APPENDIX A – UPDATED SAFETY ANALYSIS REPORT (USAR) SUPPLEMENT

A.1 INTRODUCTION

The application for a renewed operating license is required by 10 CFR 54.21(d) to include “an FSAR Supplement.” This appendix provides that supplement for the FCS USAR. Section 2 of this appendix contains a summarized description of the programs and activities for managing the effects of aging. Section 3 of this appendix contains a summary of the evaluation of time-limited aging analyses (TLAAs) for the period of extended operation.

A.2 PROGRAMS AND ACTIVITIES FOR MANAGING THE EFFECTS OF AGING

This section provides summaries of the programs and activities credited for managing the effects of aging, in alphabetical order. The FCS Quality Assurance Program 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 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.

A.2.1 ALLOY 600 PROGRAM

The Alloy 600 Program includes a primary water stress corrosion cracking (PWSCC) susceptibility assessment to identify susceptible components and inservice inspection (ISI) of Reactor Coolant System penetrations to monitor PWSCC and its effect on the intended function of the component. For susceptible penetrations and locations, the program includes an industry-wide, integrated, long-term inspection program based on the industry response to NRC Generic Letter (GL) 97-01, *Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations*.

A.2.2 BOLTING INTEGRITY PROGRAM

The Bolting Integrity Program includes periodic inspection of closure and structural bolting for indications of potential problems, including loss of material, crack initiation, and loss of preload. The program implements guidelines on materials selection, strength and hardness properties, installation procedures, lubricants and sealants, corrosion considerations in the selection and installation of pressure-retaining bolting, and enhanced inspection techniques. The program is based on (1) the bolting integrity program delineated in NUREG-1339, *Resolution of Generic Safety Issue 29: Bolting Degradiation or Failure in Nuclear Power Plants; (2)* industry’s recommendations delineated in ERPRI NP-5769, *Degradation and Failure of Bolting in Nuclear Power Plants*, with the exceptions noted in NUREG-1339 for safety-related bolting; (3) EPRI TR-104213, *Bolted Joint Maintenance and Application Guide,* for pressure retaining bolting and structural bolting; and (4) routine examinations and inspections performed in accordance with the requirements of ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*.

A.2.3 BORIC ACID CORROSION PREVENTION PROGRAM

The Boric Acid Corrosion (BAC) Prevention Program implements administrative controls to (1) perform visual inspections of external surfaces that are potentially exposed to borated water leakage, (2) ensure timely discovery of leak path and removal of the boric acid residues, (3) perform assessments of degradation, and (4) perform follow-up inspections for adequacy of corrective actions. The program is implemented in response to NRC GL 88-05, *Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants*.

A.2.4 BURIED SURFACES EXTERNAL CORROSION PROGRAM

The Buried Surfaces External Corrosion Program provides for inspection of buried piping, tanks, and valves whenever they are uncovered due to excavation for maintenance or modifications. Piping and component coatings and wrappings will be inspected for degraded conditions that could be indicative of possible surface corrosion of the protected metal beneath. The scope and periodicity of inspections will be established and/or adjusted based on the inspection results.

A.2.5 CHEMISTRY PROGRAM

The FCS Chemistry Program controls water chemistry to minimize contaminant concentration and provide chemical additions, such as corrosion inhibitors and biocides, to mitigate aging effects due to corrosion. The program includes specifications for chemical species, limits, representative sampling and analysis frequencies, and corrective actions for control of water chemistry. The program is based on EPRI Guidelines TR-105714, *PWR Primary Water Chemistry Guidelines*, for primary water chemistry, TR-102134, *PWR Secondary Water Chemistry Guideline*, for secondary water chemistry, and TR-107396, *Closed Cooling Water Chemistry Guideline*, for closed-cycle cooling water corrosion inhibitor concentration.

