTS event. The other option is to perform a thermal-annealing treatment of the reactor vessel pursuant to 10 CFR 50.61(b)(7) to recover fracture toughness. 10 CFR 50.61 requires the details of the selected alternative be provided to the NRC three years prior to when the reactor vessel is projected to exceed the PTS screening criteria. At the appropriate time, prior to exceeding the PTS screening criteria, Palisades will select the optimum alternative to manage PTS in accordance with NRC regulations and make relevant submittals to obtain NRC review and approval.

Palisades has selected the following method to manage PTS for the reactor pressure vessel.

Subsequent to the license renewal application, the NRC published 10 CFR 50.61a, "Alternate Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events." This alternative for PTS management involves inspecting the reactor vessel beltline region, determining limiting RT_{MAX-X} values for each axial and circumferential weld, plate, and forging, and submitting a report to NRC to justify continued operation using the PTS screening criteria in Table 1 of 10 CFR 50.61a. The regulation requires that an application for implementation of 10 CFR 50.61a be submitted at least three years before the limiting RT_{PTS} value calculated under 10 CFR 50.61 is projected to exceed the PTS screening criteria.

An updated PTS evaluation performed in accordance with 10 CFR 50.61 was submitted in 2010 and was subsequently approved by the NRC in 2011 (References 99 and 100). The PTS evaluation determined that the reactor vessel limiting material, which are the axial welds fabricated with weld wire heat W5214, will not reach the PTS screening criterion limit until April 2017.

An updated reactor vessel fluence evaluation that reflected recent actual reactor operation subsequently determined that the PTS screening criterion would be reached in August 2017 rather than April 2017 (References 104 and 105).

In 2014, a license amendment request to implement 10 CFR 50.61a was submitted to the NRC (Reference 109). The alternate PTS evaluation accompanying the submittal concluded that the reactor vessel materials remain below the 10 CFR 50.61a screening criteria through the period of extended operation (42.1 effective full power years) (Reference 110). The NRC approved the license amendment request in 2015 (Reference 111).

This issue will be dispositioned using the method of 10 CFR 54.21(c)(1)(iii), the effects of aging on the intended function will be adequately managed for the period of extended operation.

c. Pressure-Temperature (P-T) Limits

10 CFR Part 50 Appendix G requires that the reactor pressure vessel be maintained within established pressure-temperature limits including during heatup and cooldown. These limits specify the maximum allowable pressure as a function of reactor coolant temperature, and are contained in Technical Specifications. As the reactor pressure vessel becomes embrittled and its fracture toughness is reduced, the allowable pressure (given the required minimum temperature) is reduced.

Low temperature overpressure protection limits and setpoints are determined as part of the calculation of pressure/temperature operating limit curves.

Analysis

The current pressure/temperature analyses were determined to be valid to the end of the period of extended operation (Reference 97). License Amendment 245 was issued which extends the validity of the current pressure/temperature operating limit curves through the period of extended operation (Reference 98).

Disposition

This issue will be dispositioned using the method of 10 CFR 54.21(c)(1)(iii), the effects of aging on the intended function will be adequately managed for the period of extended operation. The Pressure-Temperature operating limits contained in Technical Specifications will be updated as required by either Appendices G or H of 10 CFR 50, or as operational needs dictate.

d. Low Temperature Overpressure Protection (LTOP) PORV Setpoints

Low temperature overpressure protection limits and setpoints are determined as part of the calculation of pressure/temperature operating limit curves. See the Pressure-Temperature (P-T) Limits section.

1.9.2.2 Metal Fatigue

a. Reactor Vessel Fatigue Analyses

The Palisades Reactor Vessel was designed, constructed, and analyzed to the ASME Boiler and Pressure Vessel Code, Section III, Subsection 4 for Class A vessels, 1965, with addenda through Winter, 1965. The original analyses have been corrected and amended to address issues that have arisen since fabrication. The current design basis highest calculated fatigue usage factors, based on the number of design basis load cycles assumed by the vessel analyses, have been determined. The number of design basis load cycles for each event was selected to be adequate for the originally-licensed 40-year design life.

Analysis

This section addresses the calculated fatigue usage factors for the reactor vessel:

- Shell and bottom head
- Inlet and outlet nozzles
- Internal welded attachments
- Instrument nozzle shroud tube
- Vessel head CRDM nozzles
- Instrument flange bolts (on the instrument nozzle on the vessel head)

The worst-case calculated usage factor for the set of design basis load events in these locations is 0.4516 on the outlet nozzle, well within the code limit of 1.0.

The number of each of the design basis events that affect the reactor pressure vessel is not expected to approach its design basis limit during the extended licensed operating period. Therefore, the actual fatigue usage factors are not expected to approach their calculated values during the extended licensed operating period. The calculated maximum usage factor for these locations on the reactor vessel is 0.4516, well within the analytical limit of 1.0.

Disposition

This issue is dispositioned using the method of 10 CFR 54.21(c)(1)(i), the analysis remains valid for the period of extended operation.

In addition, the Fatigue Monitoring Program will ensure that the effects of aging will be adequately managed for the period of extended operation under 10 CFR 54.21(c)(1)(iii) by assuring that a reanalysis or other appropriate corrective action is taken if a design basis cycle count limit is reached at any time during the extended licensed operating period.

b. Reactor Vessel Head Closure Stud Fatigue Analysis

The highest fatigue usage factor calculated by the reactor vessel fatigue analysis is in the vessel head studs.

Analysis

The calculated lifetime usage factor in the head closure studs for the set of design basis load events is 0.8346, within the code limit of 1.0. The number of design basis transient events is not expected to approach the number assumed by the analysis during the extended licensed operating period.

Disposition

This issue is dispositioned using the method of 10 CFR 54.21(c)(1)(i), the analysis remains valid for the period of extended operation.

In addition, the Fatigue Monitoring Program will ensure that the effects of aging will be adequately managed for the period of extended operation under 10 CFR 54.21(c)(1)(iii) by assuring that a reanalysis or other appropriate corrective action is taken if a design basis cycle count limit is reached at any time during the extended licensed operating period.

c. Control Rod Drive Mechanism (CRDM) Housing Fatigue Analyses

The reactor Control Rod Drive Mechanisms (CRDM) are enclosed in pressure housings, bolted, and seal-welded to the reactor pressure vessel CRDM nozzle flanges. The CRDM housings, their seal housings, their instrument and vent tube nozzles, the flange bolts, and the Omega seal welds between the CRDM housing flanges and the reactor vessel CRDM nozzle flanges were all replaced in 2001. Extension of the operating license to March 24, 2031, therefore, only requires a 30-year design life. The replacements are ASME III (1989) - Class 1, NPT stamped, with reconciliation to the 1965 code.

The upper CRDM housing at CRD-24 was replaced in 2012 due to through wall leakage. The upper CRDM housing replacement at CRD-24 was manufactured to ASME Boiler & Pressure Vessel Code, Section III, Subsection NB, 1998 edition up to and including the 2000 addenda, and was reconciled to the original design code.

A total of 44 CRDM upper housings were replaced in 2014 during 1R23 when a follow-up inspection to the 2012 incident revealed potential inner surface flaws in several housings. All replacements were manufactured to ASME Boiler & Pressure Vessel Code, Section III, Subsection NB 1998 edition up to and including the 2000 addenda. The replacement housings were reconciled to the original design code.

Analysis

<u>CRDM Housings</u>: The revised fatigue evaluation found that the criteria of ASME III - 1989, Paragraph NB-3222.4(d) are met and, therefore, that no fatigue analysis is required.

<u>CRDM Seal Housings and Tool Access Tube Assemblies</u>: The fatigue evaluation finds the criteria of N-415.2(d)(6) of ASME III are met. A revised fatigue analysis was not performed. The replacement material is stronger than the analyzed material.

CRDM Housing Bolts: The fatigue evaluation of record calculated a design lifetime cumulative usage factor of 0.173. This evaluation has not been revised, but will remain valid so long as the assumed number of lifetime design basis transient cycles remains valid. This analysis is very conservative since the bolts were replaced when the CRDM housings were replaced.

