15.0 SUPPLEMENT FOR RENEWED OPERATING LICENSES

I&M prepared a license renewal application (Application) for Donald C. Cook Nuclear Plant, Units 1 and 2 (Reference 15.3 #1). The Application, including information provided in supplemental correspondence, provided sufficient basis for the NRC to make the findings required by 10 CFR 54.29 (Final Safety Evaluation Report) (Reference 15.3 #2). As required by 10 CFR 54.21(d), this UFSAR supplement contains a summary description of the programs and activities for managing the effects of aging (Section 15.1) and the evaluation of time-limited aging analyses for the period of extended operation (Section 15.2). The period of extended operation is 20 years after the expiration dates of the original operating licenses. With the period of extended operations, the term of each plant operating license is 60 years.

15.1 AGING MANAGEMENT PROGRAMS AND ACTIVITIES

The integrated plant assessment and the time-limited aging analyses for license renewal identified existing and new aging management programs necessary to provide reasonable assurance that structures and components within the scope of license renewal will continue to perform their intended functions consistent with the current licensing basis (CLB) for the period of extended operation. This section describes the aging management programs and activities that are required during the period of extended operation.

The CNP Quality Assurance Program Description implements the requirements of 10 CFR 50, Appendix B, and is consistent with the summary in Section A.2 of NUREG-1800, Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants, published July 2001. The Quality Assurance Program Description includes the elements of corrective action, confirmation process, and administrative controls, and is applicable to the safety-related and non-safety-related structures, systems, and components that are within the scope of license renewal and subject to aging management review.

15.1.1 Alloy 600 Aging Management Program

This program manages aging effects of Alloy 600/690 components and Alloys 52/152 and 82/182 welds in the reactor coolant system that are not addressed by the following aging management programs:

- The Control Rod Drive Mechanism and Other Vessel Head Penetration Inspection Program, Section 15.1.9;
- The Steam Generator Integrity Program, Section 15.1.34; and
- The Reactor Vessel Internals Programs, Sections 15.1.30 and 15.1.31.

The Alloy 600 Aging Management Program detects cracking from primary water stress corrosion cracking (PWSCC) using the examination and inspection requirements specified in ASME Section XI. Guidance developed by the EPRI Material Reliability Program and the owners group was used to identify susceptibility rankings and program inspection requirements regarding Alloy 82/182 pipe butt welds. The Alloy 600 Aging Management Program inspection plan was submitted for NRC staff review prior to the period of extended operation and it was determined that the program demonstrates the ability to manage the effects of aging per 10 CFR 54.21(a)(3).

15.1.2 Bolting and Torquing Activities Program

The Bolting and Torquing Activities Program manages the loss of mechanical closure integrity for bolted connections and bolted closures in high temperature systems and in applications subject to significant vibration. This program relies on industry recommendations as delineated in EPRI guidelines for a comprehensive bolting integrity program.

15.1.3 Boral Surveillance Program

The Boral Surveillance Program monitors changes in neutron attenuation, dimensional measurements, and weight and specific gravity of representative coupon samples. The Boral coupon samples are located in the spent fuel pool, surrounded by freshly discharged fuel assemblies, to monitor performance of the absorber material without disrupting the integrity of the storage system. Coupons are removed on a prescribed schedule and their properties are measured. From this data, the stability and integrity of Boral in the storage cells are assessed.

15.1.4 Boric Acid Corrosion Prevention Program

The Boric Acid Corrosion Prevention Program relies on implementation of recommendations of NRC Generic Letter 88-05 to monitor the condition of ferritic steel and electrical components on which borated reactor water may leak. The program detects boric acid leakage by periodic visual inspection of systems containing borated water for deposits of boric acid crystals and the presence of moisture; and by inspection of adjacent structures, components, and supports for evidence of leakage.

This program manages loss of material and loss of mechanical closure integrity, and loss of circuit continuity, as applicable. This program required enhancements that were implemented prior to the period of extended operation.

15.1.5 Bottom-Mounted Instrumentation Thimble Tube Inspection Program

The Bottom-Mounted Instrumentation Thimble Tube Inspection Program detects loss of material due to wear in the bottom-mounted instrumentation (BMI) thimble tubes prior to leakage. The thimble tubes are part of the reactor coolant pressure boundary. The program monitors tube wall degradation of the BMI thimble tubes using eddy current testing. The replacement, repositioning, or isolation of the BMI tubes is based on analysis of the data obtained, using wear rate relationships. The inspection frequency is based on measured data and projected wear results.

15.1.6 Buried Piping Inspection Program

The Buried Piping Inspection Program manages the effects of corrosion on the pressure-retaining capability of buried carbon steel, copper, and copper alloy piping and tanks. This program includes periodic inspections and preventive measures to mitigate corrosion. Preventive measures are in accordance with standard industry practice for maintaining external coatings and wrappings. An inspection of a sample of buried piping included in the scope of this program will be performed within ten years after entering the period of extended operation, unless an opportunistic inspection of similar underground piping has occurred within this ten-year period. A minimum of one inspection of 10 feet of in-scope copper and copper alloy piping will be performed in each 10-year period of extended operation. Before the end of the tenth year of extended operation, an engineering evaluation will be performed to determine if sufficient inspections have been conducted to draw a conclusion regarding the ability of the underground coatings to protect the underground piping from degradation. If not, a sample of buried piping will be inspected to allow that conclusion to be reached. Soil testing will be conducted once in each 10 year period starting 10 years prior to the period of extended operation to confirm lack of soil corrosivity.

15.1.7 Cast Austenitic Stainless Steel Evaluation Program

The Cast Austenitic Stainless Steel (CASS) Evaluation Program augments the inspection of reactor coolant system components in accordance with ASME Section XI. The CASS Evaluation Program manages the effects of loss of fracture toughness in reactor coolant system CASS components susceptible to thermal aging embrittlement using a component-specific flaw tolerance evaluation. This program does not include CASS components within reactor vessel internals, which are evaluated and inspected as part of the Reactor Vessel Internals Cast Austenitic Stainless Steel (CASS) Program.

