DECAY HEAT DROP LINE

LICENSE RENEWAL EVALUATION REPORT

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ABBREVIATIONS

ARDM	- Age-Related Degradation Mechanism
ARDUTLR	- Age-Related Degradation Unique to License Renewal
B&W	- Babcock & Wilcox
B&WOG	- B&W Owners Group
CASS	- Cast Austenitic Stainless Steel
CS	- Carbon Steel
DBE	- Design Basis Event
FSAR	- Final Safety Analysis Report
GLRP	- Generic License Renewal Program
IGA	- Intergranular Attack
IPA	- Integrated Plant Assesment
ITLR	- Important to License Renewal
LAS	- Low Alloy Steel
LCO	- Limiting Condition of Operation
LER	- License Event Report
LOCA	- Loss-Of-Coolant-Accident
LPI	- Low Pressure Injection
MIC	- Microbiologically Influenced Corrosion
MOV	- Motor Operated Valve
NPRDS	- Nuclear Plant Reliability Data System
PPM	- Parts per Million
PSI	- Pounds per Square Inch
PWR	- Pressurized Water Reactor
RC	- Reactor Coolant
RCS	- Reactor Coolant System
SCC	- Stress Corrosion Cracking
SS	- Stainless Steel

1.0 Introduction

1.1 Scope

This report describes a technical evaluation of the effects of age-related degradation of the Decay Heat Drop Line (DHDL) in the PWR environment in B&W-designed nuclear power plants. The report is one of a series of reports developed through the B&W Owners Group (B&WOG) Generic License Renewal Program to be used in support of individual license renewal applications by the B&WOG member utilities. The purpose of the report is to determine what age-management measures, if any, are required to allow continued service of the drop line over the extended operating period associated with license renewal.

The B&WOG has developed an Integrated Plant Assessment (IPA) that governs Component Evaluation for the B&WOG GLRP program. The B&WOG Component Evaluation process determines the type of component aging evaluation to be performed based on the component's Important to License Renewal (ITLR) function. The approach discriminates between component functions for which it is appropriate to consider potential Age-Related Degradation Mechanisms (ARDMs) and those for which specific knowledge of ARDMs is not necessary to maintain safety over the extended operating period. The DHDL is classified as a long-lived passive component. It's failure could cause a system functional failure; therefore, under the B&WOG IPA methodology, an ARDM assessment of the DHDL is required.

The DHDL connects the Reactor Coolant System (RCS) and the Decay Heat Removal system. It runs from a nozzle on an RCS hot leg to the second isolation valve in the DHR system. The DHDL is located entirely within the reactor building. This report covers the elbows, straights, and piping welds. The decay heat nozzle and the DHDL isolation valves are not evaluated in this report.

Sections 1.2 and 1.3 of this report describe the ITLR Systems/Components and their ITLR functions. Sections 1.4 and 1.5 provide a detailed component description and description of operation for the DHDL.

1.2 System ITLR Functions

The DHR system ITLR functions for each of the B&WOG plants are as follows:

ARKANSAS NUCLEAR ONE-1

1. Transfer heat from the reactor core following a LOCA.

2. Provide borated water from the BWST under accident conditions at sufficient concentration to assure shutdown margin and in sufficient supply to fill the RB to an adequate level for recirculation from the RB sump.

3. Provide water to the Building Spray pump suction piping from the BWST and RB sump.

4. Supply water to the HPI pump suction piping from the BWST and RB sump for long term core cooling following a small break LOCA.

5. Provide long-term heat removal from the RB following a LOCA to reduce RB temperature and pressure.

6. Support spent fuel storage control by providing a pathway for SF pool makeup.

7. Provide long-term RCS heat removal by recirculation of the RCS and heat rejection to the SW system.

8. Provide RCS recirculation for boron mixing.

9. Maintain reactor coolant pressure boundary integrity and containment isolation of penetrations that do not serve any accident-consequence-limiting system function.