CRDM Flange - Reactor Vessel Nozzle Flange Bolts: The fatigue evaluation of record calculated a design lifetime cumulative usage factor of 0.624. This evaluation has not been revised, but will remain valid so long as the assumed number of lifetime design basis transient cycles remains valid. This analysis is very conservative since the bolts were replaced when the CRDM housings were replaced.

CRDM Flange - Reactor Vessel Nozzle Flange Omega Seal Welds: The fatigue evaluation calculated a design lifetime cumulative usage factor of 0.5621. This analysis used the set of design transients from the original design specification, which were based on an assumed 40-year design life.

For the components replaced in 2001 or later, their expected installed lifetime, including the extended licensed operating period (to March 24, 2031), will be only about 30 years, compared to the 40 years upon which the estimate of design basis event cycles was based. The two highest calculated maximum usage factors for a 40-year life in any of these components is 0.624 for the flange bolts and 0.5621 for the flange Omega seal welds, well within the analytical limit of 1.0.

In addition, the number of each of the design basis events that affect the reactor pressure vessel and the CRDM housings and appurtenances is not expected to approach its design basis limit during the extended licensed operating period. The actual fatigue usage factors are, therefore, not expected to approach their calculated values during the extended licensed operating period.

Disposition

This issue is dispositioned using the method of 10 CFR 54.21(c)(1)(i), the analysis remains valid for the period of extended operation.

d. Steam Generator Fatigue Analyses

The Palisades steam generators were replaced in 1990-1991. Extension of the operating license to March 24, 2031, therefore requires a 40-year design life. The replacement steam generators were designed to the ASME Boiler and Pressure Vessel Code, Section III, 1977. The primary coolant pressure boundary (tube side) of the steam generators is designed to Section III Class 1 rules. Critical components of the Class 2-design secondary side (e.g., the feedwater nozzles) were also analyzed using Class 1 methods.

Analysis

Vessel and Components, Except Manway Studs: The ASME III Class 1 fatigue analyses of the replacement steam generators used the number of event cycles assumed for the original plant design for a 40-year licensed operating period.

Except for the manway studs (below), the maximum usage factor at any location is 0.9158, on the main feedwater nozzle.

Manway Studs: The original ASME III Class 1 fatigue analyses calculated a worst-case usage factor for the manway studs of 0.10. However, the vendor (Westinghouse) later issued a Nuclear Safety Advisory Letter identifying a significant bending load on the studs due to differential thermal expansion during the heatup and cooldown transients, which resulted in predicted lifetime usage factors greater than 1.0. The revised analysis found that the fatigue limit for the studs would be reached in about 200 reactor heatup and cooldown cycles.

This problem has been addressed by a requirement in plant procedures to evaluate the number of heatup and cooldown cycles every 5 operating years, and to replace the manway studs before they can experience 200 heatup and cooldown cycles. Since the studs were installed at Plant Heatup Number 106, they should be replaced before Heatup Number 306. This five-year evaluation interval was based on a very conservative assumption that no more than 12.5 startup and shutdown cycles would occur per operating year, and, therefore, less than 63 cycles should accumulate between evaluations.

However, since the fatigue life analysis that supports the current design and licensing basis safety determination for the manway studs does not depend on the licensed operating period, the analysis is not a TLAA.

The replacement steam generator fatigue analyses qualify them for a 40-year design life, except for the manway studs. The replacement steam generators were installed in 1991. The qualified 40-year design life is therefore sufficient for the extended licensed operating period ending in 2031.

Disposition

This issue is dispositioned using the method of 10 CFR 54.21(c)(1)(i), the analysis remains valid for the period of extended operation.

e. Pressurizer Fatigue Analyses

The pressurizer was designed to the ASME Boiler and Pressure Vessel Code, Section III, Class A, 1965, Winter 1966 addenda. The code design calculation includes a fatigue analysis for those nozzles or other parts which do not meet the fatigue analysis exemption criteria of Section III Paragraph N-415.1. These include all nozzles attached by J-welds and other nozzles and parts subject to more-severe thermal transients:

- The surge, spray, and temperature element nozzles and nozzle-to-shell or nozzle-to-head junctions
- Heater sleeve-to-head junctions
- Upper level nozzles
- Relief (PORV) and safety valve nozzles
- The liquid-vapor boundary region of the shell
- Manway, head, and studs, and
- Bottom head support skirt

A revised set of external load cycles required reevaluation of fatigue in the three safety relief valve nozzles. The recalculation found that the stress intensities produced by the revised external loads are less than those calculated by the original analysis and, therefore, that the original simplified fatigue evaluation remains valid, except for the nozzle at RV-1041. The nozzle flange at RV-1041 was replaced and a new detailed fatigue evaluation was performed.

The Alloy 600 safe end to-pipe weld at the power-operated relief valve nozzle was found cracked and leaking in 1993. The Alloy 600 safe end was repaired with a short stainless steel pipe, and the nozzle was reanalyzed. The highest usage factor calculated for the modified safe end and its connections is 0.7572 at the inside of the nozzle-head juncture. This calculation assumed load cycles for a 40-year design life.

Analysis of the PORV nozzle safe end material removed in 1993 indicated primary water stress corrosion cracking (PWSCC), and prompted replacement of the remainder of the safe end with 316 stainless material with Alloy 690 welds, in 1995. The fatigue analysis for the currently-installed safe end and attachments calculated a worst-case usage factor of 0.084, at the inside wall of the safe end transition.

<u>Other TLAAs of the Pressurizer</u>: Thermal stratification phenomena in the surge line have required reanalysis of the surge nozzle, and concerns for high differential temperatures with auxiliary spray have required reanalysis of the spray nozzle.

Primary water stress corrosion cracking (PWSCC) of the Alloy 600 temperature nozzles required repair, a revised fatigue analysis, and analyses of the PWSCC effects. These failures, failure of the PORV nozzle safe end, and industry-wide cracking of Alloy 600 components have required evaluation of PWSCC effects in all Alloy 600 components.

Analysis and Disposition

Fatigue usage factors have been calculated for the pressurizer:

- Bottom head support skirt
- Heater sleeve-head junctions
- Liquid-vapor boundary region of the shell
- Upper level nozzles
- Power-operated relief valve (PORV) nozzle
- Code safety valve nozzles
- Manway, head, and studs

For these locations the worst-case calculated usage factor is 0.7572, at the inner PORV nozzle-head juncture. This issue is dispositioned using the method of 10 CFR 54.21(c)(1)(i), the analysis remains valid for the period of extended operation.

The fatigue analysis of the replaced PORV nozzle safe end was based on a nominal 20-year life beyond its 1995 installation. However, the low worst-case usage factor of 0.084 in this component permits a simple projection to the end of the extended licensed operating period, when the service life of this component would be about 36 years. A projected usage factor based on this 36-year life would be only about 0.15, compared to the allowable 1.0. This issue is dispositioned using the method of 10 CFR 54.21(c)(1)(i), the analysis remains valid for the period of extended operation.

For those portions of the original pressurizer analysis which have not been superseded, the number of each of the design basis events is not expected to approach its design basis limit during the extended licensed operating period, and, therefore, the actual fatigue usage factors are not expected to approach their calculated values during the extended licensed operating period. The calculated maximum usage factor for these locations on the pressurizer is 0.7572, within the analytical limit of 1.0, at the inner PORV nozzle-head juncture. This issue is dispositioned using the method of 10 CFR 54.21(c)(1)(i), the analysis remains valid for the period of extended operation.

In addition, the Fatigue Monitoring Program will ensure that the effects of aging will be adequately managed for the period of extended operation under 10 CFR 54.21(c)(1)(iii) by assuring that a reanalysis or other appropriate corrective action is taken if a design basis cycle count limit is reached at any time during the extended licensed operating period.

f. Regenerative Heat Exchanger Fatigue Analyses

The regenerative heat exchanger recovers energy from the letdown line to heat chemical and volume control system charging water for reactor system makeup (charging) and auxiliary pressurizer spray. The charging system has three positive-displacement pumps. The pumps cannot be throttled without lifting discharge safety valves, but one of them is variable speed. The charging system was designed to operate continuously, with flow from the fluid-drive variable-speed pump controlled by a primary coolant volume control signal. If the variable-speed pump is out of service, the system controls Primary Coolant System makeup by cycling a constant-speed pump on and off. This cycling of cold makeup water against the (approximately constant) hot letdown flow produces significant thermal transients in the regenerative heat exchanger.