15.1.8 Containment Leakage Rate Testing Program

This section supplements discussion of containment leakage testing in other UFSAR sections, including Sections 5.4.4 and 5.7.3. As described in 10 CFR 50, Appendix J, containment leakage rate tests are required to assure that:

Leakage through the primary reactor containment and systems and components penetrating primary containment does not exceed allowable leakage rate values; and

Periodic surveillance of reactor containment penetrations and isolation valves is performed so that proper maintenance and repairs are made during the service life of the containment, and systems and components penetrating primary containment.

This program manages loss of material, cracking, and change in material properties, as applicable, for the equipment constituting the containment pressure boundary.

15.1.9 Control Rod Drive Mechanism and Other Vessel Head Penetration Inspection Program

The Control Rod Drive Mechanism and Other Vessel Head Penetration Inspection Program manages primary water stress corrosion cracking (PWSCC) of nickel-based alloy reactor vessel head penetrations exposed to borated water to ensure that the pressure boundary function is maintained. The ASME Section XI, Subsection IWB, IWC and IWD Inservice Inspection (Section 15.1.17) and Water Chemistry Control Programs (Section 15.1.43) are used in conjunction with this program to manage cracking of the reactor vessel head penetrations.

15.1.10 Diesel Fuel Monitoring Program

The Diesel Fuel Monitoring Program ensures that adequate diesel fuel quality is maintained to prevent corrosion of the fuel oil systems associated with the emergency diesel engines and the diesel driven fire pumps. This program manages aging effects on the internal surfaces of diesel fuel oil tanks and components, as applicable, within the scope of license renewal. The program monitors fuel oil quality and the contaminant concentrations in the fuel oil using ASTM Standards specified in the technical specification. Visual inspections of tanks drained for cleaning ensures that significant degradation is not occurring. This program manages the loss of material and cracking, as applicable, for fuel oil system components.

15.1.11 Environmental Qualification of Electric Components Program

The Environmental Qualification of Electric Components Program manages component thermal, radiation, and cyclical aging of electrical equipment as required by 10 CFR 50.49. This program manages aging effects through the use of aging evaluations based on 10 CFR 50.49(f) qualification methods. As required by 10 CFR 50.49, environmentally qualified (EQ) components not qualified for the license term are to be refurbished or replaced, or have their qualification extended prior to reaching the aging limits established in the evaluation.

15.1.12 Fatigue Monitoring Program

The Fatigue Monitoring Program monitors and tracks the number of critical thermal and pressure transients for selected reactor coolant system components in order not to exceed the design limit on fatigue usage. The program maintains the basis for component analyses containing explicit thermal cycle count assumptions. Components managed by this program are those shown to be acceptable by analyses that explicitly addressed thermal and pressure fatigue transient limits. As discussed in Section 15.2.2, the Fatigue Monitoring Program was enhanced prior to the period of extended operation to address environmentally-assisted fatigue for the pressurizer surge line, Class 1 portions of the RHR piping, and the Class 1 charging and safety injection nozzles.

15.1.13 Fire Protection Program

The Fire Protection Program includes fire barrier and diesel-driven fire pump inspections.

Fire barrier inspections include:

- Periodic visual inspection of fire barrier penetration seals, fire barrier walls, ceilings, and floors; and
- Periodic visual inspection and functional tests of fire-rated doors to ensure that their operability is maintained.

Diesel-driven fire pump inspections include periodic pump testing to ensure that the fuel supply line can perform its intended function. This program also includes periodic inspection and testing of the halon/carbon dioxide fire suppression system. This program required enhancements that were implemented prior to the period of extended operation.

15.1.14 Fire Water System Program

The Fire Water System Program applies to water-based fire protection systems (consisting of sprinklers, nozzles, fittings, valves, hydrants, hose stations, standpipes, water storage tanks, and aboveground and underground piping and components) that are tested in accordance with the applicable National Fire Protection Association (NFPA) codes and standards.

Such testing assures the minimum functionality of the systems. These systems are normally maintained at required operating pressure and monitored such that leakage resulting in loss of system pressure is immediately detected and corrective actions initiated. Sprinkler heads are inspected using the guidance of NFPA 25, Section 2-3.1.1. This program required enhancements that were implemented prior to the period of extended operation.

15.1.15 Flow-Accelerated Corrosion Program

The Flow-Accelerated Corrosion (FAC) Program assures that the structural integrity of carbon steel pipes containing high-energy fluids and the low alloy steel steam generator main steam nozzles are maintained. This program includes:

a. An analysis to determine critical locations;

b. Limited baseline inspections to determine the extent of thinning at these locations;

c. Follow-up inspections to confirm the predictions; and

d. Component repair or replacement, as necessary.

15.1.16 Heat Exchanger Monitoring Program

The Heat Exchanger Monitoring Program manages loss of material and cracking, as applicable, on heat exchangers exposed to treated water in various systems. The Heat Exchanger Monitoring Program inspects heat exchangers for degradation using nondestructive examinations, such as eddy-current inspections or visual inspections. The Heat Exchanger Monitoring Program also includes activities to manage loss of material due to selective leaching. Brinell Hardness testing will be performed on selected heat exchanger tubes that are susceptible to selective leaching, when feasible. However, Brinell Hardness testing may not be feasible for some components due to form and configuration (e.g., heat exchanger tubes). In such cases, examinations other than Brinell Hardness testing may be used to identify the presence of selective leaching of material. Other mechanical means, such as scraping or chipping, provide an acceptable alternative method of identifying the presence of selective leaching. If degradation is found, then an evaluation will be performed to determine its effects on the heat exchanger's design functions. The Heat Exchanger Monitoring Program was implemented prior to the period of extended operation.

15.1.17 Inservice Inspection-ASME Section XI, Subsections IWB, IWC and IWD Program

The ASME Section XI, Subsections IWB, IWC and IWD Program implements the applicable requirements of ASME Section XI, approved NRC alternatives and relief requests, and other requirements specified in 10 CFR 50.55a. Every 10 years, the Inservice Inspection (ISI) Long-Term Plan is updated for each unit to the latest ASME Section XI code edition and addendum approved by the NRC in the current edition of 10 CFR Part 50. The ISI Long-Term Plan is a detailed listing and inspection schedule of components within the ISI boundary.

15.1.18 Inservice Inspection-ASME Section XI, Subsection IWE Program

ASME Code Section XI, Subsection IWE and the additional requirements specified in 10 CFR 50.55a(b)(2) constitute an existing required program applicable to managing aging of steel liners of concrete containments and other containment components. The ASME Section XI, Subsection IWE Program uses visual examination, limited volumetric examination, and surface examination, as required.