10. Provide electrical isolation of non-safety-related portions of DHR/LPI from safety-related power supplies.

11. Provide indication of DHR return line isolation.

12. Provide RB water level indication.

CRYSTAL RIVER-3

1. LPI provides borated water to the core for short term cooling and reactivity control.

2. Long term cooling and reactivity control by recirculation of borated water from RB sump.

3. Provide suction head for HPI pump in "piggyback" mode for SBLOCA.

4. Prevent boron stratification/precipitation in the core following LOCA.

5. Support RB heat removal by cooling RB sump fluid during post LOCA recirculation.

DAVIS BESSE-1

Not available.

OCONEE

1 Provide injection of borated fluid from the BWST after postulated DBEs to assure adequate shutdown margin and core cooling.

2. Provide for long term heat removal after LOCAs by recirculating fluid from the reactor building sump.

3. Provide a flow path from the BWST to the HPI and reactor building spray pumps.

4. Supply water from the reactor building sump to the HPI and reactor building spray pumps after a small break LOCA (piggy-back mode).

5. Provide long-term RCS heat removal by recirculation of the RCS fluid and heat rejection to the service water system.

6. Provide RCS circulation for boron mixing (includes dump-to-sump function) to prevent boron precipitation.

7. Provide RCS pressure control while operating in decay heat removal mode.

8. Provide RCS pressure boundary integrity.

9. Provide containment isolation of penetrations.

10. Provide source of RCS inventory addition (FP, SBO).

THREE MILE ISLAND-1

1. Provide makeup for LOCA via LPI & HPI for SBLOCA after BWST drained.

2. Provide water source for RB Spray.

3. Provide RB sump sampling and pH control.

4. Provide RB isolation.

5. Provide makeup/emergency borationvia HPI following SBLOCA.

6. Provide long-term core cooling following SBLOCA (piggyback).

7. Provide containment isolation.

8. Provide chemistry control to RCS.

Since the RCS pressure boundary extends to the second isolation valve of the DHDL, the ITLR functions of the RCS must also be considered. These are:

ARKANSAS NUCLEAR ONE-1

1. Remove core heat and decay heat during and following a DBE and transfer it out of the RCS.

2. Maintain pressure boundary.

3. Provide reactivity control by use of soluble boron.

4. Contain, align, and support the reactor core, and provide for interfaces for reactor control.

5. Provide RCS pressure control during and following a DBE.

6. Provide electrical isolation of non-safety-related portions of RCS from safety-related power supplies.

7. Provide containment isolation of penetrations that do not serve any accident-consequence-limiting system function.

8. Provide essential operator information for performance of safety-related manual actions.

9. Provide forced flow of reactor coolant during power operation for core heat removal and boron mixing.

10. Be capable of exhausting noncondensible gases and steam.

11. Provide a means of RCS leak detection, PCSV and ERV flow indication, and

ERV block valve position indication.

CYSTAL RIVER-3

1. Reactor Core heat removal via OTSG (Natural & Forced Circulation).

2. RC Pressure Boundary integrity.

3. Neutron moderator & reflector , Chemical shim reactivity control.

4. HPI/PORV Cooling flowpath.

DAVIS BESSE-1

Not available.

OCONEE

1. Transfer decay heat from the reactor core to the steam generators without exceeding core thermal limits.

2. Provide a barrier against the release of fission products from the reactor core to the environment.

3. Provide reactor core cooling during all normal plant operating modes and anticipated operating transients.

4. Act as a neutron moderator, neutron reflector, and act as a solvent for the soluble boron used for reactivity control.

5. Provide RCS temperature and pressure control.

6. Provide RCS inventory control.

7. Contain, align, and support the reactor core, and provide for interfaces for reactor control.

8. Be capable of exhausting noncondensible gases and steam (FP).

9. Oil collection and drainage (FP).

10. Provide containment isolation of penetrations (includes isolation capability from

SSF for all RCS lines penetrating containment).

11. Provide post accident fluid sample from steam generator and letdown line to PAM.

12. Provide structural integrity of QA-1 piping and components.

THREE MILE ISLAND-1

1. Transfer heat from reactor core to the secondary side following DBE.

2. Act as a moderator for thermal fission and reactivity control.

3. Provide a boundary to separate fission from environment.

4. Provide instrumentation for safe operation.

1.3 Component ITLR Functions

The DHDL must perform the following function to support RCS and DHR/LPI System functions for the various plants: maintain pressure boundary for reactor coolant. This is a passive function per the Generic Component Summary For GLR Program.