Isolation of letdown flow introduces a similar differential thermal load and has a similar effect.

Analysis

The original design included a fatigue evaluation to ASME III -1965, Paragraph N-415. The final addendum to the original analysis was based on a revised definition of the load transient set which eliminated the 15 percent per minute load-following transient for this component. The worst-location Cumulative Usage Factors (CUF) for a 40-year licensed operating life were 0.871 on the tubesheet, and 0.624 on the shell.

By 1993, operating experience had shown that the availability of the variable-speed pump was less than anticipated in the original design and, therefore, that the number of thermal cycles from cycling a constant-speed pump was greater than anticipated. The expected number of lifetime thermal cycles and the fatigue usage factor at the most limiting location (the tube sheet) were therefore reevaluated. The transient evaluation increased the lifetime number of thermal stress cycles due to this event from 5,520 to 17,822. The fatigue evaluation found that, with the increased number of thermal cycles described above plus the remainder of the design basis event set, the CUF at the limiting tube sheet location would be 1.002, slightly above the analytic limit of 1.0. However, the design basis event set included 60,500 load-following and reactor trip events (Transient I), of which the unit had experienced only about 180 by that date (1993). The fatigue evaluation therefore reduced the design basis Transient I cycles slightly (to 60,282), and demonstrated that the analytic limit of 1.0 was then met. The 1993 reanalysis retained the simplifying assumption from the original analysis, that the thermal effect of cycling a constant-speed charging pump produced the maximum stress range for all load pairings.

A revision in 1995 to permit more frequent letdown isolation re-evaluated both thermal and pressure transient effects. The evaluation of stress pairings for Transient I load-following and reactor trip events found that those with the maximum stress range could be further reduced to 32,500 from the 60,282 assumed in the 1993 analysis, even considering the increased frequency of letdown isolations. This 1995 revision again evaluated only the limiting inner ligament of the shell side of the tubesheet, for which the calculated lifetime CUF is 0.880.

Based on the plant events in the first 20 years of operation, a recent revision to the calculation estimates that the number of cycles for cycling the constant-speed charging pump can be increased to 27,062, and Transient I can be reduced to 6,240 for 60-year plant life. With the new estimated numbers of transient events at the end of extended operating period, the maximum fatigue usage factors at the two most critical locations in the regenerative heat exchanger (tubesheet and tubesheet to shell junction) is 0.439.

Disposition: Revision, 10 CFR 54.21(c)(1)(ii); and Aging Management, 10 CFR 54.21(c)(1)(iii)

Re-analysis has been performed to include additional thermal cycles from cycling the constant-speed charging pump and reduced number of transients that were over estimated in the original design analysis. The projected CUF at the end of the extended operating period remains less than 1. Therefore, the regenerative heat exchanger meets the criteria of 10 CFR 54.21(c)(1)(ii).

The fatigue management cycle count program will include the letdown isolation and variable-speed charging pump out-of-service events.

g. ASME III Class A Primary Coolant Piping Fatigue Analyses

A piping fatigue analysis was originally applied only to the main loops of the primary coolant system, the two 42-inch hot legs and the four 30-inch cold legs, and to the connecting nozzles for smaller piping. The original analyses calculated fatigue usage factors for the:

- Hot legs
- Cold legs
- Safety injection-shutdown cooling nozzles
- Hot leg to surge line nozzle
- Charging Inlet nozzles
- Hot leg temperature nozzles
- Shutdown cooling outlet nozzle
- Cold leg temperature nozzles

The hot leg to surge line nozzle has been reanalyzed to address transients not contemplated in the original analysis.

The design stress ranges of the cold-leg-to-pressurizer-spray nozzles were below the endurance limit in the original design basis analysis. However the cold-leg-to-pressurizer-spray nozzles have since been evaluated for additional transients not contemplated in the original analysis. The fatigue issue of the cold-leg-to-pressurizer-spray nozzles is addressed in FSAR Section 1.9.2.2.h, "Revised Bulletin 88-11 Fatigue Analysis of the Hot-Leg-to-Pressurizer-Surge-Line."

The following section addresses the remaining original analyses for the hot and cold legs, and for the remaining original nozzles.

Analysis

The fatigue analysis of record for the hot and cold legs uses the bounding stresses at all locations in each of hot leg and cold leg to determine the worst possible stress ranges. These usage factors are therefore considerably higher than would be calculated by a location-specific analysis. The maximum CUF is 0.07551 for the hot leg, and 0.7531 for the cold leg.

The cold leg charging nozzle was reanalyzed to account for the additional cycles due to the constant-speed charging pump recycling. CUF at the end of the extended operation period is 0.306.

The cold leg safety injection-shutdown cooling and charging inlet nozzles are also sample locations, which were reanalyzed for the effects of the reactor coolant environment on fatigue behavior, with a resulting NUREG/CR-5999, “Interim Fatigue Design Curves for Carbon, Low-Alloy, and Austenitic Stainless Steels in LWR Environments,” CUF of 0.456.

Disposition: Validation, 10 CFR 54.21(c)(1)(i), and Aging Management, 10 CFR 54.21(c)(1)(iii)

The number of each of the design basis events that affect the hot and cold legs is not expected to approach its design basis limit during the extended licensed operating period, and therefore that the actual fatigue usage factors are not expected to approach their calculated values during the extended licensed operating period. The calculated maximum usage factor for the hot and cold legs and their nozzles in these original calculations is 0.7531, well within the analytical limit of 1.0.

The predicted hot leg and cold leg usage factors are calculated on a very conservative basis. The hot leg usage factor is quite low, and the cold leg usage factor, though appearing significant at 0.7531, would be much less if calculated by a location-specific analysis.

The Fatigue Monitoring Program will ensure a reanalysis or other appropriate corrective action in the unlikely event that a design basis cycle count limit is reached at any time during the extended licensed operating period.

h. Revised Bulletin 88-11 Fatigue Analysis of the Hot-Leg-to-Pressurizer-Surge-Line Nozzle, Surge Line, and Pressurizer Surge Nozzle

NRC Bulletin 88-11, "Pressurizer Surge Line Stratification," dated December 1988, was issued to address pressurizer surge line temperature stratification concerns. The effects of thermal stratification were evaluated by the Combustion Engineering Owners Group (CEOG). The Combustion Engineering Owners Group Report concluded the structural integrity of the pressurizer surge line is acceptable for the forty year life of the Plant. The NRC issued an SER on September 13, 1993, concluding that the CEOG analysis adequately demonstrates that the bounding surge line and nozzles meet ASME Code stress and fatigue requirements for the 40-year design. CPCo provided additional information detailing completion of the required actions of Bulletin 88-11, including the requirement to update the pressurizer surge line stress and fatigue analyses. See FSAR Section 4.3.7.

Analysis

For both surge line nozzles and the surge line elbow, the calculated usage factors for the revised set of design basis load events are within the code limit of 1.0.

<u>Pressurizer Surge Line Elbow</u>: The fatigue evaluation for a typical Combustion Engineering plant with intermittent pressurizer spray and for the revised set of design basis load events, including the IEB 88-11 thermal stratification transients, calculated a maximum CUF of 0.937 at one of the surge line elbows.

A recent fatigue analysis using thermal stratification conditions under the Palisade continuous pressurizer spray operation show that the CUF is reduced significantly to 0.0135 for the expected number of cycles at the end of the 60-year operating period. If the number of cycles at the end of the extended operating period were based on 1.5 times the 40-year design basis cycles, the CUF at the surge elbow would be 0.0447.

This location is also a NUREG/CR 6260, "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," sample location for evaluation of environmental effects of the reactor coolant on fatigue effects, which calculated a CUF of 0.238.