15.1.19 Inservice Inspection-ASME Section XI, Subsection IWF Program

Inservice inspection of supports for ASME piping and components is addressed in Section XI, Subsection IWF. ASME Section XI, Subsection IWF constitutes an existing required program applicable to managing aging of ASME Class 1, 2, 3, and MC supports. The ASME Section XI, Subsection IWF Program uses visual examinations of a sample of the total support population.

15.1.20 Inservice Inspection-ASME Section XI, Subsection IWL Program

10 CFR 50.55a specifies the examination requirements of the ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWL for reinforced concrete containments (Class CC). ASME Code Section XI, Subsection IWL and the additional requirements specified in 10 CFR 50.55a(b)(2) constitute an existing required program applicable to managing aging of reinforced concrete containment systems. The ASME Section XI, Subsection IWL Program uses visual examinations of accessible portions of the containment concrete walls and domes.

15.1.21 Inservice Inspection-Augmented Inspections Program

The ASME Section XI, Augmented Inspections Program manages the effects of aging on selected components outside the jurisdiction of ASME Section XI. To the extent practical, augmented inspections are consistent with the applicable ASME requirements of ASME Section XI (i.e., selection of inspection methods, inspection frequency, percentage of components examined within a population, and acceptance criteria). This program required enhancements that were implemented prior to the period of extended operation.

15.1.22 Instrument Air Quality Program

The Instrument Air Quality Program periodically documents the control air system air quality for maximum dewpoint, particulate size, and dryer condition, pursuant to the performance requirements of ANSI Standard ISA-S7.3-1975. This program ensures that the control air supplied to components within the scope of license renewal is maintained free of water and significant contaminants. This program required an enhancement that was implemented prior to the period of extended operation

15.1.23 Non-EQ Inaccessible Medium-Voltage Cable Program

The Non-EQ Inaccessible Medium-Voltage Cable Program applies to inaccessible (e.g., in conduit or direct-buried) medium-voltage cables within the scope of license renewal that are exposed to significant moisture simultaneously with significant voltage. Significant moisture is defined as periodic exposures that last more than a few days (e.g., cable in standing water). Significant voltage exposure is defined as being subjected to system voltage for more than 25 percent of the time. Under this program, in-scope medium-voltage cables that are exposed to significant moisture and significant voltage will be tested at least once every 10 years to provide an indication of the condition of the conductor insulation. The specific type of test performed is determined prior to the initial test, and is based on technology that is state-of-the-art at the time the test is performed. The Non-EQ Inaccessible Medium-Voltage Cable Program was implemented prior to the period of extended operation.

15.1.24 Non-EQ Instrumentation Circuits Test Review Program

The Non-EQ Instrumentation Circuits Test Review Program manages aging effects for electrical cables that:

- Are not subject to the environmental qualification requirements of 10 CFR 50.49, and
- Are used in instrumentation circuits with sensitive, high-voltage, low-level signals such as radiation monitoring and nuclear instrumentation, which are exposed to adverse localized environments caused by heat, radiation, or moisture.

An adverse localized environment is defined as being significantly more severe than the specified service environment for the cable. This program detects aging effects by reviewing calibration or surveillance results for components within the program scope at a frequency of not less than once per ten years or as part of corrective actions when acceptance criteria are exceeded at the normal calibration frequency. A proven cable test for detecting insulation deterioration on in-scope instrumentation cables that are disconnected during calibration will be performed at a frequency determined by engineering evaluation, but will not be less than once per ten years. The Non-EQ Instrumentation Circuits Test Review Program was implemented prior to the period of extended operation.

15.1.25 Non-EQ Insulated Cables and Connections Program

The Non-EQ Insulated Cables and Connections Program applies to accessible insulated cables and connections installed in structures that are within the scope of license renewal and prone to adverse localized environments. An adverse localized equipment environment defined as being significantly more severe than the specified service condition for the insulated cable or connection. The program visually inspects at least once every 10 years a representative sample of accessible insulated cables and connections for cable and connection jacket surface anomalies, such as embrittlement, discoloration, cracking, swelling, or surface contamination. The Non-EQ Insulated Cables and Connections Program was implemented prior to the period of extended operation.

15.1.26 Oil Analysis Program

The Oil Analysis Program ensures that the lubricating oil environment in the mechanical systems in the scope of license renewal is maintained to the required quality. By monitoring oil quality, the Oil Analysis Program maintains oil systems free of contaminants (primarily water and particulates), thereby preserving an environment that is not conducive to loss of material, cracking, or fouling.

15.1.27 Pressurizer Examinations Program

The Pressurizer Examinations Program manages cracking of the pressurizer cladding (and items attached to the cladding) which may propagate into the underlying ferritic steel. This program also determined the condition of the internal spray head, spray head locking bar, and coupling by a one-time visual examination of these components in CNP Unit 1. This program required enhancements that were implemented prior to the period of extended operation.

15.1.28 Preventive Maintenance Program

The Preventive Maintenance (PM) Program comprises those preventive maintenance tasks that are intended to sustain plant equipment within design parameters and maintain the equipment's intrinsic reliability. PM activities provide for periodic component inspections and testing to detect the various aging effects applicable to those components included in the scope of the PM Program for license renewal. The program includes preventive maintenance tasks for expansion joints in the Condensate System that are required to implement NFPA 805 requirements. This program required enhancements that were implemented prior to the period of extended operation.

15.1.29 Reactor Vessel Integrity Program

The Reactor Vessel Integrity Program manages reduction of fracture toughness of reactor vessel beltline materials to assure that the pressure boundary function of the reactor vessel is maintained. The program is based on ASTM E-185-82, "Standard Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels," and includes an evaluation of radiation damage based on pre-irradiation and post-irradiation testing of Charpy V-notch and tensile specimens. Through the Reactor Vessel Integrity Program, reports are submitted as required by 10 CFR 50, Appendix H. The Reactor Vessel Integrity Program also encompasses other activities associated with managing the integrity of the reactor vessel, including updating the RT_{PTS} analysis, as required by 10 CFR 50.61, and maintenance of the pressure-temperature curves, as required by 10 CFR 50, Appendix G. This program required enhancements that were implemented prior to the period of extended operation.