1.4 Component Description

The DHDL is a section of stainless steel piping with a nominal size of 12 inches. It runs from a hot leg to the suction line of the DHR/LPI pumps. The piping is ASME Class 1 up to the second isolation valve. With the exception of the drop line at Crystal River 3, the piping was designed to Nuclear Power Piping, USAS B31.7 Class I. The DHDL at Crystal River 3 was designed to Power Piping, USAS B31.1-1967.

Table 1.1 lists relevant design specifications for the DHDL in the B&WOG plants.

1.5 Component Operation

Normal Operation

The unisolable portion of the DHDL serves as part of the reactor coolant pressure boundary during all modes of operation. During normal operation, isolation of the DHDL from the DHR system is provided by two electric motor operated valves located inside containment. The entire DHDL (up to the second isolation valve) is designed to withstand RCS pressure.

Cooldown

During the initial stages of cooldown, the steam generators are used to cool the RCS from operating temperatures down to temperatures between 250°F to 280°F. Decay heat removal is then initiated by taking suction from the hot leg in RCS loop A through the DHR/LPI pump and DHR cooler, then pumping back to the RCS through the core flood injection nozzles. The DHDL provides the flow path from the hot leg to the suction line of the DHR/LPI pumps and acts as part of the RCS pressure boundary.

Table 1.1 - DH Drop Line Design Specifications

Parameters	ANO-1 [2]	CR-3 [3]	DB-1 [4]	OC-1 [5], [10]	OC-2 [5], [10]	OC-3 [5], [10]	TMI-1 [6]
Design Codes	USAS B31.7 Class I	USAS B31.1-1967	USAS B31.7 Class I	USAS B31.7 Class I	USAS B31.7 Class I	USAS B31.7 Class I	USAS B31.7 Class I
Pipe ID (inch)	12, Sch 140	12, Sch 160	12, Sch 140	12, Sch 140	12, Sch 140	12, Sch 140	12, Sch 140
Design Pressure (psig)	2500	2500	2500	2500	2500	2500	2500
Design Temperature (°F)	650	650	650	650	650	650	650
Hydrotest Pressure (psig)	3125	3125	3125	3125	3125	3125	3125

2.0 Component Evaluation Basis

2.1 Component ITLR Function Classification

This section provides the basis for choosing the appropriate evaluation path (i.e., Condition Monitoring or ARDM path) for the ITLR functions of the DHDL. The classification of ITLR functions as either Passive or Active is based on the Generic Component Summary For GLR Program List [1]. The criteria used to determine whether a particular function can be assured with a condition monitoring program are found in Section 2.2 (Figure 2.3, Block 4) of the B&WOG GLRP Integrated Plant Assessment (IPA) Process Demonstration Report [14].

Passive ITLR Function(s)

The function of the DHDL is to maintain the RCS pressure boundary. This is a passive function per the Generic Component Summary For GLR Program List. Rupture of the DHDL could cause a LOCA and the failure of the RCS and DHR system to perform their ITLR functions. Therefore, this function must be evaluated using an ARDM assessment based on the criteria of Block 4 of the GLRP IPA process. Evaluation of this function is described in Section 2.3 of this report.

2.2 Component Performance History

The historical review of the DHDL utilized the following sources of information:

- PWR Reactor Coolant System License Renewal Industry Report
- Nuclear Plant Reliability Data System (NPRDS)
- Past Availability Reports (annual)
- NRC Bulletins & Generic Letters

2.2.1 PWR Reactor Coolant System License Renewal Industry Report

A review of this industry report revealed the following ARDMs that could affect PWR reactor coolant system components (such as the DHDL which was in the scope of the industry report). They are:

- Fatigue
- Thermal Embrittlement
- Neutron Embrittlement
- Stress Corrosion Cracking (SCC)
- Creep and Stress Relaxation
- Erosion & Erosion/Corrosion

- o Corrosion
- o Wear

These ARDMs were identified from a review/evaluation of nuclear power plant operating experience, relevant laboratory data, and related experience in other industries [11]. These ARDMs are outlined in the GLRP ARDM Manual [12] and considered in Section 3.0 of this report.