<u>Hot Leg to Surge Line Nozzle</u>: The fatigue evaluation for a typical Combustion Engineering plant with intermittent pressurizer spray and for the revised set of design basis load events, including the IEB 88-11 thermal stratification transients calculated a maximum CUF of 0.3818 for Palisades hot leg to surge line nozzle. With Palisades continuous pressurizer spray operation, the CUF is reduced significantly, similar to the above surge line elbow, because piping loads due to thermal stratification are the major contributor to nozzle fatigue stress.

Pressurizer Surge Nozzle: The fatigue evaluation for a typical Combustion Engineering plant with intermittent pressurizer spray and for the revised set of design basis load events, including the IEB 88-11 thermal stratification transients calculated a maximum CUF of 0.9611 for the Palisades pressurizer surge line nozzle. With Palisades continuous pressurizer spray operation, the CUF is reduced significantly, similar to the above surge line elbow, because piping loads due to thermal stratification are the major contributor to the nozzle fatigue stress.

Design Basis Thermal Transients and Expected Thermal Transients: The additional thermal stratification transients are in the pressurizer surge line and nozzles during plant heatup and cooldown at differential temperatures of 320, 250, 200, and 150º F ΔT, and hot standby at 90º F ΔT. These design transients were developed by the CE owners group for a typical plant with severe thermal transients due to intermittent pressurizer spray.

The use of modulated, continuous spray for pressure control, and control of pressurizer to primary loop ΔT, significantly mitigates this problem at Palisades. An assessment of the thermal stratification event mechanisms for Palisades operating conditions found that for almost all such events the metal ΔT would not exceed 210º F, instead of the 320º F ΔT assumed by the standard plant analysis. This is further supported by the log of pressurizer spray events at ΔT above 200º F, which has recorded only 47 through 9 January 2005.

These transients are auxiliary spray events, which have little or no effect on surge line stratification.

This moderation of the transients reduces the piping differential expansion loads and support and nozzle reactions. A recent fatigue analysis using thermal stratification conditions under the Palisade continuous pressurizer spray operation show that the CUF at the surge line elbow is reduced to less than 0.1 at the end of the extended operation period. Since piping load is the major contributor to the nozzle fatigue stress, a similar reduction is expected for the surge line nozzles.

Design Basis Cycle Count and Expected Cycle Count: The number of design cycles developed by the CE Owners Group for each of the above thermal stratification transients correspond to 500 cycles of plant heatup and cooldown. The number of transient events which might be expected to initiate these thermal stratification events will not exceed their design basis limits for the extended licensed operating period, the same is, therefore, also true for these thermal stratification events.

Disposition for Surge Line Elbow: Revision, 10 CFR 54.21(c)(1)(ii); and Aging Management, 10 CFR 54.21(c)(1)(iii)

A plant-unique calculation for the surge line shows that the fatigue usage factor of the surge line elbow remains less than 1 at the end of the extended operating period. The calculation includes the revised set of design basis load events, including the plant-unique IEB 88-11 thermal stratification transients. Therefore, the surge line elbow meets the revision criteria per 10 CFR 54.21(c)(1)(ii).

The Fatigue Monitoring Program will ensure that the effects of aging will be adequately managed for the period of extended operation by assuring that a reanalysis or other appropriate corrective action is taken if a design basis primary coolant system cycle count limit is reached at any time during the extended licensed operating period.

Disposition for Hot Leg and Pressurizer Surge Nozzles: Validation, 10 CFR 54.21(c)(1)(i); and Aging Management, 10 CFR 54.21(c)(1)(iii)

The design basis analysis calculated maximum usage factor at the hot leg surge nozzle of 0.3818 is well below the analytical limit of 1.0. The usage factor remains less than 1 at the end of the extended operating period. Therefore, the surge line elbow and the pressurizer surge nozzle meet the validation criteria per 10 CFR 54.21(c)(1)(i).

The calculated maximum usage factor for the pressurizer surge nozzle of 0.9611 is below the analytical limit of 1.0. This value is the result of a generic plant analysis, which assumed worst-case stratification through the entire surge line, and which calculated transients based on intermittent pressurizer spray. Similar to the CUF of the surge line elbow, the CUF of the pressurizer is expected to reduce considerably, if the moderation of the transients of the Palisades continuous pressurizer spray is used.

As mentioned above, the number of design cycles for surge line stratification flow developed by the CE Owners Group for each of the above thermal stratification transients correspond to 500 cycles of plant heatup and cooldown.

The Fatigue Monitoring Program will ensure that the effects of aging will be adequately managed for the period of extended operation by assuring that a reanalysis or other appropriate corrective action is taken if a design basis primary coolant system cycle count limit is reached at any time during the extended licensed operating period.

i. Revised Fatigue Analysis of Nozzles from PCS Cold Legs 1B and 2A to Pressurizer Spray and of the Pressurizer Spray Nozzle

Summary Description

Pressurizer spray is normally supplied by reactor coolant pump head through 3-inch nozzles on two of the four 30-inch Primary Coolant System (PCS) cold legs. Normal spray flow in each of these 3-inch lines is continuous, through a normally throttled 3/4-inch main spray bypass valve, and through a 3-inch main spray control valve. The charging line downstream of the regenerative heat exchanger supplies auxiliary spray from the chemical and volume control system through a 2-inch control valve. All three of these sources supply a single pressurizer spray nozzle.

The original design of the pressurizer included a fatigue analysis of the pressurizer spray nozzle. However, the normal spray piping and the auxiliary spray piping were designed to the B31.1 Code. Revised operating conditions and pressurizer cooldown rate prompted addition of a fatigue analysis for the two cold leg nozzles and for the auxiliary spray piping.

Analysis of Other Nozzles

<u>Cold Leg Nozzles to Pressurizer Spray</u>: The analysis of the auxiliary spray-reverse flow events is based on the design basis number of thermal cycles assumed for 40 years, and the calculated cumulative usage factor is 0.66.

<u>Pressurizer Spray Nozzle</u>: The revised pressurizer spray nozzle analysis determined that the calculated Cumulative Usage Factor (CUF) is 0.8214 for the design basis number of high-differential-temperature spray events and 200º F per hour cooldowns assumed for 40 years, and for all other applicable transients.

The revised analysis also estimated that the maximum CUF in the spray nozzle to that date (October, 1991) was 0.353, and that accumulation at then-current trends indicated a 40-year lifetime CUF of about 0.435. On that basis a projection to the end of a 60-year extended licensed operating period would indicate a CUF of about 0.517.

Disposition Cold Leg to Pressurizer Spray Nozzles: Validation,
10 CFR 54.21(c)(1)(i)

The number of thermal cycles for each of the design basis temperature differential ranges will not exceed the design basis limit during the extended licensed operating period. The calculated 40-year plant life usage factor of 0.66 is well below the analytical limit of 1.0. Thus the usage factor remains less than 1 at the end of the extended licensed operating period.

Disposition Pressurizer Spray Nozzles: Aging Management,
10 CFR 54.21(c)(1)(iii)

The design basis Cumulative Usage Factor (CUF) of the nozzle is 0.8214 for the original 40-year licensed operating period. The projected number of cycles of the design basis events does not exceed the design basis limit during the extended licensed operating period. Therefore, the actual fatigue usage factor is not expected to approach the calculated value at the end of the extended licensed operating period.

The Fatigue Monitoring Program will ensure that the effects of aging will be adequately managed for the period of extended operation by assuring that a reanalysis or other appropriate corrective action is taken if a design basis cycle count limit is reached at any time during the extended licensed operating period.

j. Pressurizer Auxiliary Spray Line Tee Fatigue Analysis in Response to NRC Bulletin 88-08

NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Cooling Systems," and supplements describe observed effects of thermal cycling and thermal stratification in reactor coolant system pressure boundary components due to thermally-driven cyclic inleakage at isolation valves and similar phenomena. In 1989 a conservative, bounding analysis of the section of the Palisades auxiliary spray line from check valve CK 2118 to the pressurizer spray line tee demonstrated that fatigue due to these effects would be acceptable for the then-remaining 30-year licensed operating life. The piping material is A 376 Type 316. The auxiliary spray line connects to the normal spray line vertically and below, so that the cooler auxiliary spray water does not cause a thermal stratification effect. The analysis assumed 500 lifetime Operating Basis Earthquake (OBE) cycles and 500 lifetime full-range thermal expansion (heatup and cooldown) cycles. It modeled the Bulletin 88-08 phenomena (due to inleakage through the check valve) as a thermal cycle every two minutes, between 536 and 400 ºF, or 1.84×10^5 per year at a 70 percent availability factor.