15.1.30 Reactor Vessel Internals Plates, Forgings, Welds, and Bolting Program

The Reactor Vessel Internals Plates, Forgings, Welds, and Bolting Program manages aging effects of reactor vessel internals plates, forgings, welds, and bolting. This program supplements the reactor vessel internals inspections required by the ASME Section XI Inservice Inspection Programs. This program manages the effects of:

- Crack initiation and growth due to stress corrosion cracking or irradiation-assisted stress corrosion cracking,
- Loss of fracture toughness due to neutron irradiation embrittlement, and
- Distortion due to void swelling, and
- Loss of bolted closure integrity due to stress relaxation (loss of preload).

This program provides for visual inspections and non-destructive examinations of the reactor vessel internals. I&M participates in industry-wide programs designed by the PWR Materials Reliability Project Reactor Internals Issue Task Group for investigating the impacts of aging on PWR vessel internal subcomponents. The Reactor Vessel Internals Plates, Forgings, Welds, and Bolting Program was submitted for staff review prior to the period of extended operation.

15.1.31 Reactor Vessel Internals Cast Austenitic Stainless Steel Program

The Reactor Vessel Internals Cast Austenitic Stainless Steel (CASS) Program manages aging effects of CASS reactor vessel internals components. This program supplements the reactor vessel internals inspections required by the ASME Section XI Inservice Inspection Program. The program manages cracking, reduction of fracture toughness, and dimensional changes using visual inspections and non-destructive examinations of applicable components. Applicability was determined based on the neutron fluence and thermal embrittlement susceptibility of a component. The Reactor Vessel Internals CASS Program was implemented prior to the period of extended operation.

15.1.32 Service Water System Reliability Program

The Service Water System Reliability Program relies on implementation of the recommendations of NRC Generic Letter 89-13 to ensure that the effects of aging on the essential service water (ESW) system are managed. The program includes surveillance and control techniques to manage aging effects caused by biofouling, corrosion, erosion, protective coating failures, and silting in the ESW system or structures and components serviced by the ESW system. This program required enhancements that were implemented prior to the period of extended operation. Included in these enhancements is the requirement to visually inspect susceptible materials for the presence of selective leaching and conduct physical testing, such as hardness testing or an equivalent physical test to determine if selective leaching has occurred.

15.1.33 Small Bore Piping Program

The Small Bore Piping Program manages cracking of small bore Class 1 piping (< 4 inch nominal pipe size), including pipe, fittings, and branch connections, in the reactor coolant system. The Small Bore Piping inspection is a one-time volumetric examination of susceptible items in selected locations of Class 1 small bore piping. The volumetric examinations were completed for Units 1 and 2 prior to the period of extended operation.

15.1.34 Steam Generator Integrity Program

The Steam Generator Integrity Program, which is based on guidance provided in NEI 97-06, Steam Generator Program Guidelines, uses nondestructive examination techniques to identify tubes that are defective and need to be removed from service or repaired in accordance with the Technical Specifications. In addition, the Steam Generator Integrity Program uses visual inspections to manage the effects of aging on secondary side internals needed to maintain steam generator tube integrity.

15.1.35 Structures Monitoring – Structures Monitoring Program

Implementation of the Structures Monitoring Program under 10 CFR 50.65, the Maintenance Rule, is addressed in NRC Regulatory Guide 1.160, Rev. 2, and NUMARC 93-01, Rev. 2. These two documents provide guidance for development of licensee-specific programs to monitor the condition of structures and structural components within the scope of both the Maintenance Rule and license renewal such that there is no loss of structure or structural component intended function. The program includes structural components in the Turbine Building and Screenhouse required for flood protection. This program required enhancements that were implemented prior to the period of extended operation.

15.1.36 Structures Monitoring – Crane Inspection Program

The Crane Inspection Program includes testing and monitoring to provide assurance that the structures and components of cranes in the scope of license renewal are capable of sustaining their rated loads. Crane rails and structural components will be visually inspected on a routine basis for degradation to manage loss of material. This program required enhancements that were implemented prior to the period of extended operation.

15.1.37 Structures Monitoring – Divider Barrier Seal Inspection Program

The Divider Barrier Seal Inspection Program detects cracking and change in material properties of the elastomeric divider barrier seals, divider barrier hatch, and personnel access door seals, and pressure seals for penetrations and openings through the containment divider barrier. The program detects aging effects through analysis of main divider barrier seal test coupons and visual examination of the three types of seals between the upper and lower containment compartments.

15.1.38 Structures Monitoring – Ice Basket Inspection Program

The Ice Basket Inspection Program verifies that ice condenser baskets are free of detrimental structural wear, cracks, corrosion, or noticeable damage. The Ice Basket Inspection Program detects loss of material of the ice baskets by visual inspections as required by Technical Specifications.

15.1.39 Structures Monitoring – Masonry Wall Program

The Masonry Wall Program manages cracking of masonry walls within the scope of license renewal. Masonry walls are visually inspected as part of the Structures Monitoring Program conducted for 10 CFR 50.65, the Maintenance Rule. This program required enhancements that were implemented prior to the period of extended operation.

15.1.40 System Testing Program

The System Testing Program encompasses a number of miscellaneous system and component testing activities credited for managing the effects of aging. These activities are typically surveillance activities required by the Technical Specifications or normal monitoring of plant operation (for example, plant log readings or other normal monitoring techniques). In general, these activities are conducted on a periodic basis (surveillances) or routinely (logs) during plant operation. They are intended to verify the continuing capability of safety-related systems and components to meet established performance requirements. This program required enhancements that were implemented prior to the period of extended operation.

15.1.41 System Walkdown Program

The System Walkdown Program manages loss of material, loss of mechanical closure integrity and cracking, as applicable, for systems and components within the scope of license renewal. The program includes components in the Station Drainage System for flood protection and components in the Condensate System required to implement NFPA 805 requirements. The program uses general visual inspections of readily accessible system and component surfaces during system walkdowns. This program required enhancements that were implemented prior to the period of extended operation.

15.1.42 Wall Thinning Monitoring Program

The Wall Thinning Monitoring Program manages loss of material of carbon steel piping and valves in the containment isolation and auxiliary feedwater systems. Inspections are performed to ensure wall thickness is above the minimum required in order to avoid leaks or failures. The Wall Thinning Monitoring Program was implemented prior to the period of extended operation.