2.2.2 NPRDS Data

Utilizing the Nuclear Plant Reliability Data System (NPRDS), the performance history of piping found in the RCS and DHR/LPI system at other Pressurized Water Reactors (PWRs) in the United States was examined. NPRDS indicates that there were no recorded problems with the DHDL specifically. However, NPRDS indicates that there were 28 recorded problems of Class 1 stainless steel piping found elsewhere. Of these 28 recorded problems, only 13 were found to be relevant to the DHDL (i.e., the remaining 15 problems pertained to tube fittings, gaskets, flanges, fasteners, and valves). Seven of the 13 relevant problems were found in the RCS, and 6 of the 13 relevant problems were found in the DHR/LPI system.

Problems Attributed To RCS Class 1 Stainless Steel Piping

Seven of the 13 relevant problems resulted from degradation of the piping welds. Two were detected by routine observations. The remaining five were detected by either operational abnormality, special inspection, or audible or visual alarm. The majority of degradation was attributed to either vibration induced fatigue or thermal fatigue. Of the seven piping weld problems, the following observations were made on plant effects:

- o 2 resulted in no significant effect
- o 1 resulted in a reactor trip
- o 3 resulted in unit off-line
- o 1 resulted in reduced-power operation

Corrective actions usually consisted of repairing/replacing the weld, the affected pipe or fitting, or providing improved piping support.

Problems Attributed To DHRS Class 1 Stainless Steel Piping

Six of the 13 relevant problems resulted from degradation of the piping welds. Four were detected by incidental observation, and two by operational abnormality. The majority of degradation was attributed to vibration induced fatigue. All six problems resulted in no significant effect on plant operations. Corrective actions usually consisted of repairing/replacing the weld and the affected pipe or fittting.

2.2.3 Past Availability and Plant Performance Commmittee Reports

From 1984-1989, there were no listed failures of piping in the DHDL in the B&W Owners Group Availability Experience Reports.

In the 1990 and 1991 Plant Performance Reports, which take the place of the Availability Reports, there were no listed failures of piping in the DHDL.

2.2.4 NRC Bulletins & Generic Letters

NRC Bulletin No. 88-08

Bulletin 88-08, "Thermal Stresses In Piping Connected To Reactor Coolant Systems", issued on June 22, 1988 identified a potential generic problem based on an incident which occured at the Farley Nuclear Plant. The incident which was first reported in NRC Information Notice 88-01, involved a through-wall pipe crack in a six inch diameter Emergency Core Cooling System Line. The crack was attributed to high cycle thermal fatigue resulting from valve leakage. The bulletin identified certain actions and reporting requirements for the licensees. Supplements 1, 2, and 3 to NRC Bulletin 88-08 provided additional information. Supplement 1, dated June 24, 1988, reported a pipe crack in a similar six inch diameter line of the Tihange Unit 2 reactor in Belgium. Supplement 2, dated August 4, 1988, emphasized the need for enhanced ultrasonic testing (UT) and for experienced examination personnel to detect cracks in stainless steel piping. Supplement 3, dated April 11, 1989, provided information on a through-wall crack in the eight inch diameter residual heat removal line of a foreign reactor. Supplement 3 also emphasized the need for sufficient review of the RCS to identify any connected, unisolable piping that could be subjected to thermal stratification; and, again emphasized the importance of taking action, where such piping is identified, to ensure that the piping will not be subjected to unacceptable thermal stresses. Supplements 1, 2, and 3 imposed no new requirements and the actions requested in NRC Bulletin 88-08 remained unchanged.

NRC Bulletin No. 88-08 requested that all light-water-cooled nuclear power reactor licensees respond by taking actions as paraphrased below:

1. Review systems connected to the RCS to determine whether unisolable sections of piping connected to the RCS can be subjected to stresses from temperature stratification or temperature oscillations that could be induced by leaking valves and that were not evaluated in the design analysis of the piping. For those addressees who determine that there are no unisolable sections of piping that can be subjected to such stresses, no additional actions are requested except for the

report required below.

2. For any unisolable sections of piping connected to the RCS that may have been subjected to excessive thermal stresses, nondestructively examine the welds, heat-affected zones and high stress locations, including geometric discontinuities, in that piping to provide assurance that there are no existing flaws.