The 500 lifetime OBE cycles are assumed to occur only once in a design lifetime, independent of its length. The analysis also conservatively attributed the entire 500 full-range thermal expansion to the remaining 30-year life. These OBE and full-range thermal cycles contributed a negligible 0.0063 usage factor. The assumed Bulletin 88-08 cycles contributed 0.4245 in 30 years, for an end-of-life Cumulative Usage Factor (CUF) of 0.43.

Analysis

License renewal will add an additional 20 years to the design life assumed by the analysis. Increasing the lifetime contribution of the OBE cycles and full-range thermal expansion cycles by 50% (a conservative assumption for OBE, and given the plant history, also for the full-range thermal cycles) results in a contribution of only 0.0095. Increasing the assumed remaining design life to 50 years results in a contribution from the Bulletin 88-08 cycles of 0.7075, for a 60-year CUF of 0.717, well within the allowable of 1.0.

The analysis assumed 70 percent plant availability. Recent experience has been about 90 percent, which would increase the contribution of the Bulletin 88-08 cycles to about 0.91, or a CUF of about 0.92.

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

The assumptions and methods of the analysis are otherwise very conservative, and this result is still within the allowable of 1.0.

k. Assumed Thermal Cycle Count for Allowable Secondary Stress Range Reduction

Factor in Piping and Components

This section addresses the issue of assumed thermal cycle counts which determine an allowable secondary stress range reduction factor for some Consumers Power Design Class 1 piping and components, and for non-Consumers Power Design Class 1 piping and components.

Only the primary coolant system piping and components have an ASME Class 1 fatigue analysis. Other Consumers Power Design Class 1 piping and components were designed to the ANSI B31.1 Power Piping Code, ASME Section VIII, or ASME III, Class 2 and 3, which requires a stress range reduction factor to the allowable stress range for secondary (expansion and displacement) stresses to account for thermal cycling. For ANSI B31.1 the allowable secondary stress range is 1.0 S_A for 7,000 equivalent full-range thermal cycles or less. The allowable secondary stress range is reduced to 0.5 S_A for thermal cycles greater than 100,000. Components designed to other codes, such as ASME VIII, have identical or very similar provisions. An increase in design life could increase the number full-range thermal cycles; therefore, design analyses under these codes are TLAAs.

Some piping within the scope of license renewal was originally designed and built to the American Standard (ASA) Code for Pressure Piping, Section 1, "Power Piping Systems," 1955 edition. However, during the implementation of IE Bulletin 79-14 work, CP Co Design Class 1 piping, except the main primary coolant piping, was designed to the USAS B31.1.0 (1967) Power Piping Code. In 1992, as a result of discussions between CP Co and the NRC, for new and existing CP Co Design Class 1 piping (except the main primary coolant piping), the code of record was changed to ANSI B31.1 (1973) Power Piping Code with the Summer (1973) Addenda (FSAR 5.10.1.1).

With regard to the stress range reduction factors and corresponding thermal cycle count assumptions, the Consumers Power Class 2 and 3 piping systems and components are designed to the same requirements as CP Co Design Class 1 piping.

The review of possible TLAAs found no Palisades piping and components design analyses to the B31.1 rules, which invoke lower stress range reduction factors for an increase in the equivalent full-range thermal and displacement cycles.

Analysis

The number of lifetime (full-range and equivalent) thermal and other displacement cycles applicable to most of the Palisades B31.1 piping and components are expected to be similar to the plant design basis events. Therefore, so long as the assumed number of the plant design basis event cycles is not exceeded, the secondary stress range reduction factors assumed for these B31.1 piping and components, and similar code designs remain valid. Results of the TLAA fatigue review for B31.1 piping and similar code designs for mechanical systems within the scope of license renewal and with operating temperature in excess of 220º F for carbon steel or 270º F for austenitic stainless steel, revealed only two piping systems that have additional cycles that exceed the 7000-cycle limit of the B31.1 Code. These systems include the charging lines inboard of the regenerative heat exchanger, which experience an increase in partial-range thermal cycles due to cycling of the fixed-speed pumps. The original 6,000 events increased to 18,000 for 40-year and 27,000 for 60-year life. The effects of additional cycles have been evaluated for the regenerative heat exchanger fatigue and for the charging inlet nozzle. The calculation provides reasonable assurance that the charging nozzle is the limiting location in the chemical and volume control system. The other system is the PCS hot leg sampling piping, which may exceed 7,000 cycles during the period of extended operation. A calculation was performed on the PCS hot leg sample line demonstrating that, with 14,000 design cycles, stresses are below the ANSI B31.1 code allowables.

Disposition for the Charging Lines Inboard of the Regenerative Heat Exchanger: Revision, 10 CFR 54.21(c)(1)(ii), and Aging Management, 10 CFR 54.21(c)(1)(iii).

The effect of the increase in variable speed charging pump out-of-service events on the charging lines inboard of the regenerative heat exchanger has been evaluated (Reference 96). The evaluation concluded that the lines will continue to meet licensing bases design criteria throughout the period of extended operation.

The Fatigue Monitoring Program will ensure that the effects of aging will be adequately managed for the period of extended operation by assuring that a reanalysis or other appropriate corrective action is taken if the variable speed charging pump out-of-service event count limit is reached at any time during the extended licensed operating period.

Disposition for Other Piping and Components: Validation, 10 CFR 54.21(c)(1)(i), and Aging Management, 10 CFR 54.21(c)(1)(iii)

The number of lifetime (full-range and equivalent) thermal and other displacement cycles applicable to most of the Palisades B31.1 piping and components are expected to be similar to the plant design basis events. The number of each of these events is not expected to exceed the existing 40-year design basis for the 60-year extended licensed operating period.

The Fatigue Monitoring Program will ensure reanalysis or other appropriate corrective action in the unlikely event that a design basis cycle count limit is reached at any time during the extended licensed operating period.

I. Effects of Reactor Coolant System Environment on Fatigue Life of Piping and Components (Generic Safety Issue 190)

The effects of the reactor coolant environment may need to be included in the calculated fatigue life of components. GSI-190, "Fatigue Evaluation of Metal Components for 60-Year Plant Life," addressed this issue. Although the parent GSI-190 safety issue has been resolved, NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," Section 4.3.1.2, states that “The applicant's consideration of the effects of coolant environment on component fatigue life for license renewal is an area of review.” The GSI-190 review requirements are therefore imposed by the Standard Review Plan and do not depend on the individual plant licensing basis.

Analysis

NUREG/CR 6260, Table 5-43, identifies seven sample locations for older Combustion Engineering plants:

- Reactor Vessel (Lower Head to Shell Transition)
- Primary Coolant Inlet Nozzle
- Primary Coolant Outlet Nozzle
- Surge Line Elbow
- Charging Nozzles
- Safety Injection Nozzles
- Shutdown Cooling Line Inlet transition

Of these:

- The Palisades safety injection and shutdown cooling nozzles are common. The two most limiting locations on the common nozzle were selected to represent the safety injection nozzles and the shutdown cooling inlet transition.

- NUREG/CR-6260 evaluated a long-radius elbow in the surge line because, in the sample plant, this was the highest usage factor location in the surge line and nozzles subject to NRC Bulletin 88-11 reanalysis for thermal stratification cycles. In the NUREG/CR-6260 sample plant this component is SA-376 Type 316; at Palisades the surge line material is a similar Type 316. The only Palisades location on this line with a plant-specific fatigue analysis is the hot leg nozzle to the surge line. The analysis for the surge line elbow is for a typical C-E PWR with intermittent pressurizer spray, with thermal stratification transients; and the pressurizer surge nozzle analysis uses pipe loads from the same source. The results are therefore more conservative than would be expected for Palisades, which has a continuously-modulated pressurizer spray and less-severe thermal transients.