15.1.43 Water Chemistry Control – Primary and Secondary Water Chemistry Control Program

The Primary and Secondary Water Chemistry Control Program mitigates damage caused by corrosion and stress corrosion cracking (SCC). The program relies on monitoring and control of water chemistry based on EPRI guidelines. This program required enhancements that were implemented prior to the period of extended operation.

15.1.44 Water Chemistry Control – Closed Cooling Water Chemistry Control Program

The Closed Cooling Water Chemistry Control Program includes preventive measures that manage loss of material, cracking, and fouling, as applicable, for the component cooling water (CCW) system and components cooled by CCW that are in the scope of license renewal. These chemistry activities provide for monitoring and controlling closed cooling water chemistry using procedures and processes based on EPRI guidelines.

15.1.45 Water Chemistry Control – Auxiliary Systems Water Chemistry Control Program

The Auxiliary Systems Water Chemistry Control Program manages loss of material and fouling, as applicable, of components in the scope of license renewal that are exposed to treated water environments. The program implements sampling activities and analyses to monitor and control relevant chemistry conditions of these environments.

15.1.46 Water Chemistry Control – Chemistry One-Time Inspection Program

The Chemistry One-Time Inspection Program verifies the effectiveness of the Water Chemistry Control Program to ensure that aging effects will be effectively managed during the period of extended operation. Using a combination of non-destructive examinations (NDE), including visual, ultrasonic and surface techniques, a representative sample of components that credit the Water Chemistry Control Programs are inspected prior to entering the period of extended operation for the respective Unit to determine if additional age management activities were required. The Chemistry One-Time Inspection Program was completed prior to the period of extended operation for Unit 1 and will be completed prior to the period of extended operations for Unit 2.

15.2 EVALUATION OF TIME-LIMITED AGING ANALYSES

As part of the application for a renewed license, 10 CFR 54.21(c) requires an evaluation of TLAAs for the period of extended operation. The following TLAAs have been identified and evaluated to meet this requirement.

15.2.1 Reactor Vessel Neutron Embrittlement

Three analyses that address the effects of neutron irradiation embrittlement of the reactor vessels have been identified as TLAAs. These analyses address:

- Charpy Upper-Shelf Energy (C_VUSE),
- Pressurized Thermal Shock (PTS), and
- Pressure-Temperature (P-T) Limits.

The analyses were updated to 48 effective full power years (EFPY), which represents the end of the period of extended operation (60 years) using an assumed capacity factor of 80%. The Reactor Vessel Integrity Program described in Section 15.1.29 ensures that time-dependent parameters in these TLAAs remain valid through the period of extended operation. The reactor vessel neutron embrittlement TLAAs are projected to the end of the period of extended operation in accordance with 10CFR54.21(c)(1)(ii).

Charpy Upper-Shelf Energy

Appendix G of 10 CFR 50 requires that reactor vessel beltline materials maintain a C_VUSE of no less than 50 ft-lb throughout the life of the vessel. The 48 EFPY C_VUSE values for the reactor vessel beltline materials for Unit 1 are reported in Table A-2 of WCAP-15879 (Reference 15.3 #6); and for Unit 2, in Table A-2 of WCAP-13517 (Reference 15.3 #7). The C_VUSE values were calculated using Regulatory Guide 1.99, Revision 2, Position 1. The C_VUSE is maintained above 50 ft-lb for all base metal (plates and forgings) and welds at 48 EFPY for both units. Therefore, Charpy upper-shelf energy has been evaluated in accordance with 10CFR54.21(c)(1)(ii).

Pressurized Thermal Shock

10CFR50.61(b)(1) provides for the protection of pressurized water reactors against pressurized thermal shock. The projected values of reference temperature for pressurized thermal shock (RT_{PTS}) are required to be assessed upon request for a change in the expiration date for operation of the facility.

10 CFR 50.61(b)(2) establishes screening criteria for RT_{PTS}: 270°F for plates, forgings, and axial welds; and 300°F for circumferential welds. The values for RT_{PTS} at 48 EFPY for Unit 1 are provided in Table 6 of WCAP-15879 (Reference 15.3 #6); and for Unit 2, in Table 6 of WCAP-13517 (Reference 15.3 #7). For both units, the projected RT_{PTS} values for 48 EFPY are within the established screening criteria. Therefore, RT_{PTS} for Units 1 and 2 have been evaluated in accordance with 10 CFR 54.21(c)(1)(ii).

Pressure-Temperature Limits

Appendix G of 10 CFR 50 requires operation of the reactor pressure vessel within established pressure-temperature (P-T) limits. The P-T limits for Unit 1 are documented in WCAP-15878 (Reference 15.3 #8); and for Unit 2 in WCAP-15047 (Reference 15.3 #9). The 48 EFPY P-T results are reported in Section 9.0, Figures 9-3 and 9-4 of each respective WCAP. The operating window at 48 EFPY is sufficient to conduct normal heatup and cooldown operations for both Units 1 and 2. Therefore, P-T limits for Units 1 and 2 have been projected to the end of the period of extended operation in accordance with 10CFR54.21 (c)(1)(ii).

15.2.2 Metal Fatigue

The analysis of metal fatigue is a TLAA for Class 1 and selected non-Class 1 mechanical components within the scope of license renewal.

Class 1 Metal Fatigue

Fatigue evaluations performed in the design of the Class 1 reactor coolant system (RCS) components were based on a number of design cycles assumed for the life of the plant. The RCS design transients used in the fatigue evaluations for the Class 1 components were reviewed for both units. The numbers of actual RCS design transients from plant operating history were extrapolated to 60 years of operation. Except for auxiliary spray line piping thermal cycling transient described in WCAP-14070, Page 6-3, the number of RCS design transients assumed in the original design was greater than the extrapolated number for 60 years of operation. Therefore, except for auxiliary spray line piping thermal cycling transient fatigue evaluation, the fatigue evaluations for the Class 1 components remain valid for the period of extended operation in accordance with 10CFR54.21(c)(1)(i). The RCS design transients are monitored through the Fatigue Monitoring Program, which is discussed in Section 15.1.12.