A.2.6 CONTAINMENT INSERVICE INSPECTION PROGRAM

The Containment Inservice Inspection Program implements the examination requirements of ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components,* Subsection IWE, *Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants,* and Subsection IWL, *Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants,* for the containment structure and support components. The ASME Section XI, Subsection IWL program consists of periodic visual inspection of concrete surfaces and periodic visual inspection and sample tendon testing for signs of degradation, assessment of damage, and corrective actions. Measured tendon lift-off forces are compared to predicted tendon forces calculated in accordance with NRC Regulatory Guide (RG) 1.35, *Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments*. The ASME Section XI, Subsection IWE program consists of periodic visual, surface, and volumetric inspection of pressure retaining components for signs of degradation, assessment of damage and corrective actions. This program is in accordance with the requirements of 10 CFR 50.55a and ASME Section XI, Subsections IWE and IWL, 1992 edition including 1992 addenda.

A.2.7 CONTAINMENT LEAK RATE PROGRAM

The Containment Leak Rate Program implements the requirements of 10 CFR Part 50, Appendix J, as well as those examination requirements needed to comply with ASME Section XI, Subsection IWE, RG 1.163, *Performance-Based Containment Leak-Test Program,* and NEI 94-01, *Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50 Appendix J,* Rev. 0 for the containment structure and pressure retaining components. The program consists of monitoring of leakage rates through containment liner/welds, penetrations, fittings, and other access openings for detecting degradation of the containment pressure boundary. Corrective actions are taken if leakage rates exceed acceptance criteria.

A.2.8 COOLING WATER CORROSION PROGRAM

The Cooling Water Corrosion Program monitors and detects aging effects through inspection and nondestructive evaluations. The program also involves some mitigation activities of periodic flushing and draining. The program's aging management activities are based on EPRI TR-107396, *Closed Cooling Water Chemistry Guideline,* for closed-cycle cooling water systems and NRC GL 89-13, *Service Water System Problems Affecting Safety-Related Equipment,* for open-cycle cooling water systems.

A.2.9 DIESEL FUEL MONITORING AND STORAGE PROGRAM

The FCS Diesel Fuel Monitoring and Storage Program monitors and controls diesel fuel quality regarding water and other contaminants in accordance with the guidelines of ASTM Standards D2709, *Standard Test Method for Water and Sediment in Middle Distillate Fuels by Centrifuge,* and D4057, *Standard Practice for Manual Sampling of Petroleum and Petroleum Products*. Exposure to fuel oil contaminants such as water and microbiological organisms is minimized by removal of water and sediment from tanks and by verifying the quality of new fuel oil before its introduction into the storage tanks.

A.2.10 FATIGUE MONITORING PROGRAM

The Fatigue Monitoring Program provides for the monitoring of reactor coolant and associated systems thermal fatigue, pressurizer surge line thermal stratification, and thermal fatigue of selected Class II and III components over the life of the plant to ensure that their operation does not result in exceeding the number of design basis transients included in the design basis of their respective design codes. It will be centered on the industry's automated cycle counting software, FatiguePro. Plant locations that cannot be counted automatically will continue to be counted manually. An FCS site specific evaluation is being performed to address environmental fatigue. Appropriate program enhancements will be made prior to the period of extended operation based on the evaluation results.

A.2.11 FIRE PROTECTION PROGRAM

The FCS Fire Protection Program provides administrative requirements for ensuring the operability of fire protection equipment required to ensure plant safe shutdown. The program includes visual inspections, system flushing, and performance tests of fire barriers (penetration seals, fire doors, walls, ceilings, and floors), fire suppression system components (piping, valves, nozzles, yard hydrants and hose stations, sprinkler heads, and halon systems and cylinders), and the diesel fire pump. The FCS Fire Protection Program includes the requirements identified in Appendix A to NRC Branch Technical Position APCSB 9.5-1 and 10 CFR 50 Appendix R, Section III.G, J, and O and is further described in Section 9.11 of the USAR.

A.2.12 FLOW ACCELERATED CORROSION PROGRAM

The FCS Flow Accelerated Corrosion (FAC) Program implements administrative controls to conduct appropriate analysis and baseline inspections, determine extent of thinning, replace/repair components, and perform follow-up inspections to confirm or quantify and take longer-term corrective actions. The program relies on implementation of EPRI guidelines of NSAC-202L-R2, *Recommendations for an Effective Flow-Accelerated Corrosion Program*.