- The Palisades charging nozzles are SB 166 Ni-Cr-Fe Alloy 600 instead of the austenitic stainless of the NUREG/CR-6260 sample plant.

Of the seven NUREG/CR-6260, Section 5.2, sample locations for an older Combustion Engineering plant, six are therefore applicable to Palisades. See FSAR Table 1-8, "Summary of Fatigue Usage Factors at NUREG/CR-6260 Sample Locations Applicable to Palisades."

Environmental effects on cracking in the charging and other Alloy 600 nozzles are also addressed in FSAR Section 1.9.2.5.b, "Alloy 600 Nozzle and Safe End Life Assessment Analyses," fatigue in the charging nozzles in FSAR Section 1.9.2.2.g, "ASME III Class A Primary Coolant Piping Fatigue Analyses," and fatigue in the surge nozzles in FSAR Section 1.9.2.2.h, "Revised Bulletin 88-11 Fatigue Analysis of the Hot-Leg-to-Pressurizer-Surge-Line."

All of the primary coolant system at Palisades is stainless steel, Alloy 600, or carbon steel with stainless or Alloy 600 clad. Fatigue in clad components is evaluated using base material properties only; that is, as if the coolant is in contact with the base material, consistent with NUREG/CR-6260.

Disposition: Revision, 10 CFR 54.21(c)(1)(ii); and Aging Management, 10 CFR 54.21(c)(1)(iii)

A plant-specific calculation was performed for the sample locations identified in NUREG/CR-6260 for older-vintage Combustion Engineering plants. Detailed environmental fatigue calculations use the appropriate F_{en} relationships from NUREG/CR-6583 for carbon and low-alloy steels and from NUREG/CR-5704 for stainless steels, as appropriate for the material at each location. The F_{en} for Alloy 600 material comes from Chopra, Omesh K, "Status of Fatigue Issues at Argonne National Laboratory," presented at EPRI Conference on Operating Nuclear Power Plant Fatigue Issues & Resolutions, August 22-23, 1996.

The calculation determines an appropriate $F_{en(i)}$ for each individual load pair in the governing fatigue calculation, so that an overall F_{en} multiplier on cumulative usage factor (CUF) for environmental effects can be determined for each location. The analysis shows that the fatigue usage factors at all NUREG/CR-6260 sample locations, including the effects of the reactor coolant environment, will remain less than 1.0 for the extended operation period.

1.9.2.3 Environmental Qualification of Electrical Equipment

Under 10 CFR 54.21(c)(1)(iii), a Plant EEQ Program which implements the requirements of 10 CFR 50.49 is viewed as an Aging Management Program (AMP) for license renewal. Re-analysis of an aging evaluation to extend the qualification of components under 10 CFR 50.49(f) is performed on a routine basis as part of the EEQ Program. Important attributes for the reanalysis of an aging evaluation include analytical methods, data collection and reduction methods, the underlying assumptions, the acceptance criteria, and corrective actions (if acceptance criteria are not met).

Analytical Methods: The analytical models used in the reanalysis of an aging evaluation are the same as those previously applied during the prior evaluation. The analytical method used for a radiation aging evaluation is to demonstrate qualification for the total integrated dose (that is, normal radiation dose for the projected installed life plus accident radiation dose). For license renewal, one acceptable method of establishing the 60-year normal radiation dose is to multiply the 40-year normal radiation dose by 1.5 (that is, 60 years/40 years). The result is added to the accident radiation dose to obtain the total integrated dose for the component. For cyclical aging, a similar approach may be used. Other models may be justified on a case-by-case basis.

Data Collection and Reduction Methods: Reducing excess conservatism in the component service conditions (for example, temperature, radiation, and cycles) used in the prior aging evaluation is a typical method used for a reanalysis. Plant temperature data can be obtained in several ways, including monitors used for technical specification compliance, other installed monitors, measurements made by plant operators during rounds, and temperature sensors on large motors (while the motor is not running). When used, a representative number of temperature measurements are conservatively evaluated to establish the temperatures used in an aging evaluation. Plant temperature data may be used in an aging evaluation in different ways, such as (a) directly applying the plant temperature data in the evaluation, or (b) using the plant temperature data to demonstrate conservatism when using plant design temperatures for an evaluation. Any changes to material activation energy values as part of a reanalysis are justified on a case-specific basis. Similar methods of reducing excess conservatism in the component service conditions used in prior aging evaluations may be used for radiation and cyclical aging.

Underlying Assumptions: EQ component aging evaluations contain sufficient conservatism to account for most environmental changes occurring due to plant modifications and events. When unexpected adverse conditions are identified during operational or maintenance activities that affect the normal operating environment of a qualified component, the affected EQ component is evaluated and appropriate corrective actions are taken, which may include changes to the qualification bases and conclusions.

Acceptance Criteria and Corrective Actions: The reanalysis of an aging evaluation could extend the qualification of the component. If the qualification cannot be extended by reanalysis, the component is maintained, replaced, or re-qualified prior to exceeding the period for which the current qualification remains valid.

Regulatory Issue Summary (RIS) 2003-09

On May 2, 2003, the staff issued NRC Regulatory Issue Summary (RIS) 2003-09, “Environmental Qualification of Low-Voltage Instrumentation and Control Cables,” providing the results of the staff's technical assessment of GSI-168, following completion of the NRC-sponsored cable test research. The staff's technical assessment of GSI-168 in RIS-2003-09 is stated as follows:

> For license renewal, a re-analysis (based on the Arrhenius methodology) to extend the life of the cables by using the available margin based on a knowledge of the actual operating environment compared to the qualification environment, coupled with observations of the condition of the cables during walk-downs, was found to be an acceptable approach. Monitoring I&C cable condition could provide the basis for extending cable life.

The Palisades Plant Electrical Equipment Qualification Program allows re-analysis for maintaining qualification using the methods described above. In addition, the EEQ Program has procedural requirements in place to monitor and track aging effects of EQ equipment including cables. The requirements are listed below:

- Monitoring equipment condition and equipment performance
- Monitoring environmental conditions of plant areas
- Incorporating the results of testing and analysis into the plant maintenance and surveillance program

Disposition

The EEQ Program will continue to be implemented for the extended operating period in accordance with 10 CFR 50.49. Continuing the existing EEQ Program provides reasonable assurance that the aging effects will be managed and that the EQ components will continue to perform their intended functions for the period of extended operation. Therefore, this TLAA is dispositioned under 10 CFR 54.21(c)(1)(iii), in that continuation of the existing EEQ Program will adequately manage aging of affected components for the period of extended operation.

1.9.2.4 Containment Liner Plate, Metal Containments and Penetrations Fatigue Analysis

a. Concrete Containment Tendon Prestress Analysis

The original design included a calculation of expected loss of prestress for the plant design life In accordance with ACI 318-63. The calculation evaluated loss of prestress due to friction and initial seating loss, tendon relaxation, concrete elasticity, concrete shrinkage, and concrete creep. FSAR Section 5.8.5.3.1 lists the predicted values for remaining prestress at the end of the 40-year design life. This original analysis was conservative, as demonstrated by a regression analysis of tendon surveillance data from the twentieth and twenty-fifth-year tendon surveillances. This regression analysis indicated that the effective dome, hoop, and vertical tendon forces would remain significantly higher than values predicted by the original relaxation estimates beyond the 40-year licensed operating period. Periodic surveillances of containment tendons for degradation are required by 10 CFR 50.55a and Palisades Technical Specification 5.5.5. See FSAR Section 5.8.8 for additional information on the existing surveillance program requirements and results.