3. Plan and implement a program to provide continuing assurance that unisolable sections of all piping connected to the RCS will not be subjected to combined cyclic and static thermal and other stresses that could cause fatigue failure during the remaining life of the unit. This assurance may be provided by (1) redesigning and modifying these sections of piping to withstand combined stresses caused by various loads including temporal and spatial distributions of temperature resulting from leakage across valve seats, (2) instrumenting this piping to detect adverse temperature distributions and establishing appropriate limits on temperature distributions, or (3) providing means for ensuring that pressure upstream from block valves which might leak is monitored and does not exceed RCS pressure.

2.3 Component Parts for Passive ITLR Function Evaluation

2.3.1 Component Refurbishment

The DHDL is not subject to a routine refurbishment program as defined in Section 2.2 of the B&WOG GLRP IPA Process Demonstration Report.

2.3.2 Identification of Part/Material Combinations

As per Section 2.2.2 of the B&WOG GLRP IPA Process Demonstration Report, the parts of the DHDL which support the pressure boundary function have been identified. Table 2.1 lists the material of construction for these piping parts.

2.3.3 Part Replacement Program

Component parts which are included in a replacement program that meets the criteria in Section 2.2 of the B&WOG GLRP IPA Process Demonstration Report are not subject to ARDUTLR. The DHDL parts are not included in a replacement program to preclude age-related failure, and therefore require an ARDM evaluation.

Table 2.1 - Material Specifications For DH Drop Line Pressure Boundary Parts

DHDL Part	Material of Construction						
	ANO-1 [7]	CR-3 [8]	DB-1 [9]	OC-1	OC-2	OC-3	TMI-1
12" Elbows	SS A-403, WP316	SS A-403, WP316 & WP304	SS A-403, WP316	SS A-403, WP316	SS A-403, WP316	SS A-403, WP316	SS A-403, WP316
12" Straights	SS A-376, Tp.316	SS A-376, Tp.316 & Tp.304	SS A-376, Tp.316	SS A-376, Tp.316	SS A-376, Tp.316	SS A-376, Tp.316	SS A-376, Tp.316
Piping Welds	SS - see note (1)	SS - see note (1)	SS - see note (1)	SS - see note (1)	SS - see note (1)	SS - see note (1)	SS - see note (1)

ABBREVIATIONS: SS - Stainless Steel

NOTES: (1) Information concerning the material specifications for the welds could not be located. However, it is assumed that the welds have corrosion resistant properties equivalent to the base material.

3.0 ARDM Assessment

This section refers to Block 10 of the GLRP IPA process. Block 10 contains the component and program evaluations for the ARDM path. The process focuses the program evaluation on those component parts that directly support the component ITLR function that is being evaluated. Parts that do not support the ITLR function will not be evaluated.

The ARDMs that were evaluated for the DHDL are outlined in the GLRP ARDM Manual (to be completed in 1994). These ARDMs were selected so as to encompass known metallurgical degradation phenomena that can be evaluated on a component part basis. The review of industry data, sections 2.2.1 through 2.2.4 of this report, was evaluated to reveal aging considerations besides material and environment.

Referring back to Table 2.1, the DHDL is fabricated of austenitic stainless steel.

In sections 3.1 through 3.14, the susceptibility of the DHDL to each ARDM is evaluated based on materials of construction, operating conditions, and information contained in the ARDM manual. An ARDM is considered to be a potential ARDM if the component is fabricated from a susceptible material, and is exposed to conditions which would promote the ARDM.

3.1 Stress Corrosion Cracking

SCC is of concern for austenitic stainless steel in the presence of tensile stress and an aggresive environment that includes dissolved oxygen and chlorides [12]. RCS chemistry controls maintain the oxygen and chloride content of the reactor coolant below levels conducive to SCC. Therefore, SCC is not considered a potential ARDM for the DHDL parts.

At the B&W plants, there have been no reported occurrences of SCC in the stainless steel clad of the primary coolant piping.