A.2.13 GENERAL CORROSION OF EXTERNAL SURFACES PROGRAM

The General Corrosion of External Surfaces Program implements systematic inspections and observations to detect corrosion of external surfaces and conditions that can result in corrosion such as damaged coatings and fluid leaks. Inspections and observations include (1) rounds by operators, (2) system engineer walkdowns, and (3) refueling interval inspections inside containment in accordance with RG 1.54, *Quality Assurance Requirements for Protective Coatings Applied to Water - Controlled Nuclear Power Plants*.

A.2.14 INSERVICE INSPECTION PROGRAM

The Fort Calhoun Station Inservice Inspection Program implements the examination requirements of the ASME Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, Subsections IWB, IWC, IWD, IWF. The program consists of periodic volumetric, surface and/or visual examination of components and their supports for assessment, signs of degradation, and corrective actions. This program is in accordance with ASME Section XI, 1995 edition through the 1996 addenda.

A.2.15 NON-EQ CABLE AGING MANAGEMENT PROGRAM

The FCS Non-EQ Cable Aging Management Program establishes a service life value for the Non-EQ cable in a similar fashion as the FCS EQ Program establishes a Qualified Life for the safety related equipment, components, and cable. Corrective actions for Non-EQ Cable, determined not to meet the operational (Service Life) requirements established for the full period of extended operation, will consider using: (1) state of the art analytical techniques to determine if the service life can be further extended; (2) industry accepted and regulatory approved cable inspection techniques that provide aging related data; and/or (3) state of the art, in-situ, non-destructive testing of cable performance, and/or laboratory testing of cable to extend life. Cable replacement will be considered should the aforementioned methodologies not succeed in extending the required service life.

A.2.16 ONE-TIME INSPECTION PROGRAM

The FCS One-Time Inspection Program is a new program that will implement a one-time inspection of internal surfaces of selected components to verify the effectiveness of mitigating programs such as the chemistry and diesel fuel oil programs. Inspections will be performed using suitable techniques at the most susceptible locations to verify that aging effects are not occurring or that the aging effect is progressing at such a slow rate it will not impact the intended function during the period of extended operation.

A.2.17 OVERHEAD LOAD HANDLING SYSTEMS INSPECTION PROGRAM

The Overhead Load Handling Systems Inspection Program implements FCS commitments made in response to NRC GL 81-07, *Control of Heavy Loads at Nuclear Power Plants,* and the maintenance monitoring requirements of 10 CFR 50.65. The program includes assessment of crane lift capabilities, periodic inspections of structural components, and functional tests to assure their integrity.

A.2.18 PERIODIC SURVEILLANCE AND PREVENTIVE MAINTENANCE PROGRAM

The Periodic Surveillance and Preventive Maintenance Program provides for periodic inspections and examinations of specific system and structural components using established NDE techniques. The inspection and examination techniques used and the periodicity of their performance provide reasonable assurance that age related degradation will not compromise the structure or component intended function(s) before the next scheduled inspection.

A.2.19 REACTOR VESSEL INTEGRITY PROGRAM

The Reactor Vessel Integrity Program monitors the extent of changes in material properties and loss of fracture toughness of irradiated reactor pressure vessel materials by periodic removal and testing of surveillance capsules located within the reactor vessel in accordance with RG 1.99, *Radiation Embrittlement of Reactor Vessel Materials,* Rev. 2. The surveillance capsule removal and evaluation is an NRC-approved program that meets the requirements of 10 CFR 50, Appendix H. The program includes revising the FCS surveillance capsule removal schedule in order to optimize the program through the end of the period of extended operation. In addition, the program verifies 10 CFR 50, Appendix G and 10 CFR 50.61 requirements.

A.2.20 REACTOR VESSEL INTERNALS INSPECTION PROGRAM

The Reactor Vessel (RV) Internals Inspection Program includes the following elements for cast austenitic stainless steel (CASS) and other reactor vessel internal components: (a) determination of the susceptibility of CASS components to thermal aging and neutron irradiation embrittlement, (b) identification of the most susceptible or limiting items, (c) development of appropriate inspection techniques to permit detection and characterizing of the feature (cracks) of interest and demonstrate the effectiveness of the proposed technique, and (d) implementation of required inspections prior to the period of extended operation.