Disposition

This issue is dispositioned under 10 CFR 54.21(c)(1)(iii), the effects of aging on the intended function(s) will be adequately managed for the extended operating period, in that the existing Palisades tendon surveillance program activities will be continued in accordance with 10 CFR 50.55a.

b. Containment Liner Plate Load and Penetrations Load Cycles

The containment liner plate and penetrations were conservatively designed, in part, to the rules of the ASME Boiler and Pressure Vessel Code, Section III-1965. This code edition classifies containment as a Class B vessel. A fatigue analysis is required under this code edition only for Class A vessels (reactor coolant pressure boundary, etc.). However the Palisades containment liner and penetration designs use some of the methods and data from Section III, Article 4, for design of Class A vessels for fatigue loads.

The Palisades containment design relies on the liner only to maintain a leak-tight containment. There are no design conditions under which the liner plate is relied upon to assist the concrete in maintaining the integrity of the structure. Containment penetrations are designed to maintain the leak tightness of the containment structure under normal and accident conditions.

Analysis

The Palisades containment design relies on the liner to maintain a leak-tight containment. However, there are no design conditions under which the liner plate is relied upon to assist the concrete in maintaining the integrity of the structure. Forces are transmitted between the liner plate and the concrete through the anchorage system and through direct contact (pressure). At times, forces may also be transmitted by bond and/or friction. These forces cause, or are caused by, liner plate strains. The liner plate is designed to withstand the predicted strains. The effect of concrete cracking on the liner plate has also been considered.

The allowable liner plate strains/stress was conservatively based on the ASME B&PV Code, Section III, Article 4, 1965. Specifically, the following sections were adopted as guides in establishing allowable strain limits:

1. Paragraph N-412(m) - Thermal Stress, Subparagraph 2

2. Paragraph N-412(n) - Operational Cycle

3. Paragraph N-414.5, Table N-413, Figures N-414 and N-415(a) - Peak Stress Intensity

4. Paragraph N-415.1 - Vessels Not Requiring Analysis for Cyclic Operation

The liner strains/stresses due the (non-DBA) loads are relatively small such that the number of environmental and operational load cycles is insignificant compared to the allowable number of cycles on the fatigue curve in code Fig. N-415 (a), or compared to 3 times the S_m value of code of Table N-421 at the operational temperature. The results of the analysis confirm that the design of the containment liner complies with the provisions of code paragraph N-415.1 for not requiring a fatigue analysis for design load cycles.

Disposition: Validation, 10 CFR 54.21(c)(1)(i)

Of the design basis fatigue load cycles of the containment liner, only the number of environmental and operational load cycles would increase due to the 60-year extended licensed operating period.

Of these two events, the effect of the assumed summer-winter annual cycles is negligible, and will remain negligible on increase from 40 to 60 cycles for the 60-year extended licensed operating period.

The assumed 500 containment interior operational heatup and cooldown cycles is very conservative, since it corresponds to an average of 8 1/3 cycles per year, or a PCS cooldown and heatup every 6 weeks. This is more than adequate to accommodate the 60-year extended licensed operating period.

Therefore, there will be negligible change in the fatigue resistance of the containment liner for the 60-year extended licensed operating period.

In addition, periodic inservice LLRTs required by Plant Technical Specifications monitor the continued inservice leaktight functionality of each individual penetration through the extended licensed operating period.

1.9.2.5 Other Plant-Specific Time-Limited Aging Analyses

a. Fuel Handling Crane Load Cycles

A crane evaluation to the Crane Manufacturers Association of America Standard CMAA-70 assumes a number of rated lifts in the design lifetime in order to establish the design Service Level, and hence the allowable stresses. At Palisades, two cranes have been reanalyzed to CMAA-70 design criteria. The NUREG-0612 heavy loads evaluation of the reactor building polar crane was performed to CMAA 70. A redesign of the Spent Fuel Pool Crane for dry fuel storage also included a NUREG-0612 evaluation to CMAA-70 design criteria. The limiting components of the containment polar crane (135 tons) and the redesigned spent fuel pool crane (110 tons) are now rated for CMAA-70 “Service Level A - Standby or Infrequent Service.” Service Level A cranes are designed to stress limits which assume either 20,000 to 100,000, or 200,000, rated lifts in a design lifetime.

Analysis

Containment Polar Crane: The polar crane was originally designed to Electric Overhead Crane Institute Specification 61. The subsequent NUREG-0612 heavy loads evaluation of the polar crane was performed to CMAA 70 (1975). The minimally-rated components are CMAA 70 Service Level A. Since the minimally-rated components are Service Level A, the effective crane design life for fatigue or allowed number of rated lifts depends on this classification, which assumes 20,000 to 100,000 rated lifts in a design lifetime.

Separate evaluations have been performed of polar crane planned engineered lifts (over the rated capacity). The evaluations were done to ANSI/ASME Standard B30.2 (1996). Lifts have been evaluated and approved up to 140 T, less than 4 percent over the 135 T rating.

Spent Fuel Pool Crane: The redesign to 110 T for dry cask storage included a NUREG-0612 evaluation to CMAA-70 Service Level A design criteria, and other considerations; and replacement of the trolley with a 110 T single-failure-proof trolley meeting NUREG-0554 guidelines. The structural redesign and evaluation considered the load combinations of ASME NOG 1, Section NOG 4140, “Load Combinations.”

Disposition: 10 CFR 54.21(c)(1)(i)

Polar Crane (L-1): Polar crane rated or near-rated lifts are limited to the reactor head plus CRDMs, insulation, and shielding, in accordance with site procedures. Only a few rated lifts are performed each refueling outage, and none during operation. Therefore this machine cannot realistically approach the 20,000 to 100,000 rated lifts, assumed for components evaluated to CMAA 70 (1975) Service Level A, during a 60-year licensed operating period.

Spent Fuel Pool Crane: Approximately 11 dry cask storage campaigns are expected between rerating and the end of the 60-year extended license. This will require loading about 64 casks. Each will require about two lifts of 100 T or more per cask, and some additional lifts of between 50 and 100 T. The total for 64 casks and 11 campaigns is about 140 lifts of 100 T or more, and about 162 lifts between 50 and 100 T. Other lifts, and lifts prior to rerating the crane, were determined to be inconsequential. Therefore, this machine can not realistically approach the 20,000 to 100,000 rated lifts assumed for its design evaluation during the 60-year extended licensed operating period.

b. Alloy 600 Nozzle and Safe End Life Assessment Analyses

Alloy 600 (Ni-Cr-Fe alloy) was used to clad the pressurizer lower head and surge nozzle, for the pressurizer heater sleeves, and for smaller nozzles and safe ends and flanges on larger nozzles of the Palisades reactor vessel head, primary coolant system loop piping, and pressurizer. There are 250 Alloy 600 heater sleeves, nozzles, safe ends, and flanges in the Palisades primary coolant system.

Analysis

The inspection methods and intervals of the Palisades Alloy 600 aging management program were determined from evaluations of the susceptibility of all 250 remaining heater sleeves, nozzles, safe ends, and flanges to primary water stress corrosion cracking (PWSCC).

Disposition for Fatigue Analyses of the Weld Pad Repairs Installed in 1993 for the Pressurizer Temperature Element Nozzles: 10 CFR 54.21(c)(1)(iii)

Entergy will monitor the cumulative number of pressurizer temperature element nozzle fatigue cycles within the Fatigue Monitoring Program, and maintain a special action level to ensure that appropriate actions are taken if at any time the cycle count for any design basis event since 1993 reaches the number assumed by these analyses.

For this purpose the fatigue management program will compare cycle counts since the repair in 1993 to appropriate action levels. Since the fatigue analyses were based on half of the 40-year pressurizer design basis event cycles, the action levels for cycles since then will be about half of the 40-year pressurizer design basis event cycles for each event.

Disposition for Corrosion Life Assessment of the TE 0101 Temperature Element Nozzle Bore in the Carbon Steel Pressurizer Wall:
10 CFR 54.21(c)(1)(i)

The evaluation estimated a repair lifetime of 52.3 years for this effect following initiation of leakage. Leakage was first detected in 1993, after 22 years of operation, which indicates that the pressurizer wall can withstand this effect for a total plant life of over 70 years. Therefore, the current analyses remain valid for the period of extended operation.