3.2 Intergranular Attack (IGA)

Austenitic stainless steels are susceptible to IGA in the presence of an aggresive environment that includes dissolved oxygen and chlorides [12]. RCS chemistry controls maintain the oxygen and chloride content of the reactor coolant below levels conducive to IGA. Therefore, IGA is not considered a potential ARDM for the DHDL.

3.3 Pitting & Crevice Corrosion

The DHDL material is susceptible to pitting and crevice corrosion. However, due to strict RCS chemistry control, pitting and crevice corrosion are not considered potential ARDMs [12].

3.4 Uniform Attack/General Corrosion

The DHDL is not a concern in relation to Uniform Attack/General Corrosion [12]. This is due to the fact that austenitic stainless steels resist corrosive attack by quickly oxidizing to form a protective film.

3.5 Erosion and Erosion-Corrosion

The DHDL is not an erosion and erosion-corrosion concern for two reasons. First, stainless steels are highly resistant to erosion and erosion-corrosion and are considered immune [12]. Second, fluid velocities are low and the RCS is subject to strict chemistry control.

3.6 Microbiological Influenced Corrosion (MIC)

Microbiological Influenced Corrosion is not considered a potential ARDM for the DHDL. The organisms which cause MIC cannot survive at temperatures greater than 140°F. The operating temperature of the RCS and DHR system, along with strict chemistry controls, precludes the growth of organisms which cause MIC [12].

3.7 Neutron Irradiation Embrittlement

The DHDL is not subjected to a sufficiently high neutron flux to allow irradiation embrittlement to occur. Neutron irradiation embrittlement in PWR plants is primarily of concern for the reactor vessel beltline region [12]. Therefore, neutron irradiation embrittlement is not considered a potential ARDM.

3.8 Thermal Embrittlement

For the conditions of PWR operation, the only materials that are susceptible to thermal embrittlement are cast austenitic stainless steels [12]. The DHDL is not constructed of cast austenitic stainless steel. Therefore, thermal embrittlement is not considered a potential ARDM.

3.9 Hydrogen Damage

Hydrogen damage is not considered a potential ARDM for the DHDL. Austenitic stainless steels are immune to hydrogen embrittlement and hydrogen blistering occurs predominantly in low-strength carbon and low-alloy steels [12].

3.10 Fatigue

The Cumulative Usage Factor (CUF) of the DHDL must be updated to address the number of observed cycles. Cycle assessment is included as part of the GLRP Fatigue Plan (i.e., in the scope of 1994-1995 activities).

Due to the fact that the DHDL is exposed to thermal transients, fatigue is considered a potential ARDM for the DHDL parts.

NRC Bulletin No. 88-08 and supplements revealed that stratified flows and thermal striping have caused through-wall thermal fatigue cracks in the welds and stainless steel base metal of the safety injection and residual heat removal piping (see Section 2.2.4 for discussion). To date, no cases of thermal fatigue cracking due to thermal stratification or valve leakage have been reported for the DHDL at the B&W plants.

The B&WOG plants were required by NRC Bulletin No. 88-08 to review their RCS designs to identify any connected, unisolable piping that could be subjected to temperature distributions which would result in unacceptable thermal stresses. Through the staff's acceptance of actions taken by the B&WOG plants in response to Bulletin 88-08, fatigue due to thermal stratification is not a concern for the DHDL (i.e., the plants are within their current licensing bases (CLB)).

3.11 Wear

Since the DHDL contains no moving parts, wear is not considered a potential ARDM.

3.12 Creep

The approximate temperatures where creep occurs are 700°F for low alloy steels, 1000°F for austenitic alloys, and 1800°F for nickel-based alloys. Since RCS and DHR/LPI system temperatures are below the creep range, creep is not considered a potential ARDM [12].

3.13 Stess Relaxation

The DHDL at the B&W-designed plants does not contain bolted or other preloaded parts;

therefore, stress relaxation is not considered a potential ARDM.

3.14 ARDM/DHDL Parts Matrix

Table 3.1, the Potential ARDM/DHDL Parts Matrix, summarizes the ARDM/DHDL part combinations and the classification of the ARDM as potential or non-potential for the part being considered. Unshaded areas designate potential ARDM/DHDL part combinations. Shaded areas designate non-potential ARDM/DHDL part combinations.