Class 1 piping has been qualified in accordance with USAS B31.1. The allowable stress limits for the piping implicitly assumes a limit of 7000 equivalent full-temperature thermal cycles. To identify the specific locations where extended operation could invalidate the stress limits, the design temperatures and operating conditions of the Class 1 piping systems were reviewed. This review determined that, based on assumptions of fewer than 7000 equivalent full-temperature thermal cycles, the analyses for all locations are valid for the period of extended operation in accordance with 10CFR54.21(c)(1)(i).

In response to NRC Bulletin 88-08, "Thermal Stresses in Piping Connected to Reactor Coolant Systems," and its supplements, a fatigue evaluation of the auxiliary spray line was performed and reported in WCAP-14070 (Reference 15.3 #10). The fatigue evaluation is based on CNP design transients.

The transient development assessment in WCAP-14070 evaluated the effects of a cyclic leak in an auxiliary spray isolation valve. The WCAP-14070 evaluation assumed the cyclic leakage would continue throughout the 40 years of plant operation. Therefore, this frequency is time-dependent and constitutes a TLAA.

The WCAP-14070 auxiliary spray line piping thermal fatigue TLAA was addressed prior to the period of extended operation by performing a plant-specific fatigue reanalysis of the auxiliary spray line piping prior to entering the period of extended operation to ensure that cumulative usage factors (CUFs) are below 1.0.

A plant-specific structural analysis of the pressurizer surge line performed and reported in WCAP-12850 (Reference 15.3 #11) supports the conclusion that CNP is in compliance with the requirements of NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification." The surge line stratification analysis was based on the CNP design transients. As described above, the number of RCS design transients assumed in the original design was greater than the extrapolated number for 60 years of operation. Thus, the existing pressurizer surge line thermal stratification analysis and its results are valid for the period of extended operation in accordance with 10CFR54.21(c)(1)(i).

Non-Class 1 Metal Fatigue

Non-Class 1 piping within the scope of license renewal was designed to USAS B31.1. Piping components that may have Normal or Upset Condition operating temperatures in excess of 220^{o}F for carbon steel, or 270^{o}F for austenitic stainless steel, were evaluated for fatigue. These piping components were evaluated for their potential to exceed the limiting number of equivalent full-temperature cycles used for the original design in 60 years of plant operation. With one exception, the review determined that none of the piping or components would exceed the limit of equivalent full-temperature thermal cycles. Thus, for all but the one exception, fatigue considerations for the original piping and component design are valid for the period of extended operation in accordance with 10CFR54.21(c)(1)(i). For the exception, RCS sampling system piping, a calculation was prepared to justify a new limit to support RCS sampling for 60 years of operation.

Only non-Class 1 pressure vessels, heat exchangers, storage tanks, and pumps designed and fabricated in accordance with the ASME Boiler and Pressure Vessel (B&PV) Code, Section VIII, Division 2 or Section III, NC-3200 (Class B) require evaluation for thermal fatigue. Of these components, consideration of thermal fatigue is not required unless specifically directed by the equipment specification. A review of the components designed to the above Code requirements determined that the components with equipment specifications requiring consideration of thermal fatigue used design transients identified consistent with the RCS transients defined in Table 4.1-10 of the UFSAR. As described for Class 1 metal fatigue in this section, the assumed number of RCS design transients is acceptable for 60 years so the fatigue evaluation considered in the original design of these components will remain valid during the period of extended operation in accordance with 10CFR54.21(c)(1)(i).

Environmentally-Assisted Fatigue

Recent test data indicates that certain environmental effects (such as temperature, oxygen, and stress rate) in the primary systems of light water reactors could result in greater susceptibility to fatigue than would be predicted by fatigue analyses based on the ASME Section III design fatigue curves. Although the NRC has concluded that the environmental effects associated with fatigue life are not safety significant through the end of the initial license term, they also determined that the effects of fatigue should be addressed for license renewal.

The effects of environmentally-assisted thermal fatigue for the limiting locations identified in NUREG/CR-6260, *Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components*, have been evaluated for CNP in accordance with 10CFR54.21(c)(1)(i and ii). The evaluations determined that the reactor vessel shell, lower head, and inlet and outlet nozzles are acceptable for the period of extended operation.

Prior to the period of extended operation, the Fatigue Monitoring Program, which is discussed in Section 15.1.12, was enhanced to address environmentally-assisted fatigue of the pressurizer surge line, Class 1 portions of the RHR piping, and the Class 1 charging and safety injection nozzles, in accordance with 10CFR54.21(c)(1)(iii). The approach for addressing environmentally-assisted fatigue for these components was to complete a plant-specific fatigue analysis that included environmental effects to ensure that cumulative usage factors remain below 1.0.

15.2.3 Environmental Qualification of Electric Components

The CNP Environmental Qualification of Electric Components Program, discussed in Section 15.1.11, manages component thermal, radiation, and cyclical aging, as applicable, through the use of aging evaluations based on the applicable qualification methods:

- 10CFR50.49(f),
- NUREG-0588, Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment, or
- Enclosure 4 to IE Bulletin 79-01B, "Guidelines for Evaluating Environmental Qualification of Class 1 Electrical Equipment in Operating Reactors."

Aging evaluations for EQ components that specify a qualification of at least 40 years are considered TLAAs for license renewal. The Environmental Qualification of Electric Components Program ensures that these EQ components will be maintained within the bounds of their qualification bases. The effects of aging will thus be managed in accordance with 10CFR54.21(c)(1)(iii).

15.2.4 Containment Liner Plate and Penetration Fatigue Analyses

TLAAs applicable to the containment structure are the containment liner plate and the containment penetration fatigue analyses.

Containment Liner Plate Fatigue

The fatigue life of the liner was evaluated in 1999 after discovery of localized thinning of the liner. The evaluation, based on testing, determined a fatigue cyclic loading limit for the uncorroded liner plate, of 180,000 cycles at an amplitude of ± 20 ksi. The amplitude of a thermal stress cycle based on an enveloping assessment of the liner design cyclic loads (UFSAR, Section 5.2.3) is well within the amplitude of the evaluated limit. Additionally, the number of containment load and thermal cycles expected during the plant life including the period of extended operation is insignificant compared to 180,000 cycles. Therefore, the analysis of fatigue for the containment liner will remain valid for the period of extended operation in accordance with 10CFR54.21(c)(1)(i).