A.2.21 SELECTIVE LEACHING PROGRAM

The FCS Selective Leaching Program implements inspection requirements for susceptible components for indication of selective leaching through dezincification or graphitization.

A.2.22 STEAM GENERATOR PROGRAM

The FCS Steam Generator Program consists of inspection scope, frequency, and acceptance criteria for various steam generator components, including the plugging and repair of flawed tubes in accordance with the plant Technical Specifications and the guidance of NEI 97-06, *Steam Generator Program Guidelines*.

A.2.23 STRUCTURES MONITORING PROGRAM

The Structures Monitoring Program provides for periodic visual inspection of designated FCS structures and component supports to ensure that aging degradation will be detected, evaluated, and repaired prior to any loss of intended functions. The inspection requirements are based on the following industry documents: NRC Bulletin 80-11, *Masonry Wall Design;* NRC IN 87-67, *Lessons Learned from Regional Inspections of Licensee Actions in Response to IE Bulletin 80-11;* NUMARC 93-01, *Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants (Line-In/Line-Out Version)*, Rev. 2; and NRC RG 1.160, *Monitoring the Effectiveness of Maintenance at Nuclear Power Plants*, Rev. 2.

A.2.24 THERMAL AGING EMBRITTLEMENT OF CAST AUSTENITIC STAINLESS STEEL PROGRAM

The Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS) Program includes evaluation of the reactor coolant piping as bounded by the Leak-Before-Break (LBB) analysis, assessment of other CASS components for susceptibility to thermal embrittlement, and performance of volumetric inspection of piping or component-specific flaw tolerance evaluation for susceptible components.

A.3 EVALUATION OF TIME-LIMITED AGING ANALYSES

A.3.1 REACTOR VESSEL NEUTRON EMBRITTLEMENT

There are four analyses affected by neutron embrittlement that have been identified as TLAAs:

- Pressure/Temperature (P/T) Curves
- Low Temperature Overpressure Protection (LTOP) Power Operated Relief Valve (PORV) Setpoints
- Pressurized Thermal Shock (PTS)
- Reactor Vessel Upper Shelf Energy

A.3.1.1 PRESSURE/TEMPERATURE (P/T) CURVES

Appendix G to 10 CFR 50 requires that PT limits be established during all phases of reactor operation and that thermal stresses be limited by determining maximum heatup and cooldown rates. The current pressure/temperature analyses are valid out to 40 effective full power years, which extends beyond the current operating license period but not to the end of the period of extended operation. The Technical Specifications will continue to be updated as required by either Appendices G or H of 10 CFR 50, or as operational needs dictate. This will assure that operational limits remain valid for current and projected cumulative neutron fluence levels. Therefore, the analyses will be projected to the end of the period of extended operation.

A.3.1.2 LOW TEMPERATURE OVERPRESSURE PROTECTION (LTOP) PORV SETPOINTS

Low temperature overpressure protection limits are considered as part of the calculation of pressure/temperature curves. Loss of ductility at low temperatures due to neutron embrittlement must be evaluated during the period of extended operation. Therefore, the LTOP analyses will be projected to the end of the period of extended operation.

A.3.1.3 PRESSURIZED THERMAL SHOCK (PTS)

10 CFR 50.61 addresses another issue related to embrittlement and thermal stress called Pressurized Thermal Shock (PTS). Irradiation makes the vessel's beltline more susceptible to cracking during a pressurized thermal shock event. The parameter describing this fracture potential is called the transition temperature (or RT_{PTS}) and it corresponds to the nil ductility reference temperature for the most limiting beltline material. It is a function of the projected fluence values and is calculated using guidance in Regulatory Guide 1.99, revision 2. Applicants are obligated to project the values of the increasing transition temperature into the period of extended operation.