Table 3.1 - Potential ARDM/DHDL Parts Matrix

ARDM	DHDL Part - Material		
	Elbows - SS A-403, WP316 & WP304	**Straights - SS A-376, Tp.316 & Tp.304**	**Welds - SS - see note (2)**
SCC			
IGA			
Pitting & Crevice Corrosion			
Uniform Attack/General Corrosion			
Erosion & Erosion-Corrosion			
MIC			
Neutron Irradiation Embrittlement			
Thermal Embrittlement			
Hydrogen Damage			
Fatigue			
Wear			
Creep			
Stress Relaxation			

Notes: (1) Unshaded areas designate potential ARDM/DHDL part combinations.

(2) Information concerning the material specifications for the welds could not be located. However, it is assumed that the welds have corrosion resistant properties equivalent to the base material.

4.0 Existing Age Management Programs

Potential ARDMs were identified in Section 3.0 of this report. Existing programs that serve to manage ARD for the DHDL are discussed in this Section. The 1989 ASME Section XI Inservice Inspection Requirements are discussed in Section 4.1. Additional programs that are in place at some of the B&W plants are briefly discussed in Section 4.2.

4.1 ASME Section XI Inservice Inspection Requirements

Pertaining to all B&W plants, the ASME Section XI requirements listed in Table IWB-2500-1 (any edition) make up the ISI program that addresses this segment of piping. The requirements listed in Table IWB-2500-1 that are relevant to the DHDL are as follows:

Category	Item	Parts Examined	Examination Requirements	Examination Method	Frequency
B-J	B9.10 B9.11	Circumferential Welds	IWB-2500-8	Surface and volumetric	- Each inspection interval
	B9.12	Longitudinal Welds	IWB-2500-8	Surface and volumetric	- Each inspection interval
	B9.30 B9.31	B anch Pipe Connections Welds	IWB-2500-9, -10, and -11	Surface and volumetric	- Each inspection interval
B-K-1	B10.10	**Piping** Integrally Welded Attachments	IWB-2500-13, -14 and -15	Volumetric or surface, as applicable	- Each inspection interval
B-P	B15.50 B15.51	Pressure Retaining Boundary Pressure Retaining Boundary	IWB-5221 IWB-5222	Visual, VT-2 Visual, VT-2	- Refueling - One test per Inspection Interval

4.1.1 Examination Methods

The relevant requirements show that the three types of examination methods used during inservice inspection are visual, surface, and volumetric.

Visual Examination

The relevant requirements of ASME Section XI require that a VT-2 visual examination be conducted once per inspection interval on the DHDL. The VT-2 visual examination is conducted to the following requirements:

(a) The VT-2 visual examination shall be conducted to locate evidence of leakage from pressure retaining components, or abnormal leakage from components with or without leakage collection systems as required during the conduct of system pressure or functional test.

(b) The VT-2 visual examination shall be conducted by examining the accessible external exposed surfaces of pressure retaining components for evidence of leakage.

(c) For components whose external surfaces are inaccessible for direct visual examination VT-2, only the examination of surrounding area (including floor areas or equipment surfaces located underneath the components) for evidence of leakage shall be required.

Surface Examination

A surface examination indicates the presence of surface cracks or discontinuities. It may be conducted by either a magnetic particle or a liquid penetrant method. The magnetic particle method is conducted in accordance with Article 7 of Section V, and the liquid penetrant method in accordance with Article 6 of section V.

Volumetric Examination

A volumetric examination indicates the presence of discontinuities throughout the volume of material and may be conducted from either the inside or outside surface of a component. It may be conducted by any of the following methods: radiographic, or ultrasonic.