Containment Penetration Fatigue

Analyses for the main steam and residual heat removal (RHR) penetrations were developed using the operating transients listed in Table 4.1-10 of the UFSAR. The analyses determined that the requirements of ASME Code, Section III, paragraph N-415.1 (exemption from fatigue) were met and that fatigue evaluations were not required for the main steam and RHR penetrations. The analyses supporting the exemption-from-fatigue analyses are TLAAs, since the evaluation is based on selected design thermal and loading cycles. As described for Class 1 metal fatigue in Section 15.2.2, the assumed number of RCS design transients is acceptable for 60 years. Therefore, the exemption-from-fatigue evaluations for the main steam and RHR containment penetration analyses remains valid for the period of extended operation in accordance with 10CFR54.21(c)(1)(i).

An analysis was developed to compare the remaining hot penetrations to the fatigue exemption provisions of ASME Code, Section III, paragraph N-415.1. The penetrations were grouped in the analysis based on their duty cycle during normal operation including inservice testing duty. The pressure, temperature, cycle, and load stress data of the grouped penetrations were compared to the fatigue exemption provisions of ASME Code, Section III, paragraph N-415.1. The exemption-from-fatigue analysis is a TLAA, since the evaluation is based on selected design thermal and loading cycles extrapolated to 60 years of operation. The analysis determined that all penetrations except the feedwater and blowdown penetrations met the fatigue exemption provisions. Therefore, the exemption-from-fatigue evaluation for all containment penetrations, except feedwater and blowdown, remains valid for the period of extended operation in accordance with 10CFR54.2l(c)(1)(i).

Explicit fatigue analyses were developed for the feedwater and blowdown penetrations to ensure that aging effects due to fatigue would be managed throughout the period of extended operation. The analyses determined that the cumulative usage factors for the feedwater and blowdown penetrations would remain below 1.0 throughout the period of extended operation. These analyses are TLAAs, since the evaluation is based on selected design thermal and loading cycles. As described for Class 1 metal fatigue in Section 15.2.2, the assumed number of RCS design transients is acceptable for 60 years. Therefore, the fatigue evaluations for the feedwater and blowdown containment penetration analyses remains valid for the period of extended operation in accordance with 10CFR54.21(c)(1)(i).

15.2.5 Other Plant-Specific Time-Limited Aging Analyses

Other CNP-specific TLAAs include:

- Leak-before-break (LBB) analyses,
- Thermal aging evaluation of the reactor coolant pump (RCP) casing,
- Ice condenser lattice frame fatigue analysis,
- Underclad cracking evaluation,
- The fatigue analysis of cranes, and
- Reactor Coolant Pump Flywheels

Reactor Coolant System Piping Leak-Before-Break

The leak-before-break analyses include WCAP-15131 (Reference 15.3 #12) for RCS primary loop piping and WCAP-15435 (Reference 15.3 #13) for pressurizer surge line piping. For both analyses, the only consideration that could be influenced by time is the accumulation of actual fatigue transient cycles. Both analyses were evaluated for fatigue design transients defined in Table 4.1-10 of the UFSAR. As described for Class 1 metal fatigue in Section 15.2.2, the assumed number of RCS design transients is acceptable for 60 years so these analyses will remain valid during the period of extended operation in accordance with 10CFR54.21(c)(1)(i).

ASME Code Case N-481

Compliance of the reactor coolant pump casings to ASME Code Case N-481 was evaluated generically for all Westinghouse plants in WCAP-13045 (Reference 15.3 #14). The CNP-specific Code Case N-481 evaluation, which references WCAP-13045, is reported in WCAP-13128 (Reference 15.3 #15). The evaluation uses transient cycle assumptions included in the CNP fatigue design transients. As presented in Section 15.2.2 for Class 1 metal fatigue, the assumed number of RCS design transients is acceptable for 60 years, so the Code Case N-481 evaluation will remain valid during the period of extended operation in accordance with 10CFR54.21(c)(1)(i).

Ice Condenser Lattice Frame

UFSAR Table 5.3.5.3-2, which contains a summary of results of fatigue analysis for the lattice frame, is based on 400 operational basis earthquakes (OBEs). Based on past operating experience at CNP and other facilities, this OBE limit will not be surpassed during the period of extended operation. Therefore, the lattice frame fatigue analysis will remain valid for the period of extended operation in accordance with 10CFR54.21(c)(1)(i).

Reactor Vessel Underclad Cracking

A detailed analysis of underclad cracks in SA-508, Class 2 reactor vessel forgings is provided in topical report WCAP-7733 (Reference 15.3 #16), which presented a fracture mechanics analysis to justify the continued operation of Westinghouse units for 32 EFPY with underclad cracks in the reactor pressure vessels. WCAP-15338 (Reference 15.3 #17) evaluates the impact of cracks beneath austenitic stainless steel weld cladding on reactor pressure vessel integrity for 60 years of operation.

The CNP reactor vessels do not contain SA 508, Class 2 forgings in the beltline regions. Only the vessel flange and inlet and outlet nozzles are fabricated from SA 508, Class 2 forgings. The Unit 1 and Unit 2 Reactor Vessel Closure Head (RVCH) forgings are fabricated from SA 508 Grade 3, Class 1. The evaluation contained in WCAP-15338 has been used to demonstrate that fatigue growth of the subject flaws will be minimal over 60 years and the presence of the underclad cracks are of no concern relative to the structural integrity of the vessels. The design transients assumed are listed in Table 4.1-10 of the UFSAR. As described for Class 1 metal fatigue in Section 15.2.2, the assumed number of RCS design transients is acceptable for 60 years. The numbers of design cycles and transients assumed in the WCAP-15338 analysis bound the numbers of design cycles and transients projected for 60 years of operation. This result demonstrates that the analysis of underclad cracking remains valid for the period of extended operation in accordance with 10CFR54.21(c)(1)(i).

Cranes

In response to NUREG-0612, *Control of Heavy Loads at Nuclear Power Plants Resolution of Generic Technical Activity A-36,* I&M stated that the polar cranes, auxiliary building cranes, and screenhouse crane were in compliance with the design standards of CMAA-70, "Specification for Electric Overhead Traveling Cranes," with limited exceptions. This position was approved in an NRC safety evaluation dated September 20, 1983 (Reference 15.3 #22). Conservative estimates of the number of cycles that could be achieved in 60 years of operation for these five cranes do not exceed the limit in CMAA-70. As a result, the crane designs will remain valid for the period of extended operation in accordance with 10CFR54.21(c)(1)(i).