OPPD has completed the projected calculation and the NRC has concluded that RT_{PTS} for the FCS reactor vessel will remain below the 10 CFR 50.61 PTS screening criteria until 2033, the end of the proposed license renewal period. Therefore, the analyses have been projected to the end of the period of extended operation.

A.3.1.4 REACTOR VESSEL UPPER SHELF ENERGY

Upper shelf energy is a measure of fracture toughness at temperatures above RT_{PTS} when the vessel is exposed to neutron radiation. The screening criteria for the increase in transition temperature are found in 10 CFR 50.61. The screening criterion for the decrease in upper shelf energy is found in 10 CFR 50, Appendix G.

Preliminary calculations have shown that the vessel beltline Charpy upper-shelf energy for the limiting weld will be approximately 54.6 ft-lbs, based on position 1.2 of RG 1.99. This value remains above the regulatory approved minimum of 50 ft-lbs through the period of extended operation. The existing Appendix G analysis will be finalized and formally revised to reflect that it bounds the minimum approved fluence value at the end of plant life. Therefore, the analyses will be projected to the end of the period of extended operation.

A.3.2 METAL FATIGUE

There are three distinct issues considered separately under the TLAA for Metal Fatigue:

- Reactor Coolant and associated systems thermal fatigue,
- Pressurizer Surge Line Thermal Stratification, and
- Fatigue of Class II and III components.

Each of these issues is managed by the Fatigue Monitoring Program which is addressed in Section A.2.10 of this Appendix.

A.3.3 ENVIRONMENTAL QUALIFICATION OF ELECTRICAL EQUIPMENT

10 CFR 50.49 requires that certain safety related and non-safety related electrical equipment remains functional during and after identified Design Basis Events (DBEs). For the period of extended operation, Environmental Qualification (EQ) is a TLAA affecting all equipment in the scope of the EQ program, with a qualified life longer than the original license period but shorter than the combined original license period plus the period of extended operation, whether active or passive.

The FCS Electrical Equipment Qualification (EEQ) Program has been demonstrated to be capable of programmatically managing the qualified lives of EQ components within the scope of license renewal. The NRC has determined that the EEQ Program is an acceptable program to address environmental qualification in accordance with 10 CFR 54. The FCS EEQ Program will provide for extension of the qualification to the end of the period of extended operation. Therefore, the effects of aging on the intended functions will be adequately managed for the period of extended operation.

A.3.4 CONCRETE CONTAINMENT TENDON PRE-STRESS

The containment wall and dome were pre-stressed by means of unbonded post-tensioned tendons. The pre-stress on the containment tendons decreases over plant life as a result of elastic deformation, creep and shrinkage of concrete, anchorage seating losses, tendon wire friction, stress relaxation and corrosion. Pre-stressing tendon integrity is monitored and confirmed by a regular program of tendon surveillance. Curves showing anticipated variation of tendon force with time, together with the lower limit curves to be applied to surveillance readings are shown in the FCS USAR. The calculated pre-stress at end of plant life exceeds by a reasonable margin the intensity required to meet the design criteria.

The USAR curves will be extended to 60 years of plant life to cover the period of extended operation. This will also show that the pre-stressing force is acceptable for continued service at the end of the period of extended operation considering the assumed time dependent nature of pre-stress losses. The tendon surveillance program will be continued into the period of extended operation using the updated curves. Therefore, the effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

A.3.5 CONTAINMENT LINER PLATE AND PENETRATION SLEEVE FATIGUE

The containment liner and penetration sleeves were designed to be leak-tight under all postulated loading combinations by limiting strains to those values that have been shown to result in leak-tight pressure vessels. The results of the containment fatigue analysis indicated that when the maximum compressive strain in the liner was reached under operating conditions and subsequent cyclic temperature variations were applied to the liner, there was no significant change in stress and strain in concrete or steel for the second cyclic load indicating that shakedown had occurred during the first cycle of loading. Also, the investigation for 500 cycles of loading for the liner steel, anchor steel and anchor welds resulted in a computed cumulative usage factor of 0.05 as compared with an allowable usage factor of 1.0. Consideration of 60 years of operation as opposed to 40 would have no relevant impact on these results. However, the observed buckling of the liner is slightly larger than was assumed in the original analyses. This condition has been evaluated and found adequate for the current term. FCS will complete an analysis considering the actual bulges for a 60-year life. The analysis will be completed before the beginning of the period of extended operation. Therefore, the analysis will be projected to the end of the period of extended operation.