4.1.2 Acceptance Standards

Flaw indications revealed as a result of the inservice examination are evaluated by comparing the examination results to the acceptance standards specified in Table IWB-3410-1 of ASME Section XI. If the flaw indications do not exceed the standards of Table IWB-3410-1, the component is acceptable for service. The standards of Table IWB-3410-1 that are relevant to the DHDL are IWB-3514, IWB-3516, and IWB-3522. Flaws detected by visual examinations are corrected in accordance with IWB-3142 prior to continued service, with the following options available: (1) acceptance by supplemental examinations, such as additional surface or volumetric examination intended to further characterize the indication; (2) acceptance by corrective measures or repairs, such as repair welding; (3) acceptance by analytical evaluation, which may involve more frequent inspections of the item; and (4) acceptance by replacement of the item. Flaws detected by volumetric and surface examinations are corrected in accordance with IWB-3132 prior to continued service, with the following options available: (1) acceptance by repair, which requires additional examinations in conjunction with flaw removal or repair of the portion of the component containing the flaw; (2) acceptance by replacement of the item; and (3) acceptance by analytical evaluation, which may involve more frequent inspections of the item.

4.2 Additional Programs

The additional programs for some of the B&WOG plants are as follows:

Arkansas Nuclear One-1

1. Augmented inspection of the DHDL due to IE Bulletin 88-08.

Davis-Besse Unit 1

1. Visual leakage inspection of part of the piping every 18 months (DB-SP-03131).
2. Containment integrated and local leak rate testing.

4.3 Adequacy of Programs

Fatigue is the only potential ARDM that must be considered when evaluating the DHDL for an extended life. Fatigue arises from stresses imposed by plant operating cycles. Fatigue damage is characterized by cracks which grow as the component is subjected to additional alternating stress cycles.

ASME Section XI ISI programs adequately manage the effects of crack propagation and growth and will continue to do so over the extended operating period associated with license renewal. This determination is based on the following considerations:

1. The aforementioned examination methods (visual, surface, and volumetric) used during inservice inspection of the DHDL will detect cracks.

2. Examination results of the DHDL are compared to the acceptance standards established in ASME Section XI prior to continued service.

3. ASME Section XI ISI programs are periodically updated and must be approved for use by the NRC. The updates are written by experts in the appropriate technical fields and take into account operating experience and known aging effects that may not have been considered in the initial design.

The ASME Sectin XI ISI programs will not ensure that the allowable transient cycles are not exceeded over an extended component life. The calculated CUF must be acceptable for the extended period of operation.

5.0 Recommendations

Each plant should update the CUF of the DHDL based on evaluation of transient cycles for its extended life. Cycle assessment is included as part of the GLRP Fatigue Plan (in the scope of 1994-1995 activities).

6.0 References

1. BWNT Document No. 51-1224979-04, "Generic Component Summary For GLR Program".

2. Safety Analysis Report, Arkansas Nuclear One Unit 1, Revision 10, Chapter 4, "Reactor Coolant System," Table 4-6, "Reactor Coolant System Piping Design Data".

3. Final Safety Analysis Report, Crystal River Unit 3, Chapter 4, "Reactor Coolant System," Table 4-6, "Reactor Coolant System Piping Design Data".

4. Updated Safety Analysis Report, Davis-Besse Nuclear Power Station No.1, Chapter 5, "Reactor Coolant System," Table 5.1-7, "Reactor Coolant System Piping Design Data".

5. Final Safety Analysis Report, Oconee Nuclear Station, Chapter 5, "Reactor Coolant System," 1985 Update.

6. Final Safety Analysis Report (Updated - Version), Three Mile Island Nuclear Station - Unit 1, Chapter 4, "Reactor Coolant System".

7. Arkansas Nuclear One Unit 1 Drawing No. 7-DH-3 SH. 1, Revision 11, Large Pipe Isometric Decay Heat Removal From Reactor.

8. Requirement Outline Fabricated Piping, Crystal River Unit 3.

9. Toledo Edison/Davis-Besse Drawing No. Plant Design Standard M-601, Piping Class Sheet.

10. Oconee Nuclear Station Unit 1 Drawing No. OFD - 102A - 1.1, Flow Diagram Of Low Pressure Injection System.

11. EPRI RP-2643-32, "PWR Reactor Coolant System License Renewal Industry Report," May 1992.

12. BWNT Document, "B&WOG GLRP ARDM Manual," Working Draft.

13. NRC Bulletin No. 88-08, "Thermal Stresses In Piping Connected To Reactor Coolant Systems," June 22, 1988.

14. BWNT Document No. 47-1228802-00, "B&W Owners Group Generic License Renewal Program Integrated Plant Assessment Process Demonstration (Summary Report)".