Reactor Coolant Pump Flywheels

The RCP motors are large, vertical, squirrel cage, induction motors. The motors have flywheels to increase rotational-inertia, thus prolonging pump coastdown and assuring a more gradual loss of main coolant flow to the core in the event that pump power is lost. The flywheel is mounted on the upper end of the rotor, above the upper radial bearing and inside the motor frame. The aging effect of concern is fatigue crack initiation and growth in the flywheel bore keyway from stresses due to starting the motor.

To reduce the RCP flywheel inspection frequency and scope, I&M submitted a license amendment request in 1996 based on WCAP-14535, "Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination" (Reference 15.3 #23). This topical report includes a stress and fracture evaluation that addresses fatigue crack growth for 60 years. The NRC approved this request via CNP Units 1 and 2 License Amendments 217 and 201 (Reference 15.3 #24). Therefore, the RCP flywheel stress and fracture evaluation remain valid for the period of extended operation in accordance with10CFR54.21(c)(1)(i).

15.2.6 TLAAs Contained in Exemptions

I&M requested that the NRC staff exempt Units 1 and 2 from application of specific requirements of Appendix G to 10CFR50 (References 15.3 #18 and 15.3 #19, respectively). The proposed exemption requests were granted (References 15.3 #20 and 15.3 #21, respectively). These exemptions are used for the calculation of the P-T limits detailed in WCAP-15878 (Reference 15.3 #8) and WCAP-15047 (Reference 15.3 #9) for Units 1 and 2, respectively. As discussed in Section.15.2.1, these evaluations have been updated in accordance with 10CFR54.21(c)(1)(ii) to include 60 years (48 EFPY) for both units. Therefore, the continuation of these exemptions is justified for the period of extended operation.

15.3 REFERENCES FOR CHAPTER 15

1. D.C. Cook, Units 1 & 2, Letter transmitting Application for Renewed Operating Licenses., AEP:NRC:3034 (ML033070177).
2. NUREG-1831, Safety Evaluation Report Related to the License Renewal of the Donald C. Cook Nuclear Plant, Units 1 and 2, (ML052230442).
3. NUREG-1800, Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, U.S. Nuclear Regulatory Commission, July 2001.
4. NUREG-1801, Generic Aging Lessons Learned (GALL) Report, U.S. Nuclear Regulatory Commission, July 2001.
5. NUREG-1743, Safety Evaluation Report Related to the License Renewal of Arkansas Nuclear One, Unit 1, U.S. Nuclear Regulatory Commission, April 2001.
6. WCAP-15879, "Evaluation of Pressurized Thermal Shock for D. C. Cook Unit 1 for 40 Years and 60 Years," Revision 0.
7. WCAP-13517, "Evaluation of Pressurized Thermal Shock for D. C. Cook Unit 2," Revision 1.
8. WCAP-15878, "D.C. Cook Unit 1 Heatup and Cooldown Limit Curves for Normal Operation for 40 Years and 60 Years," Revision 0.
9. WCAP-15047, "D.C. Cook Unit 2 WOG Reactor Vessel 60-Year Evaluation Minigroup Heatup and Cooldown Limit Curves for Normal Operation," Revision 2.
10. WCAP-14070, "Evaluation of Donald C. Cook Units 1 and 2 Auxiliary Spray Piping Per NRC Bulletin 88-08," July 1994.
11. WCAP-12850, "Structural Evaluation of Donald C. Cook Nuclear Plant Units 1 and 2 Pressurizer Surge Lines, Considering the Effects of Thermal Stratification," January 1991.
12. WCAP-15131, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the D. C. Cook Units 1 and 2 Nuclear Power Plants," Revision 1, September 1999.

13. WCAP-15435, "Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis For D. C. Cook Units 1 and 2 Nuclear Power Plants (non-proprietary)," August 2000.

14. WCAP-13045, "Compliance to ASME Code Case N-481 of the Primary Loop Pump Casings of Westinghouse Type Nuclear Steam Supply Systems," September 1991.

15. WCAP-13128, "Demonstration of Compliance of the Primary Loop Pump Casings of D. C. Cook Units 1 and 2 to ASME Code Case N-481," March 1992.

16. WCAP-7733, "Reactor Vessels Weld Cladding – Base Metal Interaction," July 1971.

17. WCAP-15338, "A Review of Cracking Associated with Weld Deposited Cladding in Operating PWR Plants," March 2001.

18. Letter from J. E. Pollock, I&M, to NRC Document Control Desk, "D. C. Cook Nuclear Plant, Unit 1, License Amendment Request for Unit 1 Reactor Coolant System Pressure-Temperature Curves, and Request for Exemption from Requirements in 10CFR50.60(a) and 10 CFR 50, Appendix G," AEP:NRC:2349-01, dated December 10, 2002.

19. Letter from J. E. Pollock, I&M, to NRC Document Control Desk, (D. C. Cook Nuclear Plant, Unit 2, License Amendment Request for Unit 2 Reactor Coolant System Pressure-Temperature Curves, and Request for Exemption from Requirements in 10CFR50.60(a) and 10 CFR 50, Appendix G," AEP:NRC:2349-01, dated July 23, 2002.

20. Letter from J. F. Stang, NRC, to A. C. Bakken III, I&M, "Donald C. Cook Nuclear Plant Unit 1, Issuance of Amendment," dated July 18, 2003.

21. Letter from J. F. Stang, NRC, to A. C. Bakken III, I&M, "Donald C. Cook Nuclear Plant Unit 2, Issuance of Amendment," dated March 20, 2003.

22. Letter from S. A. Varga, NRC, to J. Dolan, I&M, "Control of Heavy Loads (Phase 1) – NUREG-0612 Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2," dated September 20, 1983.

23. WCAP-14535, "Topical Report on Reactor Coolant Pump Flywheel Inspection Elimination," January 1996.

24. Letter from J. B Hickman, NRC, to E. E. Fitzpatrick, I&M, "Issuance of Amendments 217 for Unit 1 and 201 for Unit 2 Re: Reactor Coolant Pump Flywheel Inspection Frequency (TAC Nos. M97888 and M97889)," August 8, 1997.