A.3.6 PLANT-SPECIFIC TIME-LIMITED AGING ANALYSES

A.3.6.1 REACTOR COOLANT PUMP FLYWHEEL FATIGUE

A.3.6.1.1 GENERAL ELECTRIC RCP FLYWHEELS

The resistance to rupture of the reactor coolant pump flywheels has been examined at 120 percent overspeed. The conclusion was that over 185,000 complete cycles from zero to 120 percent overspeed would be required to cause a crack to grow to critical size.

This number of cycles will not be exceeded if the licensing period is extended to 60 years. To do so would require in excess of 8 pump starts per day, which far exceeds actual and projected pump use. Since the cycle limit will not be exceeded, the analysis for the General Electric produced RCP flywheels remains valid for the period of extended operation.

A.3.6.1.2 ASEA BROWN BOVERI (ABB) MOTOR FLYWHEEL

During the 1996 refueling outage, reactor coolant pump RC-3B motor was replaced with a motor manufactured by ABB Industries. A crack growth analysis was performed which demonstrated that critical flaw growth would not occur with fewer than 10,000 complete cycles from zero to 120 percent overspeed.

This number of cycles will not be exceeded if the licensing period is extended to 60 years. To do so would require approximately 1 pump start every 2 days, which far exceeds actual and projected pump use. Since the cycle limit will not be exceeded, the analysis for the ABB produced RCP flywheel remains valid for the period of extended operation.

A.3.6.2 LEAK BEFORE BREAK (LBB) ANALYSIS FOR RESOLUTION OF USI A-2

There are two TLAA aspects to LBB, crack growth and thermal aging. While transient cycle fatigue crack growth is a TLAA for FCS and also a design consideration, thermal aging was not evaluated for FCS by either the original design code or the LBB analysis. Consequently, OPPD will perform a plant-specific LBB analysis prior to the period of extended operation. This analysis will consider a 60-year life and thermal aging effects of the CASS RCS and will be completed before the beginning of the period of extended operation. Therefore, the analysis will be projected to the end of the period of extended operation.

A.3.6.3 HIGH ENERGY LINE BREAK (HELB)

The High Energy Line Break (HELB) analysis is a potential TLAA because postulated fatigue cumulative usage factors (CUFs) based on 40 years of operation may be used as screening criteria to determine piping locations that require further analysis regarding the effects of pipe ruptures outside the Containment Structure. For FCS, the Main Steam (MS) and Main Feedwater (MFW) systems contain piping for which CUFs have been evaluated for screening.

Fatigue analyses were previously performed for the B31.7 Class I portions of MS and MFW outside the Containment Structure to identify locations with cumulative usage factors greater than 0.1 as one of the criteria for selecting postulated break locations. The Class I portions encompass the piping from the Containment Structure penetrations to the first isolation valves outside the Containment Structure.

For the Class I MFW piping, projection of the CUFs for the period of extended operation does not require either any additional pipe break analysis to be performed or hardware to be installed on the Class I portions of MFW outside the Containment Structure. Also, for the Class I MS piping, projection of the CUFs for the period of extended operation will not require any additional pipe break analysis to be performed or hardware to be installed on the Class I portions of MS outside the Containment Structure.

The circumferential breaks, already postulated, are bounding for all nodes with respect to direction and magnitude of force. Consideration of the period of extended operation will not impact the selection of the bounding locations. The barrel slats, which cover the piping segments, restrain longitudinal movements and jets along the length of the Class I pipe, not just at the postulated break points. In summary, projection of the CUFs used as HELB screening criteria for the period of extended operation will not require any additional pipe break analysis to be performed or hardware to be installed on the Class I piping. The CUFs are in fact not part of the actual analysis, but only represent screening criteria used to select bounding locations. Therefore, the analysis remains valid for the period of extended operation.