Disposition for Cycle-Dependent Aspects of the Bounding Fracture Mechanics Analysis of the Hot Leg, Piping RTD and Sampling Nozzles, Pressurizer Instrument Nozzles, and Pressurizer Heater Sleeves; and Disposition for Fatigue Portions of All Other Alloy 600 Fracture Mechanics Analyses for a 40-Year Design Life: 10 CFR 54.21(c)(1)(i) and
10 CFR 54.21(c)(1)(iii)

The Palisades plant-specific bounding fracture mechanics analysis demonstrates the validity of the cycle-dependent aspects of the generic bounding fracture mechanics analysis (WCAP-15973-P) by demonstrating that the plant-specific load and thermal events are within those assumed by the generic bounding analysis. The basis for the safety determination of the fracture mechanics evaluation calculation will therefore remain valid so long as the numbers of these events do not exceed the design basis values.

The fatigue cycle count program described in the Palisades Nuclear Plant Application for Renewed Operating Licenses, Appendix B, "Aging Management Programs," Fatigue Monitoring Program, will ensure a reanalysis or other appropriate corrective action if a design basis primary coolant system cycle count limit is reached at any time during the extended licensed operating period.

Disposition for All Alloy 600 Heater Sleeves, Nozzles, Safe Ends, and Flanges: 10 CFR 54.21(c)(1)(iii)

The Palisades Alloy 600 Program identifies the Alloy 600 components in the primary coolant system, ranks them according to PWSCC susceptibility, and establishes a program for inspection, repairs, and mitigation. All 250 remaining Alloy 600 heater sleeves, nozzles, safe ends, and flanges are subject to the inspection program. At all 250 locations the program requires at least an insulated VT-2 visual inspection for leakage every refueling outage. Locations which are more susceptible to PWSCC, or whose failure could result in a more-significant safety hazard, are also subject to initial or periodic bare-metal VT-2, volumetric, or penetrant inspections.

c. Primary Coolant Pump Flywheel Fatigue or Crack Growth Analysis

A primary coolant pump flywheel could theoretically burst because of centrifugal stresses, which could produce missiles inside containment, and could also damage pump seals or other pressure boundary components. This concern is the subject of Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity." The flywheels may therefore be subject to crack growth or fatigue.

Early technical specifications required periodic, relatively frequent, inspections of primary coolant pump motor flywheels. To justify a longer inspection frequency, the Combustion Engineering Owners Group prepared report SIR-94-080. This report used a crack growth analysis of Palisades' primary coolant pump flywheels to establish acceptable limits for the flywheel inspection interval. The evaluation determined that the primary coolant pump would be subject to approximately 500 startup/shutdown cycles, and the crack growth fatigue analysis assumed 4000 cycles. It was concluded that a ten-year inspection interval was acceptable since an assumed preservice flaw would not grow to a critical flaw size during the period between inspections. (In fact, the report justified that the assumed preservice flaw would not grow to critical flaw size during the entire licensed operating period, or extended licensed operating period.) The NRC approved the change to the inservice inspection requirements to extend the flywheel examination frequency to once each ten years.

Analysis

The primary coolant pump is assumed to experience approximately 500 startup/shutdown cycles, and the crack growth fatigue analysis assumed 4000 cycles. The expected number of startup/shutdown cycles for the 60-year extended licensed operating period is substantially less than the 500 assumed cycles. The number of primary coolant pump starts cannot realistically approach 4000 during a 60-year licensed operating period.

Disposition: 10 CFR 54.21(c)(1)(i)

The analysis remains valid through the period of extended operation.

d. Reactor Vessel Underclad Cracking

The issue of underclad cracking in certain reactor vessels has been identified since 1970 when it was first discovered at a European vessel fabricator. Underclad cracking has occurred in the low alloy steel base metal Heat-Affected Zone (HAZ) beneath the austenitic stainless steel weld overlay that is deposited to protect the ferritic material from corrosion. Two types of underclad cracking have been identified. Reheat cracking has occurred as a result of postweld heat treatment of austenitic stainless steel cladding applied using high-heat-input welding processes on ASME SA-508, Class 2 forgings. Cold cracking has occurred in ASME SA-508, Class 3 forgings after deposition of the second and third layers of cladding, when no pre-heating or post-heating was applied during the cladding procedure. The cold cracking was determined to be attributable to residual stresses near the yield strength in the weld metal/base metal interface after cladding deposition, combined with a crack-sensitive microstructure in the HAZ, and high levels of diffusible hydrogen in the austenitic stainless steel or Inconel weld metals. The hydrogen diffused into the HAZ and caused cold (hydrogen-induced) cracking as the HAZ cooled.

Westinghouse report WCAP-15338-A, "A Review of Cracking Associated With Weld Deposited Cladding in Operating PWR Plants," summarizes the original disposition of the issue as follows,

> "In 1971 Westinghouse submitted an assessment of the underclad reheat cracking issue to the regulatory authorities, then the Atomic Energy Commission, evaluating underclad cracks for an operating period of 40 years. The commission reviewed the assessment, and issued the following conclusion:
>
> SUMMARY OF REGULATORY POSITION:
>
> 'We concur with Westinghouse's finding that the integrity of a vessel having flaws such as described in the subject report would not be compromised during the life of the plant. This report is acceptable and may be referenced in future applications where similar underclad grain

boundary separations have been detected. However, such flaws should be avoided, and we recommend that future applicants state in their PSARs what steps they plan to take in this regard' ".

WCAP-15338-A notes that the 1983 inservice inspection identified two small clusters of underclad indications, classified as reheat cracks, that were determined to be within the allowable limits of ASME B&PV Code, Section XI, IWB-3500. When these locations were again inspected in 1995 there was no evidence that the indications had expanded in number or size.

Analysis

A generic fracture mechanics evaluation of Westinghouse plants initially demonstrated that the growth of underclad cracks during a 40-year plant life was insignificant. The evaluation was extended to 60 years, using fracture mechanics analysis based on a representative set of design transients with the occurrences extrapolated to cover 60 years of service life. The 60-year evaluation (WCAP-15338-A, "A Review of Cracking Associated with Weld Deposited Cladding in Operating Pressurized Water Reactor [PWR] Plants") showed insignificant growth of the underclad cracks, and concluded that the cracks were of no concern relative to structural integrity of the reactor vessel. The NRC reviewed and approved the evaluation (WCAP-15338-A) for application to all Westinghouse RPVs.

Palisades is not a Westinghouse plant that was specifically addressed in WCAP-15338-A. However, the Palisades reactor vessel was fabricated using similar processes and materials as those used in reactor vessels fabricated by Combustion Engineering for Westinghouse. Therefore, a Palisades-specific evaluation has been performed and documented in WCAP-16605-NP, "A Review of Cracking Associated with Weld Deposited Cladding at Palisades Nuclear Plant." This evaluation demonstrates that any potential growth of underclad cracks during a 60-year plant life would be insignificant, and concludes that underclad cracks are of no concern relative to structural integrity of the reactor vessel. This is the same conclusion reached previously in WCAP-15338-A and accepted by the NRC.

Disposition: 10 CFR 54.21(c)(1)(ii)

Reactor vessel underclad cracking is dispositioned under 10 CFR 54.21(c)(1)(ii), the analysis has been projected to the end of the period of extended operation.

1.9.3 NEWLY IDENTIFIED STRUCTURES, SYSTEMS, AND COMPONENTS

After a renewed license is issued, 10 CFR 54.37(b) requires that FSAR updates include any Systems, Structures, or Components (SCCs) newly identified that would have been subject to an aging management review or evaluation of time-related aging analyses in accordance with 10 CFR 54.21. The FSAR update must describe how the effects of aging will be managed such that the intended functions in 10 CFR 54.4(b) will be effectively maintained during the period of extended operation.

In accordance with 10 CFR 54.37(b), Table 1-10 provides a listing of newly identified SSCs and describes how the effects of aging will be managed such that intended functions will be maintained during the period of extended